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Final Safety Evaluation Report

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

Volume 3 (Chapters 9 – 15)

Office of New Reactors

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ABSTRACT

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

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

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

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9.1.1 Criticality Safety of Fresh and Spent Fuel Storage and Handling

9.1.1.1 *Regulatory Criteria*

The staff reviewed the economic simplified boiling-water reactor (ESBWR) criticality safety of fresh and spent fuel storage and handling capability in accordance with NUREG–0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants (LWR Edition),” (hereafter referred to as the SRP), Section 9.1.1, Revision 3. The acceptance criteria for the criticality safety of fresh and spent fuel storage and handling are based on compliance with the following requirements:

- Title 10 of the Code of Federal Regulations (10 CFR) Part 50, “Domestic Licensing of Production and Utilization Facilities,” Appendix A, “General Design Criteria for Nuclear Power Plants,” General Design Criterion (GDC) 62, “Prevention of criticality in fuel storage and handling,” as it relates to the prevention of criticality by physical systems or processes preferably by geometrically safe configurations.

- 10 CFR 50.68 as it relates to preventing a criticality accident and to mitigating the radiological consequences of a criticality accident.

Acceptance criteria adequate to meet the above requirements include:

- The criteria for GDC 62 are specified in American National Standards Institute (ANSI)/American Nuclear Society (ANSI/ANS) 57.1, “Design Requirements for Light Water Reactor Fuel Handling Systems”; ANSI/ANS 57.2, “Design Requirements for Light Water Reactor Spent Fuel Storage Facilities in Nuclear Power Plants”; and ANSI/ANS 57.3, “Design Requirements for New Fuel Storage Facilities at Light Water Reactor Plants,” as they relate to the prevention of criticality accidents in fuel storage and handling.

- ANSI/ANS 57.1, ANSI/ANS 57.2, ANSI/ANS 57.3, and Regulatory Guide (RG) 1.13, Revision 2, “Spent Fuel Storage Facility Design Basis,” provide guidance acceptable to the staff for meeting the requirements associated with spent fuel storage and handling.

- 10 CFR 50.68(a) requires that the licensee either maintain monitoring systems capable of detecting a criticality accident, as described in 10 CFR 70.24, thereby reducing the consequences of a criticality accident, or comply with the requirements specified in 10 CFR 50.68(b), thereby reducing the likelihood that a criticality accident will occur.

9.1.1.2 *Summary of Technical Information*

Design control document (DCD), Tier 2, Revision 9, Section 9.1 describes the fuel storage and handling design bases of the ESBWR design. Fresh fuel is intended to be stored in new fuel racks in the Reactor Building (RB) buffer pool, and can also be stored in the spent fuel racks in the Fuel Building (FB), along with spent fuel assemblies. A small array of spent fuel assemblies can be stored in the RB buffer pool deep pit storage area during refueling. Both the new and spent fuel storage areas are designed to maintain a subcritical storage configuration during

normal storage and accident conditions. DCD Tier 2, Revision 9, Section 9.1, references licensing topical report (LTR) NEDC-33374P-A, Revision 4, "Safety Analysis Report for Fuel Storage Racks Criticality Analysis for ESBWR Plants," to document the analyses of storage rack criticality. NEDC-33374P-A, Revision 4, provides the detailed discussion of the criticality analyses and results for the ESBWR spent fuel and buffer pools for the storage of fuel bundles in the new and spent fuel storage racks.

9.1.1.2.1 New Fuel Storage

DCD Tier 2, Revision 9, Section 9.1.1, provides the design bases, a description, and a safety analysis of the new fuel storage arrangement for the ESBWR design. The new fuel storage racks in the RB buffer pool can store 476 new fuel assemblies. The fresh fuel assemblies are stored in underwater storage racks located adjacent to the reactor well. The racks have double rows of storage positions for assemblies that are side loaded into the storage racks. The new fuel storage racks in the buffer pool are designed with sufficient separation between new fuel bundles to ensure that the fully loaded array has an effective multiplication factor (k_{eff}) that does not exceed 0.95. Monte Carlo techniques are employed in the calculations performed to assure that k_{eff} does not exceed 0.95 under all normal and abnormal conditions.

The design of the new fuel storage racks provides for a k_{eff} for storage conditions equal to or less than 0.95. To ensure that design criteria are met, the applicant analyzed the following normal and abnormal new fuel storage conditions:

- Normal positioning in the new fuel array
- Eccentric positioning in the new fuel array

9.1.1.2.2 Spent Fuel Storage

DCD Tier 2, Revision 9, Section 9.1.2, provides the design bases, a description, and a safety analysis of the spent fuel storage arrangement for the ESBWR design. The fuel storage racks provided in the spent fuel pool (SFP) in the FB provide for the storage of 3,504 irradiated fuel assemblies. An additional 154 spent fuel assemblies can be temporarily stored in the RB buffer pool deep pit during refueling. Combined, the spent fuel storage capacity is sufficient for 10 calendar years of plant operation, plus one full core offload. The racks are composed of borated stainless steel plates forming individual cells, with an outer stainless steel frame.

The same criteria utilized in the design of the new fuel storage racks were applied to the spent fuel racks. That is, the design provides for a k_{eff} for storage conditions equal to or less than 0.95. To ensure that the design criteria are met, the applicant analyzed the following normal and abnormal spent fuel storage conditions:

- Normal positioning in the spent fuel array
- Eccentric positioning in the spent fuel array

The applicant also evaluated the effects of pool moderator temperature on criticality.

To control SFP reactivity, borated stainless steel storage racks are used as part of a strategy to maintain a k_{eff} that does not exceed 0.95 for all normal and abnormal loading scenarios including earthquake and load drop. The fuel storage cells are also spaced such that they are less than one fuel assembly apart to preclude inadvertent assembly insertion between the racks.

9.1.1.3 *Staff Evaluation*

The staff verified that the design complied with the requirements of GDC 62. The applicant committed to meet the guidance of RG 1.13, ANSI/ANS 57.1-1992, and ANSI/ANS 57.2-1983 for spent fuel storage and ANSI/ANS 57.3-1983 for new fuel storage.

DCD Tier 2, Revision 9, Section 9.1, references NEDC-33374P-A, Revision 4, for detailed discussion of the criticality analyses and results for the ESBWR spent fuel and buffer pools for the storage of fuel bundles in the new and spent fuel storage racks. The report includes sufficient detail on the methodology and analytical models utilized in the criticality analysis to verify that the storage rack systems have been accurately and conservatively represented.

The safety evaluation (SE) for NEDC-33374P provides the detailed staff review. The staff review included assessment of the applicant's proposed criticality analysis methodology, analysis model inputs and assumptions, the criticality analysis results, computer code qualification using relevant benchmarks, and the biases and uncertainties considered in the analyses.

To confirm that the analyses used appropriate fuel assembly and storage rack data, the staff reviewed design specifications and drawings for both the new and the spent fuel storage racks during a February 11-12, 2009, audit held at the applicant's Washington, DC, facility. A summary of the audit, including participants and audit activities may be found in the Agencywide Documents Access and Management System (ADAMS) at Accession Number ML101450301. The staff also reviewed detailed design calculations and computer program documentation. The staff finds that the applicant had appropriately documented and utilized the detailed design data included in the calculations. The staff finds in the SE for NEDC-33374P that the applicant's methodology is consistent with that approved for new and spent fuel storage criticality evaluations for operating boiling-water reactors (BWR).

During the course of the DCD review, the staff determined that the DCD contained no Tier 1 requirement to maintain subcriticality in the new fuel pool and SFP. Additionally, the DCD did not include a provision to verify that the installed racks would be within acceptable tolerances, consistent with the analyses. In request for additional information (RAI) 14.3-457, the staff requested that the applicant identify parameters important to the criticality safety analyses and specify acceptance criteria. In response, the applicant provided DCD markups that added the Tier 1 new and spent fuel rack subcriticality design requirement and provided inspection, test, analysis and acceptance criteria (ITAAC) in DCD Tier 1, Table 2.5.6-1. As requested by the staff in the RAI, NEDC-33374P was designated as a Tier 2* document, thus requiring that any changes to the design or analysis input be provided to the NRC for review. The staff finds that the applicant's response is acceptable since the ITAAC are based on the essential parameters for criticality safety identified in Appendix A to NEDC-33374P, Revision 3. Accordingly, based on the above and the applicant's response, RAI 14.3-457 is resolved. The staff confirmed that DCD Revision 7 incorporated the changes.

The scope of the criticality safety analyses presented in NEDC-33374P-A, Revision 4, is limited to the analysis of fuel storage racks in the FB and in the buffer pool in the RB, and no analysis is provided in the topical report for handling of fresh and spent fuel. Section 9.1.6 of DCD Tier 2, Revision 9, includes Combined License (COL) Information Item 9.1-4-A, which requires a COL applicant to address the criticality safety of fresh and spent fuel handling. Criticality safety of fuel handling need not be evaluated in the design certification application, but may be evaluated in the COL application. The scope of the analyses in NEDC-33374P-A, Revision 4, and COL

Information Item 9.1-4-A includes all applicable criticality safety issues for new and spent fuel storage and handling. Therefore, the staff finds COL Information Item 9.1-4-A acceptable.

In the SE for NEDC-33374P, the staff finds that the applicant demonstrated by analyses in NEDC-33374P, Revision 3, that the fuel to be stored in new or spent fuel racks remains subcritical under all normal and credible abnormal conditions. The racks are designed and located within the spent fuel and buffer pools such that sufficient separation is maintained between fuel bundles to preclude criticality under all normal and credible abnormal conditions. Additionally, the spent fuel racks are composed of borated stainless steel. Therefore, the staff finds that the applicant has addressed the requirements of GDC 62 regarding the criticality of new and spent fuel storage.

Based on the above, the staff finds that the criticality safety of fresh and spent fuel storage and handling meets the requirements of GDC 62.

The staff verified that the design complied with the requirements of 10 CFR 50.68. The applicant has addressed the requirements of 10 CFR 50.68 by demonstrating compliance with the additional design and analysis requirements specified in 10 CFR 50.68 (b)(2) or 10 CFR 50.68(b)(4). No credit is taken for soluble boron, so the k_{eff} of the new and spent fuel racks must not exceed 0.95 at a 95 percent probability and a 95 percent confidence level, if flooded with unborated water. The analyses provided in NEDC-33374P-A, Revision 4, demonstrate this. As discussed above, COL Information Item 9.1-4-A would require the COL applicant to address the criticality safety of fresh and spent fuel handling. Accordingly, the staff finds that the criticality safety of fresh and spent fuel storage and handling meets the requirements of 10 CFR 50.68.

9.1.1.4 *Conclusion*

Based on the above discussion, the staff concludes that the ESBWR criticality safety of fresh and spent fuel storage and handling conforms to the requirements of GDC 62 and 10 CFR 50.68.

9.1.2 New and Spent Fuel Storage

9.1.2.1 *Regulatory Criteria*

The staff reviewed the ESBWR spent fuel storage capability in accordance with SRP Section 9.1.1, Revision 3, and SRP Section 9.1.2, Revision 3. The staff performed a comparison of the SRP version used during the review (i.e., the 1981 version) with the 2007 version of the SRP Section 9.1.2 (Revision 4). The following are the major review areas included in the 2007 version of the SRP, but not in the 1981 version: (1) the new fuel vault, (2) new fuel storage racks, (3) new fuel criticality monitoring requirements, (4) as low as reasonably achievable (ALARA) considerations, (5) thermal-hydraulic considerations, (6) ways in which the design would preclude load drops on new and spent fuel, (7) radiological shielding of personnel by maintaining adequate water levels in the SFP and buffer pool, (8) the ability to maintain adequate coolant inventory in the SFP and buffer pool under accident conditions, (9) avoidance of high density storage racks for hot fuel, (10), methods of preventing pool draining, (11) ability to place a fuel assembly around the periphery of the SFP or the buffer pool, (12) increased minimum amount of fuel that can be stored, and (13) use of appropriate monitoring systems to detect the SFP and buffer pool water levels, pool temperatures, and building radiation levels. The 2007 version added regulatory requirements from 10 CFR 20.1101 and 10 CFR 50.68.

Section 9.1.2 in the 2007 version of the SRP included the discussion of new fuel storage, which was previously in Section 9.1.1 of the 1981 version of the SRP.

Although these items were not included in the SRP version used by the staff, the staff did address the additional items from SRP Section 9.1.2 Revision 4. Section 9.1.2.3 of this report discusses the evaluation of the new fuel vault and the new fuel storage racks. Note that the ESBWR does not have a facility designated as a new fuel vault because new fuel may be stored either in the SFP or the RB buffer pools. The staff addressed the SRP Section 9.1.2 guidelines regarding new fuel vaults for these facilities in Section 9.1.2.3 of this report. Sections 9.1.1.3 and 9.1.2.3 of this report include the evaluation of new fuel criticality monitoring requirements. The applicant addressed the new fuel criticality monitoring requirements of 10 CFR 50.68 by complying with the additional design and analysis requirements specified in 10 CFR 50.68 (b)(2) as described in Section 9.1.1.3 of this report. Chapter 12 of this report includes the evaluation of ALARA considerations. Section 9.1.2.3 of this report evaluates thermal-hydraulic considerations. Sections 9.1.4 and 9.1.5 of this report evaluate how the design would preclude load drops on new and spent fuel. Sections 9.1.2.3, 9.1.3.3, and 9.1.4.3 of this report present the evaluation of provisions for radiological shielding of personnel by maintaining adequate water levels in the SFP and buffer pool. Section 9.1.2.3 and 9.1.3.3 of this report evaluate the ability to maintain adequate coolant inventory in the SFP and buffer pool under design-bases accident conditions. Sections 9.1.2.3 and 9.1.3.3 of this report discuss the evaluation of methods of preventing pool draining. Section 9.1.2.3 of this report evaluates the ability to place a fuel assembly around the periphery of the SFP or the buffer pool. This section also evaluates the increased minimum amount of fuel that can be stored in an SFP. The staff considers the use of appropriate monitoring systems to detect the SFP and buffer pool water levels in Section 9.1.3.3.3.1 of this report.

Section 9.1.2 in the 2007 version of the SRP deleted the reference to GDC 62 because the evaluation of criticality with respect to fuel storage was moved in its entirety to SRP Section 9.1.1.

The acceptance criteria for the new and spent fuel storage facilities are based on compliance with the following requirements:

- GDC 2, “Design bases for protection against natural phenomena,” as it relates to the ability of SSCs important to safety to withstand the effects of natural phenomena, such as earthquakes, tornados, and hurricanes

- GDC 4, “Environmental and dynamic effects design bases,” as it relates to the protection of SSLs important to safety from dynamic effects, including the effects of external missiles and internally generated missiles, pipe whip, jet impingement forces, and adverse environmental conditions associated with pipe breaks

- GDC 5, “Sharing of structures, systems, and components,” as it relates to whether the ability of shared structures, systems, and components SSCs important to safety to perform safety functions is not significantly impaired

- GDC 61, “Fuel storage and handling and radioactivity control,” as it relates to the facility design provisions for safe fuel storage and handling of radioactive materials

- GDC 63, “Monitoring fuel and waste storage,” as it relates to monitoring systems provided to detect conditions that could result in the loss of decay heat removal capabilities, to detect excessive radiation levels, and to initiate appropriate safety actions

- 10 CFR 20.1101(b) as it relates to provisions to achieve public and occupational doses that are ALARA

- 10 CFR 20.1406, as it relates to the minimization of contamination

- 10 CFR 50.68 as it relates to criticality

9.1.2.2 *Summary of Technical Information*

9.1.2.2.1 New Fuel Storage

In DCD Tier 2, Revision 9, Section 9.1.1, the applicant provided the design bases, a description, and a safety evaluation (SE) of the new fuel storage arrangement for the ESBWR design. Upon receipt of the new fuel bundles at the reactor site, the fuel bundle containers are uncrated from the shipping crate, and the fuel bundle container is raised to the refueling floor in the FB. The fuel bundles are removed from the container and moved to the new fuel inspection stand where they are inspected and the fuel channels are installed. Once the fuel bundles are assembled, they are placed in the SFP in the FB or in the inclined fuel transfer system (IFTS) for transfer to the RB. The channeled fuel assemblies are then moved to the new fuel storage racks in the RB buffer pool until it is time to move them into the reactor.

The new fuel storage racks are constructed of stainless steel plates which form a 14 x 2 array of storage cells. These racks are located underwater in the RB Buffer Pool adjacent to the reactor well and hold up to 476 new fuel assemblies. Fuel assemblies will be loaded from the side of the racks and stored horizontally. Because the racks are open on the side to allow side loading, the weight of the fuel assemblies placed in the storage position actuates a mechanism that restrains the assemblies in position. The racks are floor mounted. Since only fresh fuel will be stored in the new fuel racks, and no decay heat will be generated by this fuel, cooling is not needed for the new fuel racks. Hence, a thermal-hydraulic analysis is not necessary.

Two fuel preparation machines are mounted on the wall of the SFP and are used to assist in the loading of new fuel into the spent fuel storage pool racks and for channeling and rechanneling of new and spent fuel assemblies.

The new fuel inspection stand is a vertical frame mounted in a pit that supports two fuel bundles, which are contained in a mechanically-driven inspection carriage. In the carriage, the lower tie plate of each fuel bundle rests on a bearing seat, and at the top each fuel assembly is supported in a separate bearing assembly. The fuel assemblies can be individually rotated about their longitudinal axis to permit viewing of all sides. The fuel channel is placed on the fuel bundle in the new fuel inspection stand. To facilitate fuel inspection, the stand is set into an inspection pit designed to allow the carriage to be lowered and raised, permitting eye-level viewing by inspecting personnel on the refueling floor.

9.1.2.2.2 Spent Fuel Storage

The fuel storage racks provided in the SFP in the FB provide for storage of 3,504 irradiated fuel assemblies. In addition, a small array of spent fuel assemblies (154) can be stored temporarily

in the RB Buffer Pool during refueling. Combined, the spent fuel storage capacity is sufficient for 10 calendar years of plant operation, plus one full core offload. The racks comprise of borated stainless steel plates forming individual cells, with an outer stainless steel frame. Cooling water enters the pool at the bottom, near the corners opposite the racks. The racks are located on the side of the SFP opposite the cooling water inlet diffusers. The rack design allows sufficient natural circulation upflow through individual storage cells to remove the decay heat generated. The fuel and auxiliary pools cooling system (FAPCS) recirculation flow removes bundle decay heat to maintain the SFP temperature below 48.9 degrees Centigrade (C) (120 degrees Fahrenheit [F]) during normal conditions (defined as 10 years of spent fuel accumulation). For abnormal conditions (defined as 10 years of spent fuel accumulation plus a full core offload), the SFP temperature will be maintained below 60 degrees C (140 degrees F).

The fuel storage racks in the RB buffer pool and in the SFP in the FB contain storage space for fuel assemblies (with channels) or bundles (without channels). A standard dynamic analysis using the appropriate response spectra is performed to demonstrate conformance to design requirements. The applicant performed a dynamic loads analysis to determine the capability of the spent fuel storage racks to withstand the combined loads of the (1) deadweight plus buoyancy load, (2) fuel handling loads, (3) thermal effect, (4) safe-shutdown earthquake (SSE), (5) safety relief valve discharge (SRVD) load, and (6) loss-of-coolant accident (LOCA) load. Furthermore, the racks are designed to protect the fuel assemblies and bundles from excessive physical damage that may cause the release of radioactive materials in excess of the requirements of 10 CFR Part 20, "Standards for Protection Against Radiation," and the guidelines in 10 CFR Part 52.47(a)(2)(iv) under normal and abnormal conditions caused by impact from fuel assemblies, bundles, or other equipment.

The SFP is a reinforced concrete structure with a stainless steel liner. The fuel storage racks and pool liner are designed to meet seismic Category I requirements (i.e., they must remain functional during and after ground motion up to the SSE). The bottoms of the pool gates are higher than the design basis minimum water level required over the spent fuel storage racks to provide adequate shielding and cooling. Pool fill and drain lines enter the pool above the safe shielding water level. Redundant anti-siphon vacuum breakers are located in the pool circulation lines to preclude a pipe break from siphoning the water from the pool to a point lower than the safe shielding level. The racks include individual solid tube storage compartments, which provide lateral restraints over the entire length of the fuel assembly or bundle. The weight of the fuel assembly or bundle is supported axially by the rack fuel support. Lead-in guides at the top of the storage spaces provide guidance of the fuel during insertion.

To control SFP reactivity, borated stainless steel storage racks are used as part of a strategy to maintain a k_{eff} equal to or less than 0.95 for all normal and abnormal loading scenarios, including earthquake and load drop. The fuel storage cells are also spaced such that they are less than one fuel assembly apart to preclude inadvertent assembly insertion in the racks.

9.1.2.2.3 Thermal-Hydraulic Design

DCD Tier 2, Revision 9, Section 9.1.2.5, discusses the spent fuel rack cooling design. The fuel storage racks are designed to allow sufficient natural convection coolant flow through the rack and fuel bundles to remove decay heat without exceeding the temperature limit for stress properties of the various fuel rack materials, which is 121 degrees C (250 degrees F). FAPCS recirculation flow provides rack cooling in the spent fuel and buffer pools to maintain the bulk pool temperature below 48.9 degrees C (120 degrees F) during normal conditions and 60

degrees C (140 degrees F) during the abnormal conditions previously defined. The fuel storage racks are designed to prevent nucleate boiling in the event both FAPCS cooling trains are lost.

9.1.2.2.4 Storage Rack Cooling Analyses

Section 5 of NEDO-33373, Revision 4, "Dynamic, Load-Drop, and Thermal-Hydraulic Analyses for ESBWR Fuel Racks," summarizes the applicant's computational fluid dynamics (CFD) analyses performed by the applicant for the spent fuel storage racks to determine the maximum peak temperatures at the exit of the fuel racks resulting from both normal and abnormal conditions, as defined above. The maximum pool inlet temperature from the FAPCS is computed for a steady-state, steady-flow process. This is calculated for both a "normal" and "abnormal" case by assuming a steady-state condition in which the pool bulk temperature (also equivalent to the pool outlet temperature) is fixed at its maximum value. The approach is intended to result in a higher-than-normal bulk pool temperature to use the minimum allowed heat removal capability of the FAPCS.

The decay heat generated by the fuel elements accumulated during 10 years of operation and the decay heat resulting from a full core offload are determined in a separate referenced calculation. The maximum inlet temperature of the water in each of the cases is determined using the maximum bulk temperature, the flow rate provided by the FAPCS, and the corresponding decay heat load.

Directional flow losses as a function of velocity through the racks and fuel are input to the CFD model, and are developed to bound all BWR fuel bundle designs. An 8 percent safety factor was applied to the calculation.

To simulate the heat generation produced by the fuel assemblies stored inside the racks, a volumetric heat generation was applied to the fully-loaded racks within the SFP. To bound potential loading configurations, the applicant assumed that the most recently discharged bundles (those producing the most decay heat) are located together in the SFP racks. The temperature reached with this configuration is greater than the temperature that would be reached if the discharged assemblies were distributed uniformly among all of the racks in the SFP.

The CFD model represents the SFP water and rack configuration loaded with 10 years of fuel accumulation. Two FAPCS inlets have been modeled at the bottom of the SFP in the corners opposite the racks. Two FAPCS outlets have been modeled at the top of the SFP above the exit of the storage racks. For the abnormal conditions case, the recently discharged full-core offload is assumed to be located in the racks farthest from the cooling inlets to bound potential loading configurations.

The CFD code used to perform the thermal-hydraulic analyses solves the momentum and energy equations in the storage racks as a function of mass flow rate through the racks and fuel bundles, the external pressure gradient, internal heat generation from the spent fuel decay heat, and pressure drop across the racks and stored fuel. Mass flow rate and inlet temperatures are entered into the CFD model, and the code calculates the velocity distribution in the SFP and through the racks.

The applicant evaluated the effects of modeling assumptions and methods (turbulence model selection, buoyancy treatment, or mesh density) through sensitivity studies. These studies included the variation of input parameters, such as inlet mass flow rate, inlet temperature, loss

coefficient, turbulence model, reference temperature for buoyancy model, and inlet turbulence intensity.

The CFD results show that the rack exit temperatures can be maintained less than the design temperature for both the normal and abnormal cases. The maximum local temperature can also be maintained less than the design value. A significant margin between the maximum allowable temperature for both the normal and the abnormal loading cases is calculated. Based on the CFD results, the maximum fuel cladding temperature remains below the boiling point of water at the depth of the spent fuel racks. This prevents bulk boiling in the racks and nucleate boiling on the fuel assemblies.

Although not required by regulations, the applicant provided additional analyses assuming an 80 percent blockage of the storage rack exit flow area. This condition would represent the expected blockage resulting from a collapsed pool liner plate section or other foreign object. The resulting temperatures are below the acceptance temperatures of 48.9 degrees C (120 degrees F) during normal conditions and 60 degrees C (140 degrees F) during a full core offload.

NEDO-33373 includes a conservative calculation of the maximum local fuel cladding temperature. Algebraic conservation of energy equations are solved for the pool water to fuel rod heat transfer. The decay heat load is increased by 20 percent to provide margin, and a foulant layer (crud deposit) is assumed on the fuel cladding surface.

9.1.2.3 *Staff Evaluation*

The staff reviewed the new fuel storage facilities for the ESBWR standard design in accordance with the guidance of SRP Section 9.1.1, Revision 2, with supplementary information from SRP Section 9.1.2, Revision 4. Compliance with GDC 2 is based in part on adherence to the guidance of Regulatory Position C.1.1 of RG 1.29, Revision 4, “Seismic Design Classification,” as it relates to the seismic classification of facility components. The ESBWR design is a single-unit station, and the requirements of GDC 5 are not applicable to the single-unit design. In accordance with SRP Section 9.1.1, specific criteria necessary to meet the requirements of GDC 61 and 62 are ANSI/ANS 57.1-1992 and ANSI/ANS 57.3-1981, as they relate to preventing criticality and to other aspects of the radiological design.

The staff identified that DCD Tier 2, Revision 4, Section 9.1.1, did not have statements to indicate that the new fuel storage conforms to GDC 2, ANSI/ANS 57.1, or ANSI/ANS 57.3, thereby meeting the requirements of GDC 2, 61, and 62. In RAI 9.1-39, the staff requested the applicant to address compliance with the above GDC. In response, the applicant provided a markup of DCD Tier 2, Section 9.1.1, which addresses the required GDC. The staff finds that the RAI response and DCD markup are acceptable since the new fuel storage racks are designed to meet the requirements of GDC 2 as described below. Accordingly, based on the above and the applicant’s response, RAI 9.1-39 is resolved. RAI 9.1-39 was being tracked as a confirmatory item in the Safety Evaluation Report (SER) with open items. The staff confirmed that the applicant incorporated the above changes into DCD Tier 2, Revision 6, and the confirmatory item is resolved.

The staff reviewed the spent fuel storage facilities for the ESBWR standard design in accordance with the guidance of SRP Section 9.1.2, Revision 3, with supplementary information from SRP Section 9.1.2, Revision 4. Compliance with the requirements of GDC 2 may be demonstrated by adherence to the guidance of Regulatory Position C.2 of RG 1.13, Revision 2;

the applicable portions of RG 1.29, Revision 4; and RG 1.117, Revision 1, "Tornado Design Classification"; and paragraphs 5.1.1, 5.1.3, 5.1.12, 5.3.2, and 5.3.4 of ANSI/ANS 57.2. Compliance with the requirements of GDC 4 may be demonstrated by adherence to the guidance of Regulatory Position C.3 of RG 1.13, as well as RG 1.115, Revision 1, "Protection against Low-Trajectory Turbine Missiles"; RG 1.117, Revision 1; and the appropriate paragraphs of ANSI/ANS 57.2. The ESBWR design is a single-unit station, and the requirements of GDC 5 do not apply to the single-unit design. Compliance with the requirements of GDC 61 may be demonstrated by adherence to the guidance of Regulatory Positions C.1 and C.4 of RG 1.13, the appropriate paragraphs of ANSI/ANS 57.2, and the fuel storage capacity guidelines noted in SRP Section 9.1.2.III.1. Compliance with the requirements of GDC 63 may be demonstrated by adherence to the guidance of paragraph 5.4 of ANSI/ANS 57.2 and Regulatory Position C.7 of RG 1.13.

Compliance with 10 CFR 20.1101(b) may be demonstrated by adherence to the guidance of Regulatory Positions C.2.f(2) and C.2.f(6) of RG 8.8, Revision 3, "Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations Will Be As Low As Is Reasonably Achievable," paragraph 5.1.5 of ANSI/ANS 57.2, and appropriate regulatory positions of RG 1.13. For new fuel storage, compliance may be demonstrated by adherence to paragraphs 6.3.3.7 and 6.3.4 of ANSI/ANS 57.3. Chapter 12 of this report discusses compliance with 10 CFR 20.1101(b). Finally, 10 CFR 50.68 can be met by following 10 CFR 70.24 for criticality monitors or the requirements in section 50.68(b) described therein for significant margins of subcriticality. Compliance with 10 CFR 50.68 is discussed below and in Section 9.1.1.3 of this report.

While DCD Tier 2, Revision 3, Section 9.1.2, provided several design bases, it did not address directly, compliance with GDC 2, 4, 61, and 63. In RAI 9.1-45, the staff requested the applicant to revise the DCD to address compliance with GDC 2, 4, 61, 62, and 63 and conformance to associated RGs and industrial standards for spent fuel storage, in accordance with the SRP. RAI 9.1-45 was being tracked as an open item in the SER with open items. In response the applicant identified modifications to DCD Tier 2, Section 9.1.2 to address compliance with the GDC. The staff finds that the RAI response and DCD markup are acceptable since the applicant revised Section 9.1.2.1 to address GDC 2, 4, 61, 62 and 63. In addition, the applicant described each GDC in detail, along with the applicable guidance of RGs and other standards such as those of ANSI/ANS. Accordingly, based on the above, the applicant's responses, and DCD changes, RAI 9.1-45 is resolved.

In Section 9.1.6 of DCD Tier 2, Revision 3, the applicant identified COL holder items relating to dynamic, impact, thermal-hydraulic, and criticality analyses of the fuel storage racks. The staff determined that the above three COL holder items are analysis and design issues that the NRC staff must review in its review of a COL application, if they are not within the scope of the design certification application. The information provided by COL holder items would be available for review only after a license is issued. This is not acceptable, because the staff would not be able to conclude, at the time the license is issued, whether the design and analysis of the spent fuel storage facility satisfy regulatory requirements. In RAI 9.1-40, the staff requested that the applicant revise the three COL holder items to make them COL applicant items. In response, the applicant stated that it would submit two LTRs to provide fuel rack analyses, NEDO-33373, and NEDC-33374P. Therefore, the COL holder items are no longer required. The staff finds the response acceptable since the LTRs include the topics in the COL holder items. Furthermore, including the LTRs in the design certification addresses the need for the information at an appropriate stage of the process. Accordingly, based on the above and the applicant's response, RAI 9.1-40 is resolved. However the staff found that it was unable to

complete its evaluation of GDC 2, 61, and 62 before the submission of the fuel rack analyses. Accordingly, the review of NEDO-33373 and NEDC-33374P was being tracked as an open item in the SER with open item.

The staff SEs on NEDO-33373 and NEDC-33374P document the staff review of these reports. For NEDO-33373, the staff evaluated compliance with GDC 2, 4, and 61, which is summarized with the corresponding GDC below. For NEDC-33374P, the staff evaluated compliance with GDC 62 and 10 CFR 50.68, which is summarized in Section 9.1.1 of this report. With the submission and review of NEDO-33373 and NEDC-33374P, this open item is resolved.

9.1.2.3.1 GDC 2

The staff verified that the design complied with the requirements of GDC 2. The applicant committed to meet the guidance of the 1981 versions of SRP Sections 9.1.1 and 9.1.2, as well as RGs 1.13, 1.29, and 1.117 and ANSI/ANS 57.2-1983 for spent fuel storage and ANSI/ANS 57.3-1983 for new fuel storage. In RAI 9.1-6, the staff asked the applicant to clarify whether the SFP and buffer pool liners are designed to seismic Category I requirements. In response, the applicant stated that the FB SFP and RB buffer pool liners and liner anchors are designed to seismic Category I requirements, and the loads and load combinations are the same as those for the pool concrete structure (except load factors for all cases are equal to 1.0 and the acceptance criteria follow ASME Section III, "Rules for Construction of Nuclear Power Plant Components," Division 2, "Code for Concrete Reactor Vessels and Containments," CC-3700, of the American Society of Mechanical Engineers [ASME] Code.) The staff finds that the RAI response and DCD markup are acceptable since the applicant adequately addressed the seismic category of the SFP and buffer pool liners along with the applicable ASME Code and loading combinations. The staff confirmed that DCD Tier 2, Revision 6, included these criteria. Accordingly, based on the above and the applicant's response, RAI 9.1-6 is resolved.

The applicant stated that fuel storage racks and pool liners in the SFP and the buffer pool are designed to meet seismic Category I requirements. DCD Tier 2, Revision 9, Section 9.1.2.4, describes the loads applied to the rack. The applicant stated that stress analyses are performed by classical methods based upon shears and moments developed by a dynamic method. Using the given loads, load conditions, and analytical methods, stresses are calculated at critical sections of the rack and compared to acceptance criteria referenced in ASME Code Section III, Subsection NF, "Supports." NEDO-33373, which is evaluated by the staff in the SE for NEDO-33373, provides additional discussion of the stress analysis and documents its results.

Both the RB and the FB, which contain the fuel storage facilities, including the storage racks and pools, are designed and constructed to accommodate the dynamic and static loading conditions associated with (1) natural phenomena, such as wind, floods, tornadoes, earthquakes, rain, and snow, and (2) internal events, such as floods, pipe breaks, and missiles. Section 3.5 of this report discusses protection from flooding and missiles (external and internal).

In RAI 9.1-1, the staff requested that the applicant describe the seismic qualification of the fuel preparation machines and the new fuel inspection stand. In response, the applicant stated that the fuel preparation machine is analyzed as seismic Category II[1] to maintain its structural integrity during an SSE event to prevent possible damage to the pool structure or adjacent fuel

[1] Seismic Category II SSCs are not required to be functional following an SSE, but are required to not fail in the event of an SSE in a manner that would prevent a seismic Category I SSC from performing its intended function.

storage racks. The applicant also stated that the fuel-handling machine is only capable of handling one fuel assembly near the fuel preparation machine. DCD Tier 2, Section 9.1.2.4, identifies that the racks are designed to withstand the impact force generated by the accidental drop of the heaviest fuel assembly from the maximum possible height. The applicant also stated that, since there can be, at most, two fuel assemblies adjacent in this scenario, the array will remain subcritical. The staff finds that the RAI response and DCD Tier 2 markup of Table 9.1-4 are acceptable for the fuel preparation machine since the applicant identified it as seismic Category II. However, the staff did not find the seismic classification of the fuel inspection stand acceptable. In RAI 9.1-36, the staff requested that the applicant identify the seismic design classification for the new fuel inspection stand. In response, the applicant clarified that the new fuel inspection stand is dynamically analyzed and that the new fuel inspection stand cannot damage adjacent equipment, as no other equipment is present in the pit. The applicant further indicated that it would revise Table 3.2-1 and Table 9.1-4 to identify that the new fuel inspection stand must be seismic Category II. The staff finds this clarification acceptable. The applicant made these modifications in DCD Tier 2, Revision 5. Subsequently, Table 9.1-4 was removed from the DCD Tier 2, Revision 6 and seismic classification is now included in DCD Tier 2, Table 3.2-1. Accordingly, based on the above, the applicant’s responses, and DCD changes, RAIs 9.1-1 and 9.1-36 are resolved.

Based on the above, the staff concludes that the ESBWR new and spent fuel storage design complies with the requirements of GDC 2.

9.1.2.3.2 GDC 4

The staff verified that the design complied with the requirements of GDC 4. The applicant committed to meet the guidance of the 1981 versions of SRP Sections 9.1.1 and 9.1.2, as well as RGs 1.13, 1.115, and 1.117.

The staff confirmed that the spent fuel in the storage racks is protected during handling of the shipping cask in the vicinity of the spent fuel storage pool. In DCD Tier 2, Revision 9, Section 9.1.5.5, the applicant stated that the FB crane provides heavy-load-lifting capability for the FB floor. The main hook is used to lift new fuel shipping containers and the spent fuel shipping cask. The applicant stated that the orderly placement and movement paths of these components by the FB crane preclude transport of these heavy loads over the SFP. The FB crane is used during refueling/servicing and when the plant is online. Minimum crane coverage includes the FB floor laydown areas, the cask washdown area, and the FB equipment hatch. During normal plant operation, the crane is used to handle new fuel shipping containers and spent fuel shipping casks. The applicant stated that the FB crane is interlocked to prevent movement of heavy loads over the SFP.

Similarly, the applicant stated that the RB crane provides heavy-load-lifting capability for the refueling floor. The main hook is used to lift the drywell head, reactor pressure vessel (RPV) head insulation, RPV head, dryer, chimney head/separator strongback, and RPV head stud tensioning equipment, as described in DCD Tier 2, Revision 9, Table 9.1-7. The applicant stated that transport of these heavy loads over the spent fuel racks in the deep pit buffer pool or over the new fuel rack is prohibited by safe load paths. The RB crane is also used during refueling/servicing and when the plant is online. Minimum crane coverage includes the RB refueling floor laydown areas and the RB equipment storage. The applicant stated that the RB crane is also interlocked to prevent movement of heavy loads over the fuel pools. Section 9.1.4.3 of this report discusses light-load-handling, and Section 9.1.5.3 of this report discusses heavy-load-handling.

In RAI 9.1-15, the staff requested that the applicant describe how light-load-handling accidents (i.e., load handling accidents involving loads transported by the light load handling system, which include fuel assemblies and light loads (e.g., control rods, burnable poison rods, and flow-limiting orifices) that weigh no more than a fuel assembly) would be mitigated. In response, the applicant stated that the amount of leakage through the liner in the event of a load-handling accident can be limited by designing the pool to withstand the load drop without significant leakage from the pool area in which fuel is stored. The applicant stated that it designed the SFP liner to the requirements in DCD Tier 2, Revision 3, Section 9.1.2.4. The applicant also stated that the liner is seismic Category I and is designed to the acceptance criteria of ASME Code, Section III, Division 2, CC-3700. The staff found this response inadequate and requested, in RAI 9.1-15 S01, that the applicant provide (1) analyses demonstrating that the pool liner will retain its leak-tight integrity after impact by a dropped fuel assembly, (2) a description of an alternate method for ensuring that an adequate pool inventory will be maintained following a fuel-handling accident, or (3) redundant safety-related makeup capability. RAI 9.1-15 was being tracked as an open item in the SER with open items.

In response to RAI 9.1-15 S01, the applicant stated that the following:

> using the previous analysis methodology as a guide, an analysis of the pool liners was performed for the ESBWR. The resulting conclusion demonstrated that a liner thickness of 10.80 mm or greater is sufficient to resist damage from a dropped fuel bundle.

The staff determined that this response was inadequate and, in RAI 9.1-15 S02, asked the applicant to (1) provide the basis for the equation used to calculate the liner thickness, (2) describe the material properties assumed for the liner, (3) describe the type of impact model assumed, (4) describe how the liner is assumed to fail, and (5) describe how the evaluation considered operational experience.

In response, the applicant provided a description of the methodology that it used. However, to determine the adequacy of the alternative analysis, the staff asked the applicant, in RAI 9.1-15 S03, to (1) provide a description of the alternative analysis, (2) explain how the results of the alternative analysis compare to the original analysis, and (3) describe the structural response of the liner plate due to impact of a dropped fuel assembly. The staff noted that the evaluation provided in response to RAI 9.1-15 S03 relied on reinforcing the liner plate in the region of the leak chase channels in areas that are not covered by spent fuel racks by welding 34 millimeter (mm) (1.34 inch [in.]) thick cover plates. Since reinforcement of the liner plate is a special design feature relied on for maintaining integrity of the liner, in RAI 9.1-15 S04, the staff asked the applicant to include this design requirement in the DCD, or justify why a DCD revision is not necessary. The staff stated that any design details added to DCD Tier 2, Appendix 3G, in response to this RAI should be designated Tier 2* to be consistent with the response to RAI 3.8-128. In addition, the staff asked the applicant to clarify whether the buffer pool has comparable leak chase channels and the corresponding need for reinforcement of the liner plate. The staff also asked the applicant to include this design requirement in the DCD or to justify why a DCD revision is not necessary.

In response to RAI 9.1-15 S04, the applicant included in DCD Tier 2, Section 3.8.4.2.5, a Tier 2* description of the reinforcing liner plate. The staff finds that this description is acceptable since the applicant included the reinforcing liner plate in the description of Seismic Category I welding of pool liners and made it Tier 2*. However, the description was unclear whether areas at the bottom of the buffer pool will be constantly exposed from above.

In RAI 9.1-15 S05, the staff asked the applicant to clarify in the DCD whether there are areas in the buffer pool deep pit that are exposed from above (i.e., areas that do not have fuel racks or other equipment shielding the bottom of the pit) such that a dropped fuel bundle could impact the pit bottom without first striking the fuel racks in the deep pit. In response, the applicant proposed a modification to the DCD in Revision 7 to clarify that the RB buffer pool deep pit floor does not require reinforcing because the pit is fully occupied by high-density fuel storage racks or other equipment, and these racks will shield the RB buffer pool deep pit floor from the impact of dropped objects such as a fuel assembly. The staff finds that the RAI responses (RAI 9.1-15 including revisions up through and including RAI 9.1-15 S05) and DCD markup of Section 3.8.4.2.5 are acceptable since the applicant adequately addressed the impacts from dropped objects such as a fuel assembly. The RB buffer pool leak chases do not warrant a reinforcing strip since the buffer pool is fully occupied by fuel storage racks. The design of the SFP leak chase channels have cover plates installed in the areas not occupied by fuel storage racks or other equipment, which is also identified as Tier 2* in the DCD. Based on the analysis results, the liner is not predicted to fail. Accordingly, based on the above, the applicant's responses, and DCD changes, RAI 9.1-15 is resolved.

The staff verified that the racks have been designed to preclude damage to fuel from dropped heavy objects. The applicant stated that the storage rack structure is designed to withstand the impact resulting from a falling fuel assembly. The applicant stated that procedural fuel-handling requirements and equipment design dictate that no more than one bundle at a time can be handled over the storage racks. The structural arrangement is such that no lateral displacement of the fuel occurs; therefore, subcritical spacing is maintained. Sections 9.1.4 and 9.1.5 of this report discuss the staff's evaluation of the ESBWR light-load-handling and heavy-load-handling systems and controls.

Based on the above, the staff concludes that the ESBWR new and spent fuel storage design meets the requirements of GDC 4.

9.1.2.3.3 GDC 61

The staff verified that the design complied with the requirements of GDC 61. The applicant committed to meet the guidance of the 1981 versions of SRP Sections 9.1.1 and 9.1.2, as well as RG 1.13 and ANSI/ANS 57.2-1983 for spent fuel storage and ANSI/ANS 57.3-1983 for new fuel storage.

The guidelines in SRP Section 9.1.2, Revision 4, specify that the materials wetted in the SFP (e.g., spent fuel racks, fixed neutron poison, and the SFP liner) and, if applicable, in the new fuel vault, be chemically compatible and stable. DCD Tier 2, Revision 3, Section 9.1.2, states that the spent fuel storage racks of the ESBWR are constructed in accordance with a quality assurance (QA) program to ensure that design, construction, and testing requirements are met.

In RAI 9.1-27, the staff requested the applicant to demonstrate compatibility and chemical stability of the materials in the SFP racks that are wetted by the water in the SFP, in accordance with Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," to 10 CFR Part 50. In response, the applicant stated that fabrication of the ESBWR spent fuel racks is limited to use of stainless steel materials. The ends are fabricated from Type 304L stainless steel, which conforms to American Society for Testing and Materials (ASTM) A240, "Standard Specification for Chromium and Chromium-Nickel Stainless Steel Plate, Sheet, and Strip for Pressure Vessels and for General Applications." The interlocking panels that form the fuel element storage matrix are fabricated from Type 304B7 borated stainless steel, which

conforms to ASTM A887, “Standard Specification for Borated Stainless Steel Plate, Sheet, and Strip for Nuclear Application” (UNS Designation S30467, Grade B, 1.75 to 2.25-percent boron inclusion). There is no welding of the borated stainless steel. Fuel rack feet are fabricated from Type 630 (17- 4 PH [precipitation hardening]) age-hardened stainless steel, which conforms to ASTM A564, “Standard Specification for Hot-Rolled and Cold-Finished Age-Hardening Stainless Steel Bars and Shapes.” All these materials have been previously used in similar applications and are compatible with the spent fuel assemblies. In addition, these materials have a proven history in the SFP environment. These materials are, therefore, acceptable for use in this application.

The staff finds that the RAI response and DCD markup of DCD Tier 2, Section 9.1.2.6, are acceptable since they specified all of the materials used in the fabrication process for each type of rack (limited to stainless steels materials.) ESBWR SFP water chemistry control is such that the presence of materials that induce corrosion and degradation in stainless steel are limited. The water treatment system includes demineralizing equipment for reducing soluble impurities, such as chloride, sulfate, silica, iron, copper, and other metals. Parameters such as conductivity, dissolved oxygen, and organic impurities are also controlled. In addition, the fuel storage tube assembly is compatible with the environment of treated water and provides a design life of 60 years. Accordingly, based on the above and the applicant’s response, RAI 9.1-27 is resolved.

The guidelines in SRP Section 9.1.2, Revision 4, specify that the applicant should have a program for monitoring the effectiveness of the neutron poison present in the neutron-absorbing panels. DCD Tier 2, Revision 3, Section 9.1.2, was unclear about whether such a program would be established. In RAI 9.1-28, the staff requested that the applicant provide details of the program for monitoring the effectiveness of the neutron poison present in the neutron-absorbing panels. In response, the applicant stated that the design includes sample coupons. These coupons are provided for periodic inservice surveillance throughout the 60-year life of the spent fuel storage racks. The sample coupons are fabricated from the same borated stainless steel material used in construction of the interlocking panels. This borated stainless steel material is UNS S30467, in accordance with ASTM A887.

The staff found this response inadequate and requested in RAI 9.1-28 S01 that the applicant provide (1) plans to use composite materials such as Boral or Metamic; (2) composition and physical properties of borated stainless steel and the composite materials, the manufacturing process, the results of long-term stability and corrosion testing, the resistance to radiation damage, and minimum poison content; (3) the size and types of coupons to be used, the technique for measuring the initial elemental boron or boron carbide content of the coupons, the frequency of coupon sampling and its justification, the tests to be performed on coupons (e.g., weight measurement, measurement of dimensions (length, width, and thickness), and poison content), and the effects of any fluid movement and temperature fluctuations of the pool water on long-term stability. RAI 9.1-28 was being tracked as an open item in the SER with open items.

In response, the applicant explained that there are no plans to use composite materials, such as Boral or Metamic, as a neutron absorbing material in the spent fuel. The applicant stated that borated stainless steel is the composite material used as a neutron absorbing material in the spent fuel. The applicant also provided the chemical composition of the borated stainless steel and additional information concerning heat treatment of the material, which is necessary to meet the specified mechanical properties. The surveillance test coupons are fabricated from the same borated stainless steel material used in the construction of the interlocking panels of the

spent fuel storage racks and are also installed in the FB and RB pool water, thereby experiencing the same environment as the spent fuel storage racks. Based on industry experience of operating plants using borated stainless steel as neutron absorbing material, recording of surveillance data occurs after completion of the first cycle following installation of the racks and no less often than the completion of every third additional operational cycle thereafter. Visual comparison, thickness measurements, and weight measurements are the tests performed to detect evidence of degradation, such as blistering, bubbling, cracking, and flaking. Surveillance coupons that have been in the spent pool environment are compared to those coupons that have been exposed to the SFP water environment. The staff finds that the RAI responses acceptable since the surveillance test coupons are fabricated from the same borated stainless steel material used in the construction of the interlocking panels of the spent fuel storage racks and the surveillance coupons are visually examined to detect evidence of degradation such as blistering, bubbling, cracking, and flaking. Therefore, potential material degradation as a result of neutron irradiation of the SFP storage racks, should it occur, will be detected in time to take corrective action, thus ensuring that the spent fuel storage performs in service as designed. Accordingly, based on the above and the applicant's response, RAI 9.1-28 is resolved.

The guidance in SRP Section 9.1.2 Revision 4 specifies that the staff should evaluate the ability of the SFP configuration to maintain adequate inventory under accident conditions and to provide radiological shielding for personnel. In RAI 9.1-46, the staff asked the applicant to provide information on the depth of water in the SFP above the top of active fuel (TAF) if the pool is drained to the bottom of the transfer gates; on the volume of water in the SFP when the water level is at the bottom of the gates; and on the time to fuel uncovery if the pool has the design-basis spent fuel heat load plus one full core offload, there is no forced circulation, and the pool level is at the bottom of the transfer gates. In response, the applicant addressed the scenario it thought the staff wanted considered in which the level of the SFP was reduced by water spilling into two adjacent empty pools. The applicant stated that it had determined that there was sufficient water to accommodate 72 hours of heat-up and boiling without uncovering the fuel, assuming design basis heat loads. The staff determined this response unacceptable and, in RAI 9.1-46 S01, asked the applicant to address whether there would be a minimum water level over the top of the fuel of at least 3.05 meters (m) (10 feet [ft]) . The staff also asked the applicant to evaluate an assumed loss of forced cooling and gross failure of the transfer gate seals to determine how much water would be above the TAF at 72 hours. In response, the applicant stated that the precise geometry of the fuel pool transfer gates was not yet determined. Assuming a gross failure of the transfer gates when the adjacent pools are empty, the applicant determined that the margin would be greater than 3.05 m (10 ft) above the TAF. In addition, the applicant stated that, assuming that a loss of FAPCS cooling occurs simultaneously with a failure of the transfer gates immediately after a full core offload has been placed in the SFP, it can be shown that there is a margin of 56 mm (2.2 in.) of water above TAF at 72 hours. The staff finds that the RAI responses are acceptable, since even postulating a gross failure of the transfer gate, an adequate water-level margin remains above the TAF. In addition, adequate water-level margin is available after the loss of FAPCS with the postulated failure of the transfer gate. The staff finds that these responses address its concerns. Accordingly, based on the above and the applicant's response, RAI 9.1-46 is resolved. The staff notes that in the applicant's, "Revised Response (Revision 2) to Audit Open Items from the Summary of the June 3 and 15, 2010 NRC Regulatory Audits of the ESBWR Spent Fuel Pool Required Water Inventory," the applicant clarified that, under bounding loss of forced cooling conditions, the water level is at the top of the stored fuel assemblies (TSFA) (rather than the TAF) at 72 hours. Section 9.1.3.3 of this report presents the staff's evaluation of the revised water level and the applicant's "Revised Response (Revision 2) to Audit Open Items from the

Summary of the June 3 and 15, 2010 NRC Regulatory Audits of the ESBWR Spent Fuel Pool Required Water Inventory," in the subsection entitled, "Audit of the ESBWR Spent Fuel Pool Required Water Inventory."

The guidelines in SRP Section 9.1.2, Revision 4, specify that the minimum storage capacity in the spent fuel storage pool should equal or exceed the amount of spent fuel from 5 years of operation at full power plus one full-core discharge. DCD Tier 1, Section 9.1, Revision 9, states that the SFP (physical structure and cooling capacity) is designed to store fuel from 20 years of operation plus one full-core offload, and the RB buffer pool is designed to store 154 fuel assemblies during refueling. The SFP racks provided with the ESBWR standard design, along with those in the RB buffer pools, are designed to store fuel from 10 years of operation plus one full-core offload. The staff notes that while the physical capacity for spent fuel storage (20 years of operation plus one full-core offload) exceeds the storage capacity of the spent fuel racks in the ESBWR standard design (10 years of operation plus one full-core offload), both exceed the minimum spent fuel storage capacity in the SRP guidelines for five years of operation at full power plus one full-core discharge. Accordingly, the staff finds that the spent fuel storage exceeds the minimum storage capacity identified in the guidelines in SRP Section 9.1.2, Revision 4.

The guidelines in SRP Section 9.1.2, Revision 4, specify that the staff should evaluate the use of high-density storage racks. In RAI 9.1-3, the staff requested that the applicant describe how it verified the ability of the SFP and the buffer pool to accommodate the required storage capacity and specify how the design accounted for the reduced cooling effectiveness for high-density racks when compared to low-density racks.

In response, the applicant stated that the size of the SFP is based on typical high-density fuel storage rack designs with typical fuel-to-fuel spacing that includes the fuel assembly at expected maximum bow and bulge, associated neutron absorbers, and any additional structural material. For the FB storage pool, with a typical spacing determined, an array is developed to accommodate the required number of fuel assemblies based on the pitch and the expected number of fuel assemblies to meet the design basis for number of discharged fuel bundles. Similarly, for the RB deep pit, the size is based on the pitch.

The applicant described the racks analysis for cooling as follows:

> Using the fuel and auxiliary pools cooling system (FAPCS) capacity the racks are designed to handle the heat load from the expected number of fuel bundles to be discharged. The hydraulic resistance of the racks with fuel is determined. Natural circulation is assumed. No forced flow under the rack is assumed. Based on natural circulation and inlet conditions at the bottom of the rack, the exhaust temperature of an individual cell is determined. Additionally, the rack array in relation to the pool walls, floors, downcomers, and weir drains is determined. Based on FAPCS flow input volume, temperature, position, and output position a bulk analysis of the racks is performed.

Because of a lack of specific design information, the staff determined this description inadequate to conclude that measures have been taken to provide adequate cooling for high-density racks. The staff requested, in RAI 9.1-3 S01, that the applicant provide information such as assembly dimensions, center-to-center distance, array layouts, and location within the pool, to determine whether sufficient cooling exists for the high-density racks. RAI 9.1-3 was being tracked as an open item in the SER with open items. Subsequent to the SER with open items,

the applicant submitted thermal-hydraulic analyses of the spent fuel racks in NEDO-33373. The SE for NEDO-33373 contains the staff's evaluation of the thermal-hydraulic analyses of the spent fuel racks, which is summarized below. The SE for NEDO-33373 includes an evaluation of the analysis methods, assumptions, and analytical models utilized in the CFD analyses to verify that the storage rack systems have been accurately and conservatively represented. In the SE for NEDO-33373, the staff finds the cooling of the high-density racks adequate. Accordingly, based on the above and the applicant's response, RAI 9.1-3 is resolved. The staff finds that the DCD adequately addresses the guidelines in SRP Section 9.1.2, Revision 4, regarding the use of high-density storage racks.

To confirm that the NEDO-33373 analyses used appropriate fuel assembly and storage rack data, the staff examined referenced design specifications and drawings for both the new and the spent fuel storage racks during the during the February 11-12, 2009, audit held at the applicant's Washington, DC, facility. A summary of the audit, including participants and audit activities may be found in ADAMS at Accession Number ML101450301. The staff also reviewed detailed design calculations and computer program documentation. The staff determined that the detailed design data were appropriately documented and utilized by the applicant and that calculations used to develop input to the CFD analyses, such as pool heat loads and the flow loss coefficient as a function of velocity, employed conservative assumptions.

The staff initially considered performing independent CFD analyses. The staff issued RAIs 9.1-124 through RAI 9.1-127 to substantiate the applicant's thermal-hydraulic analyses. In RAI 9.1-124, the staff requested the applicant to provide the SFP dimensions, the corresponding fuel pool model components, and the assumptions made in laying out the fuel pool model. The staff requested this information to clarify the applicant's model and to support the potential NRC staff confirmatory CFD model. In response, the applicant provided the necessary information to produce a confirmatory CFD model, if needed. The response to RAI 9.1-124 also clarified the rack assumptions and loss coefficients. The staff finds that the applicant's response is acceptable since it provided sufficient information to independently verify the applicant's CFD model.

In RAI 9.1-125, the staff requested the applicant to describe the sensitivity studies it had performed to support its CFD modeling assumptions. In response, the applicant described a series of related sensitivity studies of the CFD model. NEDO-33373 included a comparable description of sensitivity studies. The specific mesh density studies cited relate to an unspecified model of a BWR SFP and are only considered to be qualitative. The staff finds that the applicant's response is acceptable since the margin in the peak temperature predictions bounds the range of CFD model variability shown in sensitivity studies.

In RAI 9.1-126, the staff requested the applicant to clarify NEDO-33373, Figure 5.2, Revision 2, which is the plot of pressure drop in the racks as a function of mass flow. In response, the applicant explained that these data are calculated and the mass flow refers to a single bundle. The applicant also explained that the pressure drop was bounding since it was based on fuel for existing reactors rather than the shorter ESBWR fuel. The staff finds that the applicant's response is acceptable since it clarified the information in NEDO-33373, Figure 5.2 and explained how it is used in the cooling analysis.

In RAI 9.1-127, the staff requested the applicant to clarify the basis for the peak cladding temperature prediction. In response, the applicant cited references validating the selection of the heat transfer coefficient and performed sensitivity studies on the heat transfer coefficient to demonstrate that the value could be reduced by 75 percent and still maintain temperatures

below the limit. The applicant also discussed flow rates, experimental data, and the crud layer resistance and their impact on the peak cladding temperature prediction. The staff finds that the applicant's responses are acceptable since the applicant cited standard references for its data and the staff was able to confirm the crud layer resistance sensitivity reported by the applicant. Accordingly, based on the above and the applicant's response, RAIs 9.1-124 through 9.1-127 are resolved. As discussed in the SE for NEDO-33373, the staff subsequently determined that the CFD analyses presented in the topical report are conservative and that independent CFD analyses would not be necessary.

Based on its review of NEDO-33373, the staff finds that the thermal-hydraulic analysis of the flow through the spent fuel racks is appropriate to demonstrate adequate decay heat removal from the spent fuel assemblies during all anticipated operating and accident conditions. Furthermore, the staff finds that the analyses show that adequate natural circulation of the coolant is provided during all anticipated operating conditions, including full-core offloads during refueling, to prevent nucleate boiling for all fuel assemblies. Therefore, the staff finds that the requirements of GDC 61 regarding the thermal-hydraulic analysis of the spent fuel racks are satisfied.

The guidelines in SRP Section 9.1.2, Revision 4 specify that the staff should verify that the storage racks are designed so that a fuel assembly can be inserted only in a design location. DCD Tier 2, Revision 3, Section 9.1.2, stated that the racks include individual solid tube storage compartments, which provide lateral restraints over the entire length of the fuel assembly or bundle. The weight of the fuel assembly or bundle is supported axially by the rack fuel support. Lead-in guides at the top of the storage spaces provide guidance of the fuel during insertion. The staff requested in RAI 9.1-4 that the applicant clarify how fuel assemblies are precluded from storage in unanalyzed locations within the fuel racks.

In response the applicant stated that no unanalyzed locations exist within a fuel rack or array of racks. Individual racks are spaced less than one fuel assembly apart so that a fuel assembly cannot be inserted between racks. Moreover, the applicant stated that all configurations in which an assembly is lowered adjacent to an exterior rack are analyzed. The staff finds that the RAI response and DCD markup of Section 9.1.2.4 are acceptable since there are no unanalyzed locations within a fuel rack or array of fuel racks. Accordingly, based on the above and the applicant's response, RAI 9.1-4 is resolved. RAI 9.1-4 was being tracked as a confirmatory item in the SER with open items. The staff confirmed that DCD Tier 2, Revision 5, incorporated the above changes, and the confirmatory item is closed. The staff finds that the DCD adequately addresses the guidelines in SRP Section 9.1.2, Revision 4, regarding the design of the storage racks so that a fuel assembly can be inserted only in a design location.

The guidelines in SRP Section 9.1.2, Revision 4, specify that the staff should verify whether the fuel storage racks are capable of withstanding all design loads. In RAI 9.1-5, the staff asked the applicant to clarify how it considered crane uplift forces from a stuck fuel assembly in the rack design for the SFP. In response, the applicant noted that the load combinations listed in DCD Tier 2, Section 9.1.2.4, refer to the dynamic analysis. The uplift force analysis is a separate calculation and is not combined with the dynamic analysis. The applicant modified the DCD to clarify that the design of the spent fuel storage racks and associated support structures meet the guidance of Appendix D to SRP Section 3.8.4. In RAI 3.8-69 S01, the staff identified that while the applicant revised the DCD to reference SRP Section 3.8.4, Appendix D, the loading combinations specified in DCD Tier 2, Section 9.1.2.4, did not in agree with those in SRP Section 3.8.4, Appendix D. In response, the applicant revised DCD Tier 2, Section 9.1.2.4, to include loads and load combinations consistent with SRP Section 3.8.4 Appendix D. This

revision includes the stuck fuel load-upward force on the racks caused by a postulated stuck fuel assembly. The staff finds that the applicant's response is acceptable since it included loads from SRP Section 3.8.4, Appendix D in DCD Tier 2, Section 9.1.2.4, and designated these loads as Tier 2*. The staff also verified that the applicant considered the stuck fuel load in the dynamic analyses in NEDO-33373 Revision 4. Accordingly, based on the above, the applicant's responses, and DCD changes, RAIs 9.1-5 and 3.8-69 S01 are resolved. The staff finds that the DCD adequately addresses the guidelines in SRP Section 9.1.2, Revision 4, regarding the capability of the fuel storage racks to withstand all design loads.

The guidelines in SRP Section 9.1.2, Revision 4, specify that the staff should verify that the SFP coolant water level can be maintained at a safe level for cooling and shielding. In RAI 9.1-115, the staff asked the applicant to provide an elevation diagram of the spent fuel storage pool, lower fuel transfer pool, and cask pool, including any pits in the pools and interfaces (e.g., gates or weirs) between or among the pools or pathways that could potentially lower the water level in the pools to unacceptable levels. Similarly, the staff asked the applicant to provide an elevation diagram of the buffer pool, reactor well, upper fuel transfer pool, IFTS, and equipment storage pool, as well as any interfaces or pits in the pools. In response, the applicant provided a sketch of the equipment storage pool, buffer pool, upper fuel transfer pool and reactor well, as well as the lower fuel transfer pool, SFP, and cask pool. In addition, the applicant stated that RAI 9.1-115 was essentially answered in response to RAI 9.1-46 S01. The staff determined the applicant's response to RAI 9.1-115 unacceptable. It did not address any gates, weirs, or other interfaces that potentially could lower the level of the pools and uncover the fuel. The applicant's response to RAI 9.1-46 S01 was specific to transfer gates, and the applicant indicated that the number and dimension of gates or weirs in the pools discussed above was not determined yet.

In RAI 9.1-115 S01, the staff asked the applicant to provide an ITAAC that would require that the bottom of any gates or weirs associated with these pools be at least 3.05 m (10 ft) above the TAF. In addition, the RAI asked the applicant to provide a COL information item that to instruct COL applicant to evaluate any gates, weirs, or other interfaces (e.g., piping) with these pools to confirm that they are not capable of draining the pool water level inadvertently to less than 3.05 m (10 ft) above the active fuel. In response, the applicant agreed to modify the DCD in Tiers 1 and 2 of Revision 6 to state that transfer gates that connect the SFP to adjacent pools are designed so that the bottom of the gate is at least 3.05 m (10 ft)) above the TAF. In a revised response to RAI 9.1-115 S01, the applicant clarified that the term, "transfer gates" was not meant to imply that other kinds of "gates" could be exempt from this definition. The applicant revised the DCD to refer to the transfer gates simply as "gates" to avoid confusion. The staff finds that the RAI response and DCD markup are acceptable since the applicant adequately addressed the bottom of the gate with respect to the TAF by adding a DCD Tier 1 ITAAC requiring 3.05 m (10 ft) which provides adequate shielding and cooling. In addition, the applicant clarified the term, "gates." Based on the applicant's response and incorporation of the DCD markup in Revision 6 of the DCD, RAI 9.1-115 S01 is resolved. The staff notes that in the applicant's "Revised Response (Revision 2) to Audit Open Items from the Summary of the June 3 and 15, 2010 NRC Regulatory Audits of the ESBWR Spent Fuel Pool Required Water Inventory," the applicant clarified that under bounding loss of forced cooling conditions, the water level is at the TSFA (rather than at the TAF) at 72 hours. Section 9.1.3.3 of this report presents the staff's evaluation of the revised water level, seismic category of the gates, and the applicant's "Revised Response (Revision 2) to Audit Open Items from the Summary of the June 3 and 15, 2010 NRC Regulatory Audits of the ESBWR Spent Fuel Pool Required Water Inventory."

However, in reviewing DCD Revision 6, the staff identified that the applicant's response should have included anti-siphon devices. Accordingly, in RAI 9.1-130, the staff asked the applicant to (1) revise DCD Tier 2, Section 9.1.2.4, to clarify that the anti-siphon holes preserve the water inventory such that it would be at least 3.05 m (10 ft) above top of active fuel in the event of a break in the line at a lower elevation, (2) revise DCD Tier 1, Section 2.6.2, Design Description Item 14, to state that submerged lines entering the SFP or buffer pool must be equipped with anti-siphon holes to preserve the water inventory such that it would be at least 3.05 m (10 ft) above TAF in the event of a break in the line at a lower elevation, and (3) revise the ITAAC in DCD Tier 1, Table 2.6.2-2, Item 14, to state that the anti-siphon holes in submerged lines in the SFP or buffer pool preserve water inventory such that it would be at least 3.05 m (10 ft) above the TAF in the event of a break in the line at a lower elevation. In response, the applicant agreed to revise the DCD in Revision 7, as requested. The staff finds that the RAI response and DCD markup of DCD Tier 1 and Tier 2 are acceptable since all three items were adequately addressed. Item 14 of DCD Tier 1, Section 2.6.2 describes the redundant anti-siphon holes that preserve water inventory 3.05 m (10 ft) above the TAF for safe shielding. In addition, DCD Tier 2, Section 9.1.2.4, adequately describes this design feature of redundant anti-siphon holes. Based on the above and the applicant's response, RAI 9.1-130 is resolved. The staff confirmed that DCD Revision 7 incorporated these DCD changes. Accordingly, the staff finds that the DCD adequately addresses the guidelines in SRP Section 9.1.2, Revision 4, regarding the capability of maintaining SFP coolant water level at a safe level for cooling and shielding. The staff notes that in the applicant's "Revised Response (Revision 2) to Audit Open Items from the Summary of the June 3 and 15, 2010 NRC Regulatory Audits of the ESBWR Spent Fuel Pool Required Water Inventory," the applicant clarified that, under bounding loss of forced cooling conditions, the water level is at the TSFA (rather than the TAF) at 72 hours. Section 9.1.3.3 of this report presents the staff's evaluation of the revised water level, location of the anti-siphon holes, and the applicant's "Revised Response (Revision 2) to Audit Open Items from the Summary of the June 3 and 15, 2010, NRC Regulatory Audits of the ESBWR Spent Fuel Pool Required Water Inventory."

Based on its review of DCD Tier 2, Section 9.1, Revision 5, the staff determined that DCD Tier 1 omitted several apparent design features important to safety. In RAI 14.3-442, the staff asked the applicant to explain why it did not include the following design features in the ITAAC or specified as DCD Tier 1 material:

- The SFP and buffer pool are reinforced concrete structures with a stainless steel liner.
- The SFP and buffer pool liner embedments are designed to meet seismic Category I requirements.
- The bottoms of the SFP and buffer pool gates are higher than the minimum water level over the spent fuel storage racks necessary to provide adequate shielding and cooling.
- Lines to fill and drain the SFP and buffer pool enter the pools above the safe shielding water level.
- Redundant anti-siphon vacuum breakers are located at the high point of the pool lines in the SFP and the buffer pool to preclude a pipe break from siphoning the water from the pools and jeopardizing the safe water level.
- Individual spent fuel racks are spaced less than one fuel assembly apart so that a fuel assembly cannot be inserted between racks.

- Materials used for construction of the SFP and buffer pool are specified in accordance with the latest issue of applicable ASTM specifications at the time of equipment order.

In response to RAI 14.3-442, the applicant addressed each of the items as discussed below.

Regarding SFP and buffer pool materials, the applicant stated that it would add an ITAAC to DCD Tier 1, Tables 2.16.5-2 and 2.16.7-2, in DCD Revision 6 to document that the SFP and buffer pool are to be made of reinforced concrete with a stainless steel liner. The staff finds that the applicant's RAI response for this item and DCD changes are acceptable since the DCD Tier 1 changes adequately address the materials of the SFP.

Regarding SFP and buffer pool liner embedments, the applicant stated that it would add an ITAAC to DCD Tier 1 Tables 2.16.5-2 and 2.16.7-2 in Revision 6 to the DCD to document that they are designed to meet seismic Category I requirements. The staff finds that the applicant's RAI response for this item and DCD changes are acceptable since the DCD Tier 1 changes adequately address the seismic classification of the SFP and buffer pool liner embedments.

Regarding the elevation of the bottoms of the SFP and buffer pool gates, in response to RAI 9.1-115 S01, the applicant agreed to modify the DCD to state that the gates that connect the SFP to adjacent pools are to be designed so that the bottom of the gate is at least 3.05 m (10 ft) above the TAF. In addition, the applicant stated that, since the buffer pool is a deep pit with 9.5 m (31.2 ft) of water, an ITAAC is not needed for the buffer pool gates. The staff finds that the applicant's RAI response for this item and DCD changes are acceptable since the Tier 1 changes adequately address the SFP bottom gate location with respect to the TAF. The staff also finds that an ITAAC was not needed for the buffer pool gates based on the design of the deep pit. As noted above, RAI 9.1-115 is resolved. The staff notes that in the applicant's "Revised Response (Revision 2) to Audit Open Items from the Summary of the June 3 and 15, 2010 NRC Regulatory Audits of the ESBWR Spent Fuel Pool Required Water Inventory," the applicant clarified that under bounding loss of forced cooling conditions, the water level is at the TSFA (rather than the TAF) at 72 hours. Section 9.1.3.3 of this report presents the staff's evaluation of the revised water level, seismic category of the gates, and the applicant's "Revised Response (Revision 2) to Audit Open Items from the Summary of the June 3 and 15, 2010 NRC Regulatory Audits of the ESBWR Spent Fuel Pool Required Water Inventory."

Regarding the lines to fill and drain the SFP and buffer pool, the applicant stated that it would add an ITAAC to DCD Tier 1 Table 2.6.2-2 in Revision 6 of the DCD to describe the design feature that calls for the lines to fill and drain the SFP and buffer pool enter the pools above the safe shielding water level. However, applicant did not make this revision to Revision 6 of the DCD. In its revised response to RAI 14.3-442, the applicant revised the ITAAC satisfactorily. The staff finds that the applicant's RAI response for this item and DCD changes are acceptable since the DCD Tier 1 changes adequately address the fill and drain lines in the SFP and buffer pool related to the safe water level for shielding.

Regarding redundant anti-siphon vacuum breakers, the applicant stated that it would add an ITAAC to DCD Tier 1, Table 2.6.2-2, in Revision 6 of the DCD to verify that lines that are submerged in the spent fuel pool or buffer pool are equipped with anti-siphon holes that will preserve the water inventory above the TAF in the event of a break at a lower elevation. The staff determined that the response to RAI 14.3-442 did not clearly identify the water level needed above the TAF and requested in RAI 9.1-130 that the applicant state that these lines will be equipped with anti-siphon holes to reserve the water inventory such that it would be at least 3.05 m (10 ft) above the TAF in the event of a break in the line at a lower elevation. As

discussed above, in response to RAI 9.1-130, the applicant revised the ITAAC to verify that the anti-siphon holes are 3.05 m (10 ft) above the TAF for safe shielding. The staff finds that the applicant's response to RAI 14.3-442, as modified by the response to RAI 9.1-130, is acceptable since the ITAAC verify that the anti-siphon vacuum breakers preserve a safe water level. The staff notes that in the applicant's "Revised Response (Revision 2) to Audit Open Items from the Summary of the June 3 and 15, 2010 NRC Regulatory Audits of the ESBWR Spent Fuel Pool Required Water Inventory," the applicant clarified that, under bounding loss of forced cooling conditions, the water level is at the TSFA (rather than the TAF) at 72 hours. Section 9.1.3.3 of this report presents the staff's evaluation of the revised water level, location of the anti-siphon holes, and the applicant's "Revised Response (Revision 2) to Audit Open Items from the Summary of the June 3 and 15, 2010 NRC Regulatory Audits of the ESBWR Spent Fuel Pool Required Water Inventory."

Regarding the spacing of individual spent fuel racks to be less than one fuel assembly apart so that a fuel assembly cannot be inserted between racks, the applicant responded that NEDC-33374P, Revision 3, confirms that the gaps between racks are small enough that they cannot accommodate a spent fuel bundle. In addition, in DCD Tier 1, Revision 7, Table 2.5.6-1, the applicant added ITAAC to confirm that the fuel rack spacing dimensions are within the tolerance used in the fuel storage criticality analyses. The staff finds that the applicant's response to this item is acceptable since the rack spacing assumed in NEDC-33374P, Revision 3, is less than one fuel bundle and DCD Tier 1 adequately describes the spent fuel rack spacing.

Regarding materials used for construction of the SFP and buffer pool, the applicant responded that the DCD Tier 1 is not intended to govern details such as material specifications for equipment orders. This information is found in DCD Tier 2, Section 3.8.4, which describes the design features of the RB and FB structure. After further consideration, the staff finds the applicant's reasoning acceptable.

Based on the above, the applicant's responses, and DCD changes, RAI 14.3-442 is resolved. The staff confirmed that DCD Revisions 6 and 7 incorporated the DCD changes as applicable.

Based on the above, the staff finds that the new and spent fuel storage design meets the requirements of GDC 61. Section 9.1.3.3 of this report contains additional discussion related to GDC 61.

9.1.2.3.4 GDC 63

Section 9.1.3 of this report discusses compliance with the requirements of GDC 63.

9.1.2.3.5 10 CFR 20.1406

Section 9.1.3 of this report presents the staff's evaluation of the FAPCS, in accordance with 10 CFR 20.1406.

DCD Tier 2, Section 3.8.4.2.5, identifies that after construction is finished, each isolated pool will be leak tested. The liner welds for all pools outside of the reinforced concrete containment vessel (RCCV), including the SFP, are backed by leak chase channels and a leak detection system to monitor any leakage during plant operation. The leak chase channels are grouped according to the different pool areas and direct any leakage to area drains. This allows both leak detection and determination of where leaks originate.

The staff finds that these design features minimize contamination of the facility in accordance with 10 CFR 20.1406. Sections 12.4 and 12.7 of this report discuss the staff's evaluation of the 10 CFR 20.1406 program as it relates to the FAPCS and pools used for the storage of spent fuel.

9.1.2.3.6 10 CFR 20.1101(b)

The staff verified that the design complied with the requirements of 10 CFR 20.1101(b). As required by 10 CFR 20.1101(b), the licensee must use, to the extent practicable, procedures and engineering controls based on sound radiation protection principles to maintain occupational doses and doses to the public ALARA. DCD Tier 2, Section 12.1.1, states that the ALARA philosophy is applied during the initial design of the plant and implemented via internal design reviews. DCD Tier 2, Section 12.1.1, also specifies that the ESBWR design meets the guidelines of RG 8.8, Sections C.2 and C.4, which address facility, equipment, and instrumentation design features. DCD Tier 2, Chapter 12, discusses several design features related to the storage and handling of spent fuel.

The fuel storage pools have adequate water shielding for the stored spent fuel. DCD Tier 2, Section 12.3.2.2.3, describes the fuel storage pool as being designed to ensure that the dose rate around the pool area is less than 25 microsievert per hour (μSv/hr) (2.5 millirem per hour [mrem/hr]). During fuel-handling operations, sufficient water depth (in combination with the use of integral shielding on the refueling machine,) ensures that the dose rate to operators of the refueling machine and fuel handling machine does not exceed 25 μSv/hr (2.5 mrem/hr) during the movement of a single grappled fuel bundle in either the buffer pool in the RB or the fuel pool in the FB.

The FAPCS operates continuously to reduce the radioactive contamination in the pool water for all the major pools in the ESBWR. To prevent the uncontrolled loss of contaminated pool water from the pools, the SFPs are equipped with drainage paths behind the stainless steel liner welds which direct any leakage from the pools to the liquid waste management system (LWMS). A fuel pool leak detection system monitors any leakage during plant operation and allows both leak detection and the determination of where leaks originate. The staff finds that these design features comply with the requirements of 10 CFR 20.1101(b) for ensuring that occupational doses and doses to the public are maintained ALARA; therefore, the staff finds them to be acceptable. The ALARA program is further addressed in Section 9.1.3 of this report as it relates to the FAPCS, as well as in Section 12.3.

9.1.2.3.7 10 CFR 50.68

The staff verified that the design complies with the requirements of 10 CFR 50.68. Specifically, 10 CFR 50.68 requires provisions either to monitor for criticality accidents pursuant to 10 CFR 70.24 or to follow its guidelines to ensure k_{eff} will not increase beyond safe limits. The applicant addressed the requirements of 10 CFR 50.68 by complying with the additional design and analysis requirements specified in 10 CFR 50.68(b)(2) or 10 CFR 50.68(b)(4). No credit is taken for soluble boron, so the k_{eff} of the new and spent fuel racks must not exceed 0.95 at a 95 percent probability and 95 percent confidence level, if flooded with unborated water. This is demonstrated by the analyses provided in NEDC-33374P, Revision 3, which is evaluated in the SE for NEDC-33374P. Accordingly, the staff finds that the criticality safety of fresh and spent fuel storage and handling meets the requirements of 10 CFR 50.68.

9.1.2.4 *Conclusion*

Based on the above discussion, the staff finds that the ESBWR design conforms to the requirements of GDC 2, 4, and 61. Section 9.1.3 of this report discusses GDC 63. Because the ESBWR design is a single unit, GDC 5 is not applicable. Based on the discussion above, the staff finds that the ESBWR design conforms to the requirements of 10 CFR 50.68, 10 CFR 20.1406 and 10 CFR 20.1101(b).

9.1.3 Spent Fuel Pool Cooling and Cleanup System

9.1.3.1 *Regulatory Criteria*

The staff originally reviewed the design of the ESBWR FAPCS in accordance with SRP Section 9.1.3, Revision 1, issued July 1981. The staff subsequently performed a comparison of the SRP version used during the review with the 2007 version of the SRP. The 2007 version included the following review areas not provided in the 1981 SRP guidance: (1) evaluation of ventilation systems that provide the capability to vent steam and moisture to the atmosphere to protect safety-related components from the effects of boiling in the SFP, (2) modification of the minimum operational heat removal capacity of the spent fuel pool cooling system (SFPCS) to separate the cooling system design basis from unrealistic refueling scenarios, and (3) clarification of the seismic specifications for the SFPCS makeup system and its backup. Although these items were not included in the 1981 version of the SRP which the staff used for its review, the staff did address them.

Section 9.1.3.3 of this report discusses the evaluation of ventilation systems that provide the capability to vent steam/moisture to the atmosphere in order to protect safety-related components from the effects of boiling in the SFP. This section also discusses the minimum operational heat removal capacity of the SFPCS and clarifies the seismic specifications for the SFPCS makeup system and its backup.

The staff's acceptance of the FAPCS design is based on compliance with the following requirements:

- GDC 2, as it relates to the ability of the system and the structures housing it to withstand the effects of natural phenomena, such as earthquakes, tornados, and hurricanes.

- GDC 4, as it relates to the ability of the system and the structures housing it to withstand the effects of external missiles.

- GDC 5, as it relates to whether shared structures, systems and components (SSCs) important to safety are capable of performing required safety functions.

- GDC 61, as it relates to the following system design criteria for fuel storage and handling of radioactive materials:
  - capability for periodic testing of components
  - provisions for containment
  - provisions for decay heat removal
  - capability to prevent reduction in fuel storage coolant inventory under accident conditions in accordance with Regulatory Position C.6 of RG 1.13

- capability and capacity to remove corrosion products, radioactive materials, and impurities from the pool water and to reduce occupational exposures

- GDC 63, as it relates to monitoring systems provided to detect conditions that could result in the loss of decay heat removal capabilities, detect excessive radiation levels, and initiate appropriate safety actions

- 10 CFR 20.1101(b), as it relates to radiation doses being kept ALARA

The SRP acceptance criteria are also based on conformance to the following guidelines:

- SECY-94-084, "Policy and Technical Issues Associated with the Regulatory Treatment of Nonsafety Systems in Passive Plant Designs," dated March 28, 1994, and SECY-95-132, "Policy and Technical Issues Associated with the Regulatory Treatment of Nonsafety Systems (RTNSS) in Passive Plant Designs (SECY-94-084)," dated May 22, 1995, relate to nonsafety-related active systems that are relied upon for a passive plant design for achieving and maintaining cold shutdown conditions and for performing functions that warrant regulatory treatment of nonsafety systems (RTNSS). Specifically, these systems are subject to the following:
  - Nonsafety-related systems that are relied upon for achieving and maintaining cold shutdown conditions should be highly reliable, and there should be no single failure of these systems that would result in inability to terminate use of the passive safety grade systems and achieve cold shutdown, if desired,
  - Nonsafety-related systems that are designated as regulatory treatment of nonsafety systems (RTNSS) (including their support systems) are subject to enhanced design, quality, reliability, and availability provisions.

In addition, the staff reviewed the FAPCS emergency makeup capability to the isolation condenser system (ICS)/passive containment cooling system (PCCS) pool for long-term cooling in accordance with SRP Section 5.4.7, Revision 3; Section 6.2.2, Revision 4; and Section 6.3, Revision 2. The staff's acceptance of the FAPCS design is based on compliance with the following requirements:

- GDC 34, "Residual heat removal," as it relates to the FAPCS having suitable redundancy of components to ensure that, for either a loss of offsite power (LOOP) or a loss of onsite power, the long-term cooling function of the ICS can be accomplished assuming a single failure

- GDC 38, "Containment heat removal," as it relates to the FAPCS having suitable redundancy of components to ensure that for either a LOOP or a loss of onsite power, the long-term cooling function of the PCCS can be accomplished assuming a single failure

- 10 CFR 20.1406 as it relates to the minimization of contamination

9.1.3.2 *Summary of Technical Information*

The FAPCS consists of two redundant cooling and cleanup (C/C) trains, each with a pump, a heat exchanger, and a water treatment unit for cooling and cleanup of various cooling and storage pools except for the IC and PCC pools. A separate subsystem with its own pump,

heat exchanger, and water treatment unit is dedicated for cooling and cleaning of the IC and PCC pools independent of the FAPCS C/C train operation during normal plant operation.

The primary design function of the FAPCS is to cool and clean pools located in the containment, RB, and FB during normal plant operation. The FAPCS provides flow paths for filling and makeup of these pools during normal plant operation and during post-accident conditions, as necessary. The FAPCS is also designed, if needed, to provide the following accident recovery functions (from water drawn from the suppression pool) in addition to the SFP cooling function:

- Suppression pool cooling (SPC)
- Drywell spray
- Low-pressure coolant injection (LPCI) to the RPV
- Alternate shutdown cooling (SDC)

During normal plant operation, at least one FAPCS C/C train is available for continuous operation to cool and clean the water of the SFP, while the other train can be placed in standby or another mode for cooling the gravity-driven cooling system (GDCS) pools and suppression pool. If necessary during a refueling outage, both trains may be used to provide maximum cooling capacity for cooling the SFP. Each FAPCS C/C train has sufficient flow and cooling capacity to maintain SFP bulk water temperature below 48.9 degrees C (120 degrees F) under normal SFP heat-load conditions. During the maximum SFP heat-load conditions of a full-core offload plus irradiated fuel in the SFP resulting from 20 years of plant operations, both FAPCS C/C trains are needed to maintain the bulk temperature below 60 degrees C (140 degrees F). All operating modes are manually initiated and controlled from the main control room (MCR), except the SPC mode, which is initiated either automatically upon a high suppression pool water temperature signal or is initiated manually. Instruments are provided to indicate operating conditions to aid the operator during the initiation and control of system operation.

The FAPCS is a nonsafety-related system with the exception of the piping and components required for containment isolation, the interface with safety-related reactor water cleanup (RWCU)/SDC piping, and the piping and components providing the flow path for post-accident refilling of the ICS/PCCS pools and SFPs with emergency water supplies from the fire protection system (FPS) or another onsite or offsite source. The FAPCS piping and components that are required to support safety-related or accident recovery functions have Quality Group B or C[2] and seismic Category I or II classification. Provisions are taken to prevent inadvertent draining of the pools.

FAPCS components located outside the RB support FAPCS makeup to the ICS/PCCS pools and the SFP for the period from 72 hours to 7 days following an accident. These FAPCS components are designed to seismic Category I standards, but do not fulfill a fire protection function although they are connected directly to the FPS. No fire hydrants, stand pipes, or other large lines can be attached to this dedicated portion of the FPS. The FAPCS also contains a separate, dedicated motor-driven pump located in the FPS pump enclosure that can provide direct injection of water from the FPS to the reactor vessel through the FAPCS via the RWCU system and a feedwater line. The ESBWR probabilistic risk assessment (PRA) credits this vessel injection function.

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[2] See RG 1.26, Revision 4, “Quality Group Classifications And Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants”

9.1.3.3 *Staff Evaluation*

Compliance with the requirements of GDC 2 may be based in part on adherence to the guidance of RG 1.13, Regulatory Positions C.1, C.2, C.6, and C.8, as well as RG 1.29, Regulatory Positions C.1 and C.2. Compliance with the requirements of GDC 4 may be based on adherence to the guidance of RG 1.13, Regulatory Position C.2. The ESBWR design is a single-unit station; therefore, the requirements of GDC 5 are not applicable. Compliance with the requirements of 10 CFR 20.1101(b) depends on adherence to the guidance of Regulatory Positions C.2(f)(2) and C.2.f(3) of RG 8.8. The applicant has identified adherence to RG 8.8 as a COL information item in DCD Tier 2, Section 12.1-4-A. Chapter 12 of this report discusses this further. The staff finds this acceptable.

9.1.3.3.1 GDC 2 and GDC 4

The staff verified that the design complied with the requirements of GDC 2 and 4, as they relate to the system's ability to remain functional in the event of adverse natural phenomena, such as earthquakes, tornadoes, hurricanes, and floods, as well as related effects, including missile strikes. The applicant stated that the RB and FB are designed and constructed to accommodate the dynamic and static loading conditions associated with the various loads and load combinations that form the structural design basis. The loads are those associated with (1) natural phenomena, such as wind, floods, tornadoes, earthquakes, rain and snow, and (2) internal events, such as flooding, pipe breaks, and missiles.

DCD Tier 2, Revision 9, Table 3.2-1, states that the RB and FB are designed to seismic Category I requirements. This statement is consistent with Regulatory Position C.2 of RG 1.13 and the design criteria specified in SRP Section 3.5.3, Revision 2. Section 3.7 of this report provides details of the staff's review of the seismic design of the RB and FB. The staff finds the applicant's declaration that the RB and FB are seismic Category I to be acceptable for meeting the requirements of GDC 4 for those portions of FAPCS located inside these buildings. Section 3.5 of this report discusses protection against external missiles.

DCD Tier 2, Revision 3, Table 9.1-3, stated that piping and components outside containment that are required for SFP cooling, SPC, LPCI, and drywell spray modes of operation, including skimmer lines and all components of the C/C trains, are built to Quality Group B standards and classified as seismic Category II. Since such portions of the system are not designed to seismic Category I requirements, the staff reviewed the SFP cooling loop based on Regulatory Position C.9(b), of RG 1.13 to confirm that it is constructed to Quality Group C standards and that the SFP water makeup system and the building ventilation and filtration system are designed to seismic Category I requirements, are protected from the effects of tornados, and meet the single-failure requirements. The applicant stated that the FAPCS is a nonsafety-related system, with the exception of the piping and components required for containment isolation, the interface with the RWCU/SDC piping, and piping components providing the flow path for post-accident refilling of the IC/PCC, and SFPs with safety-related emergency water supplies.

Consistent with Regulatory Position C.8 of RG 1.13 and Criterion III.1.f of SRP Section 9.1.3, the staff verified that the ESBWR design provides a seismic Category I makeup system and an appropriate backup method to add coolant to the SFP. DCD Tier 2, Revision 3, Section 9.1.2.4, states that the SFP and buffer pool are reinforced concrete structures with a stainless steel liner and the fuel storage racks and pool liner embedments are designed to meet seismic Category I requirements. In DCD Tier 2, Revision 1, Section 9.5.1, the applicant stated that the FPS is

designed to provide an emergency backup source of makeup water for auxiliary refueling pools and reactor water inventory control through a piping connection to the FAPCS. The applicant also indicated that the fire protection piping was designed to Quality Group D[3] standards or lower and the fire pump enclosure is non-seismic.

In RAI 9.1-12, the staff requested that the applicant address quality classification and seismic categorization of makeup water supplies, since the SRP and RG 1.13 specify that the primary SFP makeup system is designed to the seismic Category I, Quality Group C standards. In response to RAI 9.1-12, the applicant stated that the FPS provides the makeup water capability from 72 hours to 7 days following an accident, after which time either additional onsite or offsite makeup sources can be utilized. The applicant stated that this function of the FPS is considered to be an RTNSS function rather than a safety-related function because it is not required for the first 72 hours following an accident. On this basis, the applicant assigned the components associated with providing makeup water from the FPS to Quality Group D. The applicant noted that it will modify the quality group classification for the seismic Category I FPS components supporting the SFP makeup water function to Quality Group C.

The applicant noted that the fire pumps are mounted on a seismic Category I concrete slab, and the enclosures are classified as seismic Category II. While the staff accepted the quality classification and seismic categorization of the FPS, the staff determined that the categorization of the FPS enclosures was unacceptable and requested that the applicant classifies the enclosure as seismic Category I, consistent with its response to RAI 9.1-16. In response to RAI 9.1-12 S01, the applicant agreed to upgrade the enclosure and revise Table 3.2-1 accordingly. The applicant separately agreed to change Table 3.2-1 in response to RAI 3.2-48 S01. RAI 3.2-48 was being tracked as a confirmatory item in the SER with open items. The staff finds these proposed changes acceptable because the applicant agreed to reclassify the FPS enclosure as seismic Category I and Quality Group D, which is a reliable makeup water source for the FAPCS. The applicant also stated in the response to RAI 9.1-12 S01 that it will designate the FPS Quality Group D instead of Quality Group C as a result of further investigation of RTNSS QA requirements. The applicant stated that DCD Tier 2, Table 1.9-9, identifies this classification as a deviation from Criterion II.1.a of SRP Section 9.1.3. The staff finds this response acceptable because the applicant has described the FPS makeup as RTNSS. The ESBWR design is not similar to currently operating plants (such as those described in SRP Section 9.1.3) in that it does not rely on makeup water to provide cooling and shielding (to spent fuel) for the first 72 hours following an accident. The passive design credits the water inventory contained in the SFP to perform these functions. Therefore, the staff does not expect the FPS makeup line to conform to all of the SRP acceptance criteria expected for pools that rely on active components to provide makeup during this period. The staff confirmed that the applicant made these modifications in DCD Tier 2, Revision 5, and the confirmatory item related to RAI 3.2-48 is closed. Accordingly, based on the above, the applicant's responses, and DCD changes, RAI 9.1-12 S01 is resolved.

In RAI 9.1-16, the staff asked the applicant to detail how the safety-related SFP makeup water supplies and water supplies to the ICS/PCCS pools would be protected from the effects of tornados and other natural phenomena. In response, the applicant stated that the only safety-related components of the FAPCS that exist outside of the RB are the emergency fill-up valves attached to the RB structure. The applicant stated that the valves are designed to seismic Category I standards, as evidenced in DCD Tier 2, Table 3.2-1. The staff determined that this response was acceptable, but noted that it conflicted with the applicant's response to RAI 9.1-

[3] See Regulatory Guide 1.26, Revision 4.

12. In RAI 9.1-16 S01, the staff asked the applicant to clarify its position. In response, the applicant stated that it concurred with the staff, and noted that its response to RAI 9.1-12 S01 addressed this inconsistency. The staff noted in RAI 9.1-16 S02 that additional apparent inconsistencies existed in the level of protection afforded FAPCS makeup regarding tornado missiles, and documented its concern about fire hydrants, standpipes, or other large lines that could be attached at some point to the dedicated portion of the FPS connection to the FAPCS for makeup. In response, the applicant reiterated that FPS components located outside the RB which are needed for FAPCS makeup will be designed to seismic Category I standards and will be designed to withstand tornados and other natural phenomena. The applicant stated that the dedicated line from the FPS to the FAPCS is not designed to the National Fire Protection Association (NFPA) standards and will not fulfill a fire protection function. Fire hydrants, stand pipes, or other large lines will not be attached to the dedicated portion of the FPS designed to provide long term makeup to pools in the RB. The staff finds that the response is acceptable, but requested in RAI 9.1-16 S03, that the applicant modify DCD Tier 2 documentation to state this directly. In response, the applicant proposed a modification to Tier 2 in Revision 6 to the DCD. The staff finds that the RAI response and markup of DCD Tier 2 are acceptable since the applicant adequately addressed the FPS components (not designed to NFPA standards) and the FPS relationship to the FAPCS in supporting long term makeup to pools in the RB. The staff confirmed the DCD Tier 2, Revision 6, incorporated the modifications. Accordingly, based on the above, the applicant’s responses, and DCD changes, RAI 9.1-16 is resolved.

In reviewing Revision 6 to the DCD, the staff determined that the FPS diagram in DCD Tier 1, Figure 2.16.3-1 seemed to show both the seismic and non-Seismic Category I lines exiting the Fire Protection Enclosure. These lines appeared to have interfaces with the CB, auxiliary diesel building, RB, and FB. Failure of any of these lines could divert flow from potential refill of the fuel pools or ICS/PCCS pools. In response to RAI 9.1-16 S03, the applicant stated that the FPS components located outside the RB supporting FAPCS makeup are designed to seismic Category I standards and will not fulfill a fire protection function. Fire hydrants, stand pipes, or other large lines are not to be attached to the dedicated portion of the FPS designed to provide long term makeup to pools in the RB. However, DCD Tier 2, Section 19A.3.1.1 stated the following:

> RTNSS functions to support core cooling have permanently installed piping in FAPCS, which connects directly to the FPS. This allows the IC/PCCS pools and SFP to be filled with water from the FPS to extend the cooling period. Water stored in the FPS tank is sufficient to provide combined cooling from 72 hours through 7 days. The dedicated FPS equipment for providing makeup water and the flow paths to the pools is nonsafety-related.

It is the staff’s understanding that there is to be a dedicated, seismic Category I line that will have no firefighting function and will only be used to refill the pools as an RTNSS backup. In RAI 9.1-142, the staff requested the applicant to identify the dedicated line on DCD Tier 1, Figure 2.16.3-1. In response, the applicant proposed a modification to DCD Tier 1 and DCD Tier 2, Revision 7, documentation to reflect this level of detail, including modifications to DCD Tier 2, Figure 9.5-1, and DCD Tier 1, Figure 2.16.3-1. The staff finds that the RAI response and markups of DCD Tier 1 and Tier 2 are acceptable since the applicant provided sufficient details on DCD Tier 1, Figure 2.16.3-1, and DCD Tier 2, Figure 9.5-1, including identification of the dedicated fire protection seismic Category I line to FACPS. This dedicated fire protection line will not be utilized for water supply to other fire protection components such as hose stations or hydrants. Accordingly, based on the above, the applicant’s responses, and DCD changes, RAI

9.1-142 is resolved. The staff confirmed that applicant incorporated the identified changes into DCD Revision 7.

In RAI 9.1-7, the staff asked the applicant to clarify the capability of the RWCU system to provide backup cooling to the SFP. In response, the applicant stated that the RWCU does not support cooling of the SFP. The applicant deleted reference to such capabilities from the DCD. The staff finds that the applicant’s response is acceptable since the applicant removed references to the RWCU as backup cooling for the SFP and related inconsistencies in the DCD. Accordingly, based on the above, the applicant’s responses, and DCD changes, RAI 9.1-7 is resolved.

The staff identified that DCD Tier 2, Revision 4, did not identify the SFP water inventory necessary to support SFP boiling for 72 hours without relying on makeup. In RAI 9.1-44, the staff requested that the applicant provide an analysis to demonstrate that the water volume provided in the SFP is sufficient to provide cooling and shielding without makeup for 72 hours. RAI 9.1-44 was being tracked as an open item in the SER with open items. In response, the applicant referenced a detailed analysis of the most limiting SFP boil-off scenario. The applicant reported that the calculated SFP water level would be approximately 5.5 m (18.0 ft) 72 hours after loss of pool cooling. NEDO-33373, Section 1.4.1 specifies the height of the spent fuel racks of 3.85 m (12.63 ft). Therefore, the height of the spent fuel water above the top of the fuel racks 72 hours after loss of pool cooling would be approximately 1.65 m (5.41 ft). Since 1.65 m (5.41 ft) is below the safe shielding level of 3.05 m (10.0 ft), the applicant further specified that plant personnel would not be allowed in close proximity of the SFP during a loss of cooling event and that pool makeup is achieved from outside the FB. The staff finds the applicant’s response to be acceptable since a water level of 1.65 m (5.41 ft) above the top of the fuel racks is sufficient to keep the active fuel covered at 72 hours. Accordingly, based on the above and the applicant’s response, RAI 9.1-44 is resolved. The staff notes that in the applicant’s “Revised Response (Revision 2) to Audit Open Items from the Summary of the June 3 and 15, 2010 NRC Regulatory Audits of the ESBWR Spent Fuel Pool Required Water Inventory,” the applicant clarified that under bounding loss of forced cooling conditions, the water level is at the TSFA (rather than the TAF) at 72 hours. The staff evaluation of the revised water level is below.

The DCD states that dual-mode operation of the FAPCS is prohibited when only one train of the FAPCS is operating. In RAI 9.1-98, the staff asked the applicant to explain how this action was prohibited. In response, the applicant explained that operators will implement operation of the FAPCS trains through logic functions in the nonsafety-related distributed control and information system (N-DCIS) and provided a corresponding markup of DCD Tier 2, Section 9.1.3.2. The staff finds that the RAI response and DCD Tier 2 markup are acceptable since the FACPS will be instrumented such that any configuration or alignment can be achieved or precluded as necessary. Prohibited modes of operation will be alarmed. The staff confirmed that the applicant incorporated the DCD markup of Section 9.1.3.2 into Revision 6 of the DCD. Accordingly, based on the above, the applicant’s responses, and DCD changes, RAI 9.1-98 is resolved.

In RAI 9.1-47, the staff requested that the applicant describe the design features that prevent drainage of water from the suppression pool or the GDC pools into the FB if these cooling paths are operating at the same time. In response, the applicant indicated that the flow paths are normally isolated and only opened if the FAPCS is in the SPC or GDCS pool cooling mode. The FAPCS pumps will trip on low water level in these pools and, coincident with the trip signal, a closure signal is sent to the safety-related containment isolation valves so that these lines to

the suppression pool would be isolated. There are also anti-siphoning provisions in the discharge lines to these pools and to the suction line to the GDCS pools. The staff finds that the RAI response is acceptable since the flow paths from these pools are normally isolated, trip on sensed low water level, and interlocks are designed to prevent crosstie. If such crosstie events occur, water volume would be preserved and be limited to the equivalent of the volume of water between the minimum and maximum pool levels associated with the design of anti-siphoning provisions. Accordingly, based on the above and the applicant's response, RAI 9.1-47 is resolved.

DCD Tier 2, Revision 3, Section 9.1.3, identified the FAPCS piping and components credited for emergency makeup as safety-related, while DCD Tier 2, Revision 3, Chapter 19, identified similar portions of the FAPCS as RTNSS. In RAI 9.1-42, the staff requested that the applicant clarify whether it will consider the FAPCS to be RTNSS. In response to RAI 9.1-42, the applicant stated that the FAPCS is safety-related in some locations and RTNSS in others and provided a description of these differences. However, the staff determined that the description was insufficient and requested, in RAI 9.1-42 S01 that the applicant provide a schematic that identifies the RTNSS and safety-related portions of the FAPCS and include this diagram in DCD Tier 2. RAI 9.1-42 was being tracked as an open item in the SER with open items. In response to RAI 9.1-42 S01, the applicant clearly delineated the portions of the FAPCS that are safety-related including (1) containment isolation, (2) refilling of the IC/PCC pools and SFP with post-accident water supplies from the FPS, and (3) high pressure interface with RWCU/SDC used for LPCI.

The applicant also identified that the RTNSS functions of the FAPCS include SPC and LPCI which encompasses the suction line from the suppression pool, all of the piping and components in the C/C trains (except the water treatment units), and the discharge lines to the suppression pool and the LPCI interface up to the safety-related isolation valves. In addition, the applicant provided a DCD markup which clarified the RTNSS components. The staff finds that the RAI response and DCD revision are acceptable since they clarify the portions of the FAPCS which are safety-related versus those that are RTNSS. The staff confirmed that the applicant incorporated the DCD markups into DCD revision 6. Accordingly, based on the above, the applicant's responses, and DCD changes, RAI 9.1-42 is resolved.

The staff reviewed DCD Revision 6 and determined that safety-related external connections for the FAPCS for emergency refill of the ICS/PCCS pools and the SFP are inconsistently described in DCD Tier 2 and incorrectly identified as nonsafety-related in DCD Tier 1. In RAI 9.1-132, the staff asked the applicant to revise the description of these safety-related connections in DCD Tier 1 and Tier 2 to be consistent. In response, the applicant clarified that the function to refill the pools is a RTNSS function, but the piping used in this function is safety-related. The applicant committed to revise DCD Tier 2, Section 19A.3.3, in Revision 7 of the DCD to reflect this distinction. The staff finds that the RAI response and DCD markup of Section 19A.3.3 are acceptable since the representative SSCs that meet the RTNSS Criterion B (applies to SSC functions relied upon to resolve long-term safety (beyond 72 hours) and to address associated seismic capabilities) are now listed in a consistent manner and ambiguous language concerning safety classification has been removed. In addition, the applicant clarified the piping used for the dedicated FPS makeup water supply to the SFP and ICS/PCCS pools. The staff confirmed that the applicant incorporated the DCD markups into DCD Revision 7. Accordingly, based on the above, the applicant's responses, and DCD changes, RAI 9.1-132 is resolved.

In its review of DCD Revision 6, the staff noted that DCD Tier 2, Section 19A.3.1.2, needed to be revised to be consistent with a previous response to RAI 7.1-140. The wording seemed to imply that a PCCS function was safety-related as well as RTNSS. In RAI 9.1-136, the staff asked the applicant to clarify this apparent confusion and to revise any other parts of DCD Tier 2, Section 19A, that incorrectly described RTNSS functions for containment integrity. In response, the applicant provided markups for DCD Tier 2, Section 19A.3.1.2, Revision 7, to more clearly indicate that the PCCS is technically a passive system dependent on active portions of the ICS. The applicant also provided markups to clarify the RTNSS makeup function provided by the FPS via the FAPCS to replenish the water boiled off from the SFP and the ICS/PCCS pools after 72 hours. The staff finds that the RAI response and DCD revision are acceptable since the applicant clarified the safety-related and RTNSS functions related to containment integrity in DCD Tier 2, Section 19A.3.1.2. The staff confirmed that the applicant incorporated the DCD markups into DCD Revision 7. Accordingly, based on the above, the applicant's responses, and DCD changes, RAI 9.1-136 is resolved.

The applicant stated that the FB does not house any safety-related equipment that may be subject to flooding. Section 3.4.1 of this report provides a detailed review of protection from the effects of flooding.

In its review of DCD Tier 2, Revision 6, Section 9.1.3.2, the staff noted a statement in the paragraph referring to, "A reactor makeup water discharge line," stated the following:

> A pressure relief valve is located upstream of the motor-operated shutoff valves. Any leakage of high-pressure coolant through the safety-related check valves and motor-operated shutoff valves is discharged through the pressure relief valve and measured before being sent to the Liquid Waste Management System.

However, on DCD Tier 2, Revision 6, Figure 9.1-1, the pressure relief valve was downstream of the motor operated valves (MOVs). While leakage past the MOVs from the FAPCS system might open the relief valve, it was not clear that relief valve leakage was to be measured at this location. In RAI 9.1-138, the staff asked the applicant to either modify the figures in DCD Tier 1 and 2 regarding the placement of the pressure relief valve, relative to the safety-related shutoff valves, or modify the description in DCD Tier 2 of the relief valve and the leakage it monitors. In response, the applicant agreed and stated that it would correct figures in DCD Tier 1 and Tier 2, Revision 7. The staff finds that the RAI response and DCD markup of Tier 1 and Tier 2 figures are acceptable since the applicant correctly placed the relief valves upstream of MOVs F332A/B in DCD Tier 1, Figure 2.6.2-1, and DCD Tier 2, Figure 9.1-1. The staff confirmed that the applicant incorporated the DCD markups into DCD Revision 7. Based on the above, the applicant's responses, and DCD changes, RAI 9.1-138 is resolved.

In its review of DCD Revision 6, the staff noted that DCD Tier 2, Section 9.1.3.2, discusses the existence of piping separate from the FAPCS pool cooling piping that provides flow paths to refill the ICS/PCCS pool and the SFP. The DCD describes this as "post-accident makeup water transfer from offsite water supply sources." This description does not appropriately describe the expected uses of these flow paths. Post-72 hours, on-site resources are to be used to provide makeup water to these pools. These resources are to include pumps, hoses, pipes, and the like that will exist or be stored on site to provide alternative pathways to refill the pools. For example, operators might use the FPS diesel driven pump and a fire hose to refill a pool via the FAPCS external connections or the operator might hook up a portable pump to take suction from the cooling tower basin and inject the water through fire hoses into the connections external to the RB to achieve pool refill. In RAI 9.1-139, the staff asked the applicant to expand

its discussion in DCD Tier 2 of the potential uses of the external FAPCS pool refill hookups. In response, the applicant stated that it will clarify DCD Tier 2, Revision 7, Section 9.1.3.2, to indicate that the FAPCS makeup water for the SFP and ICS/PCCS pools can be supplied from onsite or offsite sources. The staff finds that the RAI response and DCD markup of Section 9.1.3.2 are acceptable since the applicant clarified that the FAPCS makeup water to the SFP and ICS/PCCS pools can be supplied from onsite (i.e., the FPS) or offsite sources via flanged connection in the yard area. The staff confirmed that the applicant incorporated the DCD markups into DCD Revision 7. Based on the above, the applicant's responses, and DCD changes, RAI 9.1-139 is resolved.

Based on the above discussion, the staff finds that the ESBWR design complies with the requirements of GDC 2 and 4.

9.1.3.3.2 GDC 61

The staff verified that the SFP and cooling systems meet the requirements of GDC 61. The staff confirmed that essential portions of the system are correctly identified and are isolable from the nonessential portions of the system. DCD Tier 2, Revision 3, Section 9.1.3, states that a manifold of four MOVs is attached to each end of the FAPCS C/C trains. These manifolds are used to connect the FAPCS C/C train with one of the two pairs of suction and discharge piping loops to establish the desired flow path during FAPCS operation. One loop is used for the SFP and auxiliary pools, and the other loop for the GDCS pools and suppression pool and for injecting water to the drywell spray sparger and reactor vessel via RWCU/SDC and feedwater pipes. The use of manifolds with proper valve alignment and separate suction-discharge piping loops serves two purposes: (1) it allows operation of one train independent of the other to permit online maintenance or dual-mode operation using separate trains, if necessary, and (2) it prevents inadvertent draining of the pool and mixing of contaminated water in the SFP with cleaner water in other pools.

In RAI 9.1-8, the staff requested that the applicant describe how the SFP decay heat is transferred to an ultimate heat sink (UHS) under accident conditions (i.e., pool boiling) and how essential equipment is protected against the environmental effects. In response, as well as design clarifications to DCD Tier 2, Revision 6, Section 9.1, the applicant stated that the SFP upon loss of SFP cooling is designed to dissipate fuel decay heat through heat up and boil off of the pool water for 72 hours. The applicant stated that pool water performs the safety-related heat removal function stipulated in GDC 44, "Cooling Water" (which in this review is considered a subset of the requirements of GDC 61). Upon loss of power, the FB heating, ventilation, and air conditioning (HVAC) system isolates the FB, as described in DCD Tier 2, Revision 3, Section 9.4.2.5. Steam generated by boiling of the SFP water is released to the atmosphere (the UHS) from the FB through nonsafety-related passive relief devices so as to prevent the FB from exceeding its maximum design pressures. Similarly, steam generated by boiling of the buffer pool and reactor well, upon loss of pool cooling during refueling, is released to the atmosphere from the RB through nonsafety-related passive relief devices to prevent the RB from exceeding its maximum design pressures. The nonsafety-related passive reliefs are normally closed as a precaution against radiological releases in the event of a fuel handling accident. The setpoints for both FB and RB prevent the relief devices from opening during a full tornado pressure drop. The applicant provided markups identifying changes to DCD Tier 2, Revision 6, Sections 9.1.3.2 and 6.2.3.2, to add pressure relief devices to the FB and RB.

The staff finds that the applicant's response to RAI 9.1-8 and the design clarifications to DCD Tier 2, Revision 6, are acceptable since the applicant adequately addressed the environmental

effects of pool boiling with the addition of relief devices in the RB and FB. The RB devices will open upon a differential pressure during refueling outages between the RB and the environment and will not open during a tornado or design-bases event described in DCD Tier 2, Chapter 15. The FB devices will open upon a differential pressure between the FB and the environment and will not open during a tornado or design-bases event described in Chapter 15. The relief devices are designed as nonsafety-related and are not credited for protecting the safety-related structures from overpressure. Radioactivity releases through an open relief device during pool boiling is bounding by other fuel handling accidents. The staff confirmed that the applicant incorporated the DCD markups into DCD Revision 7. Based the above, the applicant's response, and DCD changes, RAI 9.1-8 is resolved with respect to transferring SFP decay heat to an UHS. The safety classification of the relief devices is further discussed below.

The applicant stated that the EBSWR design does not provide engineered safety feature atmosphere cleanup systems and associated guidance, as described in RG 1.52, "Design, Inspection, and Testing Criteria for Air Filtration and Adsorption Units of Post-Accident Engineered-Safety-Feature Atmosphere Cleanup Systems in Light-Water-Cooled Nuclear Power Plants." The staff requested that the applicant justify its decision not to include an atmosphere cleanup system. RAI 9.1-8 was being tracked as an open item in the SER with open items. In response to RAI 9.1-8 S01, the applicant stated that design basis accidents (DBAs) associated with the FB are limited to the fuel handling accident and spent fuel cask drop accident. Dose consequences for the fuel handling accident are calculated assuming instantaneous release of noble gas and iodine radionuclides without credit for atmospheric cleanup. The spent fuel cask drop accident does not result in any radionuclide release. The applicant further indicated that safety-related FB HVAC atmospheric cleanup capability to mitigate and further reduce radiological consequences is not required since no DBAs associated with operations in the FB are identified that require atmospheric cleanup to limit dose consequences within of the guidance of RG 1.183, "Alternative Radiological Source Terms for Evaluating Design Bases Accidents at Nuclear Power Plants," and the limits of GDC 19, "Control Room." The staff determined that this justification was unacceptable.

In RAI 9.1-8 S02, the staff communicated to the applicant that, when evaluating SFP accidents on pools that have nonsafety-related cooling systems, the staff's position is that the DBA should be assumed coincident with the loss of forced cooling. The staff clarified that conformance with RG 1.183 can be shown by demonstrating that the SFP water level will be more than 7.0 m (23 ft) above the TAF for at least 2 hours following the DBA. In addition, the staff asked the applicant to state how many feet of water would be above the top of the fuel during refueling operations, as well as the time to boil in the SFP following the loss of forced cooling. In response, the applicant stated that SFP water level during refueling is the same as in normal operation (i.e., approximately 10.9 m [35.6 ft] of water above the TAF), and it would take the SFP approximately 8.9 hours to reach boiling, given loss of forced cooling with the greatest possible heat load in the pool. The staff finds this acceptable since the SFP water level will be more than 7.0 m (23 ft) above TAF during refueling and the time for boiling in the SFP is approximately 8.9 hours during a loss of forced cooling event, and thus conforms to the RG 1.183 criteria. Accordingly, based on the above and the applicant's response, RAI 9.1-8 is resolved with respect to SFP boiling and SFP water level. The staff notes that in the applicant's "Revised Response (Revision 2) to Audit Open Items from the Summary of the June 3 and 15, 2010 NRC Regulatory Audits of the ESBWR Spent Fuel Pool Required Water Inventory," the applicant clarified that under bounding loss of forced cooling conditions, the water level is at the TSFA (rather than the TAF) at 72 hours. The staff evaluation of the revised water level is below.

Additional information was provided by the applicant in a supplement to the response for RAI 9.1-8 S01 related to the safety classification of the RB and FB relief panels. In revision 7 of the DCD, the RB and FB relief panels were classified as nonsafety-related. In the applicant's supplement to the response, a markup of DCD Tier 2, Table 3.2-1, Sections 6.2.3.2 and 9.1.3.2, and DCD Tier 1, Sections 2.16.5 and 2.16.7, was provided. The DCD markup changed the classification of the RB and FB relief panels to safety-related, Safety Class 3, and Seismic Category I since the relief panels are designed to open to prevent the FB or RB from exceeding their maximum design pressure. The applicant also provided a DCD markup of ITAACs for the FB and RB relief panels.

The staff finds the applicant's response to be acceptable since the relief panels perform a safety function to prevent building over pressurization during a loss of FAPCS event with pool boiling and thus should be classified as safety-related, Seismic Category I. The staff finds that the FB and RB relief devices are properly classified. The staff confirmed that the applicant incorporated the changes to the safety and seismic classification of the relief panels for the FB and RB into Revision 8 of DCD Tier 2, Table 3.2-1 and the DCD Tier 1, ITAAC. Accordingly, based on the above and the DCD changes, RAI 9.1-8 is resolved.

The staff requested, in RAI 9.1-9, that the applicant describe how adequate cooling is provided for fuel stored in the RB buffer pool under accident conditions. In response, the applicant stated that the spent fuel is only stored in the buffer pool for very brief periods when fuel assemblies are being shuffled to different locations in the core. The buffer pool is designed to hold a maximum of 154 spent fuel assemblies. The applicant stated that, during an outage, the available water inventory is increased by opening gates that allow the buffer pool to communicate with the water in the reactor well and dryer/separator pool. The applicant stated that this effectively increases the pool surface area to more than twice that of the SFP. The buffer pool would have to boil off a larger margin of water volume than the SFP to reach the minimum water level. The applicant stated that, if the FAPCS cooling were lost during an outage, the large water inventory would provide ample time for transferring this fuel from the buffer pool to the SFP.

The staff found the applicant's response inadequate to determine the acceptability of the design with regards to adequate cooling. The staff requested that the applicant supplement its response by describing the controls that will be used to ensure that the required volume of water will be maintained at all times. In response to RAI 9.1-9 S01, the applicant described how the FAPCS is designed to withstand a single failure. However, the intent of the RAI was to clarify how sufficient coolant inventory will be maintained in the RB buffer pool during accident conditions, such as the loss of the nonsafety-related forced cooling system for 72 hours. The response did not address the conditions identified in the RAI. The staff requested that the applicant provide an analysis to demonstrate that the volume provided by the buffer pool is sufficient to provide cooling and shielding without makeup for 72 hours. If the analysis relies on additional water inventory in the RB (e.g., from the reactor well and the dryer storage pool), the applicant should describe the controls relied upon to ensure this inventory is available to the buffer pool. RAI 9.1-9 was being tracked as an open item in the SER with open items.

In response to RAI 9.1-9 S02, the applicant provided references to calculations that show that, if both trains of the FAPCS are lost and no additional water is credited beyond that in the buffer pool, there is sufficient water to allow 72 hours of passive cooling without reducing the water level below 3.05 m (10 ft) above the TAF. This level is considered adequate for shielding based on guidance in RG 1.13, which sets a minimum water level of 3.05 m (10 ft) above the TAF. The staff finds that the applicant's response was acceptable because an adequate water level

will be maintained in the buffer pool for 72 hours. Accordingly, based on the above and the applicant's response, RAI 9.1-9 is resolved. The staff notes that in the applicant's "Revised Response (Revision 2) to Audit Open Items from the Summary of the June 3 and 15, 2010 NRC Regulatory Audits of the ESBWR Spent Fuel Pool Required Water Inventory," the applicant clarified that under bounding loss of forced cooling conditions, the water level is at the TSFA (rather than the TAF) at 72 hours. The staff evaluation of the revised water level is below.

As described in Criterion III.1.d of SRP Section 9.1.3, with normal cooling systems in operation and assuming a single active failure, the temperature of the pool should be kept at or below 60 degrees C (140 degrees F) and the liquid level in the pool should be maintained. For the full-core offload condition, the temperature of the pool water should be kept below boiling and the liquid level maintained with normal systems in operation. The calculation for the maximum amount of thermal energy to be removed by the spent fuel cooling system should be made in accordance with Branch Technical Position (BTP) Auxiliary Systems Branch 9-2, "Residual Decay Energy for Light Water Reactors for Long-Term Cooling."

DCD Tier 2, Revision 2, Section 9.1.3.2, stated that each FAPCS C/C train has sufficient flow and cooling capacity to maintain SFP bulk water temperature below 48.9 degrees C (120 degrees F) under normal SFP heat load conditions. During the maximum SFP heat load conditions of a full-core offload, plus irradiated fuel in the SFP resulting from 20 years of plant operations, both FAPCS C/C trains are needed to maintain the bulk temperature below 60 degrees C (140 degrees F).

In RAI 9.1-10, the staff requested that the applicant specify how adequate decay heat removal capacity will be demonstrated for normal operating (i.e., non-accident) conditions. In response, the applicant stated that SFP decay heat power as a function of time after shutdown is calculated based on a computer code developed using the standards in ANSI/ANS-5.1-1994, "Decay Heat Power in Light Water Reactors." The validation of code outputs is done through regeneration of the tables in ANSI/ANS-5.1-1994. The applicant stated that the scope of the calculation covers all guidelines contained in Criterion III.1.h of SRP Section 9.1.3.

The applicant stated that the FAPCS equipment heat removal capacity will be verified by performing a calculation to demonstrate that the pumps and heat exchangers are sized to accommodate the expected maximum heat loads and the required temperature limits. The staff determined that the applicant's response to RAI 9.1-10 was inadequate to determine the acceptability of the FAPCS C/C as it relates to GDC 61. The applicant neither provided specific performance requirements (heat transfer capacity and flow rate) nor described a method for calculating the required cooling capacity. The staff requested that the applicant provide these performance requirements. In response to RAI 9.1-10 S01, the applicant stated that the FAPCS C/C trains are not used to satisfy GDC 44 (which in this review is considered to be a subset of the requirements of GDC 61) and that GDC 44 is satisfied by passive pool boiling for 72 hours and subsequent makeup. The staff did not agree with this statement. GDC 61 requires an evaluation of the system under both normal operating and accident conditions. The water inventory may be credited for accident conditions; however, during normal conditions, the FAPCS provides forced cooling to the SFP and RB pools. The staff requested, in RAI 9.1-10 S02, that the applicant provide a summary heat balance of the FAPCS, including initial assumptions and performance requirements. RAI 9.1-10 was being tracked as an open item in the SER with open items. In response, the applicant provided a summary that included FAPCS design and performance parameters, as well as a heat balance summary. The applicant added performance values of the FAPCS C/C trains to the DCD in Table 9.1-8 of DCD Tier 2, Revision 5. The staff finds that the RAI response is acceptable since the applicant provided the heat load

data and balance summary and added the FAPCS design and performance parameters to the DCD. The DCD revision also clarified that one train of the FAPCS is capable of removing 9.6 megawatt thermal (MWt) at its design conditions; thus, the FAPCS can accommodate the most limiting heat load conditions in the SFP.

The applicant later changed the design capability of FAPCS from 9.6 MWt to 8.3 MWt in response to RAI 9.1-20, which is discussed below. In design clarifications for DCD Tier 2, Revision 7, the applicant proposed to modify DCD Tier 2, Table 9.1-8 to clarify that the design heat removal capacity for the FAPCS heat exchanger is 8.3 MWt per train. The nominal capacity of one train is equivalent to the heat load of 20 years of discharged fuel. The staff finds that the clarified nominal heat transfer capacity of the FAPCS system (8.3 MWt per train at rated conditions) was acceptable since the nominal capacity of two trains exceeds 0.3 percent of the rated thermal power for the ESBWR reactor, which is consistent with the guidelines of SRP Section 9.1.3, Revision 2. The staff confirmed that the applicant incorporated the DCD markups into DCD Revision 7. Based on the above, the applicant's responses, and DCD changes, RAI 9.1-10 is resolved.

The staff verified that design provisions exist to permit appropriate inservice inspection and functional testing of system components. DCD Tier 2, Revision 3, Section 9.1.3.4, states that the FAPCS is designed to permit surveillance testing and inservice inspection of the safety-related components, in accordance with ASME Code Section XI. Additionally, the FAPCS is designed to permit leak-rate testing of its components that are required to perform a containment isolation function in accordance with Appendix J, "Primary Water Containment Leakage Testing for Water-Cooled Power Reactors," to 10 CFR Part 50. Sections 6.6 and 3.9.6 of this report further discuss inservice inspection and inservice testing.

The staff confirmed that design provisions exist to address Regulatory Position C.6 of RG 1.13, such that systems have been designed to ensure that, in the event of failure of inlets, outlets, piping, or drains, the pool level will not be inadvertently drained below a point approximately 3 m (10 ft) above the TAF. DCD Tier 2, Revision 5, Section 9.1.2.4, stated that the bottoms of the pool gates are higher than the minimum water level required over the spent fuel storage racks to provide adequate shielding and cooling. Poolfill and drain lines enter the pool above the safe shielding water level. Redundant anti-siphon vacuum breakers are located at the high point of the pool circulation lines to preclude a pipe break from siphoning the water from the pool. In addition, as noted above, in response to RAI 9.1-115 S01, the applicant modified DCD Tiers 1 and 2 to state that the transfer gates in the SFP that connect to adjacent pools are designed so that the bottom of the gate is at least 3.05 m (10 ft) above the TAF.

In RAI 9.1-11, the staff requested that the applicant clarify how the safety of stored spent fuel is ensured following a piping failure in lines that extend below the surface of the SFP. In response, the applicant stated that the common emergency makeup header will not be submerged below the surface of the pool. The applicant stated that cooling system return lines are submerged below normal water level, but these lines include anti-siphoning provisions as described above. Anti-siphon holes are located at the normal water level for all FAPCS cooling system discharge lines, thus preventing any significant draining in the event of a pipe break.

The applicant stated that because the SFP does not contain suction piping, these anti-siphon holes would ensure that the water level would not drop below the normal elevation in the event of a piping failure. In addition to the cooling return lines, the FAPCS has suction lines for the GDC pools, suppression pool, and IC/PCC pools. The applicant stated that these lines will also have anti-siphoning provisions. The applicant also stated that suction lines cannot have holes

at the normal water level; therefore, the anti-siphon holes will be included on all suction lines at the elevation of the minimum water level for each respective pool. However, the applicant did not include all of these details in the DCD. Therefore, in RAI 9.1-11 S01, the staff requested that the applicant reflect in the DCD that the makeup header will not be submerged below the surface of the pool. In response to RAI 9.1-11 S01, the applicant agreed to make this change. The staff finds that the RAI responses are acceptable since the applicant clarified the emergency makeup header location and explained that anti-siphon holes are located at the normal water level for all cooling discharge lines. Furthermore, the applicant committed to make the corresponding changes to the DCD. Accordingly, based on the above, and the applicant's response, RAI 9.1-11 is resolved. RAI 9.1-11 was being tracked as a confirmatory item in the SER with open items. The staff confirmed that the applicant incorporated the above changes into DCD Tier 2, Revision 4 and the confirmatory item is closed. The staff notes that in the applicant's "Revised Response (Revision 2) to Audit Open Items from the Summary of the June 3 and 15, 2010 NRC Regulatory Audits of the ESBWR Spent Fuel Pool Required Water Inventory," the applicant clarified that, under bounding loss of forced cooling conditions, the water level is at the TSFA (rather than the TAF) at 72 hours. The staff evaluation of the revised water level, location of the anti-siphon holes, and the applicant's "Revised Response (Revision 2) to Audit Open Items from the Summary of the June 3 and 15, 2010 NRC Regulatory Audits of the ESBWR Spent Fuel Pool Required Water Inventory," is below.

In RAI 9.1-13, the staff requested that the applicant clarify how the redundancy requirements of GDC 61 are satisfied with respect to makeup water supplies to pools necessary for residual heat removal. In response, the applicant stated that it would modify the design to include two parallel valves in the makeup water supply line from the FPS to the FAPCS for both the ICS/PCCS and SFPs. This change ensures that onsite water sources remain available as makeup for the ICS/PCCS and SFPs for the first 7 days, even if a single active failure were to occur. The addition of these parallel valves ensures that the ICS and PCCS condensers can provide sufficient heat removal capability at and beyond 72 hours to satisfy the requirements of GDC 34 and 38 considering a single failure.

The applicant stated that the ESBWR design originally addressed a single active failure by having separate makeup connections to the FPS and to an alternate water supply connection point in the yard area. The new parallel valve the applicant added in response to this RAI provides further assurance that the design can withstand a single active failure. The staff found this acceptable. However, in RAI 9.1-13 S01, the staff requested that the applicant show how the proposed total makeup flow rate of 46 cubic meters per hour (m^3/h) (200 gallons per minute [gpm]) is bounding for accidents shortly after a refueling outage. RAI 9.1-13 was being tracked as an open item in the SER with open items. In response to RAI 9.1-13 S01, the applicant provided a bounding estimate of the flow rate needed to be supplied to the SFP to remove decay heat from the SFP 3-days post-shutdown. The staff finds that the RAI response is acceptable since the applicant determined the minimum makeup water flow rate based on the highest heat load, which occurs at 3-days post-accident point, using the heat of vaporization of water with the decay heat from the core and SFP. Accordingly, based on the above and the applicant's response, RAI 9.1-13 is resolved.

In RAI 9.1-14, the staff requested that the applicant describe the necessary capacity of the emergency makeup lines and how the capacity of the makeup lines will be confirmed. In response, the applicant stated that the capacity of the makeup lines will accommodate the boil-off rates associated with the maximum post-72-hour heat loads expected for the SFP and the IC/PCC pools. The applicant stated that the value for the boil-off rate is calculated based on the most limiting condition, which includes the decay heat from 10 years of accumulated spent fuel

in the SFP, as well as the shutdown power from the full core discharged to the ICS immediately following a scram. The heat output at the end of the 72-hour period will be converted to a boil-off rate, which will be taken as the required makeup rate for these pools. Because the makeup rate will remain constant as the heat loads continue to drop, the makeup rate at 72 hours will be sufficient to refill the pools in the long term. The applicant stated that the ability to transfer water from the FPS to both pools will be confirmed during plant pre-operational testing. The applicant also stated that it will update DCD Tier 1, Table 2.16.3-1, to include a requirement for performance of this test, and the corresponding ITAAC. RAI 9.1-14 was being tracked as an open item in the SER with open items. The staff finds that the RAI response is acceptable since a test will be performed to demonstrate that the diesel-driven fire pump will supply a minimum of 46 m^3/h (200 gpm) flow rate to the ICS/PCCS pools and SFP. The applicant added this test to Table 2.16.3-2, ITAAC 7a. Accordingly, based on the above, the applicant's responses, and DCD changes, RAI 9.1-14 is resolved.

In RAI 9.1-31, the staff requested that the applicant clarify the discrepancy between its response to RAI 9.1-14 and a statement in DCD Tier 2, Revision 3, Section 9.1.3.2, which identifies the maximum heat conditions as resulting from 20 years of operation. In response to RAI 9.1-31, the applicant stated that, at the time the response to RAI 9.1-14 was submitted, the reference to 10 years of spent fuel was correct. Since then, a design change augmented the cooling requirements for the SFP such that, under its most limiting conditions, it now has the capacity to dissipate the decay heat from 20 years of spent fuel plus one full-core offload. The applicant further stated that the change to a 20-year cooling capacity was not significant enough to affect the values for rate of boil-off and makeup that were contained in the response to RAIs 9.1-14 and 9.1-12. This statement was unclear since several other RAI responses indicated that there was an approximately 0.7 megawatt (MW) increase in the heat loads from 10 and 20 years of spent fuel. However, the response to RAI 9.1-13 S01 discussed above shows that, because of margin in the designated FPS flow rate for 10 years of spent fuel, an FPS flow rate of 46 m^3/h (200 gpm) to the ICS/PCCS pools or SFP is sufficient for 20 years of spent fuel. The staff finds that the RAI 9.1-31 response, when augmented by the RAI 9.1-13 S01 RAI response, is acceptable since the FPS flow is sufficient to cool 20 years of spent fuel. Accordingly, based on the above and the applicant's response, RAIs 9.1-31 and 9.1-13 S01 are resolved.

The guidelines in SRP Section 9.1.3 specify that the SFP cleanup system must have the capacity and capability to remove corrosion products, radioactive materials, and impurities so that water clarity and quality will enable safe operating conditions in the pool. DCD Tier 2, Revision 3, Section 9.1.3, states that the spent fuel cleanup system contains a prefilter, a demineralizer, and a poststrainer.

In RAI 9.1-29, the staff requested that the applicant provide a more detailed description of the SFP cleanup system. In response, the applicant stated that each train of the FAPCS is equipped with prefilters upstream of a deep bed demineralizer with mixed bead resin. The filter/demineralizer (F/D) units are designed for a minimum of 90 days between resin changes. The cooling portion of the FAPCS is designed for temperatures up to 100 degrees C (212 degrees F). However, the F/D units will be limited to a lower design temperature to preserve the integrity of the resin. An automatic bypass valve opens to reroute coolant flow around the F/D units, if a high temperature setpoint is exceeded. The F/D units on both trains are flushed to a common backwash receiving tank, which is drained to the LWMS. The cleanup system reduces radioactive materials and other contaminants from the SFP, auxiliary pools, suppression pool, and GDCS pools. The capacity of the FAPCS is sufficient to achieve two water changes per day of all the pools served by the system. The water quality requirements vary depending on the pool. Therefore, the specific water quality requirements for the FAPCS F/D units are

determined using guidance from several sources, including RG 1.13, SRP Section 9.1.3, and the Electric Power Research Institute's (EPRI's) "Advanced Light Water Reactor Utility Requirements Document" (hereafter referred to as the EPRI URD), Revision 8, Volume III, Section 2.2.3.2. The staff finds that the applicant's response is acceptable because the DCD description of the cleanup system adequately addressed the necessary water cleanup equipment including filters and demineralizers, along with water changes approximately every 12 hours. Accordingly, based on the above and the applicant's response, RAI 9.1-29 is resolved.

The guidelines in SRP Section 9.1.3 specify that the applicant should have provisions in place to preclude the inadvertent transfer of spent filter and demineralized media to any place other than the radwaste facility. The discussion in DCD Tier 2, Revision 3, Section 9.1.3, was unclear as to whether the applicant considered such provisions. The staff requested, in RAI 9.1-30, that the applicant provide a description of the provisions. In response, the applicant stated that each F/D unit is equipped with a poststrainer or resin trap that is designed to prevent the inadvertent transfer of contaminants to any location other than the intended radwaste system. The staff finds the applicant's response acceptable because poststrainers or resin traps are currently used in similar applications and are an acceptable way to prevent radwaste from transferring to any place other than the radwaste facility. Accordingly, based on the above and the applicant's response, RAI 9.1-30 is resolved.

Audit of the ESBWR Spent Fuel Pool Water Inventory

On June 3 and 15, 2010, the staff conducted regulatory audits of the supporting information for the SFP minimum water inventory as described in DCD Tier 2, Section 9.1 and Appendix 19A, "Availability Controls Manual" (hereafter referred to as the ACM). A summary of the audit, including participants and audit activities, may be found in the ADAMS at Accession Number ML101680660. Before the audit, the staff identified that the SFP water level in Availability Control (AC) 3.7.4 was potentially inconsistent with information provided in multiple RAI responses, including the responses to (but not limited to) RAIs 9.1-10, 9.1-11, 9.1-18, 9.1-44, 9.1-46, and 9.1-115. The June 3, 2010, audit was primarily focused on understanding the technical basis for AC 3.7.4 through the review of the applicant's supporting calculations. The applicant stated that it would make changes to AC 3.7.4 and the corresponding sections of the DCD to address the issues identified during the audit.

On June 15, 2010, the staff reviewed the applicant's updated analysis of the SFP minimum water inventory, which the applicant revised based on the open items identified during the June 3, 2010, audit. In addition, as a result of the June 3, 2010, audit, the applicant made changes to AC 3.7.1. During the June 15, 2010, audit, the staff reviewed the supporting information for the minimum volume and delivery rate of makeup water to be supplied from 72 hours to 7 days following an accident. The staff also indicated that the applicant should clarify the technical basis for the minimum water inventory of the buffer pool.

The staff identified 11 open items during the June 3 and June 15, 2010, audits. Open items (1) through (9) were identified during the June 3, 2010, audit and open items (10) and (11) were identified during the June 15, 2010, audit. These open items are as follows:

1. Impact of a seismic event on the SFP to maintain SFP cooled and covered with water for 72 hours without any makeup water

2. SFP water level and volume as part of the thermal analysis and boil off calculation

3. Specific anti-siphon devices locations with respect to fuel uncover

4. Technical specifications (TS) were not defined versus availability controls

5. Thermal analysis specific boil off rate from the SFP at 72 hours

6. Seismic events consideration for the buffer pool

7. Thermal analysis and core thermal power considerations

8. Availability Controls related to the FPS and its bases for water makeup

9. Apparent inconsistency between the latest thermal analysis results and the AC B 3.7.1, involving emergency makeup water (1,921 cubic meters (m3) versus 1,151 m3)

10. Clarification of any non-seismic Category I and II connections that could provide a potential drain paths form the SFP and buffer pool

11. Need to identify a water level at 72 hours that meets the requirements of GDC 61, including providing the justification for the water level, and modify the TS limit accordingly

The applicant addressed the open items in its "Revised Response (Revision 2) to Audit Open Items from the Summary of the June 3 and 15, 2010 NRC Regulatory Audits of the ESBWR Spent Fuel Pool Required Water Inventory."

For items (1), (3), and (10), the applicant clarified that both the SFP transfer gates and buffer gates are seismic Category I. The location of anti-siphon holes on piping submerged in the SFP and buffer pool was redefined (previously no lower than 3.05 m (10 ft) above TAF for safe shielding) and these anti-siphon holes are no lower than 10.26 m (33.7 ft) above the TSFA to provide safe shielding in the event of a break at a lower elevation. The applicant also stated that there are no drainage paths or any other pathways by which pool water could be reduced below the minimum level during a seismic event. In addition, the applicant provided a markup to DCD Tier 2, Table 3.2-1 and Section 9.1.3.2, and DCD Tier 1, Table 2.6.2-2, for incorporation into Revision 8 of the DCD.

The staff finds the applicant's response to items (1), (3) and (10) acceptable since anti-siphon holes are no lower than the pool elevation credited in the analysis, which determined the minimum water level at the beginning of the loss of an FAPCS event necessary to support 72 hours of pool heat up. In addition, there are no potential drain paths through which water inventory may be lost during a seismic event and the pool gates are not expected to fail since the gates are designed to seismic Category I requirements. The staff confirmed that the changes identified in the response were incorporated into DCD Revision 8. Accordingly, the staff finds that items (1), (3), and (10) are resolved.

For items (2), (4), and (8), the applicant stated that DCD Tier 2, Revision 7, Chapter 19, ACLCO [Availability Controls Limiting Condition of Operation] 3.7.4 requires that the SFP water level shall be no less than 8.5 m (27.9 ft) above the TSFA. This level was based on an out-of-date calculation. The SFP thermal analysis has since been revised and now shows a bounding boil-off volume of 1,962 m^3 (69,300 ft^3). Therefore, the applicant added a TS surveillance to DCD Tier 2, Chapter 16 (replacing the old ACLCO), which specifies a minimum pool level of greater than or equal to 10.26 m (33.7 ft) above the TSFA in the SFP and RB buffer pool. The minimum

pool level of 10.26 m (33.7 ft) above the fuel assembly bounds the volume of 1,962 m^3 (69,300 ft^3) credited for boil-off of the SFP. In addition, the applicant provided a DCD Tier 2 markup which it incorporated into Revision 8 of the DCD.

The staff finds the applicant's response to items (2), (4), and (8) acceptable since the applicant re-performed the boil off analyses with a bounding boil-off volume of 1,962 m^3 (69,300 ft^3) which resulted in a higher initial water level for the SFP loss of FAPCS event. The initial SFP water level for the maximum SFP heat load conditions of a full core offload plus irradiated fuel in the SFP resulting from 20 years of plant operations was inadequately captured as an ACLCO and has since been adequately described as a TS surveillance related to both an initial SFP water level greater than or equal to 10.26 m (33.7 ft) and water temperature less than or equal to 60 degrees C (140 degrees F) for the loss of SFP event without makeup for 72 hours. The SFP water level and water temperature are normally maintained at 14.35 m (47 ft) and less than 48.9 degrees C (120 degrees F). However, during the maximum SFP heat load conditions of a full core offload plus irradiated fuel in the SFP resulting from 20 years of plant operations, both FAPCS cooling and cleanup trains are needed to maintain the bulk temperature below 60 degrees C (140 degrees F). The staff confirmed that the changes identified in the response were incorporated into the TS in DCD Revision 8. Accordingly, the staff finds that items (2), (4), and (8) are resolved.

For items (5) and (7), the applicant stated that the SFP boil-off calculation determines a bounding boil-off volume for the SFP, but it is not a bounding scenario for makeup water flow. A separate calculation determined the minimum necessary makeup water flow rate at 72 hours at 102 percent core power. This calculation shows that this rate is 39.6 m^3/h (174 gpm), which is bounded by the DCD value of 46 m^3/h (200 gpm). In addition, the applicant provided a DCD Tier 2, AC B 3.7.1 markup for incorporation into Revision 8 of the DCD.

The staff finds the applicant's response to items (5) and (7) acceptable since the calculations included combined decay heat of the fuel in the reactor and the SFP for 72 hours through 7 days following a shutdown that occurs at the end of an operating cycle in which the reactor is run at 102 percent rated power. These conditions are bounding in terms of the combined decay heat of the irradiated fuel in the RPV and SFP and the combined evaporation from the ICS/PCCS pools and the SFP. The staff confirmed that the changes identified in the response were incorporated into DCD Revision 8. Accordingly, the staff finds that items (5) and (7) are resolved.

For item (6), the applicant stated that during a refueling outage, the water volume in the buffer pool communicates freely with the water in the reactor well, equipment pool, and upper fuel transfer pool. There are no potential drainage paths that can cause this pool volume to drain. In addition, the applicant provided a DCD Tier 2 markup related to buffer pool volumes and water levels which was incorporated into Revision 8 of the DCD.

The staff finds the applicant's response to item (6) acceptable since additional water inventory communicates with the buffer pool, and there are no drain paths that would inadvertently drain the buffer pool. The calculations included combined decay heat of the fuel in the reactor and the SFP for 72 hours through 7 days following a shutdown that occurs at the end of an operating cycle in which the reactor is run at 102 percent rated power. These conditions are bounding in terms of the combined decay heat of the irradiated fuel in the RPV and SFP and the combined evaporation from the ICS/PCCS pools and the SFP. In addition, the buffer pool normal water level is 6.7 m (22.0 ft); however, spent fuel is stored in a deep pit that provides an additional 9.5 m (31.2 ft) of submergence. In the buffer pool, a minimum free volume of 288 m^3 (10,200 ft^3) is

provided above the TSFA to accommodate a loss of FAPCS cooling for 72 hours. This minimum volume corresponds to a minimum water level of 7.3 m (24.0 ft) above the TSFA. The staff confirmed that the changes identified in the response were incorporated into DCD Revision 8. Accordingly, the staff that finds item (6) is resolved.

For item (9), the applicant stated that it has addressed the inconsistency between the minimum volumes from 72 hours to 7 days of 1,921 m^3 (67,840 ft^3) and 1,151 m^3 (40,650 ft^3). The value was determined to be an unnecessary detail and was removed from AC B.3.7.1 and the applicant provided a DCD Tier 2 markup related to deleting this information from Revision 8 of the DCD.

The staff finds the applicant's response to item (9) acceptable. The ACM Bases for the minimum volume for emergency makeup between 72 hours to 7 days for the SFP is superseded by the ACM Bases which describes the minimum water volume for both the ICS/PCCS pools and the SFP. This new volume is approximately 3,900 m^3 (1.03×10^6 gallons) for the 72 hours to 7 days duration, which is available in the two firewater storage tanks. The staff confirmed that the changes identified in the response were incorporated into DCD Revision 8. Accordingly, the staff finds that item (9) is resolved.

For Item (11), the applicant stated the proposed TS for the SFP water level requires a minimum water level of 10.26 m (33.7 ft) above the TSFA. The supporting calculation shows that, during a loss of cooling event in which the SFP contains the highest possible heat load, the pool level is reduced by no more than 10.26 m (33.7 ft). Therefore, the spent fuel assemblies are shown to remain covered with water up to the TSFA for 72 hours under the bounding case. The calculation supporting the TS value of 10.26 m (33.7 ft) above the TSFA considered the bounding heat load, which is an SFP that has recently received a full core offload in addition to an accumulated 20 years of spent fuel. The calculation demonstrates by a very conservative methodology that the SFP level could be reduced by no more than 10.26 m (33.7 ft) under the bounding heat load.

The applicant further explained that the TS limit of 10.26 m (33.7 ft) contains safety margin by virtue of the considerable margin built into the SFP boil-off calculation. Some of the margin is explicitly stated (no heat transfer through the pool structure or to the atmosphere), but the most significant margin is implicitly built into the calculation methodology. For example, the residual water in the SFP is not credited with absorbing any heat; whereas, in a realistic event, the entire pool (including residual water) would heat to saturation before any water boils. The assumption that this energy is not absorbed by the residual water results in a conservative overestimation of the volume of water that is vaporized. For an initial water level of 10.26 m (33.7 ft) above the TSFA, there would be significant margin after 72 hours. Therefore, the TS limit of 10.26 m (33.7 ft) is sufficient to meet the guidelines of SRP Section 9.1.3.

The applicant explained that following circumstances were also considered when developing the modifications to the SFP TS:

The normal operating level for the SFP is 14.35 m (47.1 ft) above the pool floor, which is 10.3 m (33.8 ft) above the TSFA.

- The SFP and buffer pool have no mechanism by which they can be drained below 10.26 m above the TSFA. The FAPCS discharges water into the pool, which then overflows into a surge tank. If a discharge line were to break, the anti-siphon holes would preserve the minimum 10.26 m (33.7 ft) coverage.

- If the pool level were to drop below the normal operating level, alarms are provided to alert the control room of a low level.

- The event for which a minimum initial level of 10.26 m (33.7 ft) above the TSFA is credited as highly improbable. The event consists of a refueling outage with a full core offload and an accumulated 20 years of spent fuel in the SFP, concurrent with a seismic event at the precise moment the last fuel bundle is placed in the SFP. For the heat loads associated with a normal refueling outage (i.e., no full core offload) and with less than 20 years of accumulated spent fuel, the heat loads in the SFP are much smaller and a lower initial level would be sufficient to provide cooling for 72 hours.

- Pool level instrumentation measures collapsed water level (see markup to DCD Tier 2, Section 9.1.3.5), thereby conservatively avoiding false readings because of steam vapors above the actual water level.

In summary, the applicant concluded that the proposed TS limit of 10.26 m (33.7 ft) provides adequate assurance that the fuel will remain covered for 72 hours after a loss of pool cooling, thereby meeting the guidelines of SRP Section 9.1.3 and the requirements of GDC 61.

The staff finds the applicant's response to item (11) acceptable as follows. The proposed TS change considered the bounding heat load for the 72 hour period following loss of FAPCS with 20 years of spent fuel in the SFP and a complete core offload. Under such conditions, the staff concludes that water level would still be above the TSFA, which ensures that active spent fuel is covered which is consistent with established NRC policy. SECY-98-161, "The Westinghouse AP600 Standard Design as it Relates to the Fire Protection and the Spent Fuel Pool Cooling Systems," states the following:

> The SFP is designed such that using only safety-related makeup, water is maintained above the spent fuel assemblies for at least 7 days following a loss of the SFP cooling system. In accordance with the design, the minimum water level required to achieve sufficient cooling is the sub-cooled, collapsed level (without vapor voids) required to cover the top of the fuel assemblies.

Design features such as anti-siphon devices and seismic Category I gates will limit the loss of SFP water inventory. In addition, the safety-related instrumentation for the SFP water level determination will measure collapsed water level. The staff confirmed that the changes identified in the response were incorporated into DCD Revision 8. Accordingly, the staff finds that item (11) is resolved.

Based on the above, the staff finds that the ESBWR design complies with the requirements of GDC 61.

9.1.3.3.3 GDC 63

GDC 63 requires that appropriate systems shall be provided in fuel storage and radioactive waste systems and associated handling areas to (1) detect conditions that may result in loss of residual heat removal capability and excessive radiation levels and (2) initiate appropriate safety actions. SRP Section 9.1.3 Revision 2, guidelines identify that GDC 63 is addressed through provisions to detect the loss of heat removal function through the use of loss of flow and temperature alarms and to detect conditions that would result in excessive radiation through the use of low-level alarms and radiation monitoring alarms.

Regarding the conditions that may result in the loss of residual heat removal capability which are directly related to excess radiation levels, DCD Tier 2, Revision 9, Section 9.1.3.5, describes system instrumentation which includes water levels, water temperatures, and flow and pressure for the FAPCS.

9.1.3.3.3.1 ***Surge Tank and Pool Water Level***

The normal FAPCS water source is the skimmer surge tanks, which are filled by overflow from the SFP. A level detector and transmitter mounted on a local panel monitor the skimmer surge tank level. The skimmer surge tank level is displayed in the MCR. In addition to level indication, this level signal is used to initiate low- and high-water-level alarms and to operate the makeup water control valve for the skimmer surge tank.

Panel-mounted pressure transmitters for the FAPCS pump suction and discharge pressure are provided locally. A pump trip signal is generated on low suction pressure to provide pump protection. The pressure transmitters send signals to pressure indicators in the MCR. An orifice-type flow element is located on the downstream side of each pump discharge check valve. A local panel-mounted flow transmitter sends the signals from these transmitters to flow indicators in the MCR.

The SFP and buffer pool have two wide-range, safety-related level transmitters that transmit signals to the MCR. These signals are used for water-level indication and to initiate high and low-level alarms.

The ICS/PCCS pool has two local, panel-mounted, safety-related level transmitters. Both transmitter signals are indicated on the safety-related displays and sent through the gateways for nonsafety-related display and alarms. Both signals are validated and used to control the valve in the makeup water supply line to the ICS/PCCS pool.

In RAI 9.1-18, the staff requested that the applicant describe how SFP water level instrumentation satisfies the requirements of GDC 63. In response, the applicant stated that the level instruments on the surge tank provide for automatic makeup water from the condensate storage and transfer system (CS&TS) when the forced cooling trains are being used, but they are not designed to satisfy the requirements of GDC 63. The applicant stated that, when forced cooling is not available, the surge tank level instruments become irrelevant and safety-related cooling is provided by the heating up and boiling of water in the SFP. In this situation, the safety-related SFP level instruments, which will sound an alarm in the MCR upon a low SFP water level, satisfy the requirements of GDC 63. Because the safety-related cooling is provided by passive boil-off, these level instruments are not required to initiate any additional safety actions.

The staff determined that the applicant's response to RAI 9.1-18 was partly acceptable since safety-related SFP level instruments alarm in the MCR upon low SFP water level, which is an adequate parameter to detect the loss of heat removal functions. While the staff finds the use of water-level instrumentation acceptable, the response did not fully address the SFP water level instrumentation relative to the top of the fuel. The response also did not explain how the operators respond to MCR alarms. Therefore, the staff generated RAI 9.1-18 S01.

In response to RAI 9.1-18 S01, the applicant stated that the instrumentation is redundant safety-related instruments for the SFP that provide level indication spanning the normal water level to the TAF, and that no operator action is credited during the first 72 hours because sufficient

water inventory exists to allow for 72 hours of boil-off without exposing the TAF. Following 72 hours, the operator responds by replenishing the pools as necessary through the emergency connections to the FPS or an alternative water source.

The staff determined that the applicant's response to RAI 9.1-18 S01 was unacceptable since it did not specify the amount of water between the TAF and the SFP low level alarms. The applicant stated that no operator actions are needed for 72 hours, and the low level setpoint was not determined such that there is at least 72 hours before the TAF is reached, assuming a loss of forced cooling during the maximum decay heat load conditions. For this reason, the staff generated RAI 9.1-18 S02 to address the low level setpoint.

In response to RAI 9.1-18 S02, the applicant stated that there are redundant safety-related level instruments for the SFP that provide level indication spanning the normal water level to the TAF for stored fuel assemblies with a low level alarm just below normal water level. The applicant's response also referenced calculations that conservatively predict SFP water height (i.e., approximately 2.0 m (approximately 6.5 ft) above the TAF) 72 hours after a loss of forced cooling (these calculations are discussed further with RAI 9.1-44 above). The RAI response also discussed additional alarm setpoints for the TAF and shielding (3.05 m (10.0 ft)) and DCD Tier 2, Revision 5, included the alarm setpoints.

The staff determined that the response to RAI 9.1-18 S02 was partly acceptable since the setpoints provide adequate warning to the operator that SFP forced cooling is lost or that loss of coolant level may affect adequate cooling. However, the response to RAI 9.1-18 S02 did not fully address how the buffer pool nonsafety-related water level instrumentation, as described in DCD Tier 2, Revision 5, Section 9.1.3.5, satisfies the requirements of GDC 63. The staff determined that the buffer pool, as a spent fuel storage area that may hold up to 154 spent fuel assemblies, should have safety-related water level instrumentation similar to that for the SFP; therefore, the staff generated RAI 9.1-18 S03 to address this issue.

In RAI 9.1-18 S03, the staff requested that the applicant explain how GDC 63 is satisfied for the buffer pool and designate appropriate equipment, such as the water level instrumentation, as safety-related. The staff asked the applicant to provide information regarding the alarms for the buffer pool similar to the design for the SFP in response to RAI 9.1-18 S02. RAI 9.1-18 was being tracked as an open item in the SER with open items.

In response to RAI 9.1-18 S03, the applicant stated that it will upgrade the level instruments in the buffer pool to safety-related and provide a DCD markup for Revision 6.

The staff finds that the applicant's response to RAI 9.1-18 S03 and the DCD revisions are acceptable since the applicant designated the water-level instruments in the buffer pool as safety-related. DCD Revision 9 identifies the SFP and buffer pool alarm locations. In design clarifications to DCD Tier 2, Revision 6, the applicant proposed to modify DCD Tier 2, Section 9.1.3.5, to clarify that the SFP and buffer pool water level instrumentation initiate alarms both locally and in the MCR. The staff finds that the design clarifications are acceptable since the applicant is making the clarification of the alarm locations consistent with the guidelines of SRP Section 9.1.3, Revision 2, which indicates that alarms should initiate both locally and in the MCR. The staff confirmed that the applicant incorporated these design clarifications into DCD Revision 7. In response to RAI 9.1-18 S03, the applicant modified DCD Tier 2 to state that the buffer pool has safety-related water-level instrumentation; however, the applicant did not implement this change in DCD Tier 1. In RAI 9.1-131, the staff asked the applicant to revise DCD Tier 1, Section 2.6.2, Design Description Item (9), and DCD Tier 1, Table 2.6.2-2, Item 9,

to include the buffer pool and to clarify that the water level instrumentation is safety-related. In the applicant's response to RAI 9.1-131, the applicant stated that it would revise DCD Tier 1, including Table 2.6.2-2, as requested for the buffer pool level instruments in Revision 7 to the DCD.

The staff finds that the applicant's response is acceptable since the applicant incorporated appropriate information on the buffer pool safety-related water level instrumentation into DCD Tier 1, Revision 7. Accordingly, based on the above, the applicant's responses, and DCD changes, RAIs 9.1-18 and 9.1-131 are resolved. The staff notes that in the applicant's "Revised Response (Revision 2) to Audit Open Items from the Summary of the June 3 and 15, 2010 NRC Regulatory Audits of the ESBWR Spent Fuel Pool Required Water Inventory," the applicant clarified that under bounding loss of forced cooling conditions, the water level is at the TSFA (rather than TAF) at 72 hours. The staff evaluation of the revised water level, water level instruments, and the applicant's "Revised Response (Revision 2) to Audit Open Items from the Summary of the June 3 and 15, 2010 NRC Regulatory Audits of the ESBWR Spent Fuel Pool Required Water Inventory," is above.

In RAI 9.1-41 the staff requested that the applicant describe how the performance of the safety-related water level instrumentation, which is provided for the SFP and ICS/PCCS pools, provide accurate level indication during boiling conditions. In response to RAI 9.1-41, the applicant indicated that the level instruments in the ICS/PCCS pools are located in the expansion pool area away from the heat load, which is restricted to the heat exchanger sub-compartments. Because the boil-off occurs in these sub-compartments, coolant flows from the expansion pool into these compartments. Therefore, the level instruments for these pools are not subjected to boiling conditions that could affect their accuracy. Boiling of water in the SFP may introduce some inaccuracy in level measurement. However, because boiling decreases the density of the water, the level instruments can only indicate a water level that is less than the actual level. Therefore, the instruments conservatively err on the side of safety. Setpoint methodology considers the inaccuracy in level measurement when determining the setpoints for the needed actions.

The staff determined that the applicant's response to RAI 9.1-41 was unacceptable since it was not clear how a decrease in the density of water (resulting from to an increase in water temperature) in the SFP will lead to a conservative water-level measurement. The staff requested, in RAI 9.1-41 S01, that the applicant provide a detailed description of the instrumentation to be used, including the elevation of the instrumentation taps in the SFP relative to the TAF; explain how it will be affected by the increase in temperature and the boiling conditions; and clarify why this results in a conservative estimate. RAI 9.1-41 was being tracked as an open item in the SER with open items.

In response to RAI 9.1-41 S01, the applicant stated that it had not chosen a specific instrumentation design. However, the applicant addressed as an example, the effect of boiling on level instrumentation that relies on differential pressure. The applicant explained that the measurement of water level for a boiling pool using differential pressure would be conservative since water expands with boiling; thus differential pressure instrumentation would indicate a lower than actual water level at boiling. In addition, the applicant modified DCD Tier 1, Revision 5, Table 2.6.2-2, to add an ITAAC description of the SFP level instrumentation.

The staff finds that the applicant's response to RAI 9.1-41 S01 is acceptable since the applicant added an instrumentation ITAAC addressing adequate operating ranges for the SFP and ICS/PCCS pools. In addition, the staff noted that, in the revision to the response to RAI 14.3-

449 S02, the applicant modified DCD Tier 1, Revision 6, Table 2.6.2-2, Item 9, to include a tolerance for the accuracy of the water level instrumentation of 300 mm (1ft). Based on the above, the applicant's responses, and DCD changes, RAI 9.1-41 is resolved. The staff notes that in the applicant's "Revised Response (Revision 2) to Audit Open Items from the Summary of the June 3 and 15, 2010 NRC Regulatory Audits of the ESBWR Spent Fuel Pool Required Water Inventory," the applicant clarified that under bounding loss of forced cooling conditions, the water level is at the TSFA (rather than the TAF) at 72 hours. The staff evaluation of the revised water level, water level instrumentation, and the applicant's "Revised Response (Revision 2) to Audit Open Items from the Summary of the June 3 and 15, 2010 NRC Regulatory Audits of the ESBWR Spent Fuel Pool Required Water Inventory," is above.

Related to the potential loss of water inventory, in RAI 9.1-17 the staff requested that the applicant describe how potential radioactive leakage from the fuel storage pools and the FAPCS is collected and processed.

In response to RAI 9.1-17, the applicant stated that leakage channels are provided behind each weld of the fuel pool liners to collect leakage. All leaks are channeled to headers and drain lines from which they are routed to a small collection tank with level-sensing devices. Tank level and leakage inflow information is displayed in the MCR with an alarm feature to prompt the operator for action if abnormal leakage occurs. Flow rates are monitored, and radioactive contaminated liquid is piped to the equipment and floor drainage system sumps and is then processed as described in DCD Tier 2, Revision 2, Section 9.3.3.

The staff finds that the applicant's response to RAI 9.1-17 is acceptable since the leakage from the fuel storage pools is adequately addressed with leakage channels and a collection and monitoring system; however, the staff requested that the applicant include this information in the DCD. In the response to RAI 9.1-17, the applicant stated that it revised DCD Tier 2 to include the requested information. DCD Section 9.1.3.2 states that the reactor well, equipment storage pool, buffer pool, upper and lower fuel transfer pools, cask pool, and ICS/PCCS pools are equipped with stainless steel liners and are equipped with leak detection drains as part of the FAPCS. All leak detection drains are designed to permit free gravity drainage to the LWMS. The staff finds that the applicant's response is acceptable since this requested information was described in Sections 9.1.3.2 and 9.3.3 of the DCD. Accordingly, based on the above and the applicant's response, RAI 9.1-17 is resolved. The staff confirmed that DCD Revision 3 contains the changes described above.

9.1.3.3.3.2 *Water Temperature*

The fuel and auxiliary pools have nonsafety-related temperature elements and local panel-mounted temperature transmitters that send signals to the MCR for water temperature indication and high-temperature alarms. In the IC/PCC pool, each condenser vault also has temperature elements and local panel-mounted temperature transmitters that send signals to the MCR for water temperature indication and high-temperature alarms. The upstream and downstream piping of the two heat exchangers in the C/C trains have temperature elements and local panel-mounted temperature transmitters that send signals to the MCR.

The staff finds that water temperature monitoring as described above is acceptable to support the RTNSS functions of the FAPCS since they include typical local and MCR controls and indications.

9.1.3.3.3.3 *Fuel and Auxiliary Pool Cooling System Flow and Pressure*

Panel-mounted pressure transmitters for the FAPCS pump suction and discharge pressure are provided locally. A pump trip signal is generated upon low suction pressure to provide for pump protection, with the pressure transmitters sending signals to pressure indicators in the MCR. A local panel-mounted flow transmitter sends the signals from these transmitters to flow indicators in the MCR.

The staff finds that FAPCS system flow and pressure instrumentation, as described above, is acceptable to support the RTNSS functions of the FAPCS since it includes typical local and MCR controls and indications.

In summary, the ESBWR design meets the requirements of GDC 63. The staff concludes that the buffer pool and SFP, which are designed for spent fuel storage, have adequate safety-related water level instrumentation with indications in the MCR for detection of conditions that may result in the loss of residual heat removal capability. For both the buffer pool and SFP, the water levels and free volumes are sufficient to ensure that, following a loss of forced cooling without active cooling water makeup for 72 hours, as described above, the water levels in the pools remain above the TAF and after 72 hours, firewater or another water source can be provided through safety-related connections.

9.1.3.3.4 GDC 34 and 38

As stated previously, in addition to satisfying the criteria of SRP Section 9.1.3, the staff evaluated the FAPCS emergency makeup capability to the IC/PCC pool for long-term cooling, in accordance with SRP Sections 5.4.7 and 6.2.2. The staff verified that the design complied with the requirements of GDC 34, as it relates to having suitable redundancy of FAPCS components to ensure that, for either a LOOP or a loss of onsite power, the long-term cooling function of the ICS can be accomplished assuming a single failure.

DCD Tier 2, Revision 3, Section 9.1.3, states that the FAPCS is designed to provide post-accident recovery (defense-in-depth) functions of the SPC, LPCI, drywell spray, and alternate SDC, which all take suction from the suppression pool. The staff requested, in RAI 9.1-20, that the applicant describe the water flow rate and heat removal capacity to perform these defense-in-depth functions, how those values are determined, and how the FAPCS will be designed and tested to provide those flow rates and heat removal capacities. In response, the applicant stated that the FAPCS is not required to satisfy any flow rate or heat removal requirement for these functions. The applicant stated that the FAPCS functions of SPC, low-pressure injection, drywell spray, and alternate SDC are not essential to plant safety, and no credit is taken for them in any safety analysis. The applicant stated that the FAPCS provides these functions to the extent it has available capacity, but that it is not specifically designed to perform these functions. The staff determined this response was inadequate. The ESBWR PRA described in DCD Tier 2, Revision 3, Chapter 19, credits the FAPCS in performing certain functions (e.g., low-pressure injection and SPC).

In RAI 9.1-20 S01, the staff requested that the applicant provide the basis for concluding that successful actuation of the assumed number of FACPS trains is adequate to satisfy the PRA success criterion for the respective coolant injection and heat removal functions. RAI 9.1-20 was being tracked as an open item in the SER with open items. The applicant responded that although the safety analysis does not credit either low pressure injection or SPC, they are credited in the ESBWR PRA. The applicant's response stated that a single train of the FAPCS

is capable of pumping water from the suppression pool to prevent core damage, in the event the GDCS is not providing makeup water to the reactor, or removing core decay heat from the suppression pool at a rate to prevent the containment from exceeding its design pressure.

The staff determined that this response was inadequate and, in RAI 9.1-20 S02, asked the applicant to provide design parameters for the FAPCS trains and calculations that demonstrate that these design parameters are adequate. In response, the applicant referenced computer computations performed with the MAAP computer code (a thermal-hydraulics code used by the nuclear industry) that document the FAPCS's ability to perform the above RTNSS functions. However, the RAI response was not acceptable because the applicant failed to provide the requested performance requirements. In RAI 9.1-20 S03, the staff requested the applicant to provide the performance requirements of the FAPCS. In response, the applicant committed to add FAPCS heat exchanger performance requirements to DCD Tier 1 in Revision 6. However, in its submission, the applicant did not clarify how the performance requirement parameters satisfy the PRA success criteria. In RAI 9.1-20 S04, the staff asked the applicant to (1) identify and include in the DCD and NEDO-33201, "ESBWR Design Certification Probabilistic Risk Assessment," the FAPCS performance requirements for the SPC mode during accident conditions considered in the PRA, and (2) provide assumptions and results showing that the FAPCS and reactor component cooling water system (RCCWS) can remove heat as assumed in the PRA, and clarify that the FAPCS can remove heat as assumed for all applicable scenarios evaluated in the PRA. In response, the applicant revised the DCD to provide the nominal performance requirements of the FAPCS pump and heat exchanger, discussed the assumptions and results showing that the FAPCS can remove heat as assumed in the PRA, and stated that the FAPCS is capable of providing heat removal for the scenarios in which it is credited in the PRA.

However, in reviewing Revision 6 of the DCD, the staff determined that the design specifications provided in DCD Tier 1, Table 2.6.2, and DCD Tier 2, Table 9.1-8, appear to pertain only to the ability of the FAPCS heat exchangers to remove 8.3 MW of heat from the suppression pool, while the PRA credits the FAPCS with being able to remove approximately 34 MW of heat under accident conditions. In previous RAI responses, the applicant indicated that MAAP runs have shown that if the differential temperature were high enough across the heat exchanger primary to secondary boundary, and if the flow was sufficiently high on the secondary side, then 34 MW could be removed by a heat exchanger. While this is true mathematically, it did not provide assurance to the staff that the heat exchanger physically can withstand the effects of such high temperatures (e.g., voiding, seal failure, water hammer, thermal expansion) or that the associated FAPCS pumps can handle the thermal effects (e.g., net positive suction head [NPSH] issues).

In RAI 9.1-20 S05, the staff asked the applicant to provide a write up in DCD Tier 2, Section 9.1 and Chapter 19, that provides reasonable assurance that the FAPCS heat exchangers and pumps will be capable of removing the assumed heat load credited in NEDO-33201, Revision 4. In addition, the staff asked that the applicant evaluate and modify the DCD Tier 1 plant service water system (PSWS) interface requirements, as appropriate, to be consistent with the changes made to DCD Tier 2, Section 9.2, in response to this RAI. In response, the applicant committed to clarify the DCD in Revision 7 and to provide additional assurance that the heat exchangers are capable of operating effectively given the assumed differential temperature between the hot and cool sides. In a revised response to RAI 9.1-20 S05, the applicant modified its response to state that the heat exchangers and pumps are designed to physically withstand the higher-than-normal temperatures associated with the PRA analysis. In particular, the pumps and heat exchangers will be capable of withstanding a differential temperature of 76 degrees C (136.8

degrees F) based on the maximum FAPCS temperature and the minimum RCCWS temperature. The staff finds that the RAI response is acceptable since the limiting differential temperature is based on the maximum FACPS temperature of 91 degrees C (195.8 degrees F) and the minimum RCCWS temperature of 15 degrees C (59 degrees F). Accordingly, based on the above, the applicant's responses, and DCD changes, RAI 9.1-20 is resolved. The staff confirmed that DCD Revision 7 contains the changes described above.

The staff requested, in RAI 9.1-151, that the applicant address potential gas accumulation in the FAPCS. Generic Letter (GL) 2008-01, "Managing Gas Accumulation in Emergency Core Cooling, Decay Heat Removal, and Containment Spray Systems," identifies that gas accumulation has been known to cause water hammer, gas binding in pumps, and inadvertent relief valve actuation that may damage pumps, valves, piping, and supports and may lead to loss of system functions. In response, the applicant stated that while the FAPCS does interface with a high pressure system (RWCU/SDC); this interface is normally isolated and prevented from opening by a high pressure interlock, as described in DCD Tier 2, Section 9.1.3.2. Additionally, the FAPCS is designed to minimize the risk of gas accumulation that could result from gas buildup following maintenance activities or long periods of nonuse since the FAPCS piping is sloped to minimize the number of locations where gas can accumulate, and high point vents are provided at these points to ensure that the system can be purged of any gases that are present. Also, plant operation and maintenance procedures ensure that piping and components are vented to avoid water hammer and gas binding in pumps. Water hammer and gas binding are addressed in the Plant Operating Procedure Development Plan as COL 13.5-2-A. The FAPCS is not relied upon to perform immediate, automatic, safety-related functions, as described in DCD Tier 2, Section 19A.8.4.7; therefore, adequate time is available for operators to implement these procedures to ensure that the system is properly vented. The applicant proposed a revision to Section 9.1.3.2 adding that high point vents and component vents are used to avoid gas accumulation and procedures are used to ensure that sufficient measures are taken to avoid water hammer and gas binding in pumps, with a pointer to DCD Tier 2, Section 13.5.2.

The staff finds the applicant's response to RAI 9.1-151 acceptable as follows. The applicant adequately addressed gas accumulation during operations and post maintenance, and stated that sloped lines, component vents, system vents, and operational and maintenance procedures will be utilized to prevent component or system damage. Any leakage of high pressure coolant from the RWCU/SDC through the safety-related check valves and motor operated shutdown valves into the FAPCS are relieved by a pressure relief valve. In addition, FAPCS is not immediately placed into service for either LPCI or alternate SDC modes; therefore, adequate time would be available to permit proper venting by the operators. Accordingly, based on the above and the applicant's response, RAI 9.1-151 is resolved. The staff confirmed that DCD Revision 7 contains the changes described above.

The staff requested in RAI 9.1-19 that the applicant describe how adequate NPSH is ensured for these functions, consistent with the guidance of SRP Section 6.2.2, Revision 4, assuming the respective pool is at saturation temperature for the pressure at its surface.

In response, the applicant stated that the FAPCS pumps are located approximately 14 m (approximately 46 ft) below the bottom of the suppression pool, which is a significantly higher available NPSH than exists for pumps performing these same functions in most BWRs.

In response to RAI 9.1-19 S01, the applicant provided a rationale to demonstrate that sufficient NPSH will be available to the FAPCS pumps when performing their low pressure injection and

SPC functions. However, the applicant did not provide an actual analysis for the FAPCS design parameters or a method for calculating design parameters. The NPSH required for these functions must be known in order to conclude that the pumps will be successful in performing the functions that are assumed in the PRA. RAI 9.1-19 was being tracked as an open item in the SER with open items. In RAI 9.1-19 S02, the staff requested the applicant to provide the calculations to demonstrate adequate NPSH for the FAPCS pumps. The applicant provided these calculations. The staff finds these calculations acceptable since the applicant identified limiting conditions for minimum NPSH and the available NPSH exceeds the limiting minimum NPSH. Accordingly, based on the above and the applicant's response, RAI 9.1-19 is resolved.

In accordance with SRP Section 6.2.2, Revision 4, the staff verified that the design complied with GDC 38 as it relates to having suitable redundancy of FAPCS components to ensure that, for either a LOOP or a loss of onsite power, the long-term cooling function of the PCC can be accomplished assuming a single failure.

Criterion III.20 of SRP Section 6.3, Revision 2, states that an intermediate heat transport system used to provide long-term cooling capability should be capable of sustaining a single active or passive failure without loss of function. The staff requested, in RAI 9.1-21, that the applicant describe how the long-term cooling function of the primary containment cooling system is satisfied, assuming an active failure of valve F420 (a post LOCA fillup isolation valve) or a passive failure of the emergency makeup header pressure boundary.

In response, the applicant stated that, to provide additional protection against a potential single active failure of the FPS makeup water supply, the connection of the FAPCS will be modified to include two parallel valves in the makeup water supply line from the FPS to the FAPCS for both the ICS/PCCS pools and SFPs. In DCD, Revision 2, the applicant revised DCD Tier 1, Figure 2.6.2-1, and DCD Tier 2, 9.1.1, accordingly. The staff finds this acceptable.

However, the applicant also stated that passive failures in the piping of the common header do not need to be considered for low-pressure, low-temperature piping that is seldom used. The staff addressed this issue separately in RAI 6.3-79, in which it requested that the applicant clarify whether the ESBWR design takes credit for any passive component during the long-term, post-LOCA period and confirm conformance to the SRP. RAI 9.1-21 was being tracked as an open item in the SER with open items.

The applicant's response to RAI 6.3-79 stated that the ESBWR design meets the guidance of SRP 6.3. For the ESBWR design, conformance to the requirement of adequate long term cooling (30 days) is assured and demonstrated for any LOCA in which the water level can be restored and maintained at a level above the top of the reactor core. DCD Tier 2, Section 6.3.3, presents the results of the short term (0 to 2000 seconds) emergency core cooling system (ECCS) performance evaluation and DCD Tier 2, Section 6.2.1.1.3 presents the results of the long term (0 to 72 hours) ECCS performance evaluation. The applicant considered a range of line breaks for long term cooling (72 hours to 30 days): bottom drain line break, GDCS injection line break, main steam line break, feedwater line break, isolation condensation return line break. As a result of this analysis, the applicant identified that the worst case event is due to an isolation condensation return line break. During this event, RPV water level is maintained greater than 8.5 m (27.9 ft) for a period of 30 days. At this water level, the reactor core is covered at a level above the top of the fuel and long term cooling is assured. The initiation set point to open the GDCS equalization lines is when the RPV water level drops below Level 0.5 (1.0 m [3.2 ft] above the TAF, or 8.453 m [27.7 ft] from the RPV bottom). For all credible single failures considered, the long term RPV water level following a LOCA remains higher than 8.45

m (27.7 ft) for a period of 30 days. The equalization lines are not actuated under these situations. However, if the RPV water level drops below Level 0.5, these equalization lines would be actuated. After actuation, these equalization lines provide the long term post-LOCA makeup water to the RPV from the suppression pool. The suppression pool water level is about 10 m (32.8 ft) from the RPV bottom, or 2.5 m (8.2 ft) above the TAF. The addition of the suppression pool water provides additional assurance that the reactor core is covered at a level above the TAF for at least 30 days. The staff finds the response is acceptable since it clarifies how the design provides adequate long term cooling considering a single active or passive failure. Accordingly, based on the above, the applicant's responses, and DCD changes, RAIs 9.1-21 and 6.3-79 are resolved.

In RAI 9.1-22, the staff requested that the applicant clarify how the ICS/PCCS pools are configured and how subcompartments communicate to share inventory. The staff also requested that the applicant clarify how the long-term cooling function of the PCCS is satisfied, assuming a single active or passive failure affecting the makeup line from the FAPCS. In response, the applicant stated that there are two large expansion pools on either side of the RB. These two pools are each divided into three separate compartments. The three compartments of each expansion pool are interconnected by valves that are locked open, and the three compartments of each expansion pool communicate and are treated as a single pool volume. The applicant stated that the two expansion pools are connected to each other and can share water inventory with each other through normally closed, parallel, redundant valves connecting to the equipment storage pool and reactor well. The valves are designed to open upon receiving a low-level signal from either of the two expansion pools and allow the ICS/PCCS pools to utilize the inventory in the equipment storage pool and reactor well.

The applicant also stated that within each of the two expansion pools there are five smaller subcompartments: three for PCCS heat exchangers and two for ICS heat exchangers. Each of these subcompartments also contains a locked-open maintenance valve that allows for communication to the rest of the inventory in the expansion pool. When water in the subcompartments is drawn down by boil-off, makeup water from the expansion pool will flow in through these maintenance valves. If a heat exchanger requires service, these valves can be closed, and the subcompartment can be pumped dry.

The staff finds this acceptable and considers this RAI resolved since the applicant adequately addressed the pool configuration and explained how the sub compartments are shared via valve design assuming a single failure. In addition, the applicant revised the emergency makeup line to include two parallel valves, as described in response to RAI 9.1-13 above which addressed single failures. Accordingly, based on the above and the applicant's response, RAI 9.1-22 is resolved.

The staff requested in RAI 9.1-32, that the applicant clarify how many lines actually discharge into the ICS/PCCS pools since the expansion pools are not normally connected. In response, the applicant clarified that one makeup line discharges to the pool while redundant safety-related connections allow water to flow freely between the expansion pools as well as the dryer/separator pool and reactor well. During an accident in which pool water is boiling off, a low-level setpoint in either of the ICS/PCCS expansion pools causes the redundant safety-related connections to the equipment storage pool to open. The applicant indicated that a weir will be maintained between the reactor well and the equipment storage pool that allows the inventory of the two pools to communicate down to a certain level. The applicant also explained that the one makeup line is low pressure and low temperature safety-related piping, designed to Seismic Category I requirements, which operates infrequently. As discussed in RAIs 9.1-13 and

9.1-22 above, this line has redundant active components to address single active failures. The staff confirmed that the makeup line is designated as ASME Section III, Class 3, with a seismic classification of Category I in DCD Tier 2, Table 3.2-1. The staff finds that the response is acceptable since the one safety-related makeup line can effectively supply multiple expansion pools, because of the redundant safety-related connections between the pools and the redundant active components on the makeup line. Based on the above and the applicant's response, RAI 9.1-32 is resolved.

DCD Tier 2, Revision 9, Section 9.1.3.2, describes the SFP cleanup system. The SFP cleanup system and various auxiliary systems are designated as nonsafety-related systems and are designed accordingly. These systems are evaluated to ensure that their failure cannot affect the functional performance of any safety-related system or component.

The staff verified that the cleanup system is designed with the capability to maintain acceptable pool water conditions. The staff verified that the applicant provided the following, as discussed in Criterion III.7 of SRP Section 9.1.3: (1) means for mixing to produce a uniform temperature throughout the pool, (2) capability for processing the refueling canal coolant during refueling operations, and (3) provisions to preclude the inadvertent transfer of spent filter and demineralized media to any place other than the radwaste facility.

Each water treatment unit is equipped with a prefilter, a demineralizer, and a poststrainer. A bypass line is provided to permit bypass of the water treatment unit, when necessary. The prefilter and demineralizers of the water treatment units are located in shielding cells so that radiation exposure of plant personnel is within acceptable limits.

In RAI 9.1-23, the staff requested that the applicant describe how the FAPCS is used to manage pool water inventory and how waste from the water treatment subsystem is handled. In response, the applicant stated that DCD Tier 1, Revision 1, Figure 2.6.2-1, indicates the capability to discharge water to the LWMS by way of the overboarding lines connected to valves on a FAPCS discharge line. The applicant also identified that spent resin from the FAPCS water treatment subsystem is discharged to the solid waste management system. The staff finds that the response is acceptable since the applicant identified flow paths for excess water and radioactive waste. Accordingly, based on the above and the applicant's response, RAI 9.1-23 is resolved.

In DCD Tier 2, Revision 5, Section 9.1.3.2, the applicant stated that the FAPCS "suppression pool suction line is conservatively designed to preclude a rupture between the pool and the containment isolation valves." In RAI 9.1-97, the staff asked the applicant to provide a reference to where in the DCD the design details and justification that this line cannot rupture under any circumstances can be found. In response, the applicant stated that it would modify DCD Tier 2, Section 9.1.3.2, in Revision 6 to state that an analysis would be performed consistent with DCD Tier 2, Section 3.6.2.1.2, on the suppression pool suction line to show that, the piping from the pool to the containment isolation valve as moderate energy piping, remains below the threshold limit for postulating leakage cracks. The staff finds that the RAI response is acceptable since the modifications in DCD Revision 6 support the conclusion that the failure frequency of the suppression pool line from the pool to the containment isolation valves is sufficiently small that a break in that line need not be postulated. Accordingly, based on the above, the applicant's responses, and DCD changes, RAI 9.1-97 is resolved.

Based on the above, the staff finds the ESBWR design to be in compliance with the requirements of GDC 34 and GDC 38.

9.1.3.3.5 Inspections, Tests, Analyses, and Acceptance Criteria

Based on the staff's review of DCD Tier 2, Revision 5, Section 9.1, the staff determined that DCD Tier 1 omitted several apparent design features important to safety. In RAI 14.3-443, the staff asked the applicant to explain why the FAPCS design criteria were not included in ITAAC or specified as Tier 1 material. The staff requested the applicant to address the following eight items:

1. The FAPCS consists of two physically separated cooling and cleanup trains.

2. The FAPCS is designed to provide drywell spray and alternate SDC.

3. DCD Tier 2, Revision 5, Section 9.1.3.1, describes those portions of the FAPCS that are not specifically defined as safety-related as being seismic Category II. Table 2.6.2-2 does not mention this quality.

4. All piping between the RWCU/SDC system and the nonsafety-related check valves (upstream of the MOVs) is designed to withstand the full reactor pressure.

5. With the exception of the suppression pool suction line, anti-siphoning devices are used on all submerged FAPCS piping to prevent unintended drainage of the pools.

6. The suppression pool suction line is conservatively designed to preclude a rupture between the pool and the containment isolation valves.

7. The electrical power supplies and the control and instrumentation of the two FAPCS trains and their supporting systems are electrically and physically separated.

8. Piping and components completely separate from FAPCS pool cooling piping provide flow paths for post-accident makeup water transfer.

In response, the applicant responded as follows:

<u>Item (1)</u>: Regarding the design feature that the FAPCS consists of two physically separated cooling and cleanup trains, the applicant stated and the staff confirmed that existing ITAAC cover the design commitment. This is acceptable and the staff agrees that existing ITAAC are adequate.

<u>Item (2)</u>: Regarding the design feature that the FAPCS is to provide drywell spray and alternate SDC, the applicant stated that these functions are not safety-related and are not credited in the licensing basis; therefore, they should not be included in DCD Tier 1. The staff concluded that this was acceptable since this design feature provides neither safety-related nor RTNSS functions and is not required by the regulations.

<u>Item (3)</u>: Regarding the fact that the system description in DCD Tier 2, Revision 5, describes portions of the FAPCS as being seismic Category II, but does not mention this in Table 2.6.2-2, the applicant stated that, although the FAPCS does perform certain RTNSS functions, these functions were not the reason the FAPCS is designed to seismic Category II; therefore, it does not need to be mentioned in DCD Tier 1, Table 2.6.2-2. The staff finds this response acceptable since seismic Category II is a defense-in depth measure and is not related to the RTNSS function.

Item (4): Regarding the feature that all piping between the RWCU/SDC system and the nonsafety-related check valves (upstream of the MOVs) is to be designed to withstand the full reactor pressure, the applicant stated that it would add an ITAAC for this design feature to DCD Tier 1, Table 2.6.2-2. The staff reviewed the revised table and finds this modification acceptable since it adequately addressed the pressure rating of the interface to the RWCU/SDC system out to the MOVs.

Item (5): Regarding the design feature that with the exception of the suppression pool suction line, anti-siphoning devices be used on all submerged FAPCS piping to prevent unintended drainage of the pools, the applicant noted that an ITAAC added to DCD Tier 1 in response to RAI 14.3-442 addressed this issue. The staff agrees and finds this acceptable since it adequately addressed the anti-siphoning devices on submerged FAPCS piping. Section 9.1.2.3 of this report further discusses the response to RAI 14.3-442.

Item (6): Regarding the design feature addressed in DCD Tier 2, Revision 5 that the suppression pool suction line be conservatively designed to preclude a rupture between the pool and the containment isolation valves, the applicant stated that the ITAAC in DCD Tier 1 Table 3.1-1, Item 3, covers this design commitment. The staff determined that this response was unacceptable because the referenced ITAAC item has no effect on the probability of a pipe break. However, in response to RAI 9.1-97, the applicant modified DCD Tier 2, Section 9.1.3.2, describing why the piping would not crack. The applicant clarified that it will perform an analysis on the suppression pool suction line, in accordance with DCD Tier 2, Section 3.6.2.1.2, for moderate energy piping, to show that the piping from the pool to the containment isolation valves remains below the threshold limit for postulating leakage cracks. The staff examined Revision 6 to the DCD and finds that the modifications support the conclusion that the failure frequency of the suppression pool line from the pool to the containment isolation valves is sufficiently small that a break in that line need not be postulated. Accordingly, based on the above and the applicant's response to RAI 9.1-97 discussed above, item 6 of this RAI is resolved.

Item (7): Regarding the design feature that the electrical power supplies, and the control and instrumentation of the two FAPCS trains and their supporting systems are electrically and physically separated, the applicant referred the staff to its response to RAI 14.3-394, S01 and the corresponding DCD Tier 1 markup. The staff determined that this response was unacceptable. The response to RAI 14.3-394 does not necessarily ensure that the electrical loads and cables are physically separated from one another since the RAI response addresses the separation to breakers, but not the separation of loads drawn from the breakers. In RAI 14.3-443 S01, the staff asked the applicant to provide criteria in DCD Tier 1 that ensure that the control cables, instrument cables, and power cables for equipment in the two FAPCS trains are physically and electrically separated. In response, the applicant stated that it would add a new ITAAC in DCD Tier 1, Revision 6 (Table 2.6.2-2, item 16) to test and inspect that "The nonsafety-related control cables, instrument cables and power cables for equipment in the FAPCS trains A and B are physically separated and electrically independent." The staff determined that the proposed description of physical separation of FAPCS equipment in DCD Tier 1 was unsatisfactory.

In addition, DCD Tier 2 does not identify separation criteria for non-Class-1E systems such as FAPCS. In RAI 9.1-133 the staff asked the applicant to modify the DCD Tier 1, Revision 6, Table 2.6.2-2, item 16, acceptance criteria to more specifically discuss physical separation criteria necessary to keep the FAPCS trains' electrical equipment appropriately physically separated to prevent both trains from being damaged simultaneously by a design basis event,

including load drop. In the supplemental response to RAI 9.1-133, the applicant clarified that the electrical equipment supporting the two FAPCS trains is routed through separate areas and is not routed through areas in which heavy loads could be transported. Any heavy loads that are being transported in the RB or FB that have the potential to simultaneously compromise both FAPCS trains would be handled by single failure-proof cranes. The staff finds this clarification acceptable, but determined that the applicant needed to include this clarification in an ITAAC. In a revised response to RAI 14.3-449 S02, the applicant included new design descriptions in DCD Tier 1 and new ITAAC (Table 2.6.2-2, items 18a and 18b) reflecting the physical separation criteria described in the supplemental response to RAI 9.1-133. (RAI 14.3-449 dealt with numerous ITAAC inspectability concerns, including concerns similar to those identified in RAIs 14.3-443 and 9.1-133 and thus served as a convenient means for the applicant to address these and other ITAAC related RAIs). The staff finds that cumulatively the responses are acceptable to address the concerns raised in RAI 14.3-443, Item 7 and RAI 9.1-133 since the applicant added ITAAC for the independence and physical separation of control, instrument, and power cables for FAPCS equipment and ITAAC to ensure that heavy load drops would not impact the electrical equipment of both trains of the FAPCS. Based on the above, the applicant's responses, and DCD changes, RAIs 14.3-443, Item 7, and 9.1-133 are resolved.

Item (8): Regarding the design feature that there are piping and components completely separate from the FAPCS pool cooling piping that provide flow paths for post-accident makeup water transfer, the applicant noted the separation of the piping presently shown in DCD Tier 1, Figure 2.6.2-1. The staff finds this acceptable since the DCD presently addresses post-accident makeup piping and components separated from FAPCS. Based on the above and the applicant's response to eight parts of RAI 14.3-443, RAI 14.3-443 is resolved.

DCD Tier 2, Revision 5, Section 9.1.3.1, describes the FAPCS as being a nonsafety-related system, with the exception of the piping and components relied upon for containment isolation, refilling the ICS/PCCS pools and SFP, and interface with the RWCU/SDC. DCD Tier 1, Figure 2.6.2-1, showed seismic Category I piping, but the piping was not listed in DCD Tier 1, Table 2.6.2-1. DCD Tier 1, Table 2.6.2, Design Commitments 2, 3, and 4 provide ITAAC for seismic Category I piping identified in DCD Tier 1, Table 2.6.2-1; however, no piping was so identified. In RAI 14.3-444, the staff asked the applicant to revise DCD Tier 1, Section 2.6.2, ITAAC for seismic Category I piping to reference Figure 2.6.2-1 or modify Table 2.6.2-1. In response, the applicant committed to adding the FAPCS piping described as safety-related to DCD Tier 1, Table 2.6.2-1, and DCD Tier 2, Table 9.1-3. The applicant also stated it would add the GDCS interconnecting pipes to DCD Tier 1. Accordingly, based on the above and the applicant's response, the staff concludes that RAI 14.3-444 is resolved since the safety-related portion of the FAPCS was added to DCD Tier 1 including the GDCS interconnecting pipes. However, the applicant did not fully implement the RAI response in DCD Revision 6.

In RAI 9.1-134, the staff requested the applicant to clarify the list of FAPCS safety-related items. In response to RAI 9.1-134, the applicant clarified that the GDCS interconnecting pipes are not part of the emergency water flow paths to the SFP and provided an associated DCD markup. The staff finds that the RAI response is acceptable since the applicant clarified that the FAPCS has four safety-related items, as described in the response to RAI 14.3-444. The staff confirmed that the applicant incorporated the DCD changes in DCD Revision 7. Accordingly, based on the above and the applicant's response, RAI 9.1-134 is resolved.

9.1.3.3.6 10 CFR 20.1101(b)

The staff verified that the design complied with the requirements of 10 CFR 20.1101(b). 10 CFR 20.1101(b) requires the licensee to use, to the extent practicable, procedures and engineering controls based on sound radiation protection principles to maintain occupational doses and doses to the public ALARA. Section 9.1.2 of this report previously addressed the requirements of 10 CFR 20.1101 as they relate to fuel storage pools.

DCD Tier 2, Revision 9, Section 12.3.1.4.2, describes the FAPCS as being designed to operate continuously to handle the spent fuel cooling load and to reduce pool water radioactive contamination in all of the major pools in the ESBWR. Included are two independent F/D units which serve to remove radioactive contamination. These units are the highest radiation level components in the system. Each unit is located in a concrete shielded cubicle that is accessible through a shielded hatch. Provisions are made for remotely backflushing the units when filter and resin material are spent. This removal of radioactivity from contaminated material reduces the component radiation level considerably and serves to minimize exposures during maintenance. All valves (inlet, outlet, recycle, vent, and drain) to the F/D units are located outside the shielded cubicles in a separate shielded cubicle, together with associated piping, headers, and instrumentation. The radiation level in this cubicle is sufficiently low to permit maintenance to be performed. Piping potentially containing resin is continuously sloped downward to the backwash tank. Personnel access to shielded system components is controlled to minimize personnel exposure. Shielding for the components is designed to reduce the radiation level to less than 10 μSv/hr (1 m rem/hr) in adjacent areas where normal access is permitted. Operation of the system is accomplished from the MCR and local control panels which are located where design radiation levels are less than 25 μSv/h (2.5 mrem/h) and normal personnel access is permitted.

The staff finds the FAPCS design as is relates to ALARA acceptable. Design provisions, such as equipment shielding, sloped piping, and provisions for backflushing of unit filters, incorporate ALARA principles. Section 12.3 of this report further discusses the ALARA program.

9.1.3.3.7 10 CFR 20.1406

Section 9.1.2.3 of this report presents the staff's evaluation of the fuel storage pools liner welds in accordance with 10 CFR 20.1406.

DCD Tier 2, Revision 9, Table 12.3-18, describes the provisions regarding RG 4.21, "Minimization of Contamination and Radioactive Waste Generation: Life-Cycle Planning," design objectives related to FAPCS for:

- Minimizing leaks and spills (Design Objective 1)
- Leak detection methods and early detection of leaks to avoid release of contamination from undetected leaks and to minimize contamination of the environment (Design Objective 3)
- Decreasing the spread of contaminant from the source (Design Objective 4)

With the exception of the suppression pool suction lines, anti-siphoning devices are used on all submerged FAPCS piping to prevent unintended draining of the pools. The FAPCS is designed with features, including drains, gates, and weirs, to prevent drainage of coolant inventory below an adequate shielding depth. The FAPCS is also designed to provide for the collection,

monitoring, and drainage of pool liner leaks from the SFPs, auxiliary pools, and ICS/PCCS pools to the LWMS. The SFP is equipped with drainage paths behind the liner welds. These paths are designed to prevent stagnant water buildup behind the liner plate, prevent the uncontrolled loss of contaminated pool water, and provide liner leak detection and measurement. The reactor well, equipment storage pool, buffer pool, upper and lower fuel transfer pools, cask pool, and ICS/PCCS pools are also equipped with stainless steel liners and with leak detection drains. All leak detection drains are designed to permit free gravity drainage to the LWMS. All FAPCS lines penetrating the containment that do not have a post-accident recovery function are automatically isolated upon receipt of a containment isolation signal from the Leak Detection and Isolation System (LD&IS).

The staff finds that these design provisions for the FAPCS conform to the guidelines of RG 4.21 and meets the requirement of 10 CFR 20.1406. Sections 12.4 and 12.7 of this report further address the ESBWR design, in accordance with 10 CFR 20.1406.

9.1.3.3.8 Operating Experience Considerations

The NRC issued Inspection and Enforcement (IE) Bulletin 84-03, "Refueling Cavity Water Seal," to address the potential failure of refueling cavity seals to ensure that fuel uncovery while refueling remains an unlikely event. The bulletin requested licensees to evaluate the potential for and consequences of a refueling cavity seal failure. Information Notice (IN) 84-93, "Potential for Loss of Water from the Refueling Cavity," provides additional information concerning refueling cavity seal failures. IN 84-93 notes that refueling cavities can also be drained because of failures associated with other seals and as a consequence of valve misalignments. Inadvertent drain down of the refueling cavity can result in a loss of cooling for fuel in transit and may cause a loss of water inventory and cooling for fuel in the buffer pool. Because the water inventory in the refueling cavity is also needed for shielding purposes, high radiation levels can also result from exposed fuel and reactor components. Therefore, the staff issued RAI 9.1-128 and RAI 9.1-128 S01 requesting that the applicant address operating experience considerations associated with IE Bulletin 84-03.

In response to the RAIs, the applicant made several changes to the DCD. The applicant revised DCD Tier 2, Table 1C-2, to show that information pertaining to IE Bulletin 84-03 is provided in DCD Tier 2, Sections 6.2.1.1.2, 9.1.4.21, and 12.4.4. The applicant also revised these sections of the DCD to reflect the applicant's RAI responses. Revisions to DCD Tier 2, Section 9.1.4.8, provided additional information concerning the seal plugs discussed in subsection A below. The staff's evaluation is based on the information that was provided in response to the RAIs and incorporated in Revision 7 of the DCD.

A. Refueling Seals

DCD Tier 2, Revision 9, Section 6.2.1.1.2, describes the refueling cavity bellows seal (RCBS) for the ESBWR, which are shown in Figure 6.2-35. The RCBS is a permanently installed seismic Category I mechanical component that is designed for a 60-year life. It is made of stainless steel for corrosion resistance, and RCBS fabrication and installation are in accordance with applicable codes and standards. The design includes a secondary seal and capability to continuously monitor any leakage that may occur through the primary (bellows) seal. The RCBS is physically located below the reactor vessel flange so as not to be subject to damage during refueling operations, and it is protected from dropped objects by steel cover plates. The RCBS will be monitored for leakage and periodic maintenance and inspections will be performed in accordance with vendor recommendations. The RCBS design is robust and

should not fail catastrophically during a seismic event, and it is not vulnerable to a single failure. Design provisions are included so that any leakage that occurs can be readily identified and corrected, and procedures specified in DCD Tier 2, Revision 9, Section 13.5.2 (and referred to below), for maintaining refueling cavity integrity will ensure that the RCBS is properly maintained over the life of the plant. Therefore, the RCBS is considered to be acceptable.

DCD Tier 2, Revision 9, Section 9.1.4.8, indicates that before refueling, the main steam and the depressurization valve (DPV)/ ICS line nozzles will be plugged to prevent water outflow from the reactor. The plugs that are used for this application are made of corrosion-resistant materials, are designed using a safety factor of 5 or more, and include redundant seals (one pneumatic and one mechanical). Each seal is individually leak tested before use during a refueling outage, and periodic maintenance and inspections will be performed in accordance with vendor recommendations. These plugs are typical of designs that have been used previously for similar applications. Based on operating experience, these plugs should provide reliable service, and a failure of one seal type should not result in significant leakage past the plug and cause the refueling cavity to drain catastrophically. Procedures specified in DCD Tier 2, Revision 9, Section 13.5.2, and referred to below for maintaining refueling cavity integrity will ensure that these plugs are properly maintained over the life of the plant. Therefore, these plugs are considered to be acceptable.

B. Refueling Cavity Drainage Paths

In addition to the flow paths associated with the seals discussed in Item A above, the applicant's response addressed other flow paths that could potentially cause the refueling cavity to drain. These other flow paths include manways that are provided between the reactor cavity and the drywell, IFTS, fine motion control rod drive (FMCRD) penetrations, and other flow paths that may result from valve misalignments.

Manway covers are fitted with gaskets or o-rings to establish an effective seal and, based on previous experience, are not expected to experience catastrophic failure after the refueling cavity is flooded. Any significant leakage is typically identified and corrected while the refueling cavity is being flooded and before fuel is removed from the reactor vessel. If a significant leak should occur while moving fuel, the manway cover will limit the leakage to well within the makeup capability that is available from the FAPCS and FPS. Therefore, the staff finds that there is reasonable assurance that manways and manway covers will not pose a threat to the refueling cavity water inventory.

DCD Tier 2, Revision 9, Section 9.1.4.12, describes the IFTS, and Section 9.1.4 of this report provides the staff's evaluation of the IFTS design and the potential for draining the refueling cavity. Consequently, this section provides no further evaluation of the IFTS.

DCD Tier 2, Revision 9, Section 4.6.2.1.4, discusses FMCRD maintenance. Like previous BWR product lines, reactor vessel drainage through FMCRD penetrations is prevented by back-seating the respective control rod before removing its FMCRD. Maintenance procedures specified in DCD Tier 2, Section 13.5.2, ensure that the control rods are properly back-seated before removing their respective FMCRDs. Based on operating experience, this approach is effective in preventing catastrophic drainage from BWR (CRD) penetrations. Therefore, the staff finds that there is reasonable assurance that FMCRD penetrations will not pose a threat to the refueling cavity water inventory or the inventory of water in the reactor vessel.

Valve misalignments can cause the reactor (and refueling cavity) to drain when aligning systems for operation and establishing maintenance boundaries. However, these evolutions are performed in accordance with strict procedural controls that are established as specified in DCD Tier 2, Revision 9, Section 13.5.2, and are subject to NRC inspection. Based on operating experience, this approach is effective in preventing catastrophic drainage from systems connected to the reactor vessel. Therefore, the staff finds that there is reasonable assurance that valve misalignments will not pose a threat to the refueling cavity water inventory or the inventory of water in the reactor vessel.

C. Refueling Cavity Leakage Detection

As discussed in DCD Tier 2, Revision 9, Section 6.2.1.1.2, leakage from the RCBS is readily detectable and isolable. During refueling, the refueling cavity pool level is constantly monitored and annunciation is provided for a drop in level. The dryer/separator storage pool, upper fuel transfer pool, and reactor well all have local, nonsafety-related, panel-mounted level transmitters that annunciate high and low water level in the control room. The buffer pool has two wide-range safety-related level transmitters that provide level indication and annunciation both locally and in the control room. The drywell sump will also alarm if there is significant leakage from the refueling cavity seal. Consequently, plant operators will be made aware of any significant leakage from the refueling cavity that develops while the reactor is being refueled and will be able to take corrective actions as appropriate. Therefore, provisions which are provided to enable operators to monitor refueling cavity level and for alerting operators to a loss of inventory are acceptable.

D. Impact and Mitigation of Refueling Cavity Leakage

DCD Tier 2, Revision 9, Section 12.4.4, discusses the impact and mitigation of refueling cavity leakage. As indicated in DCD Tier 2, Section 12.4.4, and based on the considerations discussed above, a rapid drain down of the refueling cavity is not likely to occur. Level indication and annunciation are provided to alert operators to any leakage from the refueling cavity that develops, and any leakage that does occur should be well within the makeup capability that is provided by the FAPCS and the FPS. Fuel in transit can be quickly placed in the deep pit of the buffer pool, which will provide at least 6 m (19.7 ft) of water above the fuel, and multiple fuel bundles in transit at the same time are not anticipated. Therefore, cooling for the fuel bundle in transit and for those stored in the deep pit of the buffer pool will not be compromised, and shielding that is needed for reactor components and spent fuel will be maintained. Section 12.5 of this report evaluates dose considerations associated with refueling operations.

E. Procedural Controls for Maintaining Refueling Cavity Integrity

DCD Tier 2, Revision 9, Section 13.5.2, specifies in COL Information Items 13.5-4-A and 13.5-5-A that COL applicants will develop a plant operating procedures development plan which will include plant operating procedures, procedures for performing maintenance, and procedures related to refueling cavity integrity (among others). For example, some of the procedures that are called for in this regard include procedures for monitoring refueling cavity seal leakage, responding to refueling cavity and buffer pool drain down events, and performing periodic maintenance and inspection of the refueling cavity seal and the main steam and ICS plugs. The procedures specified in DCD Tier 2, Revision 9, Section 13.5.2, will ensure that refueling cavity seals are periodically inspected and properly maintained, valve alignments and maintenance boundary conditions are properly specified and controlled, operators are cognizant of water

inventory in the refueling cavity and are alerted to any significant leaks that develop, and appropriate actions are specified and taken to preserve the integrity of the refueling cavity and maintain cooling for spent fuel during the conduct of refueling activities. The specified procedures are commensurate with the considerations discussed above and sufficient for maintaining refueling cavity integrity and spent fuel cooling when the reactor is being refueled. Therefore, the procedural controls that are called for in DCD Tier 2, Section 13.5.2, are necessary and appropriate, and the NRC staff considers them to be acceptable.

The considerations referred to in DCD Tier 2, Revision 9, Sections 9.1.4.8 and 9.1.4.21, and discussed above ensure that during the conduct of refueling operations, the integrity of the refueling cavity, cooling for spent fuel bundles that are in transit or located in the deep pit of the buffer pool, and shielding that is needed for reactor components and spent fuel will continue to be maintained. Therefore, the applicable requirements referred to in the above Regulatory Basis Section are satisfied. The staff finds that the RAI response and DCD changes are acceptable since they provided expected information on the refueling seals, refueling cavity drain paths, refueling cavity leakage detection, impact and mitigation of refueling cavity leakage, and procedural controls for refueling cavity integrity. Based on the above, the applicant's responses, and DCD changes, RAI 9.1-128 is resolved.

9.1.3.4 *Conclusion*

Based on the review discussed above, the staff has finds that the FAPCS design complies with GDC 2, 4, 34, 38, 61, and 63. Because the ESBWR design is a single unit, GDC 5 is not applicable. Based on the discussion above, the staff also finds that the ESBWR design conforms to 10 CFR 20.1406 and 10 CFR 20.1101(b).

9.1.4 Light-Load Handling System (Related to Refueling)

SRP Section 9.1.4, Revision 3, Subsection III, identifies for review purposes that the light-load handling system (LLHS) does not include equipment used to handle heavy loads (i.e., weights exceeding that of one fuel assembly and its handling tool). However, the LLHS section of the DCD discusses equipment designed to handle heavier loads that is also used to maneuver light loads; this section evaluates the use of such equipment for light loads.

9.1.4.1 *Regulatory Criteria*

The staff reviewed the LLHS in accordance with SRP Section 9.1.4, Revision 3. The staff's acceptance of the ESBWR design is based on meeting the relevant requirements of the following regulations:

- GDC 2, as it relates to the ability of structures, equipment, and mechanisms to withstand the effects of earthquakes

- GDC 5, as it relates to the capability of shared equipment and components to perform safety functions

- GDC 61, as it relates to radioactivity release as a result of fuel damage and the avoidance of excessive personnel radiation exposure

- GDC 62, as it relates to prevention of criticality accidents

Compliance with the requirements of GDC 2 depends on adherence to the guidance of Regulatory Positions C.1 and C.2 of RG 1.29. The ESBWR design is a single-unit station, and the requirements of GDC 5 are not applicable to the single unit. Compliance with the requirements of GDC 61 and 62 depends on adherence to the guidance of ANSI/ANS 57.1-1992.

9.1.4.2 *Summary of Technical Information*

The LLHS related to refueling consists of all components and equipment used from the handling of the new fuel from the receiving station to the loading of spent fuel into the shipping cask. The system for the ESBWR design includes the equipment designed to facilitate the periodic refueling of the reactor, specifically the FB crane, RB crane, refueling machine, fuel-handling machine, IFTS, fuel preparation machine, new fuel inspection stand, dryer/separator strongback, chimney partition strongback, head strongback/tensioner, grapples and hoists, and associated handling tools and devices. The handling of fuel during refueling is controlled by a series of interlocks to ensure that fuel-handling procedures are maintained.

Fuel transfer from the point of receipt up to inspection, storage, and placement in the reactor core is accomplished with fuel grapples. A general purpose fuel grapple is used when fuel movement is performed by the FB crane on the FB floor before placement in the fuel preparation machine and transfer to the SFP or buffer pool. During refueling operations, however, fuel movement is performed in the FB by the fuel-handling machine and in the RB by the refueling machine telescoping grapples.

Both the refueling machine and the fuel handling machine always maintains a safe water shielding depth equivalent to 3.05 m (10 ft) over the active fuel during transit.

9.1.4.2.1 FB Crane

The FB crane is required for lifting heavy components (e.g., fuel containers, fuel assemblies during inspection, and the fuel shipping cask) and tools up to and over the refueling floor. It is also used during plant maintenance activities to move light loads such as inspection equipment consoles on the FB floor. The FB crane’s required light-load lifting tasks during fuel handling include lifting the fuel bundle from the shipping container and placing it in the new fuel inspection stand and removing the channeled fuel assembly from the fuel inspection stand and placing it in the fuel preparation machine.

The FB crane, supported on its tracks on the FB wall structural columns, consists of two parallel girders along which the trolley traverses their span. It is classified as seismic Category I to maintain crane functional and structural integrity.

9.1.4.2.2 RB Crane

The RB crane is used for lifting large, heavy components and tools up to and over the refueling floor. It is also used during plant maintenance activities to move light loads, such as inspection equipment consoles on the RB floor; during plant operation, the RB crane handles small tools and equipment normally used during inspection and servicing activities. During fuel transport, the RB crane is also called upon to move and store pool gates. The RB is also classified as seismic Category I.

The RB crane consists of two parallel girders along which the trolley traverses their span. It is classified as seismic Category I to maintain crane functional and structural integrity.

9.1.4.2.3 Refueling Machine

The refueling machine located in the RB is used to transport fuel and reactor components to and from the buffer pool storage, the IFTS, and the reactor vessel. The machine spans the buffer pool on tracks that traverse the refueling floor. A telescoping mast and grapple suspended from a trolley system lifts and orients fuel assemblies for placement in the core or storage rack. A second auxiliary hoist is provided for handling smaller lightweight tools. The machine is controlled from an operator station on the refueling machine.

A position-indicating system and travel limit computer are provided to locate the grapple over the vessel core and prevent collisions with pool obstacles. Two auxiliary hoists are provided for in-core servicing. In its retracted position, the grapple provides water shielding over the active fuel during transit. The fuel grapple hoist has a redundant load path so that no single component failure will result in a fuel bundle drop. Interlocks are provided on the machine for the following purposes:

- Prevent hoisting a fuel assembly over the vessel with a control rod removed,
- Prevent collision with fuel pool walls or other structures,
- Limit travel of the fuel grapple,
- Engage the interlock grapple hook with the hoist load and hoist-up power,
- Ensure correct sequencing of the transfer operation in the automatic or manual modes.

The refueling machine has a position-indicator system to indicate to the operator which core fuel cell the fuel grapple is accessing. Interlocks and a monitor are provided to prevent the fuel grapple from operating on a fuel cell in which the control rod is not properly oriented for refueling.

A series of mechanically activated switches and relays provides monitor indications on the operator’s console for grapple limits, hoist and cable load conditions, and confirmation that the grapple’s hook is either engaged or released.

A series of load cells are installed to provide automatic shutdown whenever threshold limits are exceeded for either the fuel grapple or the auxiliary hoist units.

The refueling machine is classified as nonsafety-related seismic Category I. Except for hoisting speed, the fuel hoist is design to meet the requirements of NUREG–0554, “Single-Failure Proof Cranes for Nuclear Power Plants,” and ASME Code NOG-1, “Rules for Construction of Overhead and Gantry Cranes (Top Running Bridge, Multiple Girder).”

9.1.4.2.4 Fuel-Handling Machine

The fuel-handling machine, located in the FB, is used to transport fuel and reactor components to and from the IFTS and the spent fuel storage and equipment storage racks. It is also used to move spent fuel to the shipping cask. The machine spans the SFP on embedded tracks in the fuel handling floor. A telescoping mast and grapple suspended from a trolley system are used to lift and orient fuel assemblies for placement in the cask or storage rack. The machine is controlled from an operator station on the fuel-handling machine.

A position-indicating system and travel limit computer are provided to locate the grapple over the spent fuel racks and the IFTS and to prevent collisions with pool obstacles. An auxiliary hoist is provided for additional servicing. The grapple in its retracted position provides water shielding over the active fuel during transit. The fuel grapple hoist has a redundant load path so that no single component failure will result in a fuel bundle drop. Interlocks are provided on the machine to do the following:

- Prevent collision with fuel pool walls or other structures,
- Limit travel of the fuel grapple,
- Engage the interlock grapple hook with the hoist load and hoist-up power
- Ensure correct sequencing of the transfer operation in the automatic or manual modes.

The fuel-handling machine has a position-indicator system to indicate to the operator which core fuel cell the fuel grapple is accessing. Interlocks and a monitor are provided to prevent the fuel grapple from operating on a fuel cell in which the control rod is not properly oriented for refueling.

A series of mechanically activated switches and relays provides monitor indications on the operator’s console for grapple limits, hoist and cable load conditions, and confirmation that the grapple’s hook is either engaged or released.

A series of load cells are installed to provide automatic shutdown whenever threshold limits are exceeded for either the fuel grapple or the auxiliary hoist units.

The fuel-handling machine is classified as nonsafety-related seismic Category I. Except for hoisting speed, the fuel hoist is designed in accordance with the guidance of NUREG–0554 and ASME NOG-1.

9.1.4.2.5 Fuel Transfer System

The ESBWR is equipped with an IFTS. The arrangement of the IFTS consists of a terminus at the upper end in the RB buffer pool that allows the fuel to be tilted from a vertical position to an inclined position before transport to the SFP. There is a means to lower the transport device (i.e., a carriage), a means to seal off the top end of the transfer tube, and a control system to effect transfer. The ESBWR has a lower terminus in the FB storage pool and is able to tilt the fuel into a vertical position allowing it to be removed from the transport cart. Controls contained in local control panels effect the transfer. In the event of a power failure, the carriage and valves may be manually operated to allow completion of an initiated fuel transfer. There is a means to seal off the upper and lower end of the tube while allowing filling and venting of the tube. The IFTS provides a means of cooling fuel assemblies during fuel transfer.

The IFTS tubes and supporting structure are designed to withstand an SSE without failure of the basic structure or compromising the integrity of adjacent equipment and structures. The portion of the IFTS transfer tube assembly from where it interfaces with the upper fuel pool, the portion of the tube assembly extending through the building, the drain line connection, and the lower tube equipment (valve, support structure, and bellows) are designated as nonsafety-related and seismic Category I. The winch, upper upender, and lower terminus are designated as nonsafety-related and seismic Category II. The remaining equipment is designated as nonsafety-related and nonseismic.

The IFTS penetrates the RB at an angle down to the IFTS pit in the fuel storage pool in the FB. The lower terminus of the IFTS, which is anchored to the bottom of the inclined fuel transfer pool, allows for thermal expansion (i.e., axial movement relative to the anchor point in the RB). The lower terminus allows for differential movement between the anchor point in the RB and the fuel pool terminus and allows it to have rotational movement at the end of the tube relative to the anchor point in the RB. The lower end interfaces with the fuel storage pool with a bellows to seal the space between the transfer tube and the SFP wall.

The IFTS carriage primarily handles nuclear fuel using a removable insert and control blades in a separate insert in the transfer cart. Other contaminated items may be moved in the carriage using a suitable insert.

For radiation protection, personnel access into areas of high radiation or areas immediately adjacent to the IFTS is controlled. Access to any area adjacent to the transfer tube is controlled through a system of physical controls, interlocks, and an annunciator for the following reasons:

- Controls prevent personnel from inadvertently or unintentionally being left in those areas at the time the access doors are closed.

- During IFTS operation or shutdown, personnel are prevented from either (1) reactivating the IFTS while personnel are in a controlled maintenance area, or (2) entering a controlled IFTS maintenance area while irradiated fuel or components are in any part of the IFTS.

- Both an audible alarm and flashing red lights are provided both inside and outside any maintenance room to indicate IFTS operation.

- Radiation monitors with alarms are provided both inside and outside any maintenance area.

- A system of key locks in both the IFTS main operation panel and in the control room allows access to any IFTS maintenance area.

9.1.4.2.6 General Purpose Grapple

The general purpose grapple performs many tasks and is primarily used on the auxiliary hoist of either the refueling or fuel-handling machines. It is designed to fit a standard fuel bail, which is replicated on certain tooling for handling purposes. One example of such a purpose is handling the underwater vacuum cleaner.

The fuel grapple is equipped with a mounted television camera, lighting system, and instrumentation system consisting of mechanical switches and indicator lights. This system provides the operator with a positive indication that the grapple is properly aligned and oriented and that the grappling mechanism is either extended or retracted.

The general purpose grapple, when using an extension cable, can also be attached to the auxiliary hook of the FB crane as the need arises for handling new fuel.

9.1.4.2.7 Fuel Preparation Machine

Two fuel preparation machines are mounted on the wall of the SFP and are used to assist in the loading of new fuel into the spent fuel storage pool racks and for rechanneling spent fuel

assemblies. The machines are also used with fuel inspection fixtures to provide an underwater inspection capability.

Each fuel preparation machine consists of a work platform, a frame, and a movable carriage. The frame and movable carriage are located below the normal water level in the SFP, thus providing a water shield for the fuel assemblies being handled. The fuel preparation machine carriage has an up-travel-stop to prevent raising fuel above the safe water shield level. The operator places assembled new fuel in the fuel preparation machine, the carriage is lowered, and the fuel is removed from the fuel preparation machine using the fuel handling machine.

9.1.4.2.8 New Fuel Inspection Stand

The new fuel inspection stand is a vertical frame mounted in a pit that supports two fuel bundles contained in a mechanically driven inspection carriage. In the carriage, the lower tie plate of each fuel bundle rests on a bearing seat, and at the top, each fuel assembly is supported in a separate bearing assembly. The fuel assemblies can be individually rotated about their longitudinal axis to permit viewing all sides. The fuel channel is placed on the fuel bundle in the new fuel inspection stand.

9.1.4.2.9 Dryer Separator Strongback

The dryer/separator strongback is a lifting device used for transporting the steam dryer or the steam separators between the reactor vessel and the storage pools. The strongback structure has a hook box with two hook pins in the center for engagement with the RB crane sister hook. The strongback has a socket with a remotely operated pin on the end of each arm for engaging it to the four lift eyes on the steam dryer or shroud head.

The strongback is designed such that one hook pin and one main beam of the cruciform is capable of carrying the total load of 160 metric tons (176 tons), and no single component failure could cause the load to drop or swing uncontrollably out of the safety-related level attitude. The strongback conforms to the provisions of NUREG–0612, “Control of Heavy Loads at Nuclear Power Plants: Resolution of Generic Technical Activity A-36,” issued July 1980, and ANSI-14.6, “Standard for Special Lifting Devices.”

9.1.4.2.10 Head Strongback/Tensioner

The RPV head strongback stud-tensioning system is an integrated piece of equipment consisting of a strongback, a multi-station rotating frame with stud tensioners, nut and washer handling tools, stud-cleaning tools, a nut and washer rack, and a service platform.

The strongback structure has a hook box with two hook pins in the center for engagement with the reactor service crane sister hook. Extending from the center section are arms to connect to the circular monorail. The four arms have a lift rod for engagement to the four lift lugs on the RPV head. The rotating frame is connected to the strongback arms and four additional arms equally spaced between the strongback arms. The rotating frame positions the stations of the stud tensioning and nut and washer handling tools above the stud circle of the reactor vessel and serves to suspend stud tensioners and nut and washer handling devices. The nut and washer rack is attached to the strongback and surrounds the RPV flange. The head strongback rotating frame serves the following functions:

- Lifting of vessel head—the strongback, when suspended from the RB crane main hook, will transport the RPV head plus the rotating frame with all its attachments between the reactor vessel and storage on the pedestals.

- Tensioning of vessel head closure—the strongback with rotating frame, when supported on the RPV head on the vessel, carries multiple stations of stud tensioners; nut and washer handling tools; its own weight; the strongback; and storage of nuts, washers, and associated tools and equipment

- Storage with RPV Head—the strongback with rotating frame, when stored with the RPV head holding pedestals, carries the same load as outlined in the second bullet above.

- Storage without RPV head—during reactor operation, the strongback and rotating frame are stored on four separate pedestals.

The strongback, with its lifting components, is designed to meet the provisions of NUREG–0612 and ANSI-14.6. After completion of welding and before painting, the lifting assembly is proof load tested and all load-affected welds and lift pins are magnetic-particle inspected.

The steel structure is designed in accordance with the Manual of Steel Construction issued by the American Institute of Steel Construction. Aluminum structures are designed in accordance with the Aluminum Construction Manual written by the Aluminum Association.

The strongback is tested in accordance with paragraph 16-1.2.2.2 of ASME/ANSI B30.16, "American National Standard for Overhead Hoists," such that one hook pin and one main beam of the structure are capable of carrying the total load, and no single component failure will cause the load to drop. ASME Code Section IX, "Welder Qualification," is applied to all welded structures.

9.1.4.3 *Staff Evaluation*

The staff verified that the design complied with the requirements of GDC 2 and the guidelines of RG 1.29, Regulatory Positions C.1 and C.2. The LLHS is housed within the FB and the RB, which are seismic Category I, flood- and tornado-protected structures. Although fuel-handling system components are not required to function following an SSE, critical components of the fuel-handling system are designed to seismic Category II requirements so that they will not fail in a way that would result in unacceptable consequences, such as fuel damage or damage to safety-related equipment. The DCD indicates that standard dynamic analyses using the appropriate response spectra are performed to demonstrate compliance with design requirements for the refueling and fuel handling machine. In RAI 9.1-33, the staff requested that the applicant provide the dynamic analyses for fuel-handling system components. RAI 9.1-33 was being tracked as an open item in the SER with open items.

In response, the applicant noted that dynamic analysis of seismic Category I and II refueling equipment is not performed until the final structural configuration of the equipment has been determined as part of the normal equipment delivery for the plant. The staff finds that this approach is acceptable, but, in RAI 9.1-33 S01, requested that the applicant revise the DCD to include a reference to RG 1.29 for meeting the provisions of GDC 2 related to fuel-handling components and to confirm that ITAAC 5 and 6, described in DCD Tier 1, Revision 4, Table 2.16.1-1 will demonstrate conformance to RG 1.29. In response, the applicant committed to revise DCD Tier 2 to include a reference to RG 1.29 for meeting GDC 2 as it relates to fuel-

handling components. The applicant also noted that RB and FB cranes have been reclassified to seismic Category I in DCD Tier 1, Revision 5, and clarified that the seismic Category I RB and FB cranes described in Table 2.16.1-1, ITAAC 5 and 6 are designed to withstand the effects of an SSE condition, thus demonstrating conformance to RG 1.29. The staff finds that the response to RAI 9.1-33 is acceptable, in combination with the additional DCD changes discussed, since with these changes the design meets the guidance in RG 1.29. Accordingly, based on the above, the applicant's responses, and DCD changes, RAI 9.1-33 is resolved.

The staff's evaluation of the applicant's standard dynamic analyses is in accordance with Sections 3.7.2 and 3.7.3 of the SRP; Sections 3.7.2 and 3.7.3 of this report present this evaluation.

The refueling machine and fuel-handling machine are designed so that they will not become unstable and topple into pools during an SSE. Interlocks, as well as limit switches, are provided to prevent accidental movement of the grapple mast into pool walls.

The grapple on both the refueling machine and fuel-handling machine is hoisted to its retracted position by redundant cables inside the mast and is lowered to full extension by gravity. The retracted position is controlled by both an interlock and physical stops to prevent raising the fuel assembly above the normal stop position required for safe handling of the fuel. The operator can observe the exact grapple position over the core via a display screen at the operator console.

In DCD Tier 2, Revision 3, Section 9.1.4.12, the applicant stated that the IFTS is designed with sufficient redundancy and diversity in equipment and controls to prevent loss of load (carriage with fuel is released in an uncontrolled manner). The applicant also stated that no modes of operation will allow simultaneous opening of any set of valves in the IFTS that could cause draining of water from the upper pool in an uncontrolled manner. These provisions are also included as an ITAAC in DCD Tier 1, Revision 3, Section 2.5.10. In RAI 9.1-34, the staff requested that the applicant describe how sufficient redundancy and diversity in equipment are achieved and what controls are designed to prevent loss of load. RAI 9.1-34 was being tracked as an open item in the SER with open items.

In response, the applicant stated that the performance specification for the IFTS provides that equipment controlling or monitoring the movement of the carriage use dual input for carriage position. Both fixed proximity sensors (i.e., at selected positions) and continuous position sensors (e.g., an encoder) determine the position of the carriage. Each sensor consists of primary and backup sensors [two channels] whose position indications are compared to one another to ensure that the failure of one sensor does not result in a lack of knowledge of the carriage position. Control interlocks are provided to ensure that at selected positions there is agreement between the continuous sensor and the fixed sensor to allow carriage movement. The same logic is provided for valve control in the IFTS. Dual sensors for valve position are provided. Interlocks in the control logic prevent inadvertent movement without agreement between sensors and other inputs, such as carriage position. The staff finds that the RAI response is acceptable since it describes how diversity and redundancy in equipment is achieved. Accordingly, based on the above and the applicant's response, RAI 9.1-34 is resolved.

DCD Tier 2, Revision 3, Section 9.1.4.12 states that the IFTS tubes and supporting structure can withstand an SSE without failure of the basic structure or compromising the integrity of adjacent equipment and structures. The portion of the IFTS transfer tube assembly from where

it interfaces with the upper fuel pool, the portion of the tube assembly extending through the building, the drain line connection, and the lower SFP terminus equipment (tube, valve, support structure, and bellows) are designated as nonsafety-related and seismic Category I. The remaining equipment is nonsafety-related and non-seismic.

The staff was not able to identify the seismic design classification for the components of the IFTS. The staff requested, in RAI 9.1-35, that the applicant provide a table or diagram to show the seismic design classification for all of the IFTS components. RAI 9.1-35 was being tracked as an open item in the SER with open items. In response, the applicant indicated that it would revise DCD Tier 2, Figure 9.1-2, Table 3.2-1, and Table 9.1-4, to make the boundaries of seismic design classifications clear. The applicant made the modifications in DCD Tier 2, Revision 5, to clearly define the seismic classification of the IFTS components. The staff finds that the RAI response is acceptable since the system's seismic classification provides the ability to withstand the effects of seismic event. Accordingly, based on the above and the applicant's response, RAI 9.1-35 is resolved.

The new fuel inspection stand is a vertical frame mounted in a pit that supports two fuel bundles contained in a mechanically driven inspection carriage. The staff requested, in RAI 9.1-36, that the applicant identify the seismic design classification for the new fuel inspection stand. RAI 9.1-36 was being tracked as an open item in the SER with open items. In response, the applicant clarified that the new fuel inspection stand is dynamically analyzed and that it cannot damage adjacent equipment, as no other equipment is present in the pit. The applicant further indicated that it would revise DCD Tier 2, Tables 3.2-1 and 9.1-4, to identify that the new fuel inspection stand must be seismic Category II. The staff finds that the RAI response is acceptable since the applicant clarified the seismic design classification of the new fuel inspection stand. The staff confirmed that the applicant incorporated the modifications into DCD Tier 2, Revision 5. Subsequently, the applicant removed Table 9.1-4 from DCD Revision 6 and included the seismic classification in Table 3.2-1. Accordingly, based on the above, the applicant's responses, and DCD changes, RAI 9.1-36 is resolved.

The applicant stated that the dryer and chimney head/separator strongback and head strongback/tensioner conform to the provisions of NUREG–0612 and ANSI-14.6. However, the applicant had not described how the design of the chimney head/separator strongback and the head strongback/tensioner met the above cited NUREG–0612 and ANSI-14.6. The staff requested, in RAI 9.1-37, that the applicant demonstrate how it applied NUREG–0612 and ANSI 14.6 to specific components. RAI 9.1-37 was being tracked as an open item in the SER with open items. In response, the applicant clarified how the guidelines of NUREG–0612 and ANSI 14.6 will be met. The staff finds that the RAI response is acceptable since the applicant described how the provisions of NUREG–0612 and ANSI-14.6 are implemented, including through the use of COL Information Item 9.1-5-A, "Handling of Heavy Loads." Accordingly, based on the above and the applicant's response, RAI 9.1-37 is resolved.

Sections 3.2.1 and 3.2.2 of this report further address the staff's evaluation of the review of the seismic and quality group classifications for the fuel-handling system components.

Based on the above, the staff finds that the ESBWR design meets the requirements of GDC 2.

The ESBWR design is a single-unit station, and the requirements of GDC 5 are not applicable to the single-unit design.

The staff verified that the design complied with the requirements of GDC 61 and 62 and the guidelines of ANSI/ANS 57.1. DCD Tier 2, Revision 4, Section 9.1.4.5, stated that there are interlocks in the refueling machine to ensure that the grapple in its retracted position provides sufficient water shielding. In RAI 9.1-50, the staff requested that the applicant revise DCD Tier 2 to include the actual height of water over the fuel when the grapple is at its retracted position. In response, the applicant agreed to modify the DCD to provide this information. DCD Tier 2, Revision 4 Sections 9.1.4.1 and 9.1.4.5, state that both the refueling machine and the fuel handling machine always maintain a safe water shielding depth of at least 2,591 mm (8.5 ft) over the active fuel during transit. RG 1.13 provides guidance that the minimum safe water shielding depth associated with spent fuel assemblies is 3.05 m (10 ft). In RAI 9.1-50 S01, the staff asked the applicant to justify this discrepancy with SRP Section 9.1.2 and RG 1.13. In response to RAI 9.1-50 S01, the applicant stated that the interlock height of 2,591 mm (8.5 ft) is the actual height of water above the TAF that is provided with the normal full up interlock installed on either the refueling or fuel handling machine. The applicant stated that this interlock height has been successfully used in commercial nuclear power plant operations since the 1970s. The staff determined that this response was unacceptable.

In RAI 9.1-50 S02, the staff asked the applicant to specifically justify the use of the 2,591 mm (8.5 ft) interlock height. In response to RAI 9.1-50 S02, the applicant summarized a proprietary shielding calculation, *Dose Rate Calculation Using a GE14 Fuel Bundle During ESBWR Fuel Handling Operations* (dated 4/26/2008). This shielding calculation had been performed for the GE14 fuel bundle referenced in the ESBWR design using the interlock height to verify that 2,591 mm (8.5 ft) of water above the top of a single fuel assembly provides adequate shielding during transit. In addition, the applicant stated that it would include reference to the dose rates from the shielding calculation in DCD Tier 2, Revision 6. The proposed mark-up of DCD Revision 5, provided in the RAI response, stated that the estimated dose rate from the active fuel during transit (single grappled fuel bundle) from the reactor vessel to the spent fuel racks (or vice versa) was 267 μSv/h (27 mrem/h) at the water surface. The staff noted that although the information contained in the shielding calculation provided an estimate of the dose rate at the fuel pool water surface, it did not contain an estimate of the dose rate to refueling personnel who would be located on the bridge above the surface of the fuel pool water.

In RAI 9.1-50 S03, the staff asked the applicant to provide an estimate of the dose rate to a person standing on the fuel handling bridge deck during fuel movement and to include in this estimate the dose contribution from radionuclides in the SFP. The staff also asked the applicant to describe any design features to ensure that the dose to the refueling personnel would be maintained ALARA during refueling operations. In response, the applicant provided the estimated dose rate to an operator standing on the fuel handling machine platform and said that the dose contribution to this person from radionuclides in the SFP would be negligible. In RAI 9.1-50 S04, the staff noted that the estimated dose rate to an operator provided in response to RAI 9.1-50 S03 was roughly half of the estimated operator dose rate provided in response to RAI 12.2-27. The staff also requested that the applicant justify how the estimated operator dose provided in response to this RAI supplement meets the ANSI/ANS 57.1-1992 criteria, which states that the maximum dose rate to an operator for fuel handling equipment should not exceed 0.25 μSv/h (2.5 mrem/h).

In response to RAI 9.1-50 S04, the applicant stated that the interlock to the fuel handling machine in the FB will be reset so that the minimum depth of water over a raised fuel assembly in the FB will be 3.05 m (10 ft), thereby ensuring that the resulting dose rate to an operator will satisfy the dose rate criteria in ANSI/ANS 57.1-1992. To satisfy this dose rate criterion for the refueling pool in the RB, the applicant stated that it would increase the water coverage in the RB

refueling floor pools from 2.59 m (8.5 ft) to 2.74 m (9 ft) over a raised assembly and would provide additional shielding (equivalent to 1 foot of water) to the refueling machine design. The applicant stated that it would make these changes in DCD Revision 7. The staff finds that the RAI response is acceptable since the revised design of the refueling pools in both the RB and the FB satisfy the dose rate criteria in ANSI/ANS 57.1-1992. Accordingly, based on the above, the applicant's responses, and DCD changes, RAI 9.1-50 is resolved. The staff confirmed that the applicant incorporated these changes into DCD Revision 7.

In DCD Tier 2, Revision 5, Section 9.1.4 and 9.1.5.2 and Table 9.1-5, the applicant referenced only NUREG–0554 as containing the guidance it will follow in designing a single-failure-proof crane. SRP Section 9.1.5 Subsection 4(C)(i) calls for single-failure-proof, Type 1 cranes to be designed to the criteria of ASME Code NOG-1 2004. In RAI 9.1-96, the staff asked the applicant to modify its write up in Sections 9.1.4 and 9.1.5 and Table 9.1-5 of DCD Tier 2 to refer to the ASME Code standard for each single-failure-proof crane and to more clearly articulate which of the cranes are going to be designed to be single-failure-proof. In particular, the staff desired clarification about the status of the RB and FB cranes. In response, the applicant agreed to add ASME Code NOG-1 as a reference in DCD Tier 2, Revision 6, in Sections 9.1.4.5 and 9.1.5.2 and Table 9.1-5. The staff finds that the RAI response is acceptable since the applicant referred to the ASME Code standard for each single-failure-proof crane. Accordingly, based on the above, the applicant's responses, and DCD changes, RAI 9.1-96 is resolved.

SRP Section 9.1.4, Subsection III(1) also states that the LLHS' physical arrangements for stored fuel and fuel-handling areas are to be sufficiently described to establish that the various handling operations can be performed safely. The applicant did not provide figures in DCD Tier 2, Revision 5, showing the overall system arrangement, including the reactor well, buffer pool, upper fuel transfer pool, inclined fuel transfer pool, the fuel building storage pool, the spent fuel storage pool, the lower fuel transfer pool, cask pool, and IFTS. In RAI 9.1-106, the staff asked the applicant to either modify DCD Tier 2 to address the functional geometric layout of the fuel-handling equipment and areas or show how the proposed alternatives to the SRP acceptance criteria provide acceptable methods of compliance with NRC regulations. In response, the applicant stated that, in response to RAI 14.3-441 (discussed below), it would list the FB and RB overhead cranes, as well as the refueling machine and FB machine hoists, as single-failure proof with an ITAAC in DCD Tier 1. The staff confirmed that the applicant incorporated this modification in Revision 6 to the DCD. The applicant also clarified that DCD Tier 2, Figures 1.2-1 through 1.2-11 include the nuclear island plan figures for the different RB and FB elevations. These figures show the overall LLHS arrangement related to refueling. The staff finds that the RAI response is acceptable since the use of single-failure proof cranes is an acceptable alternative to providing the layout of the fuel-handling area. Accordingly, based on the above and the applicant's response, RAI 9.1-106 is resolved.

DCD Tier 2, Revision 5, Table 9.1-5, states that NUREG–0554 applies to the RB and FB overhead cranes and to the hoist on the refueling and fuel-handling machines that handles the combined fuel support and control blade grapple. DCD Tier 1, Section 2.16.1, and Table 2.16.1-1, did not list "single-failure-proof" as certified design information with the ITAAC for the RB crane, the FB crane, the hoist for the refueling machine, or the hoist for the fuel-handling machine. In RAI 14.3-441, the staff asked the applicant to justify not including single-failure-proof design criteria and ITAAC in Tier 1 of the DCD. In response, the applicant stated that it would revise the DCD in Revision 6. Subsequently, the staff, in RAI 14.3-441 S01, requested that the applicant enhance its response by providing a greater level of detail in the ITAAC for single-failure-proof cranes. In response, the applicant stated that it would revise DCD Tier 1 in

Revision 6. The staff finds that the RAI response is acceptable since the applicant committed to revise DCD Tier 1, Table 2.5.5-1, to specify greater level of details for the refueling machine and the FB machine hoists to provide reasonable assurance that they are single-failure proof. The staff confirmed that the applicant incorporated these changes into DCD Revision 6. Accordingly, based on the above, the applicant's responses, and DCD changes, RAI 14.3-441 is resolved.

Section 12.4 of this report discusses the staff's evaluation of whether the designs of the fuel-handling system and the spent fuel transfer process will result in ALARA occupational radiation exposures during spent fuel handling.

Section 15.4.2 of this report discusses the staff's evaluation of the radiological consequences of fuel-handling accidents. Section 15.4.10 of this report discusses why neither the staff nor the applicant needed to evaluate the radiological consequences of spent fuel cask drop accidents.

DCD Tier 2, Revision 3, Section 9.1.4.1, states that both the refueling machine and the fuel-handling machine have telescoping masts with integral grapples mounted from a trolley structure. Section 9.1.4.1 also states that the machines are equipped with auxiliary hoists and jib cranes to which other grapples are attached when required. Both have redundant safety features and indicators that ensure positive engagement with fuel bundles. In RAI 9.1-24, the staff requested that the applicant describe the design of grapples used to handle fuel and how that design reduces the probability of a fuel assembly drop. The staff also requested that the applicant to identify any loads handled over stored fuel that could have greater kinetic energy than a fuel assembly dropped from its normal handling elevation.

In response, the applicant stated that the fuel grapple is designed with dual interlocking deep "J" shaped hooks. With the hooks open, the first hook is to one side of the bail handle, and the second hook is to the other side of the bail handle. When closed, each hook passes under the bail handle. As the fuel assembly is raised, the bail handle rests within the radius of the "J" hooks. The "T" hooks and the bail handle are captured inside the grapple head. The fuel bail handle is completely captured. In the event that a grapple open signal is sent and the "J" hook actuator is energized, the hook cannot move because the bail handle is captured down inside the pair of "Js" and they cannot be pulled apart. At the same, time the bail is captured in part by the grapple head. The hooks cannot move. If one "J" does not close, the second will capture the bail handle providing a level of redundancy.

In response to the request to identify loads handled over stored fuel, the applicant stated that, for normal refueling and RPV maintenance operation, no components are raised and transferred over spent or new fuel. The layout of the building pools is such that components (e.g., a control blade) can be moved within the RB from the RPV to the IFTS and within the FB from the IFTS to a storage position without passing over fuel. Interlocks are in place on the refueling and fuel-handling machines such that, when a heavy load is sensed on an auxiliary hoist, the fuel-handling machine controls enforce pre-established heavy-load boundary zones, thereby limiting the travel of the refueling and fuel-handling machines. The staff finds that the response is acceptable since the applicant clarified the redundant nature of the grappling devices. In addition, DCD Tier 1 Revision 9, Table 2.16.1-1, provides ITAAC to ensure that heavy load handling equipment is designed or interlocked such that movement of heavy loads is restricted to areas away from stored fuel. Accordingly, based on the above and the applicant's response, RAI 9.1-24 is resolved.

In RAI 9.1-25, the staff requested that the applicant describe the necessary scope of the administrative controls with regard to restrictions on loads handled over stored fuel and monitoring LLHS components for degradation covered by an associated COL Holder Item. In response, the applicant stated that administrative controls are applied to the tabulated listing of the cranes and refueling equipment provided in DCD Tier 2, Table 9.1-6. The applicant stated that the development of the site-specific procedures to govern these administrative controls is a COL Holder Item. The applicant also identified information the COL holder will provide. The staff found this response to be unacceptable.

In RAI 9.1-25 S01, the staff requested that the COL Holder Items be changed to COL Information Items, which can provide information to allow the staff to conclude whether safe load paths, routing plans, and administrative controls satisfy the regulatory requirements before issuance of a COL. In DCD Tier 2, Revision 4, the applicant modified the text to state that these are to be provided by the COL applicant and modified the COL Information Items 9.1.4-A (fuel handling operations) and 9.1.5-A (handling of heavy loads) to describe the programs that address fuel handling operations and handling of heavy loads. The staff finds that the RAI response is acceptable since the proposed action items are consistent with the guidance of RG 1.206, "Combined License Applications for Nuclear Power Plants (LWR Edition)," Regulatory Positions C.I.9.1.4 and C.I.9.1.5. Accordingly, based on the above, the applicant's responses, and DCD changes, RAI 9.1-25 is resolved.

DCD Tier 2, Revision 1, Section 9.1.4.1, states that, where applicable, DCD Tier 2, Table 9.1-5, provides the appropriate ASME, ANSI, and industrial and electrical codes. In RAI 9.1-26, the staff requested that the applicant describe how industry codes and standards identified in DCD Tier 2, Table 9.1-5, apply to specific components in the light and overhead heavy-load handling systems.

In response, the applicant stated that specific standards are selected as appropriate for the device or piece of equipment and are invoked in the associated design or procurement documents. The standard is used in part or in total depending upon the equipment and application. The applicant provided a revised markup of Table 9.1-5 to clarify which codes are applicable to the load handling equipment. The staff finds that the RAI response is acceptable since the applicant clarified which codes are applicable to the load handling equipment in a revised DCD Tier 2, Table 9.1-5. Accordingly, based on the above, and the applicant's response, RAI 9.1-26 is resolved. RAI 9.1-26 was being tracked as a confirmatory item in the SER with open items. The staff confirmed that the applicant incorporated the changes in DCD Tier 2, Revision 2, and the confirmatory item is closed.

DCD Tier 2, Revision 4, Section 9.1.4, did not contain a statement to indicate that the fuel-handling system conforms to the industry standards of ANSI/ANS 57.1, thereby meeting the requirements of GDC 61 and 62. In RAI 9.1-43, the staff requested that the applicant revise the DCD to include such a statement. In response, the applicant indicated that it would add references to the ANSI/ANS standard, which it did in DCD Tier 2, Revision 5. The staff finds that the applicant's response is acceptable since it addressed conformance to ANSI/ANS 57.1. Accordingly, based on the above, the applicant's responses, and DCD changes, RAI 9.1-43 is resolved.

In SRP Section 9.1.4, acceptance criteria for meeting the relevant requirements of GDC 61 and 62 are based on meeting the guidelines of ANSI/ANS 57.1-1992. Table 1, "Required Interlock Protection," in ANSI/ANS 57.1-1992 provides interlock protection guidelines for each component of a fuel handling system. The interlocks described in the DCD did not include a number of

interlocks listed in Table 1 above. Additionally, Table 1 lists interlock guidelines for equipment, such as the FB crane, RB crane, fuel preparation machine, control component change mechanism, IFTS, and the upenders, which are not described in the application. In RAI 9.1-107, the staff asked the applicant to describe in the DCD how each interlock specified in Table 1 of ANSI/ANS 57.1-1992 is applied for each of the components listed in Table 1 and to provide a markup in DCD Tier 2 showing the above requested information. In response, the applicant stated that it would revise DCD Tier 2, Section 9.1.4, to clarify that the interlocks discussed in the DCD are only a partial list of those listed in ANSI/ANS 57.1. The staff determined that the revised wording proposed by the applicant was unacceptable since it was not clear that all interlocks listed in Table 1 of the standard would be implemented. In its revised response to RAI 9.1-107, the applicant clarified that it was revising DCD Tier 2, Section 9.1.4.1, to clearly state that the interlocks listed in Table 1 of ANSI/ANS 57.1 are applicable to the ESBWR fuel handling system, except for the interlocks associated with the new fuel elevator, which is not a part of the ESBWR fuel handling system design. The staff finds that the RAI response is acceptable since the applicant addressed the ANSI/ANS 57.1-1992 guidelines for interlocks. The staff confirmed that the applicant incorporated the changes into Revision 6 of the DCD. Accordingly, based on the above, the applicant's responses, and DCD changes, RAI 9.1-107 is resolved.

The fuel handling machine, as described in DCD Tier 2, Revision 5, Section 9.1.4.5, transports spent fuel assemblies over and above the spent fuel racks. If a raised fuel assembly is too close to the water surface of the SFP, excessive radiation levels might occur on the fuel handling floor. The depth of the water over the fuel shields workers from radiation. GDC 61 requires the avoidance of excessive personnel radiation exposure. DCD Tier 2, Section 9.1.4.5, states that, "The grapple in its retracted position provides sufficient water shielding of at least 2,591 mm (8.5 ft) over the active fuel during transit." In RAI 9.1-108, the staff asked the applicant to explain the operating interlocks for the fuel handling machine that ensure a spent fuel assembly is not raised above a specified water level in the SFP. In response to RAI 9.1-108, the applicant stated that the interlock referred to in DCD Tier 2, Section 9.1.4.18, was the "normal up" interlock for both the fuel handling and refueling machines. For this interlock, power to the main hoist is interrupted when the fuel grapple hook is at its normal retracted position and provides the "normal up" indicator light. The staff finds that the RAI response is acceptable since the applicant clarified where the necessary interlock was included in the DCD. Accordingly, based on the above and the applicant's response, RAI 9.1-108 is resolved.

DCD Tier 2, Revision 5, Section 9.1.4.9, discusses moving the instrument strongback with the RB auxiliary hoist and the instrument handling tool with the refueling platform auxiliary hoist. In RAI 9.1-109, the staff asked the applicant to modify DCD Tier 2, Table 9.1-5, in the next revision to DCD Tier 2 to identify the standards and codes to which these hoists are to be constructed and operated. In response, the applicant discussed that the Crane Manufacturer's Association of America (CMAA) – 70, "Specifications for Electric Overhead Traveling Cranes," applies to the construction and operation of the refueling machine auxiliary hoist used for lifting light incore servicing tools that are not heavy loads. The RB overhead crane auxiliary hoist is constructed and operated in the same manner as the main hoist of the RB overhead crane, thus meeting the same standards listed in Table 9.1-5 of the DCD. The staff finds that the RAI response is acceptable since the use of CMAA-70 conforms to the guidance of SRP Section 9.1.5, Revision 1. Accordingly, based on the above and the applicant's response, RAI 9.1-109 is resolved.

DCD Tier 2, Revision 5, Section 9.1.4.12, states that there is a means to seal off the upper and lower ends of the transfer tube while allowing filling and venting of the tube. In RAI 9.1-110, the staff asked the applicant to explain how this is to be accomplished and to discuss the

implications of failure of these seals (i.e., valve failure) in such a manner as to drain the tube while fuel is being transported in it. In response, the applicant stated that the sealing of the upper and lower ends is done with the upper (top and fill) valves and lower (bottom and drain) valves. The response did not discuss the effects of draining the transfer tube with fuel in the tube. In RAI 9.1-110 S01, the staff asked the applicant to (1) address the effects (including flooding and the possibility of loss of core cooling) that would be associated with failure of these transfer tube valves, including the implications of draining upper pools that can communicate with the transfer pool, and (2) address the effects from draining the transfer tube while fuel is being transported in it. In response, the applicant described how there is no operational alignment that permits the upper and lower valves to be in the open position simultaneously, and the failure of either a single upper or lower valve does not provide a drain path that would allow uncontrolled draining from the upper pool through the IFTS tube. Based on this, the applicant stated that draining of the upper pool which can lead to flooding or loss of core cooling is not credible because of a single IFTS upper or lower valve failure. In addition, the applicant proposed to revise DCD Tier 1, Revision 6, to include a statement that no single failure can cause the draining of water from the upper pool in an uncontrolled manner into the SFP or other areas. The staff finds that the RAI response is acceptable since the design uses redundant valves to prevent draindowns and the applicant clarified the ITAAC to confirm that no single active failure can cause a draindown. Accordingly, based on the above, the applicant's responses, and DCD changes, RAI 9.1-110 is resolved.

DCD Tier 2, Revision 5, Section 9.1.4.12, states that there are no modes of operation that allow simultaneous opening of any set of valves that could cause draining of water from the upper pool in an uncontrolled manner. In RAI 9.1-111, the staff asked the applicant to explain the engineering basis for this assertion and discuss whether this protection is single-failure-proof. In response, the applicant listed diverse and redundant sensors and interlocks that prevent the simultaneous opening of the upper and lower valves associated with filling and draining the transfer tube. The submission did not address the effects of failure of the isolation valves. The staff finds that the response is acceptable since the applicant identified multiple active failures that would need to occur to have a draindown event. However, the staff determined that the interlocks should be listed in Tier 2 of the DCD. In RAI 9.1-111 S01, the staff requested that the DCD specifically discuss these diverse and redundant sensors and interlocks. In response, the applicant stated that it would include the sensors and interlocks for opening the bottom and drain valves listed in response to RAI 9.1-111 in Revision 6 of DCD Tier 2. The staff finds that the response is acceptable since the sensors and interlocks would be added to the DCD. The staff confirmed that the applicant incorporated the DCD changes into DCD Tier 2, Revision 6. Accordingly, based on the above, the applicant's responses, and DCD changes, RAI 9.1-111 is resolved.

DCD Tier 2, Revision 5, Section 9.1.4.12, stated that the IFTS tubes and supporting structure can withstand an SSE without failure of the basic structure or compromising the integrity of adjacent equipment and structures. The applicant changed DCD Tier 2, Revision 5, Section 9.1.4.12, to state that cooling is provided for two instead of one freshly removed fuel assemblies in the IFTS. In RAI 9.1-112, the staff asked the applicant to please confirm in DCD Tier 2 whether the engineering basis for this assertion assumes that at least two fuel assemblies are contained in the transport device (i.e., carriage) during the seismic event. In response, the applicant stated that the seismic event assumes that two fuel assemblies are contained in the fuel transfer tube and committed to modify the DCD to make that clear. The staff finds that the RAI response is acceptable since the applicant added fuel assemblies to the list of items discussed in conjunction with an SSE. The staff confirmed that the applicant revised DCD

Revision 6. Accordingly, based on the above and the applicant's response, RAI 9.1-112 is resolved.

DCD Tier 2, Revision 5, Section 9.1.4.12, stated that (1) controls prevent personnel from inadvertently or unintentionally being left in high radiation areas or areas immediately adjacent to the IFTS at the time the access doors are closed, and (2) during IFTS operation or shutdown, personnel are prevented from reactivating the IFTS while personnel are in the area or entering the controlled maintenance area while irradiated fuel or components are in any part of the IFTS. In RAI 9.1-113, the staff asked the applicant to please describe these controls in the next revision to DCD Tier 2. In response, the applicant referenced its response to RAI 12.4-19 S03, questions 1, 2, and 3. The applicant identified rooms of interest for RAI 9.1-113 and explained that these rooms will be permanently closed except for maintenance that is only done when there is no fuel being transferred. The staff has reviewed the responses to various portions of RAI 12.4-19 S03 that address the same issues as those raised in RAI 9.1-113. The staff finds that the RAI response is acceptable since the response to RAI 12.4-19 S03 describes the methods used to control personnel access during fuel transfer. In addition, DCD Tier 2, Revision 9, Section 12.3.1.4.4, describes the radiation protection and access controls for the IFTS. Accordingly, based on the above and the applicant's response, RAI 9.1-113 is resolved.

Based on the above, the staff finds the ESBWR design to be in compliance with the requirements of GDC 61 and 62.

DCD Tier 2, Section 9.1.4, only indirectly addresses transfer of spent fuel to a cask. DCD Tier 2, Revision 5, Section 9.1.4.3, states that spent fuel casks are not in the ESBWR standard plant scope. In RAI 9.1-114 the staff asked the applicant to provide a COL action item or a DCD Tier 1 Interface Item that would require a COL applicant to address spent fuel casks including identifying safety and nonsafety-related components, a description of the safety function of each safety- related component, a discussion of the seismic capacity of the spent fuel cask system, a discussion of how the single-failure criterion is satisfied, a discussion of how emergency cooling is accomplished, a discussion of the need for emergency cooling of spent fuel casks, and a discussion of interlocks. In response, the applicant pointed out that the DCD states that the FB overhead crane has the capacity to lift a 165-ton load, which bounds anticipated SFP pool cask weights. The staff finds that the applicant's response is acceptable since the rated load capacity allows the single-failure proof FB overhead crane to safely lift a spent fuel cask; thus the, discussion of individual casks components is unnecessary in the DCD. Accordingly, based on the above and the applicant's response, RAI 9.1-114 is resolved.

In its review of DCD, Revision 6, the staff noticed that in DCD Tier 2, Section 9.1.4.17, a step in the vessel closure process had operators install both an equipment pool gate and buffer pool gates. However, later in the process, only the equipment pool gate was removed. In RAI 9.1-143, the staff asked the applicant to revise the DCD to clarify when the buffer pool gate is removed such that it needs to be installed during the refueling process. In response, the applicant clarified that the equipment pool gate is removed and installed to support the drain down and reflooding of the reactor well. The buffer pool gates are installed and removed to support fuel movement. The applicant indicated that DCD Tier 2, Revision 6, Section 9.1.4.15, describes these actions. The applicant also clarified that the configuration of the gates during reactor operation has the equipment pool gate removed and the buffer pool gate installed. The applicant explained that this gate configuration is maintained since the water in the reactor well and equipment pool is credited as a makeup source to the ICS/PCCS pools while the water in the buffer pool is not. The applicant indicated that it would add this clarification to DCD Tier 2, Section 9.1.4.15. The staff finds that the RAI response is acceptable since the applicant

clarified the movement and normal configuration of the equipment pool and buffer pool gates. Accordingly, based on the above, the applicant's responses, and DCD changes, RAI 9.1-143 is resolved. The staff confirmed that the applicant incorporated the changes into DCD Revision 7.

In RAI 14.3-445, the staff asked the applicant to explain why the RPV head strongback was not added to the ITAAC or specified as Tier 1 material. In response, the applicant stated that the RPV head strongback is nonsafety-related and thus the design details of the strongback do not meet the criteria for inclusion in Tier 1. The staff finds that the RAI response is acceptable since the strongback serves no safety function and need not be subject to an ITAAC. Accordingly, based on the above and the applicant's response, RAI 14.3-445 is resolved.

NRC guidance states that important to safety functions should be described in the DCD Tier 1. DCD Tier 1, Table 2.5.5-1, "ITAAC for Refueling Machine," lists a few interlocks that the FB fuel handling machine will have. In RAI 14.3-446, the staff asked the applicant to add interlocks to this list based on appropriate disposition of RAI 9.1-107, which addresses interlocks for the fuel handling system that are specified in Table 1 of ANSI/ANS 57.1-1992. In response, the applicant stated that this issue was addressed by the response to RAI 9.1-107, since the applicant clarified that DCD Tier 2, Subsection 9.1.4.1 is being revised to clearly state that the interlocks listed in Table 1 of ANSI/ANS 57.1 are applicable to the ESBWR fuel handling system except for the interlocks associated with the New Fuel Elevator, which is not a part of the ESBWR fuel handling system design. The staff finds that the RAI response is acceptable since the applicant addressed the ANSI/ANS 57.1-1992 guidelines for interlocks. In addition, since the resolution of RAI 9.1-107 did not add additional interlocks to DCD Tier 2, additional interlocks do not need to be added to DCD Tier 1. Accordingly, based on the above and the applicant's response, RAI 14.3-446 is resolved.

9.1.4.4 *Conclusion*

Based on the above, the staff finds that the ESBWR LLHS design meets the requirements of GDC 2, 61, and 62. Because the ESBWR design is a single unit, GDC 5 is not applicable. Section 9.1.6 of DCD Tier 2, Revision 9 includes COL Information Item 9.1-4-A, which requires the COL applicant to address the criticality safety of fuel handling. This is acceptable to the staff.

9.1.5 Overhead Heavy-Load Handling Systems

9.1.5.1 *Regulatory Criteria*

The staff reviewed the overhead heavy-load handling system (OHLHS) in accordance with SRP Section 9.1.5 Revision 1. The staff's acceptance of the ESBWR design is based on meeting the relevant requirements of the following GDC:

- GDC 1, "Quality standards and records," as it relates to the design, fabrication, and testing of SSCs important to safety to maintain quality standards
- GDC 2, as it relates to the ability of structures, equipment, and mechanisms to withstand the effects of earthquakes
- GDC 4, as it relates to protection of safety-related equipment from the effects of internally generated missiles (i.e., dropped loads)

- GDC 5, as it relates to the capability of shared equipment and components to perform safety functions

Compliance with the requirements of GDC 1 is based in part on NUREG–0554 for overhead handling systems and ANSI N14.6, "American National Standard for Radioactive Materials—Special Lifting Devices for Shipping Containers Weighing 10000 Pounds (4500 kg) or More," or ASME Code B30.9 for lifting devices. Compliance with the requirements of GDC 2 depends on adherence to the guidance of Regulatory Position C.2 of RG 1.29 and Section 2.5 of NUREG–0554. Compliance with the requirements of GDC 4 is based in part on Regulatory Position C.5 of RG 1.13. The ESBWR design is a single-unit station; therefore, the requirements of GDC 5 are not applicable to the single unit.

9.1.5.2 *Summary of Technical Information*

The OHLHS consists of the FB crane, the RB crane, the upper drywell servicing equipment, the lower drywell servicing equipment, the main steam tunnel servicing equipment, and other servicing equipment.

DCD Tier 2, Revision 9, Section 9.1.5.3, states that all handling equipment subject to the heavy-loads handling criteria has ratings consistent with the lifts required, and the design loading will be visibly marked. Cranes and hoists or monorail hoists pass over the centers of gravity of heavy equipment that is to be lifted. In locations where a single monorail or crane handles several pieces of equipment, the routing is such that each transported piece passes clear of other parts.

Pendant control is provided for the bridge, trolley, and auxiliary hoist to provide handling of fuel shipping containers during receipt, as well as to handle fuel during new fuel inspection. The crane control system is selected considering the long lift necessary through the equipment hatch and the precise positioning needed when handling the RPV and drywell heads, the RPV internals, and the RPV head stud tensioner assembly. The control system provides stepless regulated variable speed capability with high empty-hook speeds. The control system provides spotting control for the handling of the drywell and RPV heads and stud tensioner assembly. Because the handling of fuel shipping casks involves a long duration lift, low speed, and spotting control, the design incorporates thermal protection features.

DCD Tier 2, Revision 9, Section 9.1.5.3, also states that transportation routing drawings reflect the transportation route of every piece of heavy load removable equipment from its installed location to the appropriate service shop or building exit. Routes will be arranged to prevent congestion and to ensure safety while permitting a free flow of the equipment being serviced. The frequency of transportation and usage of route are documented based on the predicted number of times of usage, either per year and/or per refueling or service outage.

The spent fuel cask pit is intentionally located outside the areas normally confined to fuel movement. The cask and other heavy loads are not permitted to encroach within any part of any spent fuel, spent fuel storage pool, or safety-related structure.

Travel limit controls prevent inadvertent cask movement by the main FB crane over the fuel storage pools.

Heavy load equipment is also used to handle light loads and related fuel-handling tasks. Therefore, much of the handling systems and related design, descriptions, operations, and

service task information discussed in Section 9.1.4 of this report are also applicable to this system.

9.1.5.2.1 Fuel Building Crane

The FB is a reinforced concrete structure enclosing the SFP, cask-handling and cleaning facility, and other equipment. The FB crane provides heavy-load lifting capability for the FB floor. The main hook (150-metric ton [165-ton] capacity) is used to lift new fuel shipping containers and the spent fuel shipping cask.

The FB crane is used during refueling and servicing as well as when the plant is online. Minimum crane coverage includes the FB floor laydown areas, cask washdown area, and the FB equipment hatch. During normal plant operation, the crane is used to handle new fuel shipping containers and spent fuel shipping casks. The FB crane is interlocked to prevent movement of heavy loads over the SFP.

The FB crane is designed to be single-failure-proof, in accordance with NUREG–0554, and to meet ASME Code NOG-1.

9.1.5.2.2 Reactor Building Crane

The RB is a reinforced concrete structure enclosing the reinforced concrete containment vessel, the refueling floor, the new fuel storage buffer pool, the buffer pool deep pit pool for spent fuel storage, the dryer, chimney partitions, separator strongback, and other equipment. The RB crane provides heavy-load lifting capability for the refueling floor. The main hook (160-metric ton [176-ton] capacity) is used to lift the drywell head, RPV head insulation, RPV head, dryer, chimney partitions, separator strongback, and RPV head stud-tensioning equipment.

The RB crane is used during refueling and servicing as well as when the plant is online. Minimum crane coverage includes the RPV for shield block removal and the vessel servicing RB refueling floor laydown areas, RB equipment storage, refueling floor, and equipment hatches. The RB crane is interlocked to prevent movement of heavy loads over the fuel pools.

The RB crane is designed to be single-failure-proof in accordance with NUREG–0554 and to meet ASME Code NOG-1.

9.1.5.2.3 Upper Drywell Servicing Equipment

The upper drywell arrangement provides servicing access for the main steam isolation valves (MSIVs); feedwater isolation valves; SRVs, DPVs; ICS valves, GDCS valves; and drywell cooling coils, fans, and motors. Access to the space is from the RB through either the upper drywell personnel lock or the equipment hatch. Equipment is removed through the upper drywell equipment hatch. Platforms are provided for servicing the feedwater isolation valves and MSIVs, the SRVs, and the drywell cooling equipment to reduce maintenance time and operator exposure. Items such as MSIVs, SRVs, DPVs, and feedwater isolation valves weigh in excess of a fuel assembly and its handling device and therefore are considered heavy loads.

Since drywell maintenance activities are only performed during a plant outage, only GDCS piping and valves need to be protected from inadvertent load drops. This protection is provided through design or interlocks, such that movement of heavy loads above the component is restricted, or through spatial separation, such that a single inadvertent load drop cannot result in

the GDCS not meeting the TS for Modes 5 and 6. In addition, a piping support structure and equipment platform separates and shields the GDCS piping from heavy-load transport paths. This protection is such that no credible load drop can cause (1) a release of radioactivity, (2) a criticality accident, or (3) the inability to cool fuel within the reactor vessel or SFP.

9.1.5.2.4 Lower Drywell Servicing Equipment

The lower drywell arrangement provides for servicing, handling, and transportation operations for FMCRDs. The lower drywell OHLHS consists of a rotating equipment service platform, chain hoists, FMCRD removal equipment, and other special purpose tools.

The rotating equipment platform provides a work surface under the reactor vessel to support the weight of personnel, tools, and equipment and to facilitate transportation moves and heavy-load handling operations. The platform rotates 180 degrees in either direction from its stored or "idle" position. The platform is designed to accommodate the maximum weight of the accumulation of tools and equipment plus a maximum sized crew. Special hoists in the lower drywell and RB facilitate handling of these loads. No safety-related equipment is located below the FMCRD component. Inadvertent load drops by the FMCRD servicing equipment cannot cause (1) a release of radioactivity, (2) a criticality accident, or (3) the inability to cool fuel within the reactor vessel or SFP.

9.1.5.2.5 Main Steam Tunnel Servicing Equipment

The main steam tunnel is a reinforced concrete structure surrounding the main steam lines and feedwater lines. The safety-related valve area of the main steam tunnel is located inside the RB. Personnel can access the main steam tunnel during a refueling and servicing outage. At this time, MSIVs or feedwater isolation valves and/or feedwater check valves may be removed using permanent overhead monorail-type hoists. They are transported by monorail out of the steam tunnel and placed on the floor below a ceiling removal hatch. Valves are then lifted through the ceiling hatch by the valve service shop monorail. During shutdown, none of the piping and valves in the steam tunnel is required to operate. Inadvertent load drops by the main steam tunnel servicing equipment cannot cause (1) a release of radioactivity, (2) a criticality accident, or (3) the inability to cool fuel within reactor vessel or SFP.

9.1.5.2.6 Other Servicing Equipment

The applicant stated that outside of the containment, the main steam tunnel, or the refueling floor no safety-related components are susceptible to heavy-load drops capable of causing the loss of a safety-related component required to maintain the plant in a safe condition. Therefore, inadvertent load drops cannot cause (1) a release of radioactivity, (2) a criticality accident, (3) the inability to cool fuel within reactor vessel or SFP, or (4) prevent the safe shutdown of the reactor.

9.1.5.3 *Staff Evaluation*

The staff confirmed that the design conforms to the relevant requirements of GDC 1, 2, and 4. The ESBWR design is a single-unit station, and the requirements of GDC 5 are not applicable to the single-unit design.

The applicant stated the following in DCD Tier 2, Revision 6, Section 9.1.5.2:

The lifting capacity of each crane or hoist is designed to at least the maximum actual or anticipated weight of equipment and handling devices in a given area serviced. The hoists, cranes, or other lifting devices comply with NRC Bulletin 96-02, NUREG-0554, ANSI N14.6, ASME/ANSI B30.9, ASME/ANSI B30.10 and NUREG-0612 Subsection 5.1.1(4) or 5.1.1(5) and ASME NOG-1. Cranes and hoists are also designed to criteria and guidelines of NUREG-0612 Subsection 5.1.1(7), ASME/ANSI B30.2 and CMAA-70 specifications for electrical overhead traveling cranes, including ASME/ANSI B30.11, and ASME/ANSI B30.16 as applicable.

In RAI 9.1-140, the staff asked the applicant to add Section 5.1.1(6) of NUREG–0612 to the standards referenced in the above paragraph since that section is applicable to single-failure proof cranes. In response, the applicant agreed to do so and the staff verified that Revision 7 of the DCD incorporated this reference. The staff finds the RAI response acceptable since the applicant added Section 5.1.1(6) of NUREG–0612 to the DCD. Accordingly, based on the above and the applicant's response, RAI 9.1-140 is resolved.

DCD Tier 2, Revision 3, Table 9.1-5, addresses the applicability of these standards to specific components. The staff requested, in RAI 9.1-38, that the applicant describe how the design of each component in the LLHS and the OHLHS has met GDC 2, 4, and 61, and how industry codes and standards are applied to specific components. RAI 9.1-38 was being tracked as an open item in the SER with open items. In response, the applicant discussed conformance to RGs 1.13 and 1.29, ANSI/ANS 57.1, and NUREG–0612 and NUREG–0554 as the means of complying with GDC 2, 4, and 61, but did not add the RGs and ANSI/ANS 57.1 to the DCD. However, in response to RAI 9.1-33 S01, the applicant revised DCD Tier 2, Section 9.1.5.2, to clarify that the design conforms to GDC 2, 4, and 61 by meeting the guidance of RGs 1.13, 1.29, 1.115, 1.117, and ANSI/ANS 57.1. The staff finds that the response to RAI 9.1-38 is acceptable, along with the DCD changes from RAI 9.1-33 S01, since the SRP states that a design meeting these standards satisfies the noted GDC. The staff confirmed that the applicant incorporated the changes into DCD Revision 5. Based on the above, the applicant's responses and DCD changes, RAI 9.1-38 is resolved.

DCD Tier 2, Revision 3, Section 9.1.5.5 states that the RB crane is interlocked to prevent movement of heavy loads over the fuel pools. However, Section 9.1.1 states that, should it become necessary to move major loads along or over the pools, administrative controls require that the load be moved over the empty portion of the buffer pool and avoid the area of the new fuel racks. The staff requested, in RAI 9.1-2, that the applicant describe the administrative controls governing a bypass of the RB crane interlocks and handling of heavy loads over the buffer pool. In response, the applicant identified this as a COL Holder Item. The applicant stated that the COL holder will provide heavy-load handling safe load paths and routing plans, including descriptions of automatic and manual interlocks and safety devices and procedures to ensure safe load path compliance.

The staff did not agree with this position. The staff must review this information before the issuance of the license. In RAI 9.1-2 S01, the staff requested the applicant to revise this item to become a COL Applicant Item. In response, the applicant proposed to modify the COL Holder items in DCD Tier 2, Section 9.1.6, to COL Information Items 9.1-4-A (fuel-handling operations) and 9.1-5-A (handling of heavy loads). The staff finds the response is acceptable since COL 9.1.4-A and COL 9.1.5-A includes program elements for safe load paths, routing plans, and administrative controls to be described by the COL applicant. Accordingly, based on the above and the applicant's response, RAI 9.1-2 is resolved. RAI 9.1-2 was being tracked as a

confirmatory item in the SER with open items. The staff confirmed that the applicant incorporated the above changes into DCD Tier 2, Revision 4, and the confirmatory item is closed.

The acceptance criteria in SRP Section 9.1.5 for GDC 1 states that it is acceptable for an applicant to commit to meeting design, fabrication, and testing guidance in NUREG–0554 for overhead handling systems and ANSI N14.6 or ASME B30.9 for lifting devices (note that NUREG–0554 and ANSI/ASME refer to NUREG–0612 seismic guidance). DCD Tier 2, Revision 5, Section 9.1.5, did not address how the design meets the GDC 1 criteria nor did it specify conformance to GDC 1. In RAI 9.1-100, the staff asked the applicant to specifically address meeting the above criteria for GDC 1. In response, the applicant stated it would revise Subsection 9.1.5.2 of DCD Tier 2 in Revision 6 to state that the OHLHS complies with the criteria of GDC 1 and the associated guidance. The staff finds that the response is acceptable, since the applicant clarified conformance to GDC 1 and ANSI N14.6, ASME B30.9, and NUREG–0554 in accordance with SRP Section 9.1.5. The staff confirmed the DCD changes were incorporated into DCD Revision 6. Accordingly, based on the above, the applicant's responses, and DCD changes, RAI 9.1-100 is resolved.

Based on the above, the staff finds the ESBWR design to be in compliance with the requirements of GDC 1 with respect to the OHLHS.

The Acceptance Criteria in SRP Section 9.1.5 for GDC 2 state that it is acceptable for an applicant to commit to meet the relevant aspects of Position C.2 of RG 1.29 and Section 2.5 of NUREG–0554. DCD Tier 2, Revision 5, Section 9.1.5, did not address Section 2.5 of NUREG–0554 within the context of GDC 2. In RAI 9.1-101, the staff asked the applicant to address compliance with GDC 2. In response, the applicant stated it would revise DCD Tier 2, Section 9.1.5.2, in Revision 6 to commit that the OHLHS will comply with NUREG–0554, thus meeting the criteria of GDC 2. The staff finds that the applicant's response is acceptable, since the applicant clarified conformance to GDC 2, NUREG–0554, and RG 1.29, in accordance with SRP Section 9.1.5. The staff confirmed that the applicant incorporated the changes into DCD Revision 6. Accordingly, based on the above, the applicant's responses, and DCD changes, RAI 9.1-101 is resolved.

Based on the above, the staff finds the ESBWR design to be in compliance with the requirements of GDC 2 with respect to the OHLHS.

DCD Tier 2, Revision 5, Section 9.1.5, describes the applicant's heavy load drop analyses. In RAI 9.1-99, the staff asked the applicant to describe how the evaluations took into account the potential for the function of main steam line and isolation condenser nozzle plugs to be affected by heavy load drops. The RAI also asked the applicant to address the effect of heavy load drops on SSCs that form a temporary reactor coolant boundary during shutdown activities. In response, the applicant stated that the RB overhead crane and associated lifting devices used for handling heavy loads are single-failure-proof, in accordance with NUREG–0554. Also, hoists, cranes, or other lifting devices that comply with the applicable guidance of NRC Bulletin 96-02, "Movement of Heavy Loads over Spent Fuel, over Fuel in the Reactor Core, or over Safety-Related Equipment," dated April 11, 1996, ANSI N14.6, ASME/ANSI B30.9, ASME/ANSI B30.10, and NUREG–0612. NUREG–0612 allows the use of the single-failure-proof equipment, pursuant to NUREG–0612, Section 5.1.6, or the effects of load drops can be analyzed. As stated in the RAI response, the applicant has chosen to have the heavy load handling equipment designed to comply with the single-failure-proof guidelines of NUREG–0612, Section 5.1.6, such that no single failure will result in the dropping of a load and affecting

equipment such as main steam line and isolation condenser nozzle plugs, as well as other SSCs that form a temporary reactor coolant boundary during shutdown activities. The staff finds that the RAI response is acceptable since it clarified that the ESBWR design satisfies the single-failure-proof guidelines with respect to this equipment. Accordingly, based on the above and the applicant's response, RAI 9.1-99 is resolved.

SRP Section 9.1.5, Section III.1, states that an applicant should describe the physical arrangement of heavy load handling systems for stored fuel and safe-shutdown equipment in a DCD. DCD Tier 2, Revision 5, Section 9.1.5.4, did not describe the physical arrangements. In RAI 9.1-102, the staff asked the applicant to provide these descriptions or address how the proposed alternatives to the SRP acceptance criteria provide acceptable methods of compliance with NRC regulations. In response, the applicant stated that it would list the FB and RB overhead cranes, as well as the refueling machine hoists as single-failure-proof with an ITAAC in DCD Tier 1 as part of reconciling RAI 14.3-441. In addition, the applicant stated that, in lieu of drawings, it will add a COL Applicant Item 9.1-5-A to call for the COL applicant to develop heavy load safe paths and routing plans. The staff finds that the RAI response is acceptable, since COL Information Item 9.1-5-A addresses physical arrangements and is in accordance with the guidance of RG 1.206, Regulatory Position C.I.9.1.5. Accordingly, based on the above and the applicant's response, RAI 9.1-102 is resolved.

In DCD Tier 2, Revision 5, Section 9.1.5.8, the applicant listed measures for COL applicants to comply with regarding a QA program to monitor, implement, and ensure compliance with the heavy load handling program. SRP Section 9.1.5, Section III.4.C.i states that the program should include at least the following elements: (1) design and procurement document control; (2) instructions, procedures, and drawings; (3) control of purchased material, equipment, and services (see also Section 10 of NUREG–0554); (4) inspection; (5) testing and test control; (6) non-conforming items; (7) corrective action; and (8) records. In RAI 9.1-104, the staff asked the applicant to incorporate the missing guidance or address how the proposed alternatives to the SRP acceptance criteria provide acceptable methods of compliance with NRC regulations. In response, the applicant stated that it would revise DCD Tier 2, Section 9.1.5.2 in Revision 6 to address the guidance of the SRP Section 9.1.5 Section III.4.C.i. In addition, the applicant stated it would revise DCD Tier 2, Sections 9.1.5.8 and 9.1-5-A to commit that the OHLHS will meet the QA program recommendations of NUREG–0554 and the program elements added to the DCD Tier 2, Section 9.1.5.2. The staff finds that the RAI response is acceptable since the applicant revised the DCD to conform to the guidelines of SRP Section 9.1.5, Section III.4.C.i. The staff confirmed that the applicant incorporated the changes into DCD Revision 6. Accordingly, based on the above and the applicant's response, RAI 9.1-104 is resolved.

Section III.4.C.ii.(1) of SRP Section 9.1.5 states the following:

> [a] special lifting device that satisfies ANSI N14.6 should be used for recurrent load movements in critical areas (reactor head lifting, reactor vessel internals, spent fuel casks) [See also Section 5.1.6, NUREG–0612]. The lifting device should have either dual, independent load paths or a single load path with twice the design safety factor specified by ANSI N14.6 for the load.

DCD Tier 2, Revision 5, Section 9.1.5.5, was silent regarding the load paths and safety factors. In RAI 9.1-105, the staff asked the applicant to either modify the DCD Tier 2 to address lifting device criteria for the FB and RB cranes or address how the proposed alternatives to the SRP acceptance criteria provide acceptable methods of compliance with NRC regulations. In response, the applicant stated that it would revise DCD Tier 2, Section 9.1.5.5 in Revision 6 of

the DCD to identify lifting device load path and safety factor criteria based on ANSI N14.6 and NUREG–0612, Section 5.1.6, for the FB and RB cranes. The staff finds that the RAI response is acceptable since the applicant revised the DCD to be consistent with the guidelines of SRP Section 9.1.5, Section III.4.C.ii.(1). The staff confirmed that the applicant incorporated the changes into DCD Revision 6. Accordingly, based on the above and the applicant's response, RAI 9.1-105 is resolved.

In DCD Tier 2, Revision 5, Section 9.1.5.2, the applicant committed to having hoists, cranes, or other lifting devices comply with, among other standards, ASME/ANSI B30.9. Subsection III.4.C.ii.(2) of SRP Section 9.1.5 states, "[s]lings should satisfy the criteria of ASME B30.9 and be constructed of metallic material (chain or wire rope)." This criterion is supported by operating experience documented in NUREG–1774, "A Survey of Crane Operating Experience at U.S. Nuclear Power Plants from 1968 through 2002," issued July 2003. The report cites various examples where Kevlar slings failed or separated causing a load drop. In RAI 9.1-103, the staff asked the applicant to explain its decision not to specify metallic material (chain or rope) for sling construction. In response, the applicant stated that it will revise the existing COL information item related to the handling of heavy loads to ensure that the COL applicants address the issues described in Regulatory Issue Summary (RIS) 2005-25, Supplement 1, "Clarification of NRC Guidelines for Control of Heavy Loads, related to the use of non-metallic slings with single failure proof lifting devices." In addition the applicant revised DCD Tier 2, Sections 9.1.5.8 and 9.1.5-A, in Revision 6 to clarify that the heavy load handling system guidelines regarding the use of non-metallic slings with single-failure-proof lifting devices are included in the heavy load handling program. The staff finds that the RAI response is acceptable since the applicant included RIS 2005-25, supplement 1, in the heavy load handling program, which addresses SRP Section 9.1.5, Section III.4.C.ii.(2). The staff confirmed that the applicant incorporated the changes into DCD Revision 6. Accordingly, based on the above and the applicant's response, RAI 9.1-103 is resolved.

DCD Tier 2, Revision 5, Table 9.1-5, states that NUREG–0554 is "[a]pplicable to the RB and FB overhead cranes. Applicable to the hoist on the refueling and fuel handling machines that handles the combined fuel support and control blade grapple." DCD Tier 1, Section 2.16.1, and Table 2.16.1-1, did not list "single-failure-proof" as certified design information with ITAAC for the RB crane, the FB crane, the hoist for the refueling machine, or the hoist for the fuel handling machine. The staff believes that DCD Tier 1 should include the single-failure-proof design criteria for the above listed cranes and hoists. In RAI 14.3-441, the staff asked the applicant to justify why it did not include single-failure-proof design criteria and ITAAC in Tier 1 of the DCD, which are safety significant design criteria, for the RB crane, FB crane, the hoist for the refueling machine, and the hoist for the fuel handling machine. In response, the applicant agreed to place the single-failure-proof design criteria and an ITAAC in DCD Tier 1, Revision 6, for the RB overhead crane, the FB overhead crane, the refueling machine hoist, and the fuel handling machine hoist. However, the staff determined that the ITAAC for single-failure-proof cranes should include a minimum set of tests. In RAI 14.3-441 S01, the staff requested that the applicant include key tests in the ITAAC for single-failure proof cranes, including (1) nondestructive examination of critical welds, (2) static and dynamic load testing, and (3) no-load load test of two-blocking protection. In response, the applicant added the requested tests to the ITAAC, referencing ASME Code NOG-1 in the acceptance criteria. The staff finds that the applicant's response is acceptable since the included tests and the use of ASME Code NOG-1 conforms with the guidance of SRP Section 9.1.5, and the additional tests for the RB and FB overhead cranes provide reasonable assurance that they are single-failure proof. Accordingly, based on the above, the applicant's responses, and DCD changes, RAI 14.3-441 is resolved.

Based on the above, the staff finds the ESBWR design to be in compliance with the requirements of GDC 4 with respect to OHLHS.

Based on the review of DCD Tier 2, Section 9.1, the staff identified several apparent design features important-to-safety that were omitted from DCD Tier 1. In RAI 14.3-447, the staff asked the applicant to explain why the following design features for the OHLHS were not in the ITAAC:

- Cranes and hoists, or monorail hoists pass over the centers of gravity of heavy equipment that is to be lifted.
- The PCCS and GDCS piping and valves are spatially separated such that an inadvertent load drop that breaks more than one pipe or valve in the PCC or GDC is not credible.
- The arrangement of the refueling floor precludes transporting heavy loads, other than spent fuel handled by the refueling machine or fuel handling machine, over spent fuel stored in the spent fuel storage pool.

In response, the applicant stated the following:

- For cranes and hoists, or monorail hoists, that are to pass over the centers of gravity of heavy equipment that is to be lifted, a design commitment and ITAAC will be added to DCD Tier 1. The staff finds that the RAI response is acceptable since ITAAC conforms to the safety commitment in DCD Tier 2.
- The PCCS is not required to be operable during refueling and the applicant will revise DCD Tier 2, Section 9.1.5.6 will be revised to delete references to the PCCS regarding load drops. For the GDCS, an ITAAC will be added to DCD Tier 1 stating that the GDC is not susceptible to a load drop that could result in the GDCS being unable to meet TS for Modes 5 and 6. The applicant also clarified that the protection of the GDCS components could be provided by restricting the movements of heavy loads through interlocks or the spatial separation of the GDCS components. The staff finds that the RAI response is acceptable since ITAAC verifies that the GDCS is protected from load drops.
- The RB and FB overhead cranes are interlocked to prevent movement of heavy loads over new or spent fuel. The applicant will revise the DCD in Revision 6 to indicate that crane interlocks, and not floor arrangement, preclude transporting heavy loads over fuel storage pools. The staff finds that the RAI response is acceptable since the applicant clarified that it would use interlocks to prevent transporting heavy loads over fuel storage pools and added an ITAAC to verify the interlocks.

The staff confirmed that the applicant incorporated the changes into DCD Revision 6. Based on the above, the applicant's responses, and DCD changes, RAI 14.3-447 is resolved.

9.1.5.4 *Conclusion*

For the reasons set forth above, the staff finds that the OHLHS complies with the requirements of GDC 1, 2, and 4. Because the ESBWR design is only a single unit, GDC 5 is not applicable.

9.2 Water Systems

In DCD Tier 2, Revision 9, Section 3.1.1.5, the applicant states that the ESBWR design is a single-unit station, and the requirements of GDC 5 are met. However, the staff has determined that the requirements of GDC 5 are not applicable for the single-unit design.

9.2.1 Plant Service Water System

9.2.1.1 *Regulatory Criteria*

The staff reviewed the PSWS based on guidance provided in SRP Section 9.2.1, Revision 5. The SRP guidance is used to the extent that it pertains to system functionality and reliability considerations. Staff acceptance of the PSWS design and supporting information is based on conformance with the following:

- GDC 2, as it relates to the capability of the design to maintain and perform its safety function following an earthquake

- GDC 4, as it relates to the dynamic effects associated with water hammer

- GDC 5, as it relates to the capability of shared systems and components important to safety to perform required safety functions

- GDC 44, as it relates to transferring heat from SSCs important to safety to a heat sink

- GDC 45, "Inspection of cooling water system," as it relates to the design provisions to permit inspection of components and equipment

- GDC 46, "Testing of cooling water system," as it relates to the design provisions to permit operational testing of components and equipment

- 10 CFR 20.1406 as it relates to minimization of contamination

The PSWS is a nonsafety-related system; however, the system provides defense-in-depth for the ESBWR passive plant design. In addition to the SRP guidance, the NRC staff's evaluation of defense-in-depth systems also focuses on (1) confirming that design, performance, and reliability considerations are satisfied consistent with the NRC policies that are referred to in Chapter 22, of this report; (2) confirming that failure of defense-in-depth systems and components will not adversely impact safety-related SSCs; (3) confirming that ACs are established as appropriate; (4) and confirming that proposed ITAAC and initial test program specifications are adequate.

9.2.1.2 *Summary of Technical Information*

DCD Tier 2, Revision 9, Section 9.2.1, describes the PSWS. The system does not perform any safety-related function, and there is no interface with any safety-related component.

The PSWS consists of two independent and 100-percent redundant open trains that continuously circulate water through the RCCWS and turbine component cooling water system (TCCWS) heat exchangers. The heat removed is rejected to either the normal power heat sink (NPHS) or to the auxiliary heat sink (AHS). The portions of the PSWS that are not part of the

ESBWR standard plant consist of the heat rejection facilities (NPHS and AHS), which are dependent on actual site conditions. The conceptual design utilizes a natural draft cooling tower for the NPHS and mechanical draft cooling towers for the AHS, with a crosstie line to permit routing of the plant service water to either heat sink. Basin water level is monitored to ensure that sufficient NPSH at design flow is provided to the PSWS pumps. The conceptual design information (CDI) for the heat rejection facilities of the PSWS will be replaced with site-specific design information in the combined license application (COLA) Final Safety Analysis Report (FSAR).

The PSWS is designed so that neither a single active nor single passive component failure results in a complete loss of nuclear island cooling or plant dependence on any safety-related system. This is achieved by redundant components, automatic valves and piping cross-connects for increased reliability. The PSWS is designed to operate during a loss of preferred power (LOPP).

Each PSWS train consists of two 50-percent capacity vertical pumps taking suction in parallel from the plant service water basin. Discharge is through a check valve, a self-cleaning strainer, and a motorized discharge valve at each pump to a common header. Each common header supplies plant service water to each RCCWS and TCCWS heat exchanger train arranged in parallel. The plant service water is returned via a common header to the mechanical draft cooling towers AHS in each train or to the NPHS. Remotely operated isolation valves and a crosstie line permit routing of the plant service water to either heat sink. RCCWS and TCCWS heat exchangers are provided with remotely operated isolation valves. Flow control valves are provided at each heat exchanger outlet.

The PSWS has RTNSS functions as described in DCD Tier 2, Appendix 19A (which provides the level of oversight needed to meet the RTNSS functions). Performance of RTNSS functions are assured by applying the defense-in-depth principles of redundancy and physical separation to ensure adequate reliability and availability as described in DCD Tier 2, Section 19A.8.3.

In addition to the CDI referred to above, COL Information Item 9.2.1-1-A, “Material Selection,” specifies that the COL applicant will determine material selection, including the need for valve hard seat material, and provide provisions to preclude long-term corrosion and fouling of the PSWS based on site water quality analysis.

In the event of a LOPP, the PSWS supports the RCCWS in bringing the plant to cold-shutdown condition within 36 hours assuming the most limiting single active failure. DCD Tier 2, Revision 9, Figure 9.2-1, is a simplified diagram of the PSWS. DCD Tier 2, Revision 9, Tables 9.2-1 and 9.2-2, tabulate the PSWS design heat loads and component design characteristics.

9.2.1.3 *Staff Evaluation*

The staff’s review of the PSWS is based on guidance found in SRP Section 9.2.1, Revision 5, and applicable regulations such as GDC 2, 4, 5, 44, 45 and 46. The staff’s review also includes 10 CFR 20.1406. The PSWS for the ESBWR differs from that of the traditional BWR designs in that the ESBWR PSWS is a nonsafety-related system because the PSWS removes heat only from the RCCWS and TCCWS, which are not safety-related systems. Therefore, portions of SRP Section 9.2.1 that apply to safety-related systems do not apply to the PSWS.
Sections 9.2.2 and 9.2.8 of this report contain the staff evaluations of the RCCWS and TCCWS.

9.2.1.3.1 System Design Considerations

As previously stated, the PSWS has RTNSS functions. The PSWS, which is a nonsafety-related active system, should be highly reliable and capable of achieving and maintaining cold shutdown conditions. In addition, there should be no single failure of this system which would result in an inability to terminate the use of the passive safety-related systems and achieve cold shutdown. DCD Tier 2, Revision 9, Table 9.2-1, includes a design limiting condition for the PSWS to reaching cold shutdown conditions (i.e. cooling the plant to Mode 5 conditions) within 36 hours. This design limiting condition is intended to satisfy the ESBWR TS requirements in which numerous TS sections require Mode 5 entry within 37 hours. The PSWS, which is designated as RTNSS (including its support systems), is subject to enhanced design, quality, reliability, and availability provisions and is relied upon for performing functions as discussed in Tier 2 of DCD, Appendix 19A. Sufficient information needs to be included in Tier 1 and Tier 2 of the DCD to demonstrate that the PSWS is adequate for achieving and maintaining cold shutdown conditions (i.e., cooldown from Mode 4 to Mode 5), performing RTNSS functions, and satisfying applicable design consideration.

On March 19-20, 2009, the staff conducted a regulatory audit of the supporting information for DCD Tier 2, Section 9.2, including the PSWS (Section 9.2.1), RCCWS (Section 9.2.2) and nuclear island chilled water subsystem (NICWS) (Section 9.2.7). The audit was primarily focused on the review of these systems with regard to RTNSS and the ability to support cold shutdown operations. A summary of the audit, including participants and audit activities may be found in ADAMS at Accession Number ML101250439. The remainder of this section refers to this audit.

A. Plant Service Water System Classification and Quality Assurance Provisions

DCD Tier 2, Revision 9, Section 3.2, specifies the classification of SSCs based on safety importance and other considerations. Section 3.2 of this report provides the staff's evaluation of the specified classification designations. This section of the staff's evaluation confirms that the appropriate classification designations are specified for the PSWS consistent with the approach described in DCD Tier 2, Revision 9, Section 3.2, and that the designations properly reflect the regulatory oversight provisions that pertain to the PSWS (RTNSS Criterion C), as discussed in DCD Tier 2, Section 19A.8. The staff reviewed DCD Tier 2, Revision 9, Figure 9.2-1, and confirmed that the classification designations on the simplified diagrams are consistent with those that are listed for the PSWS in DCD Tier 2, Revision 9, Table 3.2-1. In particular, the following classification designations are specified in DCD Tier 2, Table 3.2-1 for the PSWS:

- The PSWS is designated as Safety Class N which is used for nonsafety-related applications. The PSWS does not perform any safety-related functions and the N designation is therefore appropriate.

- The PSWS is designated as Quality Group D. As discussed in DCD Tier 2, Section 3.2.4, this quality group generally applies to nonsafety-related SSCs that satisfy specified industry codes and design standards and are subject to one or more significant licensing requirements or commitments. The staff concludes that this is an appropriate quality group designation since the PSWS does not perform a safety-related function and does not interface with any safety-related component.

- DCD Revision 6 specifies QA Requirement S for the PSWS, as stated in the applicant's response to RAI 3.2-6 S02. Based on the RAI response, RTNSS components and systems

that were identified in Revision 5 of the DCD as QA Requirement E are to be changed to QA Requirement S in Revision 6. QA Requirement S has special QA requirements that apply during the design and procurement specification preparation processes, in accordance with procedures that will be established. The staff concludes that this is an appropriate QA group since the PSWS does not perform a safety-related function and does not interface with any safety-related component. However, the PSWS has RTNSS functions that are assured by applying the defense in depth principles of redundancy and physical separation to ensure adequate reliability and availability. In addition, the staff concluded that Revision 6 of the DCD has incorporated this RAI proposed change, which the staff determined this change to be acceptable.

The PSWS is designated as seismic Category non-seismic (NS). Seismic Category NS is used for nonsafety-related SSCs and is appropriate for those nonsafety-related SSCs that are classified as RTNSS Criterion C because augmented seismic design standards do not apply. As stated in DCD Tier 2, Revision 9, Section 19A.8.3, RTNSS C systems do not require augmented seismic design criteria. However, some RTNSS C systems are housed in seismic Category I or II structures, and some are housed in NS structures that are designed to maintain structural integrity with a margin of safety that is equivalent to a seismic Category I structure under SSE conditions. As described in DCD Tier 2, Table 3.2-1, PSWS is housed in the service water building (non-seismic) and the turbine building (seismic Category II) with the remainder of the PSWS outdoors onsite. Therefore, seismic Category NS is appropriate for the PSWS.

B. GDC 2

To meet the requirements of GDC 2 relating to structures and systems being capable of withstanding the effects of natural phenomena, SRP Section 9.2.1 indicates that acceptance depends on meeting the guidance of the portions of Regulatory Position C.2 of RG 1.29, Revision 4, regarding nonsafety-related systems. In RAIs 9.2-12 and 9.2-12 S01, the staff requested that the applicant demonstrate that the PSWS (among other water systems) satisfies Regulatory Position C.2 of RG 1.29. In responses, the applicant explained that the PSWS does not have any piping in the control room, and it is not possible for the PSWS to result in an incapacitating injury to occupants of the control room or interface with any safety-related components. The PSWS is under the RTNSS designation to provide cooling functions and post-72-hour cooling to the RCCWS. It will be designed to seismic requirements to be specified in DCD Tier 2, Revision 9, Appendix 19A and Section 3.2. Chapter 22 of this report provides the staff’s evaluation of the RTNSS systems and the associated design bases. The staff reviewed the above RAI responses and DCD Tier 2, Revision 5, Section 9.2.1.1. Based on the above, the staff finds that the PSWS meets the guidance of the portions of Regulatory Position C.2 of RG 1.29 regarding nonsafety-related systems because the failure of the nonsafety-related portions of the systems does not impact any safety-related SSCs or could it incapacitate the control room occupants. Accordingly, based on the above and the applicant’s response, the portions of RAI 9.2-12 relating to the PSWS are resolved.

C. GDC 4

SRP Section 9.2.1 provides guidance to review the PSWS against GDC 4, as it relates to the dynamic effects associated with water hammer.

Since the PSWS return flow piping to the natural draft and mechanical draft cooling towers is well above the water levels in the service water basins, the standby cooling loop (or both loops during a loss of power) can potentially drain down and create a void in the PSWS piping. If this

were to occur, a potentially damaging water hammer event could occur upon an automatic start of the affected loop or loops. Any loop voids in the PSWS that are caused by component design or piping configuration could result in a much more severe water hammer event. The system description indicates that the potential for water hammer is mitigated through the use of various system design and layout features, such as automatic air release and vacuum valves installed at high points in system piping and at the pump discharge, proper valve actuation times to minimize water hammer, procedural provisions ensuring proper line filling before system operation and after maintenance operations, and the use of a check valve at each pump discharge to prevent backflow into the pump.

The staff discussed with the applicant water hammer considerations at the March 19-20, 2009, audit and the applicant's responses to RAIs 9.2-11, 9.2-11-S01, 9.2-11-S02, 9.2-11-S03, 9.2-11-S04, and RAI 9.2-24 addressed this issue. In these RAIs the staff asked the applicant to discuss the potential for water hammer, as well as operating and maintenance procedures for the avoidance of water hammer in the PSWS and RCCWS. RAI 9.2-11 was being tracked as an open item in the SER with open items. In response to RAI 9.2-11, the applicant listed the following provisions to mitigate water hammer:

- Minimize high points in the system.
- Provide for venting at all high points.
- Have the COL applicant address procedural requirements ensuring proper line filling before system operation and following maintenance operations.
- Keep valve actuation times slow enough to prevent water hammer.
- Use check valves at pump discharge to prevent backflow into the pump.

In DCD Tier 2, Table 1.11-1, the applicant identified Task Action Item A-1, "Water Hammer," as a means of meeting the guidance of several SRP sections. SRP Section 9.2.1 is among the sections that discussed the issue. The staff determined that the response to RAI 9.2-11 did not completely address all of the water hammer issues; therefore, the staff included the issue in RAI 9.2-24 to ask the following:

  - The amount of back leakage through the pump check valves that is considered to be excessive needs to be specified and explained, the means by which excessive check valve back leakage or system voiding will be prevented from occurring over time needs to be described.
  - A description needs to be provided for how proper operation of the automatic air release and vacuum valves will be assured over time.
  - Valve actuation and stroke times that are considered to be appropriate (especially with respect to the air operated valves (AOVs)) needs to be specified and explained, and how these times will be maintained as the plant ages needs to be described.

In response to RAI 9.2-24, the applicant stated that the PSWS design provides provisions to prevent water hammer by preventing voiding in liquid lines, control valve instability and excessive valve actuation time. The applicant will perform a detailed hydrodynamic analysis

during the detailed design phase with input from the COL applicant to determine the size, location and number of vacuum breaker valves used to prevent voiding. Operational and maintenance procedures will be employed to prevent water hammer caused by improper filling of voided lines. Control valve instability will be prevented by specifying valve design parameters, such as actuator type, flow coefficient, and trim to be compatible with final designed operating conditions. For piping systems that rise more than 9.75 m (32 ft), column separation will be prevented by taking care to ensure that the pressure in any portion of the system will not be below the vapor pressure of the fluid. The valve and its control system will be designed to minimize the potential for oscillation instability by including features such as balanced trim design for all pressure drop and flow configurations, stiff actuators, moderate rate of operator response, long valve strokes, and minimal pressure drop. Proper operation of system valves under expected operating conditions including timing will be verified during pre-operational startup testing described in DCD Tier 2, Revision 9, Section 14.2.8.1.51. The detailed hydrodynamic analysis for the PSWS will ensure all valves will be designed and controlled so the opening and closing time is sufficiently long to prevent unacceptably high pressure waves. Where water hammer could be caused by a stuck-open check valve slamming shut or by an abnormal valve actuation resulting from actuator failure, the valves will be designed to allow thorough and proper inspection, testing and maintenance. In addition, the applicant in response to RAI 9.1-11 S04 revised DCD Tier 2, Section 13.5.2, to include provisions to ensure that procedures developed for RTNSS systems will address water hammer.

Based on its review of the applicant's response to RAIs 9.2-11 and 9.2-24, the staff concludes that the applicant adequately addressed water hammer since the PSWS design incorporated water hammer mitigation features and components, the hydrodynamic analysis will be performed to preclude a water hammer event, and operational procedures are to be developed addressing water hammer concerns for the RTNSS systems as part of COL Information Item 13.5-2-A. Accordingly, based on the above and the applicant's response, RAIs 9.2-11 and 9.2-24 as they relate to water hammer are resolved. Based on the above, the staff finds that the PSWS meets the requirements of GDC 4, in accordance with the guidance of SRP Section 9.2.1.

D. GDC 5, GDC 44, GDC 45 and GDC 46

The ESBWR design is a single-unit station, and the requirements of GDC 5 do not apply to the single-unit design.

Based on the requirements of GDC 44, 45, and 46 and SRP Section 9.2.1 guidance, the staff's review of the PSWS against GDC 44, 45, and 46 is based on the ability of the PSWS to remove heat from SSCs important to safety to a heat sink under normal operating and accident conditions and the availability of design provisions for inspection and operational testing.

As stated in DCD Tier 2, Revision 1, Table 1.9-9 and Section 9.2.1, the applicant determined that GDC 44, 45, and 46 were not applicable to the PSWS, among other systems. In RAIs 9.2-7, 9.2-7 S01, and 9.2-7 S02, the staff questioned this determination. In response to these RAIs, the applicant revised DCD Tier 2, Revision 5, Table 1.9-9, to address conformance to GDC 44, 45, and 46. The applicant also clarified that the PSWS satisfies the requirements of GDC 44, 45, and 46 because the design of the PSWS included the following provisions:

- Capability to transfer heat loads from SSCs to a heat sink under normal and accident conditions

- Component redundancy so the system will remain functional assuming a single failure coincident with a LOOP

- Capability to isolate components or piping so system function will not be compromised

- Design provisions to permit inspection and operational testing of components and equipment

The staff believes that those portions of the GDC 44 requirements that apply to the heat removal function under normal operating conditions also apply to the PSWS. The staff reviewed the PSWS in terms of the designed heat removal capability, component redundancy and single-failure design, plant TS shutdown cooling requirements, and testing and inspection requirements, as described in DCD Section 9.2.1, and finds that the PSWS satisfies GDC 44, 45, and 46 with respect to its normal operation function. However, in a DBA, decay heat is transferred to the ICS/PCCS pools. The portions of the GDC 44 requirements that apply to a safety-related system to remove decay heat following an accident do not apply to the PSWS. The staff finds that the design of the PSWS satisfies the applicable portions of GDC 44, 45, and 46 based on the above review. In addition, the staff finds that the response to RAI 9.2-7 was acceptable since the applicant clarified conformance of the PSWS to GDC 44, 45, and 46 and described how this is achieved. Accordingly, based on the above and the applicant's response, RAI 9.2-7 is resolved. The PSWS design attributes, including system capability, reliability, heat transfer, pump NPSH, operating experiences, testing, and instrumentation and controls (which are related to the applicable GDCs), are further addressed below for the PSWS RTNSS and cold shutdown functions.

E. Minimization of Contamination; 10 CFR 20.1406

10 CFR 52.47(a)(6) and 10 CFR 20.1406 require applicants for standard plant design certifications to describe how facility design and procedures for operation will minimize contamination of the facility and the environment. The staff's review criteria (SRP Section 9.2.1, Paragraph III.3.D) specify that provisions should be included to detect and control leakage of radioactive contamination into and out of the PSWS. The staff considers the design to be acceptable if the simplified diagrams of the PSWS show that radiation monitors are located on the PSWS discharge and at components that are susceptible to leakage, and if the components that are susceptible to leakage can be isolated

In RAIs 9.2-8, 9.2-8 S01, and 9.2-8 S02, the staff asked the applicant to demonstrate the capability to detect, control, and isolate PSWS leakage, including radioactive leakage into and out of the system, and prevention of accidental releases to the environment. In addition, the staff asked the applicant to describe allowable operational degradation (e.g., pump leakage) and the procedures to detect and correct these conditions when they become excessive. The staff also requested the applicant to clarify where the DCD stated that it requires continuous radiation monitoring. RAI 9.2-8 was being tracked as an open item in the SER with open items. In responses, the applicant stated that the flow rate reduction would indicate possible system water losses or pump degradation portions of the PSWS that have adverse flow reduction so that they could be isolated, identified, and repaired without immediately impacting plant operation; the PSWS design includes provisions for grab sampling; and the COL information item (COL Information Item 11.5-2-A in DCD Tier 2, Section 11.5.7) will make provisions for sampling cooling tower blowdown as referenced in DCD Tier 2, Table 11.5-5. The DCD requires continuous effluent monitoring either directly on the effluent of the PSWS or another downstream process effluent (i.e., one detector could monitor the combined effluent of the

PSWS and circulating water) to ensure monitoring before release to the environment. The staff finds that the RAI responses are acceptable since the applicant clarified the provisions for PSWS leakage, radiation monitoring, and sampling. Based on the above and the applicant's response, RAI 9.2-8 is resolved.

The staff noted that (1) DCD Tier 2, Section 9.2.1 did not describe radiation monitors (including alarm functions) and (2) the PSWS simplified diagrams did not show the radiation monitors. Consequently, the applicant had not adequately addressed the provisions of 10 CFR 20.1406. Therefore, the staff requested, in RAI 9.2-26, that the applicant revise DCD Tier 2, Section 9.2.1, and the simplified diagrams, as appropriate, to address the requirements of 10 CFR 20.1406.

In response, the applicant stated that radioactive leakage into the PSWS from the RCCWS can only occur following these three independent failures:

(1) RCCWS can only become contaminated by the interface with either RWCU/SDC, postaccident sampling program coolers and process sampling system (PSS) coolers or FAPCS, which could occur only by failure through the heat exchangers associated with those systems.

(2) The RCCWS is equipped with continuous radiation monitors (Reference DCD Tier 2, Revision 5, Section 11.5.3.2.6 and Table 11.5-5). If these detectors alarm, the applicable train and/or equipment will be isolated. If these alarms fail and isolation of the affected RCCWS loop is not performed, a third failure is required to contaminate PSWS.

(3) In addition to these two failures, a leak from the RCCWS process water into the PSWS cooling water at the interface in the RCCWS heat exchangers would have to occur. RCCWS is designed using plate heat exchangers and leakage through holes or cracks in the plates is not considered credible based on industry experience with plate type heat exchangers. These heat exchangers are also designed such that any gasket leakage from either RCCWS or PSWS drains to the equipment and floor drain system (Reference DCD Tier 2, Revision 5, Section 9.2.2.2). Consequently, there is essentially no potential for plate failure and cross contamination.

DCD Tier 2, Revision 9, Section 9.2.1.2, explains that the PSWS design detects any potential gross leakage and alarms in the MCR and permits the isolation of any such leak in a sufficiently short period of time so as to preclude extensive plant damage. Means are provided to detect leakage into the PSWS from the RCCWS, which may contain low levels of radioactivity.

DCD Tier 2, Revision 9, Section 9.2.2.2, states that the RCCWS provides cooling water to nonsafety-related components in the nuclear island and provides a barrier against radioactive contamination of the PSWS. DCD Tier 2, Revision 9, Section 9.2.2.5, explains that RCCWS surge tank levels are used to monitor losses of cooling water, and detect intersystem leakage intrusions into the RCCWS. The level transmitters in the surge tank standpipes, in combination with low-low surge tank level, automatically initiate a train shut down. A train shutdown signal will trip off all pumps in the train and close all isolation, bypass, and flow control valves. RCCWS radiation monitors are provided for monitoring radiation levels and alerting the plant operator of abnormal radiation levels. The PSWS and RCCWS are designed with provisions to

detect and control leakage of radioactive contamination into and out of the PSWS and to minimize contamination of the facility and the environment.

DCD Tier 2, Revision 9, Table 12.3-18, which addresses RG 4.21 design objectives and applicable DCD section information, describes similar provisions related to the PSWS for the following objectives:

- Minimizing leaks and spills (Design Objective 1)
- Provide for adequate leak detection capability to provide detection of leakage for any SSC which has the potential for leakage (Design Objective 2)
- Decreasing the spread of contaminant from the source (Design Objective 4)

The staff finds that these design provisions for the PSWS meet the requirements of 10 CFR 20.1406 and conforms to the guidelines of RG 4.21. Sections 12.4 and 12.7 of this report further address the ESBWR design in accordance with 10 CFR 20.1406. The staff finds that the RAI response is acceptable since the applicant clarified why radiation monitors do not need to be described for the PSWS and confirmed that 10 CFR 20.1406 requirements have been satisfied. Accordingly, based on the above and the applicant's response, RAI 9.2-26 is resolved.

F. Protection from Probable Hazards

In accordance with the policies referred to in Chapter 22 of this report, SSCs that are classified as RTNSS should be protected from the more probable hazards that exist. As previously discussed the PSWS is classified as RTNSS Criterion C. DCD Tier 2, Section 19A.8.3, indicates that RTNSS Criterion C systems incorporate the defense-in-depth principles of redundancy and physical separation to ensure adequate reliability and availability. DCD Tier 2, Section 19A.8.3, also indicates that RTNSS Criterion C systems and structures meet design standards to withstand wind and missiles generated from Category 5 hurricanes, and non-RTNSS systems that can adversely interact with RTNSS Criterion C systems are designed to the same seismic requirements as the affected RTNSS system. Additionally, DCD Tier 2, Section 19A.8.3, indicates that RTNSS Criterion C equipment is qualified to The Institute of Electrical and Electronics Engineers, (IEEE) Standard 344-1987, "Recommended Practice for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations-Description," to demonstrate structural integrity.

RTNSS Criterion C systems in the ESBWR design, such as the PSWS, do not require augmented design standards to ensure reliable performance in the event of hazards such as seismic events, high winds, flooding, and environmental conditions experienced during an accident. RTNSS Criterion C systems are designed to standards to withstand wind and missiles generated from Category 5 hurricanes.

As indicated in the applicant's response to RAI 9.2-24, PSWS supports plant investment protection (PIP) and defense-in-depth goals. DCD Tier 2, Revision 9, Section 9.2.1.2, describes that, in the event of a LOPP, the PSWS supports the RCCWS in bringing the plant to the cold shutdown condition within 36 hours, assuming the most limiting single active or passive component failure. Because the PSWS cooling water systems are also significant contributors to plant availability and plant investment protection, the ESBWR design is focused on ensuring that these systems are available and reliable. Therefore, design goals for plant investment

protection and defense-in-depth protection (seismic ruggedness; redundancy; and fire, missile, and flood protection) may be more restrictive than the applicable RTNSS provisions.

In summary, the PSWS is a support system to the FAPCS and is only included as an augmented system to address uncertainties in the defense in depth role of the FAPCS in providing a backup source of lower pressure injection and SPC. RTNSS Criterion C systems are not designed to the level of RTNSS Criterion B, systems in that seismic events, flooding, and environmental conditions are not considered. The staff finds that this graded design approach is acceptable considering the design function of the PSWS under the regulatory criteria for this nonsafety system.

G. Plant Service Water System Capability and Reliability

In RAI 9.2-24, the staff requested that the applicant to specifically address information concerning the PSWS functions that are subject to RTNSS, focusing on PSWS capability and reliability. This RAI included the following key points:

- The most limiting conditions upon which the PSWS design is based with the amount of excess margin built in to the design.

- Clarification in the DCD descriptions, drawings, and tables (to include valves, strainers, air interface, instrumentation logic, and installed instruments).

- PSWS pump design to include pump recirculation protection, minimum NPSH, and pump protection for debris.

- PSWS freeze protection, erosion, gross leakage detection, and component back leakage.

- PSWS basin design and minimum water level, consideration for pump clogging and silting, and cross-connect configuration.

- PSWS cooldown provisions (24 hours and 36 hours) and system alignment to support cooldown.

- PSWS vacuum breaker design and water hammer consideration.

- PSWS component testing and component reliability.

In resolving this RAI, the staff audited supporting information for the PSWS on March 19 and 20, 2009, as discussed above. The response to RAI 9.2-24 addresses both the RAI and the audit findings. The remainder of the section discusses the results of the audit and the RAI response.

(1) <u>Descriptive Information and Flow Considerations</u>

The staff reviewed the DCD Tier 2, Section 9.2.1, PSWS description and drawings to confirm that the design bases, flow paths, and components have been identified and described in sufficient detail to enable a complete understanding of the system design and operation. The staff determined that additional information was needed and requested in RAI 9.2-24 that the applicant revise Section 9.2.1 to address the following considerations:

- Nominal pipe sizes and system flow rates

- MOV and AOV design, including a discussion of valve hard seat materials
- System freezing design requirements
- Pump protection from debris
- System strainer mesh size

The RAI response addressed in detail each of the above noted items. The staff finds them acceptable since the most limited piping velocities were approximately 4.6 m per second (15 ft per second) or less. In the staff's experience and in accordance with general engineering practice, piping velocities between 1.2 and 4.6 m per second (4-15 ft per second) are reasonable. Thus, the staff expects long term internal pipe wear to be minimal. The staff reviewed the remaining items noted above as part of the RAI response and the staff concluded these items had been properly addressed. The RAI response provided a DCD mark-up related to the need of valve hard seat material, which is identified as a COL information item. The staff confirmed that Revision 6 of the DCD incorporated this RAI proposed change. The staff finds that the response to RAI 9.2-24 regarding PSWS descriptive information and flow considerations is acceptable since the applicant clarified the basis for the design parameters included in the DCD and the need for hard seat material. Accordingly, based on the above and the applicant's response, the flow consideration aspects of RAI 9.2-24 are resolved.

(2) Heat Transfer

The staff reviewed the PSWS description in DCD Tier 2, Section 9.2.1, and applicable DCD tables to confirm that the heat transfer and flow capabilities are adequately specified and that the bases for these values are fully explained. DCD Tier 2, Table 9.2-1, lists the PSWS heat loads for various operating modes and indicates that the most limiting case is a single train failure cooldown. The staff determined that additional information was needed and requested in RAI 9.2-24 that the applicant revise DCD Tier 2, Section 9.2.1, to address heat transfer and the amount of excess margin and to include uncertainties for wear and aging effects.

The applicant's response to RAI 9.2-24 provided further detailed explanation related to the PSWS heat loads. Single train failure during cooldown results in the greatest heat load per PSWS train at 80.8 MW (2.75×10^8 British thermal unit per hour [BTU/h]). This transient mode occurs with one train of the PSWS in operation (two PSWS pumps) and all heat loads are dissipated through two RCCWS heat exchangers and two TCCWS heat exchangers. LOPP cooldown with single train failure is the most limiting system heat removal design condition for the RCCWS. This mode of operation differs from single train failure during cooldown in that the TCCWS heat loads are replaced with the heat loads associated with a standby diesel generator (SDG). This transient mode occurs when a LOPP and a single train failure occur concurrently. Similar to the single train failure transient, only one train is in operation and all heat loads are dissipated using three RCCWS heat exchangers (and two PSWS pumps on the active PSWS train. Two PSWS pumps provide sufficient cooling capacity to the RCCWS heat exchangers to bring the plant to the cold shutdown condition within 36 hours. This mode of operation removes 74.8 MW (2.55×10^8 BTU/h) from RCCWS using one train of PSWS.

The RAI response regarding DCD Table 9.2-2 states that each of the PSWS cooling towers is capable of removing a minimum of 83.5 MW (2.85×10^8 BTU/h). Based on the staff's review, for the two bounding conditions noted above, there is at least an 83.5 MW/80.8 MW (2.85×10^8 BTU/h / 2.75×10^8 BTU/h) or a 3.3 percent design margin between the cooling tower capacity and the heat loads. In addition, for support of RTNSS only, the heat loads are 21.9 MW/80.8 MW (7.47×10^7 BTU/h / 2.75×10^8 BTU/h) or a 368 percent design margin between the cooling tower capacity and the heat loads. Based on the staff's review of the RAI response, the staff

finds the heat transfer capability of the PSWS of sufficient margin to support normal plant cooldown, single train failure cooldown, LOPP operation, and RTNSS support. The RAI response provided a DCD markup related to the clarification of the PSWS heat loads and PSWS component design characteristics in DCD Tier 2, Tables 9.2-1 and 9.2-2. The staff confirmed that Revision 6 of the DCD incorporated these RAI proposed changes. The staff finds that the response to RAI 9.2-24 regarding heat transfer is acceptable since the applicant clarified the basis for the heat loads in the DCD and added corresponding clarifications to DCD tables identifying PSWS heat loads and component design characteristics.

(3) Single Failure and Backup Power Considerations

As described in DCD Tier 2, Section 9.2.1, the PSWS consists of two fully redundant (train A and train B), 100 percent capacity trains with each train consisting of two 50-percent pumps powered by separate SDGs. Although the two trains are normally cross-connected via AOVs, they can be split out if necessary from the control room. The staff determined that clarification was needed for the case in which offsite power is not available and requested, in RAI 9.2-24, that the applicant revise DCD Tier 2, Section 9.2.1 to address single failure of the cooling tower basin, backup power for the self cleaning strainer functions, and AOVs.

The response to RAI 9.2-24 addressed in detail each of the above noted items. All MOVs fail as is during a LOPP and can change position once power is restored via the SDG. During a LOPP, the MOVs associated with the PSWS pump discharge system cross tie and mechanical draft cooling tower cross-tie will auto close (once power is restored), thus providing PSWS train separation. There are redundant valves at these two locations, thus PSWS train isolation will still occur if one valve fails to isolate. The PSWS basin full-flow bypass block valves, which are manually opened and closed from the MCR, fail-as-is during a LOPP, thus maintaining PSWS system flows. AOVs associated with flow control through the RCCWS and TCCWS heat exchangers fail open, thus maintaining PSWS system flows.

Based on the its review of the RAI response, the staff finds that the applicant has properly addressed the single-failure consideration because of the redundancy of the design, availability of component emergency power supply, and component failure position during a LOPP. In addition, train redundancy ensures that single failure of any AOV will not impact the other train. The staff finds that the response to RAI 9.2-24 regarding single failure and backup power is acceptable since the applicant clarified how the DCD includes the single-failure and backup power attributes of the PSWS.

(4) Plant Service Water System Pump Net Positive Suction Head

DCD Tier 2, Revision 9, Section 9.2.1 states that the PSWS pumps have sufficient NPSH under worst case conditions. Basin water level is monitored to ensure that sufficient NPSH at design flow is provided to the PSWS pumps.

To provide minimum system flow, the PSWS design should ensure that the minimum NPSH for the PSWS pumps is satisfied for all postulated conditions, including vortex formation considerations. The system description indicates that the PSWS pumps have sufficient available NPSH under worst case conditions and the water levels in the service water basins are monitored to ensure sufficient NPSH. However, the system description did not detail (1) the specific minimum NPSH for the PSWS pumps; (2) the minimum service water basin water level necessary to provide NPSH and the basis for this determination and limiting assumptions that were used (e.g., water level, maximum temperature, maximum flow rate, number of pumps

operating, vortex effects); (3) how this minimum water level compares to the minimum water level that is maintained in the service water basins to satisfy excess margin and inventory considerations; and (4) how COL applicants will know to periodically confirm that adequate levels exist in the service water basins. Therefore, the staff requested, in RAIs 9.2-23, 9.2-23 S01, and 9.2-24, that the applicant address NPSH and additional questions regarding the design alarm features in the MCR available to the operators. In addition, the staff asked the applicant to revise DCD Tier 2, Section 9.2.1, to include this information and to establish COL information items and interface requirements as appropriate.

In response to these RAIs, the applicant provided changes to DCD Revision 5 and markups to Revision 6. The applicant stated that, in Revision 5 of DCD Tier 2, Section 9.2.1.2, the design of the heat rejection facilities and PSWS pumps have sufficient available NPSH under worse-case conditions. Basin water level is monitored to ensure that sufficient NPSH at design flow is provided to the PSWS pumps. In addition, the change to DCD Tier 1, Section 4.1, stated that the PSWS pumps must have sufficient available NPSH at the pump suction location for the lowest probable water level of the heat sink. In response to RAI 9.2-23 S01, the applicant revised the description of the interface between the standard plant design for the ESBWR and the conceptual design to be addressed by COL applicants to include consideration of NPSH under worst-case conditions. In addition, DCD Tier 2, Revision 9, Section 14.2.8.1.51, describes a series of individual component and integrated system tests to demonstrate acceptable pump suction under the most limiting design flow conditions.

The staff finds that the responses to RAIs 9.2-23 and 9.2-24 regarding PSWS pump NPSH are acceptable since the applicant clarified how sufficient NPSH is assured. The applicant also added DCD Tier 1 interface requirements and clarified how testing in accordance with Section 14.2.8.1.51 addresses NPSH under the most limiting design flow conditions. Therefore, the staff's concern regarding NPSH is resolved. The staff confirmed that Revision 6 of the DCD incorporated these RAI proposed changes.

(5) <u>Operating Experience</u>

DCD Tier 2, Revision 5, Chapter 1, identifies the following generic issues as not applicable for the ESBWR:

- GL 89-13, "Service Water System Problems Affecting Safety-Related Equipment," dated July 19, 1989, is identified in DCD Tier 2, Table 1C-1.
- Supplement 1 to GL 89-13, "Service Water System Problems Affecting Safety-Related Equipment," dated April 4, 1990, is identified in DCD Tier 2, Table 1C-1.
- New Generic Issue 51, "Proposed Requirements for Improving the Reliability of Open Cycle Service Water System," is identified in DCD Tier 2, Table 1.11-1.
- New Generic Issue 153, "Loss of Essential Service Water in LWRs," is identified in DCD Tier 2, Table 1.11-1.
- IE Bulletin 81-03, "Flow Blockage of Cooling Water to Safety System," is identified in DCD Tier 2, Table 1C-2.

Related to IE Bulletin 81-03, in RAI 9.2-9, the staff asked the applicant to describe the measures provided for precluding long-term corrosion and organic fouling that would degrade PSWS

performance. In responses, the applicant stated that the type of water (e.g., fresh or sea water) and the results of water quality analysis for a COL applicant would determine the material selection for all piping and pump parts wetted by raw PSWS water. In DCD Tier 2, Revision 5, Section 9.2.1.2, the applicant stated that the COL applicant would determine material selection and make provisions to preclude long-term corrosion and fouling of the PSWS based on site water quality analysis. DCD Tier 2, Revision 5, Section 9.2.1.6, identifies a corresponding COL Information Item, 9.2.1-1-A, "Material Selection." The staff finds that the RAI response, with the addition of COL Information Item 9.2.1-1-A, is acceptable since it clarified that a COL applicant would make provisions for precluding long-term corrosion and organic fouling of the PSWS. This also addresses IE Bulletin 81-03. Accordingly, based on the above and the applicant's response, RAI 9.2-9 is resolved.

However, the staff did not agree with the applicant's position that GL 89-13 need not be considered for the ESBWR because it was issued for safety-related systems. The staff believes that, while the PSWS is not safety-related, it performs defense-in-depth functions, and there is no basis to conclude that the provisions of the GL should not apply to those systems that perform these functions. Defense-in-depth systems differ from typical nonsafety-related systems in that they are subject to regulatory oversight and are expected to be highly reliable, as reflected in the policies that are referred to in Chapter 22 of this report. The provisions of GL 89-13 were developed based on plant operating experience to ensure that the capability and reliability of service water systems to perform their functions as the plant ages, and from this perspective, the provisions of GL 89-13 apply to the PSWS. Therefore, the staff requested in RAI 9.2-24, that the applicant revise DCD Tier 2, Section 9.2.1, to describe how the provisions of GL 89-13 will be implemented to ensure the capability and reliability of the PSWS to perform its defense-in-depth functions over the life of the plant. Likewise, the applicant needs to revise its responses to the other operating experience items referred to in Chapter 1 that pertain to defense-in-depth systems and components to address the operating experience considerations as they relate to these important systems rather than inappropriately dismissing the items based on system safety classifications.

In response to RAI 9.2-24, the applicant stated that the ESBWR PSWS is not committed to meeting the recommendations of GL 89-13. DCD tier 2, Table 1C-1, states that the ESBWR has no safety-related service water and applies water quality standards to the use of water for safety functions. But, the recommendations have been integrated into the cooling water system design. The RAI response added the following:

- Conduct, on a regular basis, performance testing of all heat exchangers, which are cooled by the service water system. Testing should be done with necessary and sufficient instrumentation, though the instrumentation need not be permanently installed. The relevant temperatures should be verified to be within design limits. An example of an alternative action that would be acceptable to the NRC is frequent regular maintenance of a heat exchanger in lieu of testing for degraded performance of the heat exchanger. ESBWR PSWS design includes sufficient instrumentation to monitor performance of individual heat exchangers. The plate heat exchanger design utilized for PSWS heat loads could also be maintained through a preventative/predictive maintenance program.

- Verify that their service water systems are not vulnerable to a single failure of an active component. All ESBWR RTNSS systems are designed with

component redundancy so the system will remain functional assuming a single active failure coincident with LOPP.

- Inspect, on a regular basis, important portions of the piping of the service water system for corrosion, erosion, and biofouling. Ensure by establishing a routine inspection and maintenance program for open-cycle service water system piping and components that corrosion, erosion, protective coating failure, silting, and biofouling cannot degrade the performance of the safety-related systems supplied by service water. The maintenance program should have at least the following purposes: to remove excessive accumulations of biofouling agents, corrosion products, and silt; to repair defective protective coatings and corroded service water system piping and components that could adversely affect performance of their intended safety functions. The PSWS design incorporates features to facilitate inspection and allow for planned maintenance. Material selection for all PSWS components wetted by raw cooling water will match the corrosion resistance of the material to the water chemistry. Both operating and stagnant (shutdown) conditions will be addressed, including placing components and idle loops in wet layup. Erosion resistance will also be addressed. Pipe size and routing support remote visual inspections and repairs. The PSWS basin is equipped with a trash rack in order to prevent damage to the PSWS pumps due to ingestion of large debris and minimize macrofouling.

- Reduce human errors in the operation, repair, and maintenance of the service water system. The ESBWR Human Factors Engineering (HFE) design process integrates human capabilities and limitations into the PSWS.

The staff finds that the response to RAI 9.2-24 regarding operating experience is acceptable since the applicant properly addressed applicable operating experiences for the RTNSS, nonsafety-related PSWS. The staff finds that the applicant has addressed the major concerns of service water system degradation over time and adequately addressed in the design sufficient instrumentation to monitor performance of individual heat exchangers, component redundancy, inspections and planned maintenance, proper material selections, and human factors consideration. Accordingly, based on the above and the applicant's response, the operating experience aspects of RAI 9.2-24 are resolved.

(6) <u>Periodic Inspections and Testing</u>

As discussed in Item D above, the applicant demonstrated in DCD Tier 2, Revision 5, Section 9.2.1.1, that the PSWS satisfies GDC 44, 45, and 46 because the design of the PSWS includes design provisions to permit inspection and operational testing of components and equipment.

DCD Tier 2, Revision 9, Section 9.2.1.4, describes the applicant's provisions for periodic inspection of components to ensure the capability and integrity of the system. The pumps are tested in accordance with American National Standards Institute/Hydraulic Institute ANSI/HI 2.6 (M108), "Vertical Pump Tests." Testing is performed to simulate the various modes of operation to the greatest extent practical. MOVs are tested and inspected to ensure plant availability.

Periodic inspections and testing are important for assessing and maintaining the capability and reliability of the PSWS to perform its defense-in-depth functions over the life of the plant. The

PSWS design bases indicate that provisions are included to permit inspection of components and equipment. In addition, the system description indicates that valves are arranged for ease of in-service inspection. DCD Tier 2, Section 9.2.1.4, indicates that provisions are made for periodic inspection of components to ensure the capability and integrity of the system and that MOVs are inspected to ensure plant availability. The staff determined the periodic inspection and testing to be incomplete; therefore, the staff requested, in RAI 9.2-24, that the applicant revise DCD Tier 2, Section 9.2.1.

The applicant's response to RAI 9.2-24 noted that maintenance, testing, and operating procedures will include provisions for regular inspection testing and maintenance of valves to prevent degradation over time. As described in DCD Tier 2, Sections 19A.8, and 19A.8.4.9, all RTNSS systems are within the scope of the Design Reliability Assurance Program, as directed by DCD Tier 2, Chapter 17, which will be incorporated into the Maintenance Rule Program. The Maintenance Rule, 10 CFR 50.65, requires performance monitoring of SSCs that are not safety-related but are relied upon to mitigate accidents or transients, are used in EOPs, or whose failure could prevent safety-related SSCs from performing their safety-related function or could cause a reactor scram or actuation of a safety-related system. Such SSCs may include RTNSS components.

The staff finds that the periodic inspection and testing aspect of the RAI 9.2-24 response is acceptable since the PSWS will be monitored under the Maintenance Rule Program, which will include the maintenance of valves to prevent degradation over time. For the PSWS and other RTNSS systems covered by the Maintenance Rule Program, components are periodically tested and appropriate actions are taken if the PSWS SSCs are found degraded. Accordingly, based on the above and the applicant's response, the inspection and testing aspects of RAI 9.2 24 are resolved.

(7) Instrumentation, Controls, and Alarms

DCD Tier 2, Revision 9, Section 9.2.1.5, indicates that the PSWS is operated and monitored from the MCR, as well as from the remote shutdown panels. This section also briefly describes the PSWS automatic pump starts, pump discharge strainers operations, and PSWS header and heat exchangers instrumentation.

In RAI 9.2-10, the staff asked the applicant to identify all alarms, instruments, and controls for the PSWS. In response, the applicant explained all of the instruments, controls, and alarms in the MCR for the PSWS and revised the DCD accordingly. To address this RAI, the applicant made changes to DCD Tier 2, Section 9.2.1.5, in Revision 5 to provide the instruments and controls and alarms in the MCR. The staff finds that the RAI response and DCD changes are acceptable since DCD Tier 2, Revision 5, Section 9.2.1.5, identifies the instrumentation controls and alarms necessary for PSWS operation and indicates that they are in the MCR. Accordingly, based on the above and the applicant's response and DCD changes, RAI 9.2-10 is resolved.

In RAIs 9.2-6 and 9.2-6 S01, the staff requested that the applicant include simplified diagrams in the DCD for the PSWS and RCCWS showing system function, major equipment, components, piping classes, instrumentation, and interface systems. RAI 9.2-6 was being tracked as an open item in the SER with open items. In response, the applicant did not provide simplified diagrams in the DCD for the PSWS and RCCWS. The applicant indicated that the simplified diagrams are proprietary information and are not intended to be included in the DCD. This response did not provide sufficient bases for the staff to resolve the RAI. Subsequently, the

applicant added more details in the existing simplified diagrams of DCD Tier 2, Figures 9.2-1 and 9.2-2.

As a follow-up to RAI 9.2-6 S01, the staff generated RAI 9.2-24, which requested the following:

- Provide revised drawings in the DCD to include header temperature and pressure detectors.

- Include a more detailed description of how the PSWS detects gross leakage, and specify the instrumentation that is credited.

- DCD Tier 2, Section 9.2.1.5, indicates that, with one PSWS pump operating, the respective standby pump starts automatically upon detection of a low system pressure signal in that train, loss of electric power to the operating pump, or an operating pump trip signal. This section also indicates that starting a PSWS pump automatically opens a flow path through the RCCWS and TCCWS heat exchangers. However, no description is provided under the operation discussion in DCD Tier 2, Section 9.2.1.2, about these operating features, and there is no discussion about operation of the self-cleaning strainers.

As part of the March 19-20, 2009 audit and its review of the applicant's response to RAI 9.2-24, the staff reviewed the following:

- Available Phase 1 design drawings and PSWS proprietary drawings with regard to header temperature and pressure detectors.

- Drawings that provide monitoring of system flow in the MCR and can be used to assist in leak detection.

In response to RAI 9.2-24, the applicant noted that DCD Tier 2, Section 9.2.1.5, describes the operation of the motor operated self-cleaning strainers. The pump discharge self-cleaning strainers have remote manual override features for their automatic cleaning cycle. The pressure drop across the strainer is indicated in the MCR and a high-pressure drop is annunciated in the control room. During a LOPP, PSWS components, including the strainers and strainer blowdown valves will be powered from the two nonsafety-related on-site SDGs. This ensures that the PSWS pumps are available in case of a loss of power to one electrical train, while maintaining frequent backwashing to ensure minimal differential pressure across the strainers.

Based on its review, the staff finds that the drawings are adequate in the placement of PSWS instrumentation, including instruments used in the assistance of leak detection. The response to RAI 9.2-24 related to the self-cleaning strainer was adequate since it included operation of the strainers with backup power. In addition, the staff reviewed the PSWS pump trip based on the pump discharge valve failing to open and finds it to be adequate since it provides pump protection against a no-flow condition.

The response to RAI 9.2-24 also stated that the applicant will revise DCD Tier 2, Section 9.2.1.5, in Revision 6 to specify that a PSWS pump will trip if the pump discharge valve fails to open, thus ensuring that minimum flow conditions are maintained. The staff finds that the response to RAIs 9.2-6 and 9.2-24 as it relates to simplified diagrams is acceptable since the additional information added in the simplified diagrams of Figures 9.2-1 and 9.2-2 in DCD Revision 5 supports the PSWS RTNSS functions and is consistent with the more detailed design document reviewed during the audit. Based on the applicant's responses and DCD

changes, RAIs 9.2-6, and 9.2-24 as they relate to the simplified diagrams are resolved. The staff confirmed that Revision 6 of the DCD incorporated the RAI proposed changes.

9.2.1.3.2 COL Information

The staff reviewed DCD Tier 2, Section 13.5.3, COL Information Item 13.5-4-A, for plant operating procedure development. This section refers to Section 13.5.3.4, which in turn refers to the procedures as delineated in ANSI/ ANS-3.2, "Administrative Controls and Quality Assurance for the Operational Phase of Nuclear Power Plants." RG 1.33, "Quality Assurance Program (Operations)," endorses ANSI/ANS-3.2, and its Appendix A lists typical safety-related activities that should be covered by written procedures. Appendix A to RG 1.33 lists the service water system and component cooling water system. However, the PSWS and RCCWS in the ESBWR are not safety-related, so the generic COL information item cited above might not cover the nonsafety-related systems such as the PSWS and RCCWS in the ESBWR. In response to RAI 9.2-11 S04, the applicant revised DCD Tier 2, Section 13.5.2 to clarify that COL Information Item 13.5-2-A will include the water hammer procedures for the RTNSS systems. Therefore, the staff finds COL Information Item 13.5-2-A acceptable regarding procedure development for the PSWS.

The applicant identified one COL information item, COL Information Item 9.2.1-1-A, "Material Selection," specifically for the PSWS. Section 9.2.1.3.1 of this report discusses this item under Item D.5, with respect to GL 89-13. The staff considers this particular item, which addresses aspects of GL 89-13 related to material determinations, to be acceptable.

9.2.1.3.3 Availability Controls

As discussed in DCD Tier 2, Revision 9, Section 19A.8.1, regulatory oversight is applied to each system that is designated as RTNSS to ensure adequate reliability and availability to perform RTNSS functions. DCD Tier 2, Section 19A.8.1 also indicates that Maintenance Rule performance monitoring is specified for all RTNSS functions and that the ACM describes additional oversight for support systems. DCD Tier 2, Table 19A-2, identifies that the PSWS is a support system and that the PSWS 'Availability Controls' are the 'Maintenance Rule,' which means that Maintenance Rule performance monitoring addresses the availability of the PSWS rather than a specific ACM entry.

The PSWS is subject to the ACM through the systems it supports. DCD Tier 2, Table 19A-2 classifies the PSWS as a support system for the RCCWS, which is classified as a support system for the SDGs and for the NICWS. The NICWS supports the building HVAC, which supports the FAPCS. The FAPCS is the RTNSS system that is relied upon for active mitigation and the SDGS are support systems for the FAPCS. Of these systems, the ACM specifies ACs for the SDGs in AC 3.8.1, "Standby Diesel Generators – Operating," and AC 3.8.2, "Standby Diesel Generators – Shutdown;" and for the FAPCS in AC 3.7.2, "Fuel and Auxiliary Pools Cooling System (FAPCS) – Operating," and in AC 3.7.3, "Fuel and Auxiliary Pools Cooling System (FAPCS) – Shutdown." Therefore, the PSWS is a support system that is subject to the availability controls that are specified for the SDGs and FAPCS.

ACM Section 1.1, states that for the term "AVAILABLE-AVAILABILITY," a system, subsystem, train, division, component, or device shall be considered available or to have availability when it is capable of performing its specified risk informed function or functions and when all necessary attendant instrumentation, controls, normal or emergency electrical power, cooling and seal water, lubrication, and other auxiliary equipment that support operation of the system,

subsystem, train, division, component, or device with respect to perform its specified risk informed function or functions are also capable of performing their related support function or functions). Since the PSWS is a support system for the RCCWS, NICWS, FAPCS, and SDGs, if the PSWS were to become unavailable, then the systems it supports become unavailable and the applicable ACM action statements would apply.

Based on the above, the staff finds the ACs for the PSWS acceptable, since the PSWS is subject to the Maintenance Rule and is indirectly subject to the ACM as an RTNSS support system and the ACM definitions.

9.2.1.3.4 Inspections, Tests, Analyses, and Acceptance Criteria

DCD Tier 1, Revision 5, Section 2.12.7, provides ESBWR design certification information and ITAAC for the PSWS. Section 14.3.7 of this report evaluates DCD Tier 1 information for balance-of-plant (BOP) SSCs; evaluation of the Tier 1 information in this section is an extension of the evaluation provided in Section 14.3.7. This evaluation pertains to plant systems aspects of the proposed DCD Tier 1 information for the PSWS.

In RAI 14.3-69, the staff requested that the applicant revise DCD Tier 1, Section 2.12.7, to include a system description and system drawing, design commitment, and ITAAC scope for the PSWS. In response, the applicant recognized that the PSWS is an RTNSS system, but maintained its position that ITAAC are not required for the PSWS because the PSWS is not safety significant. The staff disagrees with the applicant’s determination because it is inconsistent with DCD Tier 2, Section 14.3.7.3, which indicates that RTNSS systems shall have Tier 1 inputs that include design descriptions and ITAAC. In DCD Tier 1, Revision 5, Section 2.12.7, the applicant provided a design description, ITAAC Table 2.12.7-1, and Figure 2.12.7-7, as requested in RAI 14.3-69. Accordingly, based on the above, the applicant’s responses, the RAI response and DCD changes, RAI 14.3-69 is resolved.

The staff reviewed the descriptive and other information provided in DCD Tier 1, Section 2.12.7, to confirm completeness and consistency with the plant design basis as described in DCD Tier 2, Section 9.2.1. The staff addressed the ITAAC details as part of the March 19-20, 2009, audit and RAI 9.2-24. The applicant’s response to the staff’s questions regarding the lack of specific details for the RTNSS Criterion C acceptance criteria was stated as follows:

> PSWS, RCCWS and NICWS provide supporting functions for FAPCS suppression pool cooling and low pressure injection modes, and thus meet RTNSS Criterion C. RTNSS C SSCs are assumed to be available at the time of the initiating event. Validation of these RTNSS functions is assured by Tier 1 ITAAC (Section 2.12.7, PSWS; Section 2.12.3, RCCWS; Section 2.12.5, NICWS) where testing of the PSWS/RCCWS /NICWS demonstrates flow to the RCCWS (nuclear island chillers, diesel generators and FAPCS island chillers, diesel generators and FAPCS). The ESBWR RTNSS Criterion C Cooling Water System ITAAC scope and detail differs from that associated with validation of RTNSS Criterion B functions. The ESBWR is designed so that safety-related passive systems are able to perform all safety functions for at least 72 hours, after initiation of a design basis event, without the need for active systems or operator actions. After 72 hours, nonsafety-related systems (RTNSS Criterion B) can be used to replenish the passive systems or to perform core cooling and containment integrity functions directly. RTNSS Criterion B ITAAC (e.g. FAPCS

section 2.6.2 Item 7 and FPS section 2.16.3 item 7) provides a greater assurance of function.

The staff finds that the RAI response is acceptable since the PSWS DCD Tier 1 information is adequate and reasonable based on the ESBWR graded approach for this RTNSS Criterion C, nonsafety-related system. For the importance of the PSWS, flow is verified to the RCCWS heat exchanges, as-built verifications are performed, selected controls from the MCR are verified, and PSWS system flow indication is available in the MCR. Accordingly, based on the above and the applicant's response, the ITAAC related aspects of RAI 9.2-24 are resolved.

9.2.1.3.5 Interface Requirements

In DCD Tier 1, Revision 3, Section 4.1, the applicant stated that the cooling tower and intake/discharge structure of the cooling water systems are not within the scope of the certified design. The cooling water systems provide the heat sink for power cycle waste heat. A specific design for this portion of the cooling water systems should be selected for any facility that has adopted the certified design. The plant-specific portion of the cooling water systems must meet the interface requirements defined in DCD Tier 1, Section 4.1. The interface requirements are necessary to support the post-72-hour cooling function of the PSWS. The PSWS is relied upon to remove 2.02×10^{7} megajoules (MJ) (1.92×10^{10} BTU) over a period of 7 days without active makeup. Consequently, verification of compliance with the interface requirements will be achieved by inspections, tests, and analyses that are similar to those provided for the certified design. The COL applicant referencing the certified design must develop these inspections, tests, and analyses, together with their associated acceptance criteria. The staff has reviewed this and agrees with the applicant that it is a COL Interface Requirement.

As previously discussed in Section 9.2.1.3.1.G.4, of this report, to provide minimum system flow, the PSWS design should ensure that the NPSH for the PSWS pumps is satisfied for all postulated conditions, including vortex formation considerations. The staff requested, in RAI 9.2-24, that the applicant revise DCD Tier 2, Section 9.2.1 to include this information and to establish appropriate COL information items and interface requirements. In the applicant's response to this RAI, the applicant provided a DCD Revision 6 markup. The change to DCD Tier 1, Section 4.1, stated that the PSWS pumps must have sufficient available NPSH at the pump suction location for the lowest probable water level of the heat sink. In response to RAI 9.2-23 S01, the applicant revised the description of the interface between standard plant design for the ESBWR and the conceptual design to be addressed by COL applicants to include consideration of minimum NPSH under worst case conditions. In addition, DCD Tier 2, Section 14.2.8.1.51, describes a series of individual component and integrated system tests to demonstrate acceptable pump suction under the most limiting design flow conditions.

The staff finds that the responses to RAIs 9.2-23 and 9.2-24 regarding PSWS pump NPSH are acceptable since the applicant clarified how sufficient NPSH is assured. The applicant also added DCD Tier 1 interface requirements and clarified how testing in accordance with Section 14.2.8.1.51 addresses NPSH under the most limiting design flow conditions. Based on the RAI responses and DCD changes, the interface requirements aspect of RAI 9.2-24 is resolved. The staff confirmed that Revision 6 of the DCD incorporated these RAI proposed changes. Based on the above, the staff finds the PSWS interface requirements acceptable.

9.2.1.3.6 Initial Test Program

Section 14.2 of this report evaluates the initial test program for ESBWR. The evaluation of the PSWS initial test program in this section is an extension of the evaluation provided in Section 14.2 of this report.

DCD Tier 2, Revision 9, Section 14.2.8.1.51, describes the pre-operational test program for the PSWS. The staff finds the objective of the PSWS pre-operational test program to be appropriate since its purpose is to verify proper operation of the PSWS and its ability to supply design quantities of cooling water to the RCCWS and TCCWS heat exchangers. While the test specifications are written in general terms to address the considerations that apply to PSWS, the approach for this nonsafety-related system is considered to be acceptable because the COL applicant will develop test procedures in accordance with COL Information Item 14.2-3-A.

During the review of DCD Tier 2, Revision 5, the staff determined that additional information and specificity was necessary in some respects and requested, in RAI 9.2-24, that the applicant revise DCD Tier 2, Section 14.2.8.1.51, to address the testing of automatic air release and vacuum valves. In response to RAI 9.2-24, the applicant provided a DCD mark-up of DCD Tier 2, Section 14.2.8.1.51, with the addition of testing of the automatic air release and vacuum valves. The staff finds that the applicant's response is acceptable since the addition of testing of the automatic air release and vacuum valves ensures a complete scope of testing. Based on the above and the applicant's response, the initial test program aspects of RAI 9.2-24 are resolved. The staff confirmed that Revision 6 of the DCD incorporated this RAI proposed change.

9.2.1.4 *Conclusion*

The staff finds that the PSWS complies with the requirements of GDC 2, 4, 44, 45 and 46. Because the ESBWR design is a single unit, GDC 5 is not applicable. The staff also finds that the design of the PSWS conforms to established NRC policies with respect to its RTNSS Criterion C function.

9.2.2 Reactor Component Cooling Water System

9.2.2.1 *Regulatory Criteria*

The staff reviewed the RCCWS based on guidance provided in SRP Section 9.2.2, Revision 4. The SRP guidance is used to the extent that it pertains to system functionality and reliability considerations. Staff acceptance of the RCCWS design and supporting information is based upon conformance with the following:

- GDC 2, as it relates to the capability of the design to maintain and perform its safety function following an earthquake
- GDC 4, as it relates to the dynamic effects associated with water hammer
- GDC 5, as it relates to the capability of shared systems and components important to safety to perform required safety functions
- GDC 44, as it relates to transferring heat from SSCs important to safety to a heat sink

- GDC 45, as it relates to the design provisions to permit inspection of components and equipment

- GDC 46, as it relates to the design provisions to permit operational testing of components and equipment

The RCCWS is a nonsafety-related system; however, the system provides defense-in-depth for the ESBWR passive plant design. In addition to the SRP guidance, the NRC staff's evaluation of defense-in-depth systems also focuses on confirming that design, performance, and reliability considerations are satisfied consistent with the NRC policies that are referred to in Chapter 22, of this report; on confirming that failure of defense-in-depth systems and components will not adversely impact safety-related SSCs; on confirming that ACs are established as appropriate; and on confirming that proposed ITAAC and initial test program specifications are adequate.

9.2.2.2 *Summary of Technical Information*

DCD Tier 2, Revision 9, Section 9.2.2, describes the RCCWS. The system does not perform any safety-related function, and there is no interface with any safety-related component. The system is designed to provide cooling water to plant auxiliary equipment during start-up, hot standby, and plant cooldown.

The RCCWS consists of two 100-percent-capacity independent and redundant trains. RCCWS cooling water is continuously circulated through various auxiliary equipment heat exchangers and rejects the heat to the PSWS. DCD Tier 2, Table 3.2.1, indicates that part of the RCCWS (P21) is a nonsafety-related system located in the RB and is designated as "Quality Group D" and seismic Category II. Other portions of the RCCWS are located in the turbine building (TB), RB, FB, and electrical building (EB) and are designated as "Quality Group D" and nonseismic. The RCCWS has RTNSS functions.

In the event of a LOPP, the RCCWS supports the FAPCS and the RWCU/SDC in bringing the plant to cold-shutdown condition in 36 hours assuming the most limiting single active failure.

In addition, the RCCWS provides cooling water to the chilled water system (CWS) nuclear island chiller-condenser and SDGs. DCD Tier 2, Tables 9.2-3 and 9.2-4, tabulate the RCCWS design heat loads and component design characteristics.

While the RCCWS is a nonsafety-related system, it performs defense-in-depth functions and is also subject to RTNSS as described in DCD Tier 2, Appendix 19A. As stated in DCD Tier 2, Section 19A.4.2, in order to address uncertainties in the performance of passive systems, an active system with the capability to provide backup functions is added to the scope of RTNSS. The portions of the FAPCS that provide low pressure injection and SPC are added to the scope for RTNSS Criterion C. Of the support systems needed for FAPCS, RCCWS is used to cool the FAPCS.

9.2.2.3 *Staff Evaluation*

The staff's review of the RCCWS is based on guidance found in SRP Section 9.2.2 and applicable regulations such as GDC 2, 4, 5, 44, 45 and 46. The RCCWS for the ESBWR differs from that of the traditional BWR designs in that the ESBWR RCCWS is a nonsafety-related system because the RCCWS removes heat only from the CWS, RWCU/SDC, FAPCS, and

SDGs, which are not safety-related systems. Therefore, the portions of SRP Section 9.2.2 that apply to safety-related systems do not apply to the RCCWS.

9.2.2.3.1 System Design Considerations

As previously stated, the RCCWS has RTNSS functions. The RCCWS, which is a nonsafety-related active system, should be highly reliable and capable of achieving and maintaining cold shutdown conditions. In addition, no single failure of this system should result in inability to terminate use of the passive safety-related systems and achieve cold shutdown in accordance with GDC 44. Nonsafety-related systems, including the RCCWS, should be capable of cooling the plant to Mode 5 conditions within 36 hours in order to satisfy ESBWR TS requirements. Numerous Technical Specification sections require Mode 5 entry (i.e. placing the plant in cold shutdown). In support of these TS requirements, DCD Tier 2, Section 9.2.2.2, states that the RCCWS supports the FAPCS and the RWCU/SDC in bringing the plant to cold shutdown condition in 36 hours, if necessary, assuming the most limiting single active failure. Nonsafety-related systems that are designated as RTNSS (including their support systems) are subject to enhanced design, quality, reliability, and availability provisions and are relied upon for performing functions as discussed in DCD Tier 2, Appendix 19A. Sufficient information needs to be included in Tier 1 and Tier 2 of the DCD to demonstrate that these systems are adequate for achieving and maintaining cold shutdown conditions (i.e., cooldown from Mode 4 to Mode 5), performing RTNSS functions, and satisfying applicable design considerations.

On March 19-20, 2009, the staff conducted a regulatory audit of the supporting information for DCD Tier 2, Section 9.2, including the PSWS (Section 9.2.1), RCCWS (Section 9.2.2), and NICWS (Section 9.2.7). The audit primarily focused on the review of these systems with regard to the RTNSS functions and the ability to support cold shutdown operations. A summary of the audit, including participants and audit activities may be found in the ADAMS at Accession Number ML101250439. This audit is referred to several times throughout the remainder of this section.

A. Reactor Component Cooling Water System Classification and Quality Assurance Provisions

DCD Tier 2, Section 3.2, specifies the classification of SSCs based on safety importance and other considerations. Section 3.2 of this report provides the staff’s evaluation of the specified classification designations. This section of the staff’s evaluation is to confirm that the appropriate classification designations are specified for the RCCWS to be consistent with the approach described in DCD Tier 2, Section 3.2, and that the designations properly reflect the regulatory oversight provisions that pertain to the RCCWS (RTNSS Criterion C), as discussed in DCD Tier 2, Section 19A.8. The staff reviewed simplified drawings, DCD Tier 2, Figures 9.2-2a and 9.2.2b, and confirmed that the classification designations on the drawings are consistent with those that are listed for RCCWS in DCD Tier 2, Table 3.2-1. In particular, the following classification designations are specified in Table 3.2-1 for the RCCWS:

- The RCCWS is designated Safety Class N which is used for nonsafety-related applications. Because the RCCWS does not perform any safety-related functions, the staff concludes the N designation to be appropriate.

- The RCCWS is designated Quality Group D. As discussed in DCD Tier 2, Section 3.2.4, this quality group generally applies to nonsafety-related SSCs that satisfy specified industry codes and design standards and are subject to one or more significant licensing

requirements or commitments. The staff concludes that this is an appropriate quality group designation since the RCCWS does not perform a safety-related function and does not interface with any safety-related component.

- Part of the RCCWS, located in the RB and FB, is designated as seismic Category II. SSCs that perform no safety-related function, but whose structural failure or interaction could degrade the functioning of a seismic Category I item to an unacceptable level of safety or could result in incapacitating injury to occupants of the MCR, are designated seismic Category II. These items are designed to structurally withstand the effects of an SSE. Other portions of the RCCWS are located in the TB, RB, FB and EB and are designated as nonseismic. The staff concludes that the RCCWS has the appropriate seismic classifications since the RCCWS does not perform a safety-related function and does not interface with any safety-related component.

- Revision 6 of the DCD specifies QA Requirement S for the RCCWS, as stated in the applicant's response to RAI 3.2-6 S02. Based on the RAI response, RTNSS components and systems that were identified in Revision 5 of the DCD as QA Requirement E are to be changed to QA Requirement S in Revision 6. QA Requirement S has special provisions that apply during the design and procurement specification preparation processes in accordance with procedures that will be established. The staff concludes that this is an appropriate QA group designation since the RCCWS does not perform a safety-related function and does not interface with any safety-related component; however, the RCCWS has RTNSS functions that are assured by applying the defense in depth principles of redundancy and physical separation to ensure adequate reliability and availability. In addition, the staff concludes that Revision 6 of the DCD incorporated this RAI proposed change and finds that this change is acceptable.

B. GDC 2

The RCCWS is a nonsafety-related system and is routed in the RB (seismic Category I and II building), FB (seismic Category I and II building), TB (seismic Category II building), and EB (nonseismic building). SRP Section 9.2.2 indicates that the requirements of GDC 2 can be met for a nonsafety-related system based on meeting Regulatory Position C.2 of RG 1.29, regarding nonsafety-related systems. In RAIs 9.2-12, and 9.2-12 S01, the staff requested that the applicant demonstrate that the RCCWS (among other water systems) satisfies Regulatory Position C.2 of RG 1.29. In response, the applicant explained that the RCCWS does not have any piping in the control room or interface with any safety-related components. The RCCWS is under the RTNSS process to provide cooling functions following an SSE. It will be designed to the seismic requirements specified in DCD Tier 2, Appendix 19A and Section 3.2. Chapter 22 of this report provides the staff's evaluation of the RTNSS systems. The staff reviewed the above RAI responses and DCD Tier 2, Revision 5, Section 9.2.2.1. Based on the above, the staff finds that the RCCWS meets the guidance of those portions of Regulatory Position C.2 of RG 1.29 regarding nonsafety-related systems because the failure of the nonsafety-related portions of the system does not impact any safety-related SSCs or incapacitate the control room occupants. Accordingly, based on the above and the applicant's response, the portions of RAI 9.2-12 relating to the RCCWS are resolved.

C. GDC 4

SRP Section 9.2.2 provides the guidance to review the RCCWS against GDC 4, as it relates to the dynamic effects associated with water hammer. As stated in DCD Revision 9, Section

9.2.2.1, the effects of missiles, jet impingement, pipe whipping, and discharged fluids are addressed by the following design considerations:

- Pipe routing.
- Piping design consideration, such as material section, pipe size, and schedule.
- Protective barrier as necessary.
- Appropriate supports and restraints.

March 19-20, 2009, audit discussed water hammer considerations, which the applicant addressed in responses to RAIs 9.2-11, 9.2-11-S01, 9.2-11-S02, 9.2-11-S03, 9.2-11-S04, and RAI 9.2-24. In these RAIs, the staff asked the applicant to discuss the potential for water hammer, as well as the operating and maintenance procedures for avoiding water hammer in the PSWS and RCCWS. RAI 9.2-11 was being tracked as an open item in the SER with open items. In response, the applicant listed the following provisions to mitigate water hammer:

- Minimize high points in the system.

- Provide for venting at all high points.

- Have the COL applicant address procedural requirements ensuring proper line filling before system operation and following maintenance operations.

- Keep valve actuation times slow enough to prevent water hammer.

- Use check valves at pump discharge to prevent backflow into the pump.

- Ensure that the surge tank location (high point of the system) provides constant pump suction.

Because the RCCWS is a closed-loop system, the mechanism and flow path for drain down of risers is not available for a properly filled and vented system. Proper system engineering design of closed-loop systems precludes system pressure from falling below vapor pressure of the fluid being transported. Surge tanks are also used in accordance with DCD Tier 2, Section 9.2.2.2, within the RCCWS, which provide a constant pump suction head and allow for thermal expansion of the RCCWS inventory. In addition, in DCD Tier 2, Section 13.5.2, the applicant clarified that elements of ANSI/ANS-3.2-1994; R1999, addressing water hammer will be applied in the development of procedures for RTNSS systems.

Based on the staff's review of the applicant's responses to RAI 9.2-11, and its supplements, and RAI 9.2-24, the staff concludes that the applicant has adequately addressed water hammer since the RCCWS design incorporated water hammer mitigation features and components and operational procedures are to be developed addressing water hammer concerns for the RTNSS systems as part of COL Information Item 13.5-2-A. Accordingly, based on the above and the applicant's response, RAIs 9.2-11 and 9.2-24 as they relate to water hammer are resolved. Based on the above, the staff finds that the RCCWS meets the requirements of GDC 4 as it relates to water hammer, in accordance with the guidance of SRP Section 9.2.2

D. GDC 5, 44, 45, and 46

The ESBWR design is a single-unit station, and the requirements of GDC 5 are not applicable to the single-unit design.

Based on the requirements of GDC 44, 45, and 46 and the guidance in SRP Section 9.2.2, the staff reviewed the RCCWS against GDC 44, 45, and 46, assuming that the RCCWS is capable of removing heat from SSCs important to safety to a heat sink under normal operating and accident conditions and that design provisions are available for inspection and operational testing.

As stated in DCD Tier 2, Revision 1, Table 1.9-9, and Section 9.2.2, the applicant determined that GDC 44, 45, and 46 do not apply to the PSWS and RCCWS. In RAI 9.2-7, RAI 9.2-7 S01, and RAI 9.2-7 S02, the staff questioned this determination. In response to these RAIs, the applicant revised DCD Tier 2, Revision 5, Table 1.9-9 and Section 9.2.2.1, to address conformance to GDC 44, 45, and 46. The applicant also stated that the RCCWS meets the intent of certain acceptance criteria of GDC 44, 45, and 46 because the design of the RCCWS includes the following provisions:

- Capability to transfer heat loads from SSCs to a heat sink under normal and accident conditions

- Component redundancy so the system will remain functional assuming a single failure coincident with a LOOP

- Capability to isolate components or piping so system function will not be compromised

- Design provisions to permit inspection and operational testing of components and equipment

The staff believes that those portions of the GDC 44 requirements that apply to the heat removal function under normal operating conditions apply to the RCCWS. The PSWS and RCCWS are nonsafety-related.

The staff reviewed the RCCWS based on the designed heat removal capability; component redundancy and single failure design; and plant TS shutdown cooling requirements, testing and inspection requirements, as described in DCD Tier 2, Section 9.2.2. The staff finds that the RCCWS satisfies GDC 44, 45, and 46 with respect to its normal operation function. However, in a DBA, decay heat is transferred to the ICS/PCCS pools. The portions of the GDC 44 requirements that apply to a safety-related system to remove decay heat following an accident do not apply to the RCCWS. The staff finds that the design of the RCCWS satisfies the applicable portions of GDC 44, 45, and 46 based on the above review. In addition, the staff finds that the response to RAI 9.2-7 is acceptable since the applicant clarified conformance of the RCCWS to GDC 44, 45, and 46 and described how this is achieved. Based on the above and the applicant’s response, RAI 9.2-7 as it relates to the RCCWS is resolved. The PSWS design attributes, including system capability, reliability, heat transfer, pump NPSH, operating experiences, testing, and instrumentation and controls (which are related to the applicable GDCs), are further addressed below for the PSWS RTNSS and cold shutdown functions.

E. Minimization of Contamination; 10 CFR 20.1406 and Radiation Monitoring

10 CFR 52.47(a)(6) and 10 CFR 20.1406 require applicants for standard plant design certifications to describe how facility design and procedures for operation will minimize contamination of the facility and the environment. The staff’s review criteria (SRP Section 9.2.2, Paragraph III.4.C) specify that provisions should be included to detect radioactive leakage or contamination from one system to another.

In RAIs 9.2-13, 9.2-13 S01, and 9.2-13 S02, the staff asked the applicant to describe design provisions to detect RCCWS leakage of radioactive or chemical contamination and the locations of radioactivity and conductivity monitors. RAI 9.2-13 was being tracked as an open item in the SER with open items. In response, the applicant stated that intersystem leakage in the RCCWS is monitored through three methods: radiation monitoring (see DCD Tier 2, Sections 9.2.2.5 and 11.5.3.2.7), RCCWS flow rate, and high level alarm from the head tank (see DCD Tier 2, Section 9.2.2.5).

First, the RCCWS has radiation monitoring in each cooling water train to detect intersystem radiation leakage into the respective RCCWS loop. Second, the flow rate of RCCWS water is constantly monitored throughout the system to provide detection of leakage to or from the RCCWS. In addition, other monitored system parameters can be used to detect intersystem leakage. Low pump discharge header pressure, high or low head tank level, and excessive makeup valve opening time are alarmed or annunciated in the MCR. The third method available to detect RCCWS leakage is the high level alarm from the head tank. A high level alarm would indicate a malfunction. The malfunction could be intersystem leakage, such as, in-leakage from one of the RCCWS cooling loads or a leaking makeup water valve. Grab sampling can be used in identifying the source of in-leakage.

In addition, the staff discussed RCCWS radiation monitoring and system gross leakage at the March 19-20, 2009, audit, which involved RAI 9.2-24. In the response to RAI 9.2-24, the applicant expanded on its previous responses to RAI 9.2-13 and its supplements. RCCWS radiation monitors are provided for monitoring radiation levels and alerting the plant operator of abnormal radiation levels. The minimum amount of monitoring is at two points in each train; after the RWCU/SDC heat exchangers to detect potential reactor coolant leakage and at the pump suction return line upstream of the cross-tie header, but downstream of the heat exchanger hot leg connections.

The RCCWS is designed such that a major line break is automatically detected through the process monitoring of flow rates. This is accomplished by monitoring flow rates at key points in the piping network and confirming that the flow rates are balanced such that the inlet and outlet flows in the given section of piping are equal. Upon receipt of an unbalanced flow in a major supply or return line, the cooling water trains will be separated and the damaged train shut down either manually or automatically. Inconsistent RCCWS flow rates based on upstream and downstream flow values that are greater than or equal to the makeup water system (MWS) instrumentation flow rate will generate an unbalanced flow signal. These flow rates will also be used by the RCCWS to determine if an automatic train separation is necessary.

During the audit, the staff noted that DCD Tier 2, Figure 9.2-2b, illustrates a radiation detector downstream of the A Train RWCU/SDC heat exchangers; however, a radiation detector was not shown downstream of B Train RWCU/SDC heat exchangers. The applicant indicated that DCD Revision 6 will correct this omission and add the radiation detector downstream of the B Train RWCU/SDC heat exchangers. The staff confirmed that Revision 6 of the DCD incorporated this RAI proposed change.

The RCCWS surge tank levels are used to monitor losses of cooling water and detect intersystem leakage intrusions into the RCCWS. The level transmitters in the surge tank standpipes, in combination with low-low surge tank level, automatically initiate the train shut down valves.

As a follow-up to RAI 9.2-24, the staff requested, in RAI 9.2-27, that the applicant address the requirements of 10 CFR 20.1406 regarding the RCCWS and how components susceptible to leakage can be isolated. In response, the applicant explained that the RCCWS has radiation monitors at the discharge of the RWCU/SDC heat exchangers to alert the plant operator of abnormal radiation levels. In addition, RCCWS surge tank levels are used to monitor losses of cooling water and detect intersystem leakage intrusions into the RCCWS. The level transmitters in the surge tank standpipes in combination with low-low surge tank level automatically initiate a train shut down. A train shutdown signal will trip off all pumps in the train and close all isolation, bypass, and flow control valves. This will isolate any leaking component and minimize train cross contamination.

In addition, DCD Tier 2, Table 12.3-18, which addresses RG 4.21 design objectives and applicable DCD section information, describes similar provisions related to the RCCWS for the following objectives:

- Minimizing leaks and spills (Design Objective 1)

- Provide for adequate leak detection capability to provide detection of leakage for any SSC which has the potential for leakage (Design Objective 2)

- Leak detection methods and early detection of leaks to avoid release of contamination from undetected leaks and to minimize contamination of the environment (Design Objective 3)

- Decreasing the spread of contaminant from the source (Design Objective 4)

The staff finds that these design provisions for the RCCWS meet the requirements of 10 CFR 20.1406 and conforms to the guidelines of RG 4.21. Sections 12.4 and 12.7 of this report further address the ESBWR design in accordance with 10 CFR 20.1406. The staff finds that the responses to RAIs 9.2-13, 9.2-24, and 9.2-27, as they relate to leakage detection, are acceptable since the applicant clarified the leakage detection and monitoring provision for the RCCWS. Based on the above and the applicant's response, RAIs 9.2-13, 9.2-24, and 9.2-27, as they relate to leakage detection, are resolved.

F. Protection from Probable Hazards

In accordance with the policies referred to in Chapter 22 of this report, SSCs that are classified as RTNSS should be protected from the more probable hazards that exist. As previously discussed, RCCWS is classified as RTNSS Criterion C. DCD Tier 2, Section 19A.8.3, indicates that RTNSS Criterion C systems incorporate the defense-in-depth principles of redundancy and physical separation to ensure adequate reliability and availability. DCD Tier 2, Section 19A.8.3, also indicates that RTNSS Criterion C systems and structures meet design standards to withstand wind and missiles generated from Category 5 hurricanes and that non-RTNSS systems that can adversely interact with RTNSS Criterion C systems are designed to the same seismic standards as the affected RTNSS system. Additionally, DCD Tier 2, Section 19A.8.3 indicates that RTNSS Criterion C equipment is qualified to IEEE Standard 344-1987, to demonstrate structural integrity. Also, DCD Tier 2, Section 19A.8.3, describes design criteria for use with the seismic standards of International Building Code – 2003 (IBC-2003) for the seismic design of RTNSS C systems and components.

As stated in the applicant's response to RAI 9.2-24, the PSWS and RCCWS support PIP and defense-in-depth. DCD Tier 2, Section 9.2.2.2, explains that, in the event of a LOPP, the

RCCWS supports FAPCS and the RWCU/SDC in bringing the plant to cold shutdown condition in 36 hours, assuming the most limiting single active failure. Because the PSWS and RCCWS cooling water systems are also significant contributors to plant availability and plant investment protection, the ESBWR design is focused on ensuring that these systems are available and reliable.

In summary, the PSWS and RCCWS are support systems to the FAPCS and are only included as augmented systems to address uncertainties in the defense in depth role of the FAPCS in providing a backup source of lower pressure injection and SPC. RTNSS Criterion C systems are not designed to the level of RTNSS Criterion B; however, RTNSS Criterion C systems are designed to the seismic standards of IBC-2003 consistent with the above SSE ground motion, which is equal to two-thirds of the certified seismic design spectra. The staff finds this graded design approach is acceptable considering the design function of the PSWS under the regulatory criteria for this nonsafety system.

G. RCCWS Capability and Reliability

In RAI 9.2-24, the staff requested the applicant to specifically address information concerning the RCCWS functions that are subject to RTNSS, focusing on RCCWS capability and reliability. The RAI included the following key points:

- The most limiting conditions upon which the RCCWS design is based with the amount of excess margin built into the design
- Clarification in the DCD descriptions, drawings and tables (to include valves, cross-tie connections between trains, instrumentation logic and installed instruments)
- RCCWS pump design to include pump recirculation protection, vortex and NPSH
- Radiation monitoring and gross leakage detection
- RCCWS cooldown requirements (24 hours and 36 hours); and system alignment to support cooldown
- RCCWS water hammer consideration
- RCCWS failure modes and effects
- RCCWS component testing and component reliability

To resolve this RAI, the staff audited supporting information for the RCCWS on March 19 and 20, 2009, as discussed above. The response to RAI 9.2-24 addresses both the RAI and the audit findings. The remainder of this section discusses the results of the audit and the RAI response.

(1) <u>Descriptive Information and Flow Considerations</u>

The staff reviewed the DCD Tier 2, Section 9.2.2 RCCWS description and drawings to confirm that the design bases, flow paths, and components have been identified and described in sufficient detail to enable a complete understanding of the system design and operation. The

staff found that additional information was needed in this regard and requested, in RAI 9.2-24, that the applicant revise DCD Tier 2, Section 9.2.2 to address the following considerations:

- Nominal pipe sizes and system flow rates
- Valve design, including a discussion of valve hard seat materials
- Pump protection and system strainers

During the March 19-20, 2009, audit and in its RAI response, the applicant addressed all of the above items. At the audit, staff reviewed the supplied system diagrams for piping sizes and adverse system velocities. The staff finds that the normal operating system velocities acceptable since system flow velocities are enveloped by the system velocity design limits. In addition, pipe size and fluid velocity were based on ensuring the RCCWS can be filled in less than 6 hours using the makeup water fill connection. The RCCWS surge tank makeup pipe sizing ensures that the system is capable of maintaining the surge tank level with a relief valve stuck open. It was pointed out at the audit that under certain conditions the RCCWS pipe sizing was based on failures of the flow paths through various heat exchangers during RCCWS cooling to the FAPCS for cooling of the suppression pool to mitigate boiling. Accordingly, the FAPCS fuel pool heat exchanger pipeline will be sized based on the sum of the normal FAPCS fuel pool heat exchanger flow plus half of the RWCU/SDC flow. The applicant viewed the increased velocities associated with failures of the RCCWS flow paths as a highly unlikely event; however, the potential higher system velocities will be allowed to exceed the recommended velocities.

The RAI response stated that valves are usually provided with hard seats to withstand erosion associated with water quality issues. Since RCCWS water is treated with corrosion inhibitors to minimize the corrosion of the RCCWS piping and components, specifying hard seats for RCCWS valves are not necessary.

The audit, as well as the response to RAI 9.2-24, discussed the RCCWS strainers. The RCCWS is a closed system with clean de-mineralized water treated with corrosion inhibitors to minimize the corrosion of the RCCWS piping and components. Therefore, RCCWS pumps are not susceptible to failure from large debris during normal operation. The RCCWS pumps are provided with temporary suction strainers designed to remove post-construction corrosion products and other debris that may have accumulated in the piping system during construction. These strainers are removed after initial plant startup.

The staff finds that the normal RCCWS velocities were adequately addressed and discussed. The staff finds them acceptable since the most limited piping velocities were approximately 4.6 m per second (15 ft per second) or less. In the staff's experience and in accordance with general engineering practice, piping velocities between 1.2 and 4.6 m per second (4-15 ft per second) are reasonable, thus long term internal pipe wear is expected to be minimal. For the condition of potential higher system velocities above recommended velocities, the staff finds this is an unlikely event associated with failures of RCCWS flow paths considering of all the RCCWS design features and the designation of the QA measures for this system as RTNSS Criterion C. The remaining items noted above were reviewed by the staff as part of the RAI response. The staff finds that the response to RAI 9.2-24 regarding RCCWS descriptive information and flow considerations is acceptable since the applicant clarified the basis for the design parameters included in the DCD. Based on the above and the applicant's response, the flow consideration aspects of RAI 9.2-24 are resolved.

(2) Heat Transfer

In RAI 9.2-20, the staff asked the applicant to explain an inconsistency in DCD Tier 2, Revision 4, Table 9.2-3, regarding the CWS heat load of 12.3 MW (42.0 million BTU [MBTU]/h) applicable for train A only. In response, the applicant stated that it would add a note DCD Tier 2, Revision 5, Table 9.2-3, to clarify for the 12.3 MW (42.0 MBTU/h) CWS heat load that the "total CWS heat load shown is applicable to Train A or Train B, or shared between the two trains." The staff finds that the RAI response is acceptable since the applicant added a note to the DCD to clarify the potential inconsistency identified in the RAI. Based on the above and the applicant's response, RAI 9.2-20 is resolved.

The staff reviewed the RCCWS description in DCD Tier 2, Section 9.2.2, as well as applicable DCD tables to confirm that the heat transfer and flow capabilities are adequately specified and that the bases for these values are fully explained. DCD Tier 2, Table 9.2-3, lists the RCCWS heat loads for various operating modes and indicates that the most limiting case is a single train failure cooldown.

The staff determined that it needed additional information in this regard and requested, in RAI 9.2-24, that the applicant revise DCD Tier 2, Section 9.2.2, to address heat transfer and the amount of margin to include uncertainties for wear and aging effects.

The applicant's response to RAI 9.2-24 stated that a LOPP cooldown with single train failure is the most limiting system heat removal design condition for the RCCWS (73.5 MW [250 MBTU/h]). This transient mode occurs when a LOPP and a single train failure occur concurrently. Similar to the single train failure transient, only one train is in operation and all heat loads are dissipated using the three RCCWS heat exchangers and three RCCWS pumps on the active RCCWS train. This mode of operation provides sufficient cooling capacity to bring the plant to cold shutdown condition within 36 hours. The most limiting condition for the RCCWS heat exchanger design is a single train failure cooldown without a LOPP, which has a design heat load of 58.5 MW (200 MBTU/h) divided between two heat exchangers. Each RCCWS heat exchanger is designed for 30.6 MW (104 MBTU/h).

Based on the above, the staff finds that for the two bounding conditions noted above, there is sufficient design margin between the capacity of the RCCWS heat exchangers and the maximum heat loads. In addition, for support of RTNSS only (FAPCS, CWS, and SDG), the heat loads are bounded by design margin between the heat exchanger capacity and the heat loads. Based on the staff's review of the RAI, the staff finds the heat transfer capability of the RCCWS of sufficient margin to support normal plant cooldown, single train failure cooldown, LOPP operation and RTNSS support. The response to RAI 9.2-24 provided a DCD markup regarding the clarification related to the RCCWS heat loads, and added two notes to Table 9.2-3 defining that normal shutdown is within 24 hours and that design limiting condition cooldown is within 36 hours. In addition, the applicant's response to RAI 9.1-20 S04, provided a markup to the FAPCS heat loads in DCD Tier 2, Table 9.2-3, which were reduced by 1.3 MW (4.5 MBTU/h). The staff confirmed that Revision 6 of the DCD incorporated these RAIs proposed changes. The staff finds that the response to RAI 9.2-24 regarding heat transfer is acceptable, with the clarification provided by the response to RAI 9.1-20, since the applicant clarified the basis for the heat loads in the DCD and added corresponding clarifications to DCD tables identifying RCCWS heat loads and component design characteristics. Based on the above and the applicant's response, the heat transfer aspects of RAI 9.2-24 are resolved.

(3) Single Failure and Backup Power Considerations

As described in Section 9.2.2, the RCCWS consists of two fully redundant (train A and train B), 100-percent capacity trains, with each train consisting of a total of three pumps and three RCCWS heat exchangers cooled by the PSWS. The pumps in each train are powered from separate buses. During a LOPP, the pumps are powered from the two nonsafety-related SDGs. Each RCCWS train consists of parallel pumps, parallel heat exchangers, one surge tank, connecting piping, and instrumentation. Both trains share a chemical addition tank. The trains are normally connected by crosstie piping during operation for flexibility, but may be isolated for individual train operation or maintenance of either train.

Although the two trains are normally cross-connected via AOVs, they can be split out if necessary from the control room. The staff determined that clarification was needed for the case in which offsite power is not available and requested in RAI 9.2-24 that the applicant revise DCD Section 9.2.2 to address single failure.

The design flow rate at the RCCWS pump rated head is specified to ensure that the pump will not operate below 85 percent or above 125 percent of its best efficiency point. RCCWS cooling water train supply valves (direct current [dc] backed, motor operated) automatically close upon a LOPP to prevent RCCWS pump runout and ensure sufficient cooling for the SDGs. These valves are opened after the SDGs are running as part of the load sequencing. As part of the response to RAI 9.2-24, the applicant provided a Revision 6 markup of changes to DCD Tier 2, Section 9.2.2.2, related to the dc MOVs. The staff confirmed that Revision 6 of the DCD incorporated this RAI proposed change. The staff finds this change to be acceptable since the applicant clarified the signals used to close the dc MOVs.

When the RCCWS train cross-tie valves are open, any four pumps and heat exchangers can be used. When the RCCWS train cross-tie valves are closed, two pumps and heat exchangers must be used on each train, with any two of the three pumps and heat exchangers on the train. Upon a train separation signal, opening the bypass line valves for the CWS, FAPCS, and RWCU/SDC is needed to keep the RCCWS pumps within their operating ranges. If the bypass line for the RWCU/SDC heat exchanger fails, then the isolation valves for that heat exchanger will automatically open to maintain an adequate flow path. Each flow path to all interfacing system heat exchangers is designed to have flow balancing features that may include fixed plate orifices or control or manual valves.

AOVs are located at the discharge of the RCCWS heat exchangers, RCCWS heat exchanger bypass line and RCCWS cross-tie line (suction and discharge.) In addition, AOVs are used for RCCWS surge tank level control, SDG cooling water return, and RCCWS RWCU/SDC heat exchanger bypass and discharge flow control valves. The RCCWS heat exchanger flow control AOVs are normally open, fail open valves. RCCWS heat exchanger bypass valves are fail closed valves upon loss of control signal or loss of power to the control signal. The RCCWS air-operated heat exchanger bypass and flow control valves function in coordination to regulate the RCCWS supply temperature. The position of these valves is regulated by the redundant discharge temperature elements. The valves are programmed such that when one valve opens, the other valve will close.

The RCCWS cross-tie valves are air-operated block valves and are automatically and manually opened and closed by the N-DCIS in the MCR. The valves are normally open and automatically close upon a train separation event and fail close. Two automatic train separation signals are used to close the cross-tie valves, which are the detection of unbalanced flow, and a LOPP

event. Manually initiated train separations also close the cross-tie valves. As part of the RAI response, the applicant provided a Revision 6 markup of changes to DCD Tier 2, Section 9.2.2.2, related to separation signals. The staff concluded that Revision 6 of the DCD incorporated this RAI proposed change, and the staff finds that this change is acceptable since the applicant adequately described the details of the RCCWS cross-tie valves and train separation signals and added to the information to DCD Tier 2, Section 9.2.2.1.

The RCCWS surge tank level is controlled by air-operated block valves. The valves are automatically opened and closed and can be manually controlled by the MCR N-DCIS. The block valve is opened when the RCCWS surge tank level drops to a predetermined low level. The block valve closes when the RCCWS surge tank level rises to a predetermined high level. A manual valve provides a backup source of makeup from the FPS. Extended makeup water supply additions indicate a leak in the RCCWS; the cooling water trains should be separated, and the damaged train repaired. The separation of trains because of extended makeup water supply addition is a manually initiated event. The RCCWS surge tank makeup water inlet block valves fail close.

The RCCWS SDG cooling water return valves are air-operated block valves, and are automatically and manually opened and closed by the MCR N-DCIS. The valves normally are closed and will automatically open upon a LOPP. The valves fail open. The RCCWS cooling water flow rate through the RWCU/SDC heat exchangers is regulated with bypass and discharge air-operated flow control valves. The RCCWS diesel generator cooling water return valves are controlled using RWCU/SDC discharge temperature process data and not the RCCWS. Control of these valves by the RWCU/SDC will prevent overcooling of the reactor coolant. The bypass and discharge valves can also be controlled manually from the MCR N-DCIS. The bypass valve will fail close and the discharge valve will fail open. Train redundancy ensures that single failure of any AOV will not impact the other train. As described in DCD Tier 2, Section 9.3.6, this system is designed to ensure that failure neither compromises any safety-related system or component nor prevents a safe shutdown.

Based on the staff's review of the applicant's response to RAI 9.2-24, the staff finds that the applicant has properly addressed the single-failure consideration through the redundancy of the design, the availability of the components' emergency power supply, and the component failure positions upon a LOPP. The design redundancy of the RCCWS provides for adequate system reliability. In addition, train independence ensures that single failure of any AOV will not impact the other train. The staff finds that the response to RAI 9.2-24 regarding single failure and backup power is acceptable since the applicant clarified how the DCD includes the single-failure and backup power attributes of the RCCWS. Based on the above, the applicant's responses and DCD changes, the single failure aspects of RAI 9.2-24 are resolved.

(4) <u>RCCWS Pump Net Positive Suction Head</u>

As described in DCD Tier 2, Revision 9, Section 9.2.2.2, surge tanks provide a constant pump suction head and allow for thermal expansion of the RCCWS inventory. The tanks are located above the highest point in the system. The MWS provides makeup to the RCCWS inventory through an automatic level control valve. A manual valve provides a backup source of makeup from the FPS.

The staff requested, in RAI 9.2-23 and RAI 9.2-24, that the applicant address NPSH and provide additional information to address the design alarm features in the MCR available to the operators.

In response to RAI 9.2-24, the applicant clarified that the surge tank level is monitored to ensure that sufficient NPSH is available for pump operation and to detect intersystem leakage intrusions into RCCWS. During cooling water train separation, low surge tank standpipe level, in combination with low-low surge tank level, automatically initiate a train shutdown. A train shutdown signal trips off all pumps in the train and closes all isolation, bypass, and flow control valves. The automatic train shutdown signal will be the only automated pump trip signal based on process conditions for the RCCWS pumps. The staff noted during the audit that the DCD does not describe surge tank level controls, train separation, and shutdown upon indication of low level. In response to RAI 9.2-24, the applicant modified Revision 6 of DCD Tier 2, Sections 9.2.2.2 and 9.2.2.5, to describe the function of RCCWS train separation and signals that initiate train shutdown, which includes low-low surge tank level. The staff confirmed that Revision 6 of the DCD incorporated this RAI proposed change.

The staff finds that the responses to RAIs 9.2-23 and 9.2-24 regarding RCCWS pump NPSH are acceptable since the applicant clarified how sufficient NPSH is assured. The applicant clarified the design features of the RCCWS to ensure NPSH, including the RCCWS surge tank and its system position (high point of the system), instrumentation which detects a low-low surge tank level, and automatic train shutdown. Available NPSH for pump performance is maintained with these design features. Accordingly, based on the above, the applicant's responses and DCD changes, RAI 9.2-23 and the NPSH aspects of RAI 9.2-24 are resolved.

(5) Operating Experience

The NRC issued GL 96-06, "Assurance of Equipment Operability and Containment Integrity During Design-Basis Accident Conditions," to address the potential for (1) water hammer and two phase flow in cooling water systems penetrating the containment and (2) thermally induced over-pressurization of isolated water-filled piping sections in containment that could jeopardize the function of accident mitigation systems and could lead to a loss of containment integrity. The staff concluded that GL 96-06 does not apply to the RCCWS since it is not routed through containment.

(6) Periodic Inspections and Testing

As discussed in Item D above, the applicant demonstrated in DCD Tier 2, Revision 5, Section 9.2.2.1, that the RCCWS satisfies GDC 44, 45, and 46 because the design of the RCCWS included design provisions to permit inspection and operational testing of components and equipment.

DCD Tier 2, Revision 9, Section 9.2.2.4, describes the applicant's provisions for periodic inspection of components to ensure the capability and integrity of the system. Indicators are provided for vital parameters necessary for testing and inspection, and provisions for grab sampling of RCCWS cooling water are provided for chemical and radiological analyses.

Periodic inspections and testing are important for assessing and maintaining the capability and reliability of the RCCWS to perform its defense-in-depth functions over the life of the plant. The RCCWS design bases indicate that provisions are included to permit inspection of components and equipment. In addition, the system description indicates that valves are arranged for ease of in-service inspection. DCD Tier 2, Section 9.2.2.4, indicates that provision is made for periodic inspection of components to ensure the capability and integrity of the system. The periodic inspection and testing was determined to be incomplete; therefore, the staff requested in RAI 9.2-24 that the applicant revise DCD Tier 2, Section 9.2.2.

The applicant's response to RAI 9.2-24 noted that maintenance, testing, and operating procedures will include provisions for regular inspection testing and maintenance of valves to prevent degradation over time. As described in DCD Tier 2, Sections 19A.8 and 19A.8.4.9, all RTNSS systems are in the scope of the Design Reliability Assurance Program, as directed by DCD Tier 2, Chapter 17, which will be incorporated into the Maintenance Rule Program. The Maintenance Rule, 10 CFR 50.65, requires performance monitoring of SSCs that are not safety-related but are relied upon to mitigate accidents or transients, are used in EOPs, or whose failure could prevent safety-related SSCs from performing their safety-related function or could cause a reactor scram or actuation of a safety-related systems. Such SSCs may include RTNSS components. In addition, the PRA model of the RCCWS (NEDO 33201, Revision 3, "ESBWR Certification Probabilistic Risk Assessment") assumes that active components other than pumps and heat exchangers are tested every 24 months during the plant shutdown for refueling. The function of these valves would be verified every refueling outage during SDG LOPP testing.

The staff finds that the RAI response is acceptable since the RCCWS will be monitored under the Maintenance Rule Program, which includes the maintenance of valves to prevent degradation over time. For the RCCWS and other RTNSS systems, the Maintenance Rule Program ensures that unacceptable risk is detected and appropriate actions are taken. Accordingly, based on the above and the applicant's response, the inspection and testing aspects of RAI 9.2-24 are resolved.

(7) Instrumentation, Controls, and Alarms

DCD Tier 2, Section 9.2.2.5, indicates that the RCCWS is operated and monitored from the MCR. Major system parameters, which include loop flow rates and heat exchanger outlet temperatures and pressures, are indicated in the MCR. Other RCCWS instrumentation that was briefly described includes that which controls RCCWS automatic pump starts based in failure of one electrical bus, RCCWS radiation monitors, and surge tank level. DCD Tier 2, Section 7.4.2, states that control of two RCCWS trains and two PSWS trains is provided on the remote shutdown system (RSS) panel.

In RAIs 9.2-6 and 9.2-6 S01, the staff determined that the simplified diagrams in DCD Tier 2, Revision 2, Figures 9.2-1 and 9.2-2 did not have sufficient detail and requested that the applicant include system drawings (piping and instrumentation drawings (P&IDs)) in the DCD for the PSWS and RCCWS showing system functions, major equipment, components, piping classes, interfacing systems, and instrumentation. RAI 9.2-6 was being tracked as an open item in the SER with open items. In response, the applicant did not provide P&IDs in the DCD for the PSWS and RCCWS. The applicant indicated that the P&IDs are proprietary information and are not intended to be included in the DCD. This response did not provide sufficient bases for the staff to resolve the RAI. The applicant subsequently added in DCD Tier 2, Revisions 3, 4, and 5, more details in the existing simplified diagrams in Figures 9.2-1, 9.2-2a, and 9.2-2b to supplement the information regarding system functions, major equipment, components, piping classification, interface systems, and instrumentation. However, as a follow-up to RAI 9.2-6 S01, the staff specifically asked, in RAI 9.2-24, the applicant to include the header temperature and pressure detectors in the diagrams.

The staff finds that the response to RAIs 9.2-6 and 9.2-24 as they relate to the simplified diagrams is acceptable since the supplemental information in the revised Figures 9.2-1, 9.2-1a, and 9.2-2b supports the RCCWS RTNSS functions. The staff also finds that the description of RCCWS header temperature and pressure detectors in DCD Tier Section 9.2.2.5 is sufficient to

support RTNSS functions and add additional instrumentation information does not need to be added to the simplified diagrams beyond that included in the response to RAI 9.2-24. Accordingly, based on the above, the applicant's responses and DCD changes, RAIs 9.2-6, 9.2-6 S01 and 9.2-24 regarding the simplified diagrams are resolved.

As previously stated in Section 9.2.2.3.1.E of this report, DCD Tier 2, Figure 9.2-2b, illustrates a radiation detector downstream of the A Train RWCU/SDC heat exchangers. This instrument is not shown downstream of B Train RWCU/SDC heat exchangers. The applicant corrected this omission in DCD Revision 6 and added the radiation detector downstream of the B Train RWCU/SDC heat exchangers

As previously stated in Section 9.2.2.3 G(3) of this report, two automatic train separation signals are used to close the cross-tie valves, which are the detection of unbalanced flow and a LOPP event.

As previously stated in Section 9.2.2.3 G(4) of this report, the DCD did not describe surge tank level controls, train separation, and shutdown upon indication of low level. In response to RAI 9.2-24, the applicant modified Revision 6 of DCD Tier 2, Section 9.2.2.2, to add a description of the function of RCCWS train separation and signals that initiate train shutdown.

Based on the above, the staff finds the RCCWS instrumentation, controls, and alarms acceptable.

9.2.2.3.2 COL Information

The staff reviewed DCD Tier 2, Section 13.5.3, COL Information Item 13.5-2-A for plant operating procedure development. This section refers to Section 13.5.4, which in turn refers to the procedures as delineated in ANSI/ ANS-3.2. RG 1.33 endorses ANSI/ANS-3.2, and its Appendix A lists typical safety-related activities that should be covered by written procedures. Appendix A to RG 1.33 lists the service water system and component cooling water system. However, the PSWS and RCCWS in the ESBWR are not safety-related; therefore, the above generic COL information item might not cover nonsafety-related systems, such as the PSWS and RCCWS, in the ESBWR. In response to RAI 9.2-11 S04, the applicant revised DCD Tier 2, Section 13.5.2 to clarify that COL Information Item 13.5-2-A will include the water hammer procedures for the RTNSS systems. Therefore, the staff finds COL Information Item 13.5-2-A acceptable regarding procedure development for the RCCWS.

The applicant identified no other COL information items in DCD Tier 2, Section 9.2.2.6. The staff finds that there are no relevant COL information items for the RCCWS that need to be developed as part of the DCD.

9.2.2.3.3 Availability Controls

As discussed in DCD Tier 2, Revision 9, Section 19A.8.1, regulatory oversight is applied to each system that is designated as RTNSS to ensure adequate reliability and availability to perform RTNSS functions. DCD Tier 2, Section 19A.8.1 also indicates that Maintenance Rule performance monitoring is specified for all RTNSS functions and that the ACM describes additional oversight for support systems. DCD Tier 2, Revision 9, Table 19A-2, identifies that the PSWS is a support system and that the PSWS 'Availability Controls' are the 'Maintenance Rule,' which means that Maintenance Rule performance monitoring addresses the availability of the PSWS rather than a specific ACM entry.

The PSWS and RCCWS are subject to the ACM through the systems they support. DCD Tier 2, Table 19A-2, classifies the PSWS and the RCCWS as support systems for the SDGs and the NICWS. The NICWS supports the building HVAC, which supports the FAPCS. The FAPCS is the RTNSS system that is relied upon for active mitigation and the SDGS are support systems for the FAPCS. Of these systems, the ACM specifies availability controls for the SDGs in AC 3.8.1, and AC 3.8.2, and for FAPCS in AC 3.7.2, and AC 3.7.3. Therefore, the PSWS and RCCWS are support systems that are subject to the ACs specified for the SDGs and the FAPCS.

ACM Section 1.1, states that for the term "AVAILABLE-AVAILABILITY," a system, subsystem, train, division, component, or device shall be considered AVAILABLE or to have AVAILABILITY when it is capable of performing its specified risk informed function or functions and when all necessary attendant instrumentation, controls, normal or emergency electrical power, cooling and seal water, lubrication, and other auxiliary equipment that support operation of the system, subsystem, train, division, component, or device with respect to its specified risk informed function or functions are also capable of performing their related support function or functions. Since the PSWS supports the RCCWS, which supports the NICWS, FAPCS, and SDGs, if the PSWS or the RCCWS become unavailable, then the systems they support become unavailable and the applicable ACM action statements would apply.

Based on the above, the staff finds the ACs for the RCCWS acceptable since the RCCWS is subject to the Maintenance Rule and indirectly subject to the ACM because the RCCWS is an RTNSS support system and its availability is indirectly covered by the ACs for the FAPCS and SDGs.

9.2.2.3.4 Inspections, Tests, Analyses, and Acceptance Criteria

In DCD Tier 1, Revision 3, Section 2.12.3, the applicant revised the RCCWS ITAAC to remove the system description and system drawings, design commitment, and scope of ITAAC. DCD Tier 2, Section 14.3.7.3, indicates that RTNSS systems will have Tier 1 inputs that include design descriptions and ITAAC. The staff determined that the removal of RCCWS ITAAC in Tier 1 was not acceptable. In RAIs 22.5-1 and 22.5-1 S01, the staff requested that the applicant review and revise DCD Tier 1 to include the RCCWS in Tier 1. The applicant responded to the RAI and provided the requested Tier 1 system description, ITAAC, and drawing for the RCCWS in the revised DCD Tier 1, Section 2.12.3. Based on the above, the applicant's responses, the RAI response and DCD changes, RAI 22.5-1 is resolved.

The applicant addressed the ITAAC details as part of RAI 9.2-24 and the March 19–20, 2009, audit. The applicant provided the following response to the staff's questions regarding the lack of specific details for the RTNSS Criterion C acceptance criteria:

> PSWS, RCCWS and NICWS provide supporting functions for FAPCS suppression pool cooling and low pressure injection modes, and thus meet RTNSS Criterion C. RTNSS C SSCs are assumed to be available at the time of the initiating event. Validation of these RTNSS functions is assured by Tier 1 ITAAC (Section 2.12.7 PSWS; Section 2.12.3 RCCWS; Section 2.12.5 NICWS) where testing of the PSWS /RCCWS / NICWS demonstrate flow to the RCCWS (nuclear island chillers, diesel generators and FAPCS island chillers, diesel generators and FAPCS). The ESBWR RTNSS Criterion C Cooling Water System ITAAC scope and detail differs from that associated with validation of RTNSS Criterion B functions. The ESBWR is designed so that safety-related

passive systems are able to perform all safety functions for at least 72 hours, after initiation of a design basis event, without the need for active systems or operator actions. After 72 hours, nonsafety-related systems (RTNSS Criterion B) can be used to replenish the passive systems or to perform core cooling and containment integrity functions directly. RTNSS Criterion B ITAAC (e.g. FAPCS section 2.6.2 Item 7 and FPS section 2.16.3 item 7) provides a greater assurance of function.

The staff finds that the RAI response is acceptable since the RCCWS DCD Tier 1 information is adequate and reasonable based on the ESBWR graded approach for this RTNSS Criterion C, nonsafety-related system. For the RTNSS functions of the RCCWS, flow to key RTNSS equipment such as chillers, FAPCS, and SDGs, is verified; as-built verification is performed, operation of selected controls from the MCR is verified, and RCCWS system flow indication is confirmed to be available in the MCR. Based on the above and the applicant's response, the ITAAC related aspects of RAI 9.2-24 are resolved.

9.2.2.3.5 Initial Test Program

Section 14.2 of this report evaluates the initial test program for the ESBWR; the evaluation of the RCCWS initial test program in this section is an extension of the evaluation provided in Section 14.2.

DCD Tier 2, Revision 9, Section 14.2.8.1.21, describes the pre-operational test program for the RCCWS. The staff finds the objective of the RCCWS pre-operational test program to be appropriate since its purpose is to verify proper operation of the RCCWS based on its ability to supply design quantities of cooling water, at the specified temperatures, to assigned loads, as appropriate, during normal, abnormal, and accident conditions. Because of insufficient heat loads during the pre-operational phase, the final system flow balancing and heat exchanger performance evaluation is performed during the startup phase. While the test specifications are written in very general terms to address the considerations that apply to the RCCWS, the approach for this nonsafety-related system is considered to be acceptable because the COL applicant will develop test procedures in accordance with COL Information Item 14.2-3-A.

During of review DCD Tier 2, Revision 5, the staff determined that it needed additional information and specificity in some respects and requested, in RAI 9.2-24, that the applicant revise DCD Tier 2, Section 14.2.8.1.21 to address the testing of the RCCWS. In response to RAI 9.2-24, the applicant clarified the basis for its pre-operational test program. Preoperational startup testing, as described in DCD Tier 2, Section 14.2.8.1.21, will verify proper operation of system valves, including timing, under expected operating conditions. Maintenance, test, and operating procedures will include provisions for regular inspection testing and maintenance of valves to prevent leakage that can cause void formation during periods of standby. RCCWS pump tests and integrated flow tests will ensure that discharge check valve leakage will not impact pump or system flow performance. This includes startup of a standby loop or actuation following a loss of power with proper operation ensuring that water hammer does not occur. The staff finds that the RAI 9.2-24 response is acceptable since the level of testing addresses system performance, minimum NPSH, instrumentation and interlocks, and water hammer. Therefore, DCD Tier 2, Section 14.2, need not describe additional testing. Accordingly, based on the above and the applicant's response, the initial test program aspects of RAI 9.2-24 are resolved.

9.2.2.4 *Conclusion*

For the reasons set forth above, the staff finds that the RCCWS complies with the requirements of GDC 2, 4, 44, 45, 46 and 10 CFR 20.1406. Because the ESBWR design is a single unit, GDC 5 is not applicable. The staff also finds that the design of the RCCWS satisfies established NRC policies with respect to its RTNSS Criterion C function.

9.2.3 Makeup Water System

9.2.3.1 *Regulatory Criteria*

The staff reviewed the MWS based on the guidance provided in SRP Section 9.2.3, Revision 2. Staff acceptance of the design is based on meeting the following requirements:

- GDC 2, as it relates to the capability of the design to maintain and perform its safety function following an earthquake

- GDC 5, as it relates to the capability of shared systems and components important to safety to perform required safety functions

- 10 CFR 20.1406 as it relates to minimization of contamination

9.2.3.2 *Summary of Technical Information*

DCD Tier 2, Revision 9, Section 9.2.3, describes the MWS. The MWS consists of two subsystems—(1) the demineralization subsystem and (2) the storage and transfer subsystem. The demineralization subsystem is a conceptual design that is dependent on the site-specific water quality of the available water source. The storage and transfer subsystem is a standard design applicable to any site.

The MWS major equipment is housed entirely in the service water/water treatment building, except for the demineralized water storage tank (which is outdoors and adjacent to this building) and the distribution piping to the interface systems. The MWS equipment and associated piping in contact with demineralized water are fabricated from corrosion-resistant materials, such as stainless steel, to prevent contamination of the makeup water. DCD Tier 2, Revision 9, Table 9.2-9, lists the major MWS components.

The flow path of the storage and transfer subsystem of the MWS is from the MWS demineralized water storage tank, through a MWS transfer pump, to the interface systems. One pump operates continuously to maintain the system pressure. Increased demand or primary transfer pump failure automatically starts the second transfer pump.

9.2.3.3 *Staff Evaluation*

The staff reviewed the design of the MWS in accordance with the applicable portions of SRP Section 9.2.3.

Piping and valves forming part of the containment boundary are designed to seismic Category I. Piping and valves inside containment or inside the RB are designed to seismic Category II. Other than the containment isolation and penetrations, the other portions of the MWS are nonsafety-related. To meet the requirements of GDC 2 as they relate to structures and systems

being capable of withstanding the effects of natural phenomena, acceptance depends on meeting the guidance of those portions of Regulatory Position C.1 of RG 1.29 applicable to the safety-related portions of the system and Regulatory Position C.2 of RG 1.29 for the nonsafety-related portions of the system.

The ESBWR design is a single-unit station, and the requirements of GDC 5 are not applicable for the single-unit design.

The MWS is a nonsafety-related system. SRP Section 9.2.3 indicates that the requirements of GDC 2 can be met for a nonsafety-related system based on meeting Regulatory Position C.2 of RG 1.29 regarding nonsafety-related systems. In RAI 9.2-12, the staff requested that the applicant demonstrate that the MWS (among other water systems) satisfies Regulatory Position C.2 of RG 1.29. In response, the applicant explained that the MWS does not have any piping in the control room or interface with any safety-related components.

The MWS does not have any safety-related functions except for containment isolation. MWS containment penetrations and isolation valves are designated as seismic Category I, and those portions within seismic Category I buildings are designed as seismic Category II. Failure of the MWS will not compromise any safety-related system or component, and will not prevent a safe shutdown. The staff reviewed the response to RAI 9.2-12 and DCD Tier 2, Revision 6, Section 9.2.3 and Table 3.2-1. Based on the above, the staff finds that the MWS meets the guidance of Regulatory Position C.2 of RG 1.29 regarding nonsafety-related systems because the failure of the nonsafety-related portions of the systems does not impact any safety-related SSCs. In addition, the MWS meets the guidance of Regulatory Position C.1 of RG 1.29 for the portions of the system (containment penetrations) that are safety-related, and the MWS meets the requirements of GDC 2. Accordingly, based on the above and the applicant's response, the portions of RAI 9.2-12 relating to the MWS are resolved.

The staff reviewed the design of the MWS for conformance to 10 CFR 20.1406. DCD Tier 2, Revision 9, Table 12.3-18, does not identify any specific MWS design features to address conformance to RG 4.21 design objectives. This is consistent with the interim staff guidance in DC/COL-ISG-06, "Evaluation and Acceptance Criteria for 10 CFR 20.1406 to Support Design Certification and Combined License Applications," which does not identify the MWS as a system that typically has the potential to release radioactive material to the facility site, or environment. Section 12.4 of this report provides the evaluation of ESBWR generic design features for conformance to RG 4.21 and 10 CFR 20.1406.

DCD Tier 2, Table 9.2-6, states that the MWS is designed to provide makeup water for the RCCWS and CWS. The response to RAI 14.3-69 identifies these systems as RTNSS systems. The staff requested the applicant in RAIs 22.5-19 and 22.5-19 S01 to clarify whether the makeup to the RCCWS and CWS provided by the MWS is required to satisfy RTNSS selection Criterion B. RAI 22.5-19 was being tracked as an open item in the SER with open items. In the responses to the RAI, the applicant stated that the MWS is available, but not relied upon, to support the RCCWS and CWS cooling functions. The RCCWS and CWS are closed loop systems and minimum leakage is expected; surge tanks should have adequate capacity to provide makeup for normal system leakage. However, if necessary, the FPS can provide a dedicated seismic makeup source to the RCCWS. This seismic FPS makeup source could also provide makeup to the CWS. Based on the above, the staff finds that the MWS does not need to be a RTNSS Criterion B system. The staff finds that the RAI response is acceptable since the applicant explained its basis for the MWS RTNSS determination. Accordingly, based on the above and the applicant's response, RAI 22.5-19 is resolved.

DCD Tier 2, Section 9.2.3.2 states that the CDI for the MWS will be replaced with site-specific design information in the COLA FSAR. In RAI 9.2-17, the staff asked the applicant to identify a COL information item for the site-specific design. In the response, the applicant stated that 10 CFR Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants," allows a DCD applicant to provide a representative conceptual design for those portions of the plant for which the application does not seek certification to aid the NRC in its review of the FSAR and to permit assessment of the adequacy of the interface requirements. The DCD addresses separately the CDI and COL information items. DCD Tier 2, Section 1.8.2, provides a summary of the BOP interfaces and references some DCD Tier 2 sections in which CDI information could possibly be found. DCD Tier 2, Section 1.10, also provides COL information items. RG 1.206 discusses the need for COL applicants to address CDI "in addition" to addressing COL information items (refer to Section C.III.1.8). RG 1.206 specifies that COL applicants who reference a certified design provide complete designs for the entire facility, including appropriate site-specific design information to replace the conceptual design portions of the DCD. Hence, it is unnecessary to assign COL information items to the CDI in the DCD, since the need to address this information is specified in RG 1.206. The staff finds that the response is acceptable since the justification for not having a COL information item to address the CDI is consistent with RG 1.206. Accordingly, based on the above and the applicant's response, RAI 9.2-17 is resolved.

9.2.3.4 *Conclusion*

Based on the above, the staff finds that the design of the MWS is acceptable and meets the requirements of GDC 2 and 10 CFR 20.1406. The site-specific CDI design will be reviewed in the COL application.

9.2.4 Potable and Sanitary Water Systems

9.2.4.1 *Regulatory Criteria*

The staff reviewed the potable and sanitary water systems based on the guidance provided in SRP Section 9.2.4, Revision 3, "Potable and Sanitary Water Systems," issued March 2007. Staff acceptance of the design is based on meeting the following requirements:

- GDC 60, "Control of releases of radioactive materials to the environment," as it relates to design provisions provided to control the release of liquid effluents containing radioactive material from contaminating

- 10 CFR 20.1406 as it relates to minimization of contamination

9.2.4.2 *Summary of Technical Information*

DCD Tier 2, Section 9.2.4, states that the potable and sanitary water systems design are dependent on the site-specific water pathways. The conceptual system is to supply up to 12.6 liters per second (l/s) (200 gpm) of potable water during peak demand periods. The potable and sanitary water systems will meet GDC 60 by including provisions to control the release of liquid effluents containing radioactive material. The potable and sanitary water systems have no interconnections to systems with the potential for containing radioactive material. The design of wastewater effluent systems properly disposes of sanitation wastes. The COLA FSAR will replace the above CDI for the potable and sanitary water systems with site-specific design information.

9.2.4.3 *Staff Evaluation*

The applicant states that the COLA FSAR will provide the site-specific design information, and the DCD only provides the CDI. The CDI in the DCD does not have a design for review. The staff agrees with the applicant that the nature of the system is site-specific and will review the design of the potable and sanitary water systems in accordance with SRP Section 9.2.4 at the COLA stage. The staff will evaluate the design in light of the requirements of GDC 60 when the plant-specific design is available.

DCD Tier 2, Section 9.2.4 states that the COLA FSAR will replace the CDI for the potable and sanitary water systems with site-specific design information. In RAI 9.2-18, the staff asked the applicant to identify a COL information item for the site-specific design. In the response, the applicant stated that it is unnecessary to assign COL action items to the CDI in the DCD, since the need to address this information is specified in RG 1.206. Similar to the evaluation for RAI 9.2-17, discussed in Section 9.2.3.3 of this report, the staff finds that the response to RAI 9.1-18 is acceptable since the justification for not having a COL information item to address the CDI is consistent with RG 1.206. Accordingly, based on the above and the applicant's response, RAI 9.2-18 is resolved.

The staff reviewed the design of the potable and sanitary water systems for conformance to 10 CFR 20.1406. DCD Tier 2, Revision 9, Table 12.3-18, does not identify any specific potable and sanitary water systems design features to address conformance to RG 4.21 design objectives. This is consistent with the interim staff guidance in DC/COL-ISG-06, "Evaluation and Acceptance Criteria for 10 CFR 20.1406 to Support Design Certification and Combined License Applications," which does not identify the potable and sanitary water systems as a systems that typically has the potential to release radioactive material to the facility site, or environment. Section 12.4 of this report provides the evaluation of ESBWR generic design features for conformance to RG 4.21 and 10 CFR 20.1406.

9.2.4.4 *Conclusion*

The staff reviewed the CDI of the potable and sanitary water systems at this stage, and will review the site-specific design in the COL applications as it relates to the requirements of GDC 60 and 10 CFR 20.1406.

9.2.5 Ultimate Heat Sink

9.2.5.1 *Regulatory Criteria*

The staff reviewed the UHS based on the guidance provided in SRP Section 9.2.5, Revision 3. Staff acceptance of the design is based on meeting the following requirements:

- GDC 2, as it relates to the capability of the design to maintain and perform its safety function following an earthquake

- GDC 5, as it relates to the capability of shared systems and components important to safety to perform required safety functions

- GDC 44, as it relates to transferring heat from SSCs important to safety to a heat sink

- GDC 45, as it relates to the design provisions to permit inspection of components and equipment

- GDC 46, as it relates to the design provisions to permit operational testing of components and equipment

- 10 CFR 20.1406 as it relates to minimization of contamination

9.2.5.2 *Summary of Technical Information*

DCD Tier 2, Revision 9, Sections 9.2.5, 5.4.6, and 6.2.2, describe the UHS. The UHS consists of the ICS and PCCS pools, the dryer/separator pool and reactor well, FPS makeup water for the ICS/PCCS pools, and SFP from the primary (seismic Category I) firewater storage tanks via the safety-related FAPCS piping, and other water sources that are credited for providing makeup water for the ICS/PCCS pools and SFP after water from the firewater storage tanks is depleted. The dryer/separator pool and reactor well provide sufficient makeup water for the ICS/PCCS expansion pools to support operation of the ICS and PCCS during the initial 72 hours following an accident. A source of makeup water for the SFP is not credited during this period. After the initial 72 hours, the FPS is relied on for supplying the necessary makeup water for the IC/PCC pools and the SFP for up to 7 days.

In the event of an accident, the UHS is provided by the ICS/PCCS pools, which provide the heat transfer mechanism from the reactor and containment to the atmosphere. The principal heat source is decay heat from the fuel. The decay heat input rate decreases with time as shown in the series of decay heat curves in DCD Tier 2, Figure 6.2-10c. Therefore, the minimum total makeup water flow rate beyond 72 hours, as well as beyond 7 days, into an event, would not exceed the minimum total makeup water flow rate at 72 hours, as shown in DCD Tier 2, Table 9.5-2. The makeup water sources meet the minimum flow rate specified in DCD Tier 2, Table 9.5-2. DCD Tier 2, Section 9.1.3.2, discusses the use of the FAPCS to provide water after 72 hours the 72-hour-period following an accident.

9.2.5.3 *Staff Evaluation*

The staff reviewed the design of the UHS in accordance with applicable portions of SRP Section 9.2.5. Staff acceptance of the UHS is based on meeting the requirements of GDC 2, 5, 44, 45, and 46.

In the event of an accident, the UHS function is provided by the ICS/PCCS pools, which provide the heat transfer mechanism for the reactor and containment to the atmosphere. In DCD Tier 2, Revision 6, Section 9.2.5, the applicant stated that the ICS/PCCS meets GDC 2, 44, 45, and 46. The ICS and PCCS are designed to seismic Category I and therefore meet GDC 2 by satisfying Regulatory Position C.1 of RG 1.29. GDC 44 is met by the heat removal capability of the IC/PCC to transfer decay heat to the heat sink. GDC 45 and 46 are met because the ICS and PCCS are subject to testing and inspection as described in DCD Tier 2, Sections 5.4.6.4 and 6.2.2.4.

SRP Section 9.2.5 identifies the requirement for 30-day water makeup capability during an accident. The ICS/PCCS pools have reserve capacity for 72 hours of heat removal without makeup. The ICS/PCCS pools, which provide the UHS function for the first 72 hours following an accident, are safety-related and are evaluated in Sections 5.4.6 and 6.2.2 of this report. The parts of the UHS that are relied upon for providing makeup water during the period from 72

hours through 7 days following an accident are not safety-related, but are readily available on-site and are subject to RTNSS as discussed in DCD Tier 2, Revision 9, Appendix 19A. Section 22.5.6 of this report provides the staff evaluation. The FPS provides post-accident makeup to the ICS/PCCS pools through safety-related FAPCS piping. DCD Tier 2, Revision 9, Section 9.5.1, discusses the FPS as a backup emergency makeup water source through the FAPCS. The FPS provides onsite makeup water capability from 72 hours to 7 days, after which time offsite makeup sources can be provided via safety-related external FAPCS connections outside the RB and FB or onsite makeup sources. Section 9.1.3 of this report discusses the external connection and emergency makeup water piping, which is part of the FAPCS.

This section evaluates the adequacy of the capability that is credited for providing makeup water to the ICS/PCCS pools, and the SFP after the initial seven days have elapsed following an accident. In RAI 9.2-19, the staff asked the applicant to clarify the minimum makeup flow beyond 72 hours. DCD Tier 2, Table 9.5-2, specified a constant of 46 m^3/h (200 gpm) at 72 hours, but not beyond 72 hours. In the response, the applicant stated that the makeup water demand decreases with time. The makeup demand at 72 hours bounds the minimum makeup demand beyond 72 hours. The staff finds that the response is acceptable since the applicant clarified that a constant makeup capacity is provided even though the demand decreases with time. Accordingly, based on the above and the applicant’s response, RAI 9.2-19 is resolved.

DCD Tier 2, Section 9.2.5, states that the COL applicant will develop procedures to use an external makeup water supply through the FAPCS to the ICS/PCCS pools and SFP beyond the 7 days following an accident. DCD Tier 2, Revision 6, Section 9.2.5.1, identifies this as COL Information Item 9.2.5-1-A, which states the following:

> The COL Applicant will include in its operating procedure development program:
>
> - Procedures that identify and prioritize available makeup sources 7 days after an accident, and provide instructions for establishing necessary connections.
> - Milestone for completing this category of operating procedures (Subsection 9.2.5).

The staff finds COL Information Item 9.2.5-1-A acceptable since available makeup sources after 7 days are expected to be site-specific. The staff will review this information during the COL application process.

The staff reviewed the design of the UHS for conformance to 10 CFR 20.1406. DCD Tier 2, Revision 9, Table 12.3-18, does not identify any specific UHS design features to address conformance to RG 4.21 design objectives. However, DCD Tier 2, Revision 9, Table 12.3-18, describes the provisions regarding RG 4.21 design objectives related to the ICS/PCCS pools, which provide the UHS function:

- Provide for adequate leak detection capability to provide detection of leakage for any SSC which has the potential for leakage (Design Objective 2)

- Leak detection methods and early detection of leaks to avoid release of contamination from undetected leaks and to minimize contamination of the environment (Design Objective 3)

The FAPCS provides cooling and cleanup of the ICS/PCCS pools and includes several design provisions to address the above design objectives. Anti-siphoning devices are used on all submerged FAPCS piping to prevent unintended draining of the pools. The FAPCS is designed

with features, including drains, gates, and weirs, to prevent drainage of coolant inventory below an adequate shielding depth. The FAPCS is also designed to provide for the collection, monitoring, and drainage of pool liner leaks from the ICS/PCCS pools to the LWMS. The ICS/PCCS pools are also equipped with stainless steel liners and with leak detection drains. All leak detection drains are designed to permit free gravity drainage to the LWMS. Accordingly, the staff finds that these design provisions for the FAPCS and the ICS/PCCS pools conform to the guidelines of RG 4.21 with respect to monitoring and minimizing leakage, and that the FAPCS and the ICS/PCCS pools, and therefore the UHS, meet the requirements of 10 CFR 20.1406 in this regard. Sections 12.4 and 12.7 of this report further address the ESBWR design in accordance with 10 CFR 20.1406. The ESBWR design is a single-unit station, and the requirements of GDC 5 are not applicable for the single-unit design.

9.2.5.4 *Conclusion*

Based on the above, the staff finds that the design of the UHS is acceptable and meets the requirements of GDC 2, 44, 45, 46, 10 CFR 20.1406. The staff will review COL Information Item 9.2.5-1-A in the COL applications.

9.2.6 Condensate Storage and Transfer System

9.2.6.1 *Regulatory Criteria*

The staff reviewed the CS&TS based on the guidance provided in SRP Section 9.2.6, Revision 3. Staff acceptance of the design is based on meeting the following requirements:

- GDC 2, as it relates to the capability of the design to maintain and perform its safety function following an earthquake
- GDC 5, as it relates to the capability of shared systems and components important to safety to perform required safety functions
- GDC 44, as it relates to transferring heat from SSCs important to safety to a heat sink
- GDC 45, as it relates to the design provisions to permit inspection of components and equipment
- GDC 46, as it relates to the design provisions to permit operational testing of components and equipment
- GDC 60, as it relates to tanks and systems handling radioactive material in liquids
- 10 CFR 20.1406 as it relates to minimization of contamination
- 10 CFR 50.63 as it related to station blackout (SBO)

9.2.6.2 *Summary of Technical Information*

DCD Tier 2, Section 9.2.6, describes the CS&TS. The CS&TS is designed to do the following:

- Operate during plant startup, power operation, and normal shutdown. The system is not required to operate following loss of power or during any design-basis event.

- Provide managed storage capacities in the condensate storage tank (CST).

- Provide a distribution system to supply condensate-quality water to equipment.

- Provide a 100-percent-redundant backup transfer pump.

- Provide the capability to maintain the water quality requirements in the CST by pumping tank contents to the liquid radwaste system when the condensate purification system is not operating.

- Provide an enclosed area to retain any tank overflow or leakage until an appropriate disposal action is taken.

- Provide sampling of the retention area sump before disposal to determine if the activity of the sump contents is within the limits set by 10 CFR Part 20.

The CS&TS is designed to seismic Category II criteria when located in seismic Category I buildings to preclude damage to safety-related equipment should a seismic event occur.

The CS&TS consists of two independent and 100-percent-redundant transfer pumps that take suction from the CST and provide water to interface systems as required. The CST provides storage capacity for condensate rejected from the condensate and feedwater system, condensate-quality LWMS effluent during normal operation, and condensate and feedwater system and condenser hotwell inventory during system maintenance outages.

The CST also provides a minimum storage capacity for the CRD system as a reserve water source for RPV makeup following a nuclear steam supply system isolation event. The CS&TS equipment and associated piping are fabricated from stainless steel to prevent contamination of the system water.

The CST is the normal source of water for makeup to selected plant systems. The condensate transfer pumps take their suction from the CST and provide makeup water for various services in the RB, TB, FB, and radwaste building. There are two 100-percent-redundant condensate transfer pumps. One of the two transfer pumps runs continuously to provide condensate-quality water, as required. Minimum flow recirculation is provided for pump protection.

9.2.6.3 *Staff Evaluation*

The staff reviewed the design of the CS&TS in accordance with SRP Section 9.2.6. Staff acceptance of the CS&TS is based on meeting the requirements of GDC 2, 5, 44, 45, 46, and 60 and 10 CFR 20.1406 and 10 CFR 50.63

The ESBWR design is a single-unit station, and the requirements of GDC 5 are not applicable for the single-unit design.

To meet the requirements of GDC 2 as they relate to the capability of structures and systems to withstand the effects of natural phenomena, acceptance depends on meeting the guidance of those aspects of Regulatory Position C.1 of RG 1.29 pertaining to the safety-related portions of the system and those aspects of Regulatory Position C.2 of RG 1.29 pertaining to the nonsafety-related portions of the system.

The staff asked the applicant to provide additional information. The staff reviewed the applicant's response and discusses its evaluation of the response below.

As a part of RAI 9.2-12, the staff asked the applicant to demonstrate that the CS&TS meets GDC 2. The CS&TS is a nonsafety-related system. Based on SRP Section 9.2.6, a nonsafety-related system satisfies the requirements of GDC 2 by meeting the guidance of those portions of Regulatory Position C.2 of RG 1.29 applicable to nonsafety-related systems. In response, the applicant stated that the CS&TS does not have any piping in the control room or interface with any safety-related components. Those portions of the system within seismic Category I buildings are designed as seismic Category II. Failure of the CS&TS will not compromise any safety-related system or component, and it will not prevent a safe shutdown. Therefore, the CS&TS satisfies the requirements of GDC 2. The staff reviewed the above RAI responses and DCD Tier 2, Revision 6, Section 9.2.6 and Table 3.2-1. Based on the above, the staff finds that the CS&TS meets the guidance of Regulatory Position C.2 of RG 1.29 regarding nonsafety-related systems because the failure of the nonsafety-related portions of the systems does not impact any safety-related SSCs. Therefore, the CS&TS satisfies GDC 2. Accordingly, based on the above and the applicant's response, the portions of RAI 9.2-12 relating to the CS&TS are resolved.

In DCD Tier 2, Revision 6, Section 9.2.6.1, the applicant stated that GDC 44, 45, and 46 are not applicable to the CS&TS. SRP Section 9.2.6 Paragraph II.3 provides guidelines for how the CS&TS can meet GDC 44 related to performing the safety functions specified in SRP Section 9.2.6 Paragraphs II.3.A, II.3.B, II.3.C, II.4, and II.5. The staff reviewed the system description of the CS&TS and found that the CS&TS does not have the safety function, as specified in SRP Section 9.2.6, to provide makeup water to safety-related cooling systems. Therefore, the staff agrees with the applicant that GDC 44, 45, and 46 are not applicable for the CS&TS.

SRP Section 9.2.6 states that the acceptance for meeting the relevant aspects of GDC 60 is based on meeting the guidance of RG 1.143, "Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants." This guidance is also applicable to meeting the requirements of 10 CFR 20.1406 for the CS&TS. DCD Tier 2, Section 9.2.6.1, states that the CS&TS complies with RG 1.143, Regulatory Position C.I.2, for provisions to prevent uncontrolled releases of radioactive materials. DCD Tier 2, Table 12.3-18, describes the provisions regarding RG 4.21 design objectives related to CS&TS for:

- Minimizing leaks and spills (Design Objective 1)
- Provide for adequate leak detection capability to provide detection of leakage for any SSC which has the potential for leakage (Design Objective 2)
- Leak detection methods and early detection of leaks to avoid release of contamination from undetected leaks and to minimize contamination of the environment (Design Objective 3)
- Decreasing the spread of contaminant from the source (Design Objective 4)

- Minimize the generation and volume of radioactive waste both during operation and during decommissioning, by minimizing the volume of components and structures that become contaminated during plant operation. (Design Objective 6)

DCD Tier 2, Section 9.2.6 discusses corresponding provisions. While DCD Tier 2, Table 12.3-18 describes conformance to RG 4.21, the staff finds that these provisions also conform to RG 1.143, Position C.I.2. Therefore, the staff finds that the CS&TS meets GDC 60 and 10 CFR 20.1406 because it will include the means to reliably control the release of radioactive liquid effluents. Section 12.4 of this report further address the ESBWR design in accordance with 10 CFR 20.1406.

Regarding 10 CFR 50.63, DCD Tier 2, Revision 9, Section 8.2.2.2, states that the ESBWR design bases do not rely upon any offsite power system to achieve and maintain safe shutdown. The staff concludes that the CS&TS for the ESBWR design is not credited in the safety analysis to support flow delivery in the event of a SBO or in recovering from an SBO. Rather, passive safety-related systems perform SBO recovery. DCD Tier 2, Revision 9, Section 15.5.5, describes the ESBWR SBO analysis. Section 15.5.5 of this report provides the staff evaluation of the ESBWR SBO analysis.

9.2.6.4 *Conclusion*

Based on the above, the staff finds that the design of the CS&TS is acceptable and meets the requirements of GDC 2 and 60 and 10 CFR 20.1406 and 10 CFR 50.63.

9.2.7 Chilled Water System

9.2.7.1 *Regulatory Criteria*

The staff reviewed the CWS based on guidance provided in SRP Section 9.2.2, Revision 4. The SRP guidance is used to the extent that it pertains to system functionality and reliability considerations. Staff acceptance of the CWS design and supporting information is based upon conformance with the following:

- GDC 2, as it relates to the capability of the design to maintain and perform its safety functions following an earthquake

- GDC 4, as it relates to the dynamic effects associated with water hammer

- GDC 5, as it relates to the capability of shared systems and components important to safety to perform required safety functions

- GDC 44, as it relates to transferring heat from SSCs important to safety to a heat sink

- GDC 45, as it relates to the design provisions to permit inspection of components and equipment

- GDC 46, as it relates to the design provisions to permit operational testing of components and equipment

- 10 CFR 20.1406 as it relates to minimization of contamination

The CWS is a nonsafety-related system; however, the system provides defense-in-depth for the ESBWR passive plant design. In addition to the SRP guidance, the staff's evaluation of defense-in-depth systems also focuses on (1) confirming that design, performance, and reliability considerations are satisfied consistent with the NRC policies that are referred to in Chapter 22 of this report; (2) confirming that failure of defense-in-depth systems and components will not adversely impact safety-related SSCs; (3) confirming that ACs are established as appropriate; and (4) confirming that proposed ITAAC and initial test program specifications are adequate.

9.2.7.2 *Summary of Technical Information*

DCD Tier 2, Revision 5, Section 9.2.7, describes the CWS. The CWS consists of two independent subsystems: the NICWS and the balance-of-plant chilled water subsystem (BOPCWS). The CWS provides chilled water to the cooling coils of air-handling units (AHUs) and other coolers in the RB, CB, TB, radwaste building (RW), EB, and FB. The chilled water absorbs the rejected heat from these coolers and is pumped through the chillers where the heat is transferred from the NICWS to the RCCWS and from the BOPCWS to the TCCWS.

The NICWS consists of two 100-percent capacity trains, with redundancy and independence for active components. The BOPCWS consists of one 100 percent capacity independent loop with crossties to the NICWS chilled water piping. DCD Tier 2, Table 9.2-11, lists the CWS component design characteristics. DCD Tier 2, Figure 9.2-3, shows the CWS simplified diagram. DCD Tier 2, Table 3.2.1, indicates that the portion of the CWS (P25) that forms part of the containment boundary is safety-related, Quality Group B, and seismic Category I. The portion of the CWS located inside the RB or containment is nonsafety-related, Quality Group "D," and seismic Category II. The balance of the CWS is located in various parts of the TB, FB, EB, CB and RW and is nonsafety-related, Quality Group D and nonseismic.

While the NICWS is a nonsafety-related system, it performs defense-in-depth functions and is also subject to RTNSS as described in DCD Tier 2, Appendix 19A. As stated in DCD Tier 2, Section 19A.4.2, to address uncertainties in the performance of passive systems, an active system with the capability to provide backup functions is added to the scope of RTNSS. The portions of the FAPCS that provide low pressure injection and SPC are added in the scope for RTNSS Criterion C. Of the support systems needed for FAPCS, NICWS is used to cool various RTNSS components via room coolers. Therefore, part of NICWS is also designated as a RTNSS Criterion C system.

9.2.7.3 *Staff Evaluation*

The staff's review of the PSWS is based on guidance found in SRP Section 9.2.1 and applicable regulations such as GDC 2, 4, 5, 44, 45 and 46 and 10 CFR 20.1406. The CWS differs from that of the traditional BWR designs in that the ESBWR CWS removes heat only from nonsafety-related areas. Therefore, the portions of SRP Section 9.2.2 that apply to safety-related systems do not apply to the CWS.

DCD Tier 2, Table 6.2-47 and Section 6.2.4.3.2, describe the safety-related portions of the CWS at the containment penetrations. Section 6.2.4, of this report provides the staff evaluation of the containment penetration.

9.2.7.3.1 System Design Considerations

On March 19-20, 2009, the staff conducted a regulatory audit of the supporting information for DCD Tier 2, Section 9.2, including the PSWS (Section 9.2.1), RCCWS (Section 9.2.2), and NICWS (Section 9.2.7). The audit was primarily focused on the review of these systems with regard to the RTNSS and the ability to support cold shutdown operations. A summary of the audit, including participants and audit activities may be found in the ADAMS at Accession Number ML101250439. This audit is referred to several times throughout the remainder of this section.

A. CWS Classification and Quality Assurance Provisions

DCD Tier 2, Section 3.2, "specifies the classification of SSCs based on safety importance and other considerations. Section 3.2 of this report presents the staff's evaluation of the specified classification designations; this section of the staff's evaluation confirms that the appropriate classification designations are specified for the CWS consistent with the approach that is described in DCD Tier 2, Section 3.2 and that the designations properly reflect the regulatory oversight provisions that pertain to CWS/NICWS (RTNSS Criterion C) as discussed in DCD Tier 2, Section 19A.8. The staff reviewed simplified drawings, shown in DCD Tier 2, Figure 9.2-3, and confirmed that the classification designations on the drawings are consistent with those that are listed for the CWS in DCD Tier 2, Table 3.2-1. In particular, the following classification designations are specified in DCD Tier 2, Table 3.2-1 for CWS:

- The portion of the CWS/NICWS that is located at the containment and RB interface, which forms part of the containment boundary, is designated as Safety Class II, seismic Category I. This portion of the CWS is designated as Quality Group B. As discussed in DCD Tier 2, Section 3.2.4, this quality group generally applies to pressure-retaining portions and supports of containment and other mechanical items, requirements for which are within the scope of ASME Code, Section III. DCD Tier 2, Table 6.2-47 and Section 6.2.4.3.2.1, describe containment penetrations of the CWS (P25). Section 6.2 of this report evaluates CWS containment penetrations. The balance of the CWS is designated Safety Class N which is used for nonsafety-related applications. The CWS does not perform any safety-related functions and the "N" designation is therefore appropriate. The balance of the CWS is designated Quality Group D. As discussed in DCD Tier 2, Section 3.2.4, this quality group generally applies to nonsafety-related SSCs that satisfy specified industry codes and design standards and are subject to one or more significant licensing requirements or commitments. The staff concludes that these are the appropriate quality groups since the CWS does not perform a safety-related function and does not interface with any safety-related component other then containment as noted above.

- The portion of the CWS that is located in the RB and containment is a nonsafety-related system and is designated as seismic Category II. SSCs that perform no safety-related function, but whose structural failure or interaction could degrade the functioning of a seismic Category I item to an unacceptable level of safety or could result in incapacitating injury to occupants of the MCR, are designated seismic Category II. These items are designed to structurally withstand the effects of an SSE. Other portions of the CWS are located in the TB, RB, FB, RW, CB and EB and are designated as nonseismic. The staff concludes that the CWS has the appropriate seismic classifications since the CWS does not perform a safety-related function and does not interface with any safety-related component. However, since the CWS location is in the RB and it is designed to withstand an SSE, its

structural failure will not affect the safety function of any safety system or the MCR occupants.

- Revision 6 of the DCD specifies QA Requirement S for the CWS, as stated in the applicant's response to RAI 3.2-6 S02. Based on the RAI response, RTNSS components and systems identified in Revision 5 of the DCD as QA Requirement E are to be changed to QA Requirement S in Revision 6. QA Requirement S has special QA measures that apply during the design and procurement specification preparation processes, in accordance with procedures that will be established. The staff concludes that this is an appropriate QA group designation since the CWS does not perform a safety-related function and does not interface with any safety-related component. However, the CWS has RTNSS functions that are assured by applying the defense-in-depth principles of redundancy and physical separation to ensure adequate reliability and availability. In addition, the staff determined that Revision 6 of the DCD incorporated the RAI proposed change, which the staff finds to be acceptable.

B. GDC 2

Section 6.2.4 of this report evaluates containment isolation valves. Other than the containment isolation, the CWS is a nonsafety-related system. SRP Section 9.2.2 indicates that the requirements of GDC 2 can be met for a nonsafety-related system based on meeting Regulatory Position C.2 of RG 1.29 regarding nonsafety-related systems.

As a part of RAIs 9.2-12 and 9.2-12 S01, the staff asked the applicant to demonstrate that the CWS meets GDC 2. In response, the applicant stated that the CWS containment penetration and isolation valves are designed as seismic Category I. The CWS does have piping in the control room, but it is not possible for these components to result in an incapacitating injury to occupants of the control room because the CWS components are designed to remain functional during and following an SSE. Those portions of the system within seismic Category I buildings are designed as seismic Category II. Failure of the CWS will not compromise any safety-related system or component, and it will not prevent a safe shutdown. Therefore, the CWS satisfies the requirements of GDC 2. The staff reviewed the above RAI response and DCD Tier 2, Revision 6, Section 9.2.7 and Table 3.2-1. Based on the above, the staff finds that the CWS meets the guidance of Regulatory Position C1 regarding the safety-related portions of the CWS and Regulatory Position C.2 of RG 1.29 regarding the nonsafety-related portions of the CWS. Therefore, the CWS satisfies GDC 2. Accordingly, based on the above and the applicant's response, RAI 9.2-12 relating to the CWS is resolved.

C. GDC 4

SRP Section 9.2.2 provides guidance to review the CWS against GDC 4, as it relates to the dynamic effects associated with water hammer. As stated in DCD Revision 5, Section 9.2.7.1, the potential for water hammer is mitigated through the use of various system design and layout features, such as high point vents, valve cycle times, and surge tanks. The DCD also stated that the following design considerations address the effects of missiles, jet impingement, pipe whipping and discharge fluids:

- Pipe routing
- Piping design consideration, such as material section, pipe size, and schedule
- Protective barrier as necessary
- Appropriate supports and restraints

In RAI 9.2-21, the staff asked the applicant to discuss how the CWS meets GDC 4. In response, the applicant stated that the CWS is designed to mitigate the possibility of water hammer, as addressed in responses to RAIs 9.2-15 and 9.2-15 S01.

The CWS/NICWS is an RTNSS system. Electrical power is assumed to be unavailable for 72 hours and then returned to service for RTNSS systems. Restarting the CWS presents an opportunity for the dynamic effects associated with water hammer. In RAI 9.2-15, the staff requested that the applicant describe how the design of the CWS addresses water hammer so that the CWS can meet its post-72-hour RTNSS cooling function. In response, the applicant stated that proper system engineering design, along with operation and maintenance procedures are used to ensure that sufficient measures are taken to avoid water hammer. Surge tanks and air separators mitigate voiding. Surge tanks are also used in accordance with DCD Tier 2, Section 9.2.7.2, within the CWS, which provide a constant pump suction head and allow for thermal expansion of the CWS inventory. The CWS is a closed-loop system that does not drain down when isolated. In addition, in DCD Tier 2, Section 13.5.2, the applicant clarified that elements of ANSI/ANS-3.2-1994; R1999, addressing water hammer will be applied in the development of procedures for RTNSS systems.

The staff discussed water hammer considerations with the applicant during the March 19–20, 2009, audit, and the applicant addressed this topic in response to RAI 9.2-24. The staff asked the applicant to discuss the potential for water hammer, as well as the operating and maintenance procedures for avoiding water hammer in the CWS/NICWS. In response, the applicant provided the following provisions to mitigate water hammer:

- System design and layout features: Each NICWS Train (A and B), each of which has an air separator located before the chilled water primary pump suction headers with a vent to the surge tank of the respective NICWS train. The air separators remove entrained air and route this air to the vented surge tank.

- Valve cycle times: The applicant has guidance for valve actuation and stroke time development during system design to prevent water hammer and control instability while minimizing operation of pumps below minimum flow while the valves stroke open to establish system flowpaths.

- The surge tank location: This is the high point of the system which provides NPSH to the CWS pumps.

- CWS operation and maintenance procedures: These procedures incorporate necessary steps, such as proper line filling, to avoid water hammer.

Based on the staff’s review of the applicant’s response to RAIs 9.2-15 and 9.2-24, the staff finds that the applicant adequately addresses water hammer since the CWS/NICWS design incorporates water hammer mitigation features and components and operational procedures addressing water hammer concerns are to be developed for the RTNSS systems as part of COL Information Item 13.5-4-A. Accordingly, based on the above and the applicant’s responses, RAIs 9.2-15, 9.2-21, and 9.2-24 as they relate to water hammer are resolved. The staff finds that the CWS meets GDC 4 in accordance with the guidance of SRP Section 9.2.2.

D. GDC 5, 44, 45, and 46

The ESBWR design is a single-unit station, and the requirements of GDC 5 are not applicable to the single-unit design.

Based on the requirements of GDC 44, 45, and 46 and the guidance of SRP Section 9.2.2, the staff reviewed the CWS/NICWS against GDC 44, 45, and 46 to determine whether the CWS/NICWS is capable of removing heat from SSCs important to safety to a heat sink under normal operating and accident conditions and whether the design provides for inspection and operational testing.

DCD Tier 2, Section 9.2.7.1, states that, although the CWS/NICWS is a nonsafety-related system, it meets the intent of certain acceptance criteria of GDC 44, 45, and 46, as clarified by the following design considerations:

- Capability of transferring heat loads from SSCs to a heat sink via the RCCWS and under normal and accident conditions

- Component redundancy so the system remains functional assuming a single active failure coincident with a LOOP

- Capability to isolate components so system function is not compromised

- Design provisions to permit inspection and operational testing of components and equipment

The staff believes that those portions of the GDC 44 requirements that apply to the heat removal function under normal operating conditions apply to the CWS/NICWS. The PSWS, RCCWS, and CWS/NICWS are nonsafety-related.

The staff reviewed the CWS/NICWS in terms of the designed heat removal capability, component redundancy and single failure design, plant TS shutdown cooling requirements, and testing and inspection requirements, as described in DCD Tier 2, Section 9.2.2. The staff finds that the CWS/NICWS satisfies GDC 44, 45, and 46 with respect to its normal operation function. However, in a DBA, decay heat is transferred to the ICS/PCCS pools. The portions of the GDC 44 requirements that apply to a safety-related system to remove decay heat following an accident do not apply to the CWS. The staff finds that the design of the CWS satisfies the applicable portions of GDC 44, 45, and 46 based on the above review. The CWS design attributes, including system capability, reliability, heat transfer, pump NPSH, operating experiences, testing, and instrumentation and controls (which are related to the applicable GDCs), are further addressed below for the CWS RTNSS and cold shutdown functions.

E. Minimization of Contamination; 10 CFR 20.1406 and Radiation Monitoring

10 CFR 52.47(a)(6) and 10 CFR 20.1406 require applicants for standard plant design certifications to describe how facility design and procedures for operation will minimize contamination of the facility and the environment. SRP Section 9.2.2, Paragraph III.4.C specifies that provisions should be provided to detect radioactive leakage or contamination from one system to another.

DCD Tier 2, Revision 5, Section 9.2.7.1, states that the heat exchangers associated with the offgas system (OGS) handle potentially radioactive material at an operating pressure lower than the pressure of the water that cools it. Any tube leakage, therefore, results in a flow from the CWS to the OGS. DCD Tier 2, Section 9.2.7.4, identifies that samples of chilled water may be obtained for chemical analyses and that the system design ensures that the chilled water does not become radioactive during normal operation.

In RAI 9.2-28, the staff requested that the applicant address the requirements of 10 CFR 20.1406, since the DCD did not adequately discuss this issue or explain in detail the CWS operating pressures relative to the system coolers. In response, the applicant clarified that the offgas cooler-condenser operates at less than 138.9 kilopascal (kPa) (20 pound-force per square inch gauge (psig), and the CWS maximum operating pressure is approximately 861.8 kPa (125 psig) with a nominal pressure greater than 413.7 kPa (60 psig). Therefore, any postulated leakage during normal operating conditions will be from the CWS to the OGS. Leakage of CWS fluid into the OGS waste stream will be detected by an increased conductivity in condensate drain stream, as described in DCD Tier 2, Revision 9, Table 11.3-3. Since the OGS consists of two redundant trains of offgas cooler-condensers, an OGS train could be isolated if leakage were to be detected at the offgas cooler condenser.

The CWS pressure will also exceed the drywell pressure associated with the drywell cooling loads for the CWS during all anticipated operations. Therefore, any intersystem leakage will be out of chilled water into the drywell. An upper or lower drywell fan cooling unit can be isolated upon CWS leakage to isolate the component. Upon the occurrence of high drywell pressure, the CWS containment isolation valves will shut, isolating the CWS from potential contamination sources. In addition, the CWS surge tank levels are used to monitor losses of chilled water and detect inter-system leakage or intrusions into the CWS. A low-low surge tank level will alarm in the MCR. This alarm indicates that system leakage has exceeded makeup water capacity. A high-high surge tank level alarms in the MCR. This alarm indicates that there is inter-system leakage into the CWS. While the CWS is not expected to become contaminated, design provisions are included to allow periodic grab samples that could be analyzed to determine CWS activity levels. The applicant concluded in its RAI response that the CWS does not require installed radiation monitors to prevent contamination of the facility and the environment.

Based on above, the staff finds the CWS provisions relating to leakage detection and 10 CFR 20.1406 to be acceptable. Accordingly, based on the above and the applicant’s response, RAI 9.2-28 is resolved.

F. Protection from Probable Hazards

DCD Tier 2, Revision 5, Section 9.2.7.1, states that the CWS/NICWS RTNSS functions are assured by applying the defense-in-depth principles of redundancy and physical separation to ensure adequate reliability and availability, as described in DCD Tier 2, Section 19A.8.3.

In accordance with the policies referred to in Chapter 22 of this report, SSCs that are classified as RTNSS should be protected from the more probable hazards that exist. As previously discussed the CWS/NICWS is classified as RTNSS Criterion C. DCD Tier 2, Revision 9, Section 19A.8.3, indicates that RTNSS Criterion C systems incorporate the defense-in-depth principles of redundancy and physical separation to ensure adequate reliability and availability. DCD Tier 2, Section 19A.8.3, also indicates that RTNSS Criterion C systems and structures meet design standards to withstand wind and missiles generated from Category 5 hurricanes and that non-RTNSS systems that can adversely interact with RTNSS Criterion C systems are

designed to the same seismic requirements as the affected RTNSS system. Additionally, DCD Tier 2, Revision 9, Section 19A.8.3 indicates that RTNSS Criterion C equipment is qualified to IEEE Standard 344-1987 to demonstrate structural integrity.

The CWS/NICWS is classified as RTNSS Criterion C; therefore, systems and components are designed to the seismic requirements of IBC-2003 consistent with the SSE ground motion equal to two-thirds of the Certified Seismic Design Spectra.

As stated in the applicant's response to RAI 9.2-24, the PSWS, RCCWS and CWS/NICWS support PIP and defense-in-depth functions. DCD Tier 2, Revision 9, Section 9.2.2.2 describes that in the event of a LOPP, the RCCWS supports the FAPCS and the RWCU/SDC in bringing the plant to cold shutdown conditions within 36 hours, assuming the most limiting single active failure. Because the PSWS and RCCWS cooling water systems and the CWS/NICWS are also significant contributors to plant availability and plant investment protection, the ESBWR design is focused on ensuring that these systems are available and reliable.

In summary, the PSWS, RCCWS and CWS/NICWS are support systems to the FAPCS and are only included as an augmented system to address uncertainties in the defense-in-depth role of the FAPCS in providing a backup source of lower pressure injection and SPC. RTNSS Criterion C systems are not designed to the level of RTNSS Criterion B; however, RTNSS Criterion C systems are designed to the seismic standards of IBC-2003 consistent with the SSE ground motion equal to two-thirds of the Certified Seismic Design Spectra. The staff finds this graded design approach is acceptable considering the design function of the NICWS under the regulatory criteria for this nonsafety system.

G. CWS Capability and Reliability

In RAI 9.2-24, the staff requested the applicant to specifically address information concerning the CWS/NICWS functions that are subject to RTNSS, focusing on CWS/NICWS capability and reliability. The RAI included the following key points:

- The most limiting conditions upon which the CWS/NICWS design is based with the amount of excess margin built in to the design
- Clarification in the DCD descriptions, drawings and tables (including valves, cross-tie connections between trains, instrumentation logic and installed instruments)
- CWS/NICWS pump design, including pump recirculation protection, vortex and NPSH
- CWS/NICWS water hammer consideration
- CWS/NICWS failure modes and effects
- CWS/NICWS component testing and component reliability

To resolve this RAI, the staff audited supporting information for the CWS/NICWS on March 19 and 20, 2009, as discussed above. The response to RAI 9.2-24 addresses both the RAI and the audit findings. The remainder of this section discusses the results of the audit and the RAI response. The CWS/BOPCWS, which does not have any safety-related or RTNSS function, was not part of the scope of RAI 9.2-24 and was not discussed as part of the audit.

(1) Descriptive Information and Flow Considerations

The staff reviewed the DCD Tier 2, Section 9.2.7, CWS description and drawings to confirm that the design bases, flow paths, and components have been identified and described in sufficient detail to enable a complete understanding of the system design and operation. The staff found that it needed additional information in this regard and requested, in RAI 9.2-24, that the applicant revise Section 9.2.7 to address the following considerations:

- Cross-connect valves between BOPCWS and NICWS
- Nominal pipe sizes and system flow rates
- System ASME Code class breaks

The applicant addressed each of the above-noted items in detail during the March 19–20, 2009, audit and in its RAI response.

The CWS is divided into two independent chilled water subsystems, the NICWS and the BOPCWS. At the March 19-20, 2009, audit and in response to RAI 9.2-24, the applicant clarified the relationship between the two subsystems. The NICWS contains two redundant trains for active components, Train A and Train B. The NICWS redundant trains share passive components (e.g., piping, supports, manual shutoff valves). The BOPCWS is a single train with three pumps and three chillers. A normally shut manual cross-tie line connects the chilled water supply and return headers of the BOPCWS and NICWS. The manual valves may be opened to support maintenance activities. The applicant revised DCD Tier 2, Figure 9.2-3, to include these clarifications. The staff confirmed that Revision 6 of DCD Tier 2, Figure 9.2-3 incorporated this RAI proposed change. The staff finds that the response is acceptable since the revised simplified diagram clarifies the relationship between the two subsystems.

NICWS Train A and Train B, and the BOPCWS are each powered by separate buses. The active components in the NICWS Train A and Train B chilled water trains are identical. Each train contains two 50-percent chillers, two 50-percent primary pumps, one surge tank, one air separator, optional secondary pumps, and a shared chemical addition skid.

In the applicant’s response to RAI 9.2-24 for NICWS, system velocities for the piping system were defined to be approximately 4.6 m per second (15 ft per second) or less.

At the audit, the staff questioned the missing ASME Code class breaks for CWS containment isolation noted on DCD Figure 9.2-3. As listed in DCD Tier 2, Table 3.2-1, NICWS piping and valves (including supports) forming part of the containment boundary are safety class 2, Quality Class B, and seismic Category I. NICWS piping and components inside containment (and the RB) are classified as nonsafety-related, Quality Group D and seismic Category II. The applicant provided in the RAI response a markup of changes to the components in the RB indicating Quality Class B, and seismic Category I components (containment penetration area) and Quality Class D, and seismic Category II for the remaining CWS components inside containment consistent with DCD Tier 2, Table 3.2-1. The staff confirmed that Revision 6 of DCD Tier 2, Figure 9.2-3 has incorporated this RAI proposed DCD change related to the CWS piping classification inside the containment. The staff finds this response is acceptable since the applicant clarified the seismic, safety, and quality classifications of the various portions of the CWS.

The staff finds that the applicant has adequately addressed the CWS/NICWS system velocities. The staff finds them acceptable since the most limited piping velocities were approximately

4.6 m per second (15 ft per second) or less. In the staff's experience and in accordance with general engineering practice, piping velocities between 1.2 and 4.6 m per second (4-15 ft per second) are reasonable, thus long term internal pipe wear is expected to be minimal. The staff reviewed the remaining items noted above as part of the RAI response. The staff finds that the response to RAI 9.2-24 regarding CWS descriptive information and flow considerations is acceptable since the applicant clarified the basis for the design parameters included in the DCD. Accordingly, based on based on the above and the applicant's response, the flow consideration aspects of RAI 9.2-24 are resolved.

(2) Heat Transfer

The staff reviewed the DCD Tier 2, Section 9.2.7, CWS description and DCD Tier 2, Table 9.2.11, to confirm that the heat transfer and flow capabilities are adequately specified and that the bases for these values are fully explained.

The staff determined that it needed additional information in this regard and requested, in RAI 9.2-24, that the applicant revise DCD Tier 2, Section 9.2.7, to address heat transfer and address the amount of excess margin and to include uncertainties for wear and aging effects

The applicant stated that there is no specific NICWS alignment specified for the 24 hour or 36 hour plant cooldown conditions. DCD Tier 2, Table 9.2-11, listed a CWS chiller heat load of 4,850 kilowatts (kW) (16.55 MBTU/h) and total system heat load of 19,110 kW (6.5 x 10^7 BTU/hr) based on conservative preliminary calculations. This CWS heat load was used to size chillers as input for the TCCWS and RCCWS heat load calculations. NICWS and BOPCWS subsystem heat loads are considered bounding with final actual heat loads determined upon completion of HVAC calculations for the nuclear island and turbine island HVAC systems. As described in the applicant's chiller heat load calculations, the NICWS and BOPCWS chillers will be sized for a heat load of 4,638 kW (1,319 tons) per chiller.

The NICWS consists of two trains with two 50-percent chillers in each train, resulting in a total NICWS heat load of 9.3 MW (31.7 MBTU/hr). The system cooling loads and chilled water flows were developed using Advanced Boiling Water Reactor (ABWR) design heat loads and applying a 25 percent additional margin to selected loads with a 10 percent margin applied to the identified loads to account for unidentified loads in both the NICWS and BOPCWS. As reflected in the RCCWS heat load calculations, the CWS bounding heat load is 12.3 MW (42.0 MBTU/hr). Significant margins have been applied to the NICWS during the design process to account for uncertainties.

It was emphasized at the audit that not all of the CWS heat loads support RTNSS. Of all the CWS heat loads, the following list, which notes the relationship to RTNSS, was developed:

CWS/NICWS

- EB other (two HVAC units); RTNSS
- SDG room (two HVAC units); RTNSS
- RCCWS room (two HVAC units); RTNSS
- NICWS room (two HVAC units); RTNSS
- CB (two HVAC units); RTNSS
- RB (two HVAC units); RTNSS
- FB (two HVAC units); RTNSS
- Technical support center (TSC) (two HVAC units); non-RTNSS

- Service air system (SAS) room (two HVAC units); non-RTNSS
- Drywell cooling system (DCS) area (two HVAC units); non-RTNSS

CWS/BOPCWS

- RW building; non-RTNSS
- TB; non-RTNSS
- Other loads; non-RTNSS

Based on the above, the staff finds that there is sufficient design margin between the capacity of the seven CWS chillers and the maximum heat loads. Of these seven chillers, four are designated as RTNSS chillers. In addition, for RTNSS support, which includes the FAPCS and the SDGs, the maximum heat loads are bounded by design margin between the heat exchanger capacity and the maximum heat loads. The staff finds that the response to RAI 9.2-24 regarding heat transfer is acceptable, since the heat transfer capability of the CWS includes sufficient margin to support normal plant operations and RTNSS support. Accordingly, based on the above and the applicant's response, the heat transfer aspects of RAI 9.2-24 are resolved.

(3) <u>Single Failure and Backup Power Considerations</u>

DCD Tier 2, Section 9.2.7.1, states that the NICWS has RTNSS functions as described in DCD Tier 2, Appendix 19A, which provides the level of oversight and additional requirements to meet RTNSS functions. Performance of RTNSS functions is assured by applying the defense-in-depth principles of redundancy and physical separation to ensure adequate reliability and availability as described in DCD Section 19A.8.3. Chapter 22 of this report documents the review of RTNSS.

DCD Section 9.2.7.1 states that the CWS/NICWS is designed so that a single active failure or malfunction of one NICWS train does not affect system functionality. In case of failure, the system automatically generates an isolation signal.

The following actions are relied upon in case of a train isolation signal:

- Close cross-tie isolation valves
- Start up the chillers and pumps on standby
- Start up the AHUs of served by the NICWS
- Start up the second fans in the drywell cooling system

In addition, the following events require the automatic train isolation signal:

- Low level signal in surge tanks (chilled water leakage exceeding makeup capacity)
- LOPP

During a LOPP, the NICWS is automatically powered from two nonsafety-related onsite SDGs.

Although the two NICWS trains are normally cross-connected, the staff determined that clarification was needed for the case in which offsite power is not available and requested, in RAI 9.2-24, that the applicant revise DCD Tier 2, Section 9.2.7, to address single failure.

In response to RAI 9.2-24, the applicant clarified that chilled water is supplied from either train to a common header, thus distributing chilled water to the NICWS loads throughout the facility via a single piping distribution loop. NICWS chilled water is supplied by both chilled water trains during normal operation with one primary pump and chiller in service on each train and the other primary pump and chiller set in standby. A normally shut manual cross-tie line connects the chilled water supply and return headers of the BOPCWS and NICWS. The manual valves may be opened to support maintenance activities. In the event of a LOCA, the only safety-related function of the NICWS is to close the NICWS containment isolation valves. The CWS automatically performs a containment isolation function by closing its containment isolation valves upon receipt of an isolation signal from the LD&IS.

As described in DCD Tier 2, Revision 9, Section 9.3.6.1, the instrument air system (IAS) is designed to ensure that failure of the IAS does not compromise any safety-related system or component and that it does prevent a safe shutdown. Pneumatically operated devices are designed fail-safe and do not rely on a continuous air supply under emergency or abnormal conditions. The importance of nonsafety-related compressed air supplies was evaluated relative to the criteria for special RTNSS in DCD Tier 2, Appendix 19A, and does not meet the criteria for special regulatory treatment.

Based on the staff's review of the response to RAI 9.2-24, the staff finds that the applicant properly addressed the single-failure consideration through the redundancy of the design, the availability of components' emergency power, and components failure position on a LOPP event. The design redundancy of the CWS/NICWS system provides for adequate system reliability. In addition, train independence ensures that single failure of any NICWS train will not impact the other train. The staff finds that the response to RAI 9.2-24 regarding single failure is acceptable, since the applicant clarified how the DCD includes the single-failure attributes of the NICWS. Accordingly, based on the above, the applicant's responses, and DCD changes, the single failure aspects of RAI 9.2-24 are resolved.

(4) <u>Chilled Water System/Nuclear Island Chilled Water System Pump Net Positive Suction Head</u>

As described in DCD Tier 2, Revision 9, Section 9.2.7.2, the surge tanks provide a constant pump suction head and allow for thermal expansion and contraction of the chilled water inventory. Surge tanks also provide NPSH to the CWS pumps and maintain system pressure above vapor pressure to mitigate voiding. The tanks are located above the highest system point and the use of sloped piping minimizes the potential for air binding. The MWS provides makeup to the chilled water inventory through an automatic level control valve to the surge tanks. In addition, DCD Tier 2, Revision 9, Section 9.2.7.5, identifies that the level transmitters in the surge tank standpipes monitor the surge tank levels to ensure that sufficient NPSH is available for pump operation.

The staff requested in RAI 9.2-24, that the applicant clarify NPSH availability and provide additional information, regarding design alarms features in the MCR available to the operators. In addition, the staff asked the applicant to revise DCD Section 9.2.2 to include this information.

In response to RAI 9.2-24, the applicant clarified that the NICWS includes one surge tank per train provided at the highest point of each NICWS train. The surge tanks are connected to each NICWS train suction header to maintain available static head and adequate NPSH for the primary pumps. The surge tanks remove air and gases coming out of solution for this closed system and are designed with sufficient makeup capacity to accommodate design leakage from

the system. The CWS surge tank levels are used to monitor losses of chilled water and detect intersystem leakage or intrusions into the CWS. Low-low surge tank level alarms in the MCR. This alarm indicates that system leakage has exceeded makeup water capacity. High-high surge tank level alarms in the MCR. This alarm indicates that there is intersystem leakage into NICWS. The level transmitters in the surge tank standpipes monitor the surge tank levels to ensure that sufficient NPSH is available for pump operation. The applicant provided in the RAI response a DCD markup of Section 9.2.7.2 indicating that the surge tanks are designed with sufficient make-up capacity to accommodate design leakage from the system. The staff confirmed that Revision 6 of the DCD incorporated this RAI proposed change. The staff finds this change to be acceptable.

The staff finds that the responses to RAIs 9.2-23 and 9.2-24 regarding CWS pump NPSH are acceptable, since the applicant clarified how sufficient NPSH is assured. The applicant clarified the design features of the CWS to assure NPSH, which include the CWS surge tank and its system position (high point of the system), and instrumentation which detects a low-low surge tank level. Accordingly, based on the above, the applicant’s responses, and the DCD changes, the NPSH aspects of RAI 9.2-24 are resolved.

(5) Operating Experience

DCD Tier 2, Revision 5, Table 1.11-1, which identifies the resolution to NUREG–0933, “Resolution of Generic Safety Issues,” Table II task action plan items, new generic issues, human factors issues and Chernobyl issues, discusses the following generic issue related to CWS:

- New Generic Issue 143, “Availability of Chilled Water System and Room Cooling,” is identified in DCD Tier 2, Table 1.11-1. This new issue is related to problems with safety system components and control systems experienced by several nuclear plants that resulted from a partial or total loss of HVAC systems. The applicant stated in the DCD that the CWS is nonsafety related and provides chilled water to the cooling coils of air conditioning units and other coolers in the RB portion of the plant, but has no safety-related function. In addition, the failure of the CWS does not compromise any safety-related system or component, and it does not prevent a safe shutdown of the plant.

For the ESBWR passive design, the staff finds that the CWS has no safety-related function (expect for containment isolation), but has RTNSS functions to provide post-72-hour cooling for HVAC. The performance of RTNSS functions is assured by applying the defense-in-depth principles of redundancy and physical separation to ensure adequate reliability and availability with proper level of oversight and the additional requirements described in DCD Tier 2, Appendix 19A. Accordingly, the staff finds that New Generic Issue 143 is resolved for the CWS.

DCD Tier 2, Revision 5, Table 1C-1, discusses GL 96-06. The NRC issued this GL to address the potential for (1) water hammer or two phase flow in cooling water systems penetrating the containment and (2) thermally induced over-pressurization of isolated water-filled piping sections in containment that could jeopardize the function of accident mitigation systems and could lead to a loss of containment integrity. The applicant clarified its resolution of GL 96-06 in response to RAI 6.2-170 and modified DCD Tier 2, Revision 5, Table 1C-1, to state the following:

> Passive containment cooling system (PCCS) provides containment air cooling during design basis accidents as described in DCD Tier 2 Sections 6.2.1,

"Containment Functional Design," and 6.2.2, "Passive Containment Cooling System," and is not subject to water hammer effects. The chilled water system provides cooling water to the drywell cooling system during normal operation, and is isolated on a LOCA signal as discussed in Sections 9.2.7.5 and 6.2.4.3.2.1, "Influent Lines to Containment." Fluid-filled piping associated with containment penetrations that automatically isolate during DBAs is designed in accordance with ASME Code Section III to accommodate thermal transient loadings as described in Section 3.9.3.4, "Other Components," and Table 3.9-2. "Load Combinations and Acceptance Criteria for Safety-Related, ASME Code Class 1, 2 and 3 Components, Component Supports, and Class CS Structures."

The staff concludes that GL 96-06 does not apply to the CWS/NICWS, since the system is isolated upon a LOCA signal. Section 6.2.2 of this report discusses GL 96-06 further.

(6) Periodic Inspections and Testing

As discussed in System Design Consideration D above, the applicant demonstrated in DCD Tier 2, Revision 5, Section 9.2.7.1, that the CWS/NICWS satisfies GDC 45 and 46 because the design of the CWS/NICWS includes design provisions to permit inspection and operational testing of components and equipment.

DCD Tier 2, Section 9.2.7.4, describes the applicant's provisions for periodic inspection of major components to ensure the capability and integrity of the system. Local display devices are provided to indicate all vital parameters during testing and inspections.

Periodic inspections and testing are important for assessing and maintaining the capability and reliability of the CWS/NICWS to perform its defense-in-depth functions over the life of the plant. The CWS/NICWS design bases indicate that provisions are included to permit inspection of components and equipment. The system description also indicates that valves are arranged for ease of in-service inspection. DCD Tier 2, Section 9.2.7.4, further states that provisions are made for periodic inspection of components to ensure the capability and integrity of the system. The determined periodic inspection and testing was determined to be incomplete; therefore, the staff requested in RAI 9.2-24 that the applicant revise DCD Tier 2, Section 9.2.7.

The applicant's response to RAI 9.2-24 noted that maintenance, testing, and operating procedures will include provisions for regular inspection testing and maintenance of valves to prevent degradation over time. As described in DCD Tier 2, Revision 9, Sections 19A.8 and 19A.8.4.9, all RTNSS systems are within the scope of the Design Reliability Assurance Program, as directed by DCD Tier 2, Revision 9, Chapter 17, which will be incorporated into the Maintenance Rule Program. The Maintenance Rule, 10 CFR 50.65, requires performance monitoring of SSCs that are not safety-related but are relied upon to mitigate accidents or transients, are used in EOPs, or whose failure could prevent safety-related SSCs from performing their safety-related function or could cause a reactor scram or actuation of a safety-related system. Such SSCs may include RTNSS components.

The staff finds that the RAI response is acceptable since the CWS/NICWS will be monitored under the Maintenance Rule Program, which includes the maintenance of valves to prevent degradation over time. For the CWS/NICWS and other RTNSS systems, the Maintenance Rule Program ensures that unacceptable risk is detected and appropriate actions are taken. Accordingly, based on the above and the applicant's response, the inspection and testing aspects of RAI 9.2-24 are resolved.

(7) Instrumentation, Controls, and Alarms

DCD Tier 2, Revision 9, Section 9.2.7.5, states that the CWS is operated and monitored from the MCR. Major system parameters are indicated in the MCR. Other instrumentation that was briefly described includes the CWS chiller controls and monitoring instruments and surge tank level instruments. The CWS may be controlled from the RSS and the chillers have local control panels.

Chiller package protective controls and monitoring instruments indicate high and low oil pressure, condenser pressure, high and low chilled water temperature and flow, high and low condenser water temperature and flow, and unit diagnostics.

In RAI 9.2-24, the staff asked the applicant to revise the DCD figures to show header temperature and pressure detectors. In response to RAI 9.2-24, the applicant clarified the alarms for the CWS. The surge tanks are provided with level controlled demineralized water makeup valves and high/low level alarms in the MCR. The CWS surge tank levels are used to monitor losses of chilled water, and detect inter-system leakage or intrusions into the CWS. Low-low surge tank level will alarm in the MCR. This alarm indicates that system leakage has exceeded makeup water capacity. High-high surge tank level alarms in the MCR. This alarm indicates that there is inter-system leakage into the CWS. The level transmitters in the surge tank standpipes monitor the surge tank levels to ensure that sufficient NPSH is available for pump operation. The applicant provided a markup of DCD Tier 2, Section 9.2.7.5, highlighting the proposed changes to the surge tank alarms and addressing NPSH. The staff confirmed that Revision 6 of the DCD incorporated this RAI proposed change. The staff finds that the RAI response is acceptable since the applicant clarified the description of the CWS alarms in the DCD. Accordingly, based on the above and the applicant's response, the instrumentation and controls aspects of RAI 9.2-24 are resolved.

Based on the above, the staff finds the CWS instrumentation, controls, and alarms acceptable.

9.2.7.3.2 COL Information

The applicant identified no COL information items in Section 9.2.7.6. The staff finds that there are no relevant COL information items that need to be developed as part of the DCD.

9.2.7.3.3 Availability Controls

As discussed in DCD Tier 2, Revision 9, Section 19A.8.1, regulatory oversight is applied to each system that is designated as RTNSS to ensure adequate reliability and availability to perform RTNSS functions. DCD Tier 2, Section 19A.8.1 also indicates that Maintenance Rule performance monitoring is specified for all RTNSS functions, and that additional oversight for support systems is described in the ACM. DCD Tier 2, Table 19A-2, identifies that the NICWS is a support system and that the NICWS 'Availability Controls' is the 'Maintenance Rule,' which means that the availability of the NICWS is addressed by the Maintenance Rule performance monitoring rather than by a specific ACM entry.

The NICWS is subject to the ACM through the systems it supports. DCD Tier 2, Revision 9, Table 19A-2 classifies the PSWS and the RCCWS as support systems for the SDGs and the NICWS. The NICWS supports the building HVAC, which supports the FAPCS. The FAPCS is the RTNSS system that is relied upon for active mitigation and the SDGs are support systems for FAPCS. Of these systems, the ACM only specifies availability controls for the SDGs in AC

3.8.1 and AC 3.8.2 and for FAPCS in AC 3.7.2 and in AC 3.7.3. Therefore, the PSWS, RCCWS, and NICWS are support systems that are subject to the ACs that are specified for the SDGs and FAPCS.

ACM 1.1 states that for the term "AVAILABLE-AVAILABILITY," a system, subsystem, train, division, component, or device shall be considered available or to have AVAILABILITY when it is capable of performing its specified risk informed function or functions and when all necessary attendant instrumentation, controls, normal or emergency electrical power, cooling and seal water, lubrication, and other auxiliary equipment that support operation of the system, subsystem, train, division, component, or device with respect to its specified risk informed function or functions are also capable of performing their related support function or functions. Since the PSWS supports the RCCWS which supports NICWS, FAPCS, and SDGs, if the PSWS or the RCCWS becomes unavailable, then the systems they support become unavailable and the applicable ACM action statements would apply.

Based on the above, the staff finds the ACs for the NICWS acceptable since the NICWS is subject to the Maintenance Rule and indirectly subject to the ACM because the NICWS is an RTNSS support system and its availability is indirectly covered by the ACs for the FAPCS and SDGs.

9.2.7.3.4 Inspections, Tests, Analyses, and Acceptance Criteria

In DCD Tier 1, Revision 3, Section 2.12.5, the applicant revised the CWS ITAAC to remove large portions of information, including a system description and system drawings, design commitment, and scope of ITAAC. The staff determined that the removal of CWS ITAAC information in Tier 1 is not acceptable. In RAIs 22.5-1 and 22.5-1 S01, the staff requested that the applicant review and revise DCD Tier 1 to include the CWS in Tier 1 for ITAAC. The applicant responded to the RAI and provided the requested Tier 1 system description, ITAAC, and drawing for the CWS in revised DCD Tier 1 Section 2.12.5. Accordingly, based on the above, the applicant's responses, and DCD changes, RAI 22.5-1 is resolved. Section 22 of this report also discusses the resolution of these RAIs.

The applicant addressed ITAAC details as part of its response to RAI 9.2-24 and during the March 19–20, 2009, audit. The applicant responded to the staff's questions regarding the lack of specific details for the RTNSS Criterion C acceptance criteria as follows:

> PSWS, RCCWS and NICWS provide supporting functions for FAPCS suppression pool cooling and low pressure injection modes, and thus meet RTNSS Criterion C. RTNSS C SSCs are assumed to be available at the time of the initiating event. Validation of these RTNSS functions is assured by Tier 1 ITAAC (Section 2.12.7 PSWS; Section 2.12.3 RCCWS; Section 2.12.5 NICWS) where testing of the PSWS /RCCWS / NICWS demonstrate flow to the RCCWS (nuclear island chillers, diesel generators and FAPCS island chillers, diesel generators and FAPCS). The ESBWR RTNSS Criterion C Cooling Water System ITAAC scope and detail differs from that associated with validation of RTNSS Criterion B functions. The ESBWR is designed so that safety-related passive systems are able to perform all safety functions for at least 72 hours, after initiation of a design basis event, without the need for active systems or operator actions. After 72 hours, nonsafety-related systems (RTNSS Criterion B) can be used to replenish the passive systems or to perform core cooling and containment integrity functions directly. RTNSS Criterion B ITAAC (e.g. FAPCS

section 2.6.2 Item 7 and fire protection system (FPS) section 2.16.3 item 7) provides a greater assurance of function.

The staff finds that the RAI response is acceptable since the Tier 1 information is adequate and reasonable based on the ESBWR graded approach for this RTNSS Criterion C, nonsafety-related system. For RTNSS functions of the NICWS, flow is verified to key RTNSS equipment such as EB HVAC units, DG room HVAC units, RCCWS room HVAC units, NICWS room HVAC units, CB HVAC units, RB HVAC units, and FB HVAC units. In addition, as-built verification is performed, selected controls from the MCR are verified, and NICWS system flow indication is verified to be available in the MCR. Accordingly, based on the above and the applicant's response, the ITAAC aspects of RAI 9.2-24 are resolved.

9.2.7.3.5 Initial Test Program

Section 14.2 of this report evaluates the initial test program for the ESBWR; evaluation of the CWS initial test program in this section is an extension of the evaluation provided in Section 14.2.

DCD Tier 2, Revision 9, Section 14.2.8.1.24, describes the pre-operational test program for the CWS. The staff finds the objective of the CWS pre-operational test program to be appropriate since its objective is to verify the ability of the CWS to supply the design quantities of chilled water at the specified temperatures to the various cooling coils of the HVAC systems serving rooms and areas that rely upon conditioned air. Because of insufficient heat loads during the pre-operational phase, it is not then possible to fully evaluate the capacity of the chiller units with inlet and outlet temperatures and flow data. The final chiller evaluation will be performed in the startup phase. While the test specifications are written in very general terms to address the considerations that apply to CWS, the staff considers the approach for this nonsafety-related to be acceptable because the COL applicant will develop test procedures in accordance with COL Information Item 14.2-3-A, "Test Procedures."

During of review DCD Tier 2, Revision 5, the staff determined that additional information and specificity was necessary in some respects and requested, in RAI 9.2-24, that the applicant revise DCD Tier 2, Section 14.2.8.1.24 to address the testing of the CWS. In response to RAI 9.2-24 and in discussions during the March 19-20, 2009, audit, the applicant clarified the basis for its pre-operational test program. Preoperational startup testing will verify proper chiller performance; operation of system valves, including timing, under expected operating conditions; and proper operation of pumps and motors in all design operating modes. This includes startup of a standby loop or actuation following a loss of power with proper operation ensuring that water hammer does not occur. Procedures will include provisions to prevent void formation during periods of standby. CWS pump test and integrated flow tests will ensure that discharge check valve leakage will not impact pump or system flow performance.

The applicant provided a markup of DCD Tier 2, Figure 9.2-3, to reflect the pump check valves located downstream of the primary and secondary pumps (as applicable). The staff confirmed that Revision 6 of the DCD incorporated this RAI proposed change.

As previously stated, in DCD Tier 2, Section 13.5.2, the applicant clarified that those elements of ANSI/ANS-3.2-1994; R1999 which address water hammer will be applied in the development of procedures for RTNSS systems.

The staff finds that the RAI 9.2-24 response is acceptable since the level of testing addresses system performance, minimum NPHS, chiller and pump performance, instrumentation and interlocks, and water hammer. Therefore, DCD Tier 2, Section 14.2, need not describe additional testing. Based on the applicant's response, the initial test program aspects of RAI 9.2-24 are resolved.

9.2.7.4 *Conclusion*

For the reasons set forth above, the staff finds that the CWS complies with the requirements of GDC 2, 4, 44, 45, and 46 and 10 CFR 20.1406. The staff also finds that the design of the CWS/NICWS satisfies established NRC policies with respect to its RTNSS Criterion C function.

9.2.8 Turbine Component Cooling Water System

9.2.8.1 *Regulatory Criteria*

The staff reviewed the TCCWS based on the guidance provided in SRP Section 9.2.2, Revision 4. Staff acceptance of the design is based on meeting the following requirements:

- GDC 2, as it relates to the capability of the design to maintain and perform its safety function following an earthquake
- GDC 4, as it relates to the dynamic effects associated with water hammer
- GDC 5, as it relates to the capability of shared systems and components important to safety to perform required safety functions
- GDC 44, as it relates to transferring heat from SSCs important to safety to a heat sink
- GDC 45, as it relates to the design provisions to permit inspection of components and equipment
- GDC 46, as it relates to the design provisions to permit operational testing of components and equipment
- 10 CFR 20.1406 as it relates to minimization of contamination

9.2.8.2 *Summary of Technical Information*

DCD Tier 2, Revision 9, Section 9.2.8, describes the TCCWS. The TCCWS is a single-loop system and consists of one surge tank, one chemical addition tank, pumps, heat exchangers connected in parallel, associated coolers, piping, valves, controls, and instrumentation. DCD Tier 2, Revision 9, Table 9.2-12, shows the system parameters, and DCD Tier 2, Figure 9.2-4, shows the system configuration. Heat is removed from the TCCWS and transferred to the nonsafety-related PSWS. The system is designed to Quality Group D guidelines.

During normal power operation, the TCCWS pumps circulate water through one side of the TCCWS heat exchangers in service. The heat from the TCCWS is rejected to the PSWS that circulates water on the other side of the parallel plate TCCWS heat exchangers.

9.2.8.3 *Staff Evaluation*

The staff reviewed the design of the TCCWS in accordance with applicable provisions of SRP Section 9.2.2. The ESBWR TCCWS is a nonsafety-related system because the TCCWS removes heat only from the nonsafety-related systems and components. Therefore, the portions of SRP Section 9.2.2 that apply to safety-related systems do not apply to the TCCWS.

The ESBWR design is a single-unit station, and the requirements of GDC 5 are not applicable for the single-unit design.

For a nonsafety-related system to meet the requirements of GDC 2, SRP Section 9.2.2 indicates that acceptance depends on meeting the guidance of the portions of Regulatory Position C.2 of RG 1.29 regarding nonsafety-related systems.

As a part of RAIs 9.2-12, 9.2-22, and 9.2-22 S01, the staff asked the applicant to demonstrate that the TCCWS meets the requirements of GDC 2. In response, the applicant stated that the TCCWS is a nonsafety-related, non-RTNSS system. The TCCWS is not relied upon to transfer heat from safety-related or RTNSS SSCs. Its failure will not prevent the performance of any safety function or result in any incapacitating injury to occupants of the MCR. The staff determined that Regulatory Position C.1 of RG 1.29 does not apply to the TCCWS. Based on the information that the TCCWS is not relied upon to transfer heat from safety-related or RTNSS SSCs and that its failure will not prevent the performance of any safety function or result in any incapacitating injury to occupants of the MCR, the staff finds that Regulatory Position C.2 of RG 1.29 is satisfied. Therefore, TCCWS meets the requirements of GDC 2. Accordingly, based on the above and the applicant's response, RAIs 9.2-12 and 9.2-22 as related to GDC 2 for the TCCWS are resolved.

The staff reviewed the TCCWS and issued RAIs 9.2-12 S01 and 22.5-2 to determine if the applicant had properly determined whether the TCCWS is not an RTNSS system. In DCD Tier 2, Revision 3, the applicant identified the TCCWS as an RTNSS system to provide post-72-hour cooling to the TB HVAC. However, the applicant stated in responses to RAIs 9.2-12 S01 and 22.5-2 that, after a reevaluation of the RTNSS, the applicant changed its determination because the TCCWS does not remove heat from any safety-related systems or from other RTNSS systems. In DCD Tier 2, Revision 6, the applicant indicated that a portion of the CWS that is cooled by the RCCWS, not the TCCWS, provides for the post 72-hour cooling function to the TB HVAC. Based on the above, the staff concluded that the TCCWS is not an RTNSS system because the TCCWS is not relied upon to remove heat from components being used for post-72-hour cooling. The staff finds that the responses RAI 9.2-12 and RAI 22.5-2 are acceptable since the applicant clarified that the TCCWS is not a RTNSS system and provided a basis for the change in classification. Accordingly, based on the above and the applicant's response, RAIs 9.2-12 and 22.5-2 relating to the RTNSS determination is resolved. In addition, since the TCCWS is not a safety-related system or a RTNSS system (i.e., it is not important to safety), GDC 4 is not applicable to the TCCWS.

In RAIs 9.2-7 S01 and 9.2-7 S02, the staff questioned the applicant as to whether or not the TCCWS met the requirements of GDC 44, 45, and 46. In responses, the applicant stated that the TCCWS is not required to meet NRC regulations, safety goal guidelines, and containment performance goal objectives. Additionally, the TCCWS is not a system used to transfer heat from SSCs important to safety that are RTNSS or safety-related. Therefore, the requirements of GDC 44, 45, and 46 do not apply to the design of the TCCWS. The staff reviewed DCD Tier 2, Table 9.2-12, and determined that the TCCWS does not provides cooling to SSCs important to

safety under normal or accident conditions. Based on the above, the staff has finds that the TCCWS is adequately designed for its function, even though GDC 44, 45, and 46 are not applicable. The staff finds that the response is acceptable since the applicant clarified that the TCCWS is not a system used to transfer heat from RTNSS or safety-related SSCs and thus the requirements of GDC 44, 45, and 46 do not apply to the TCCWS. Based on the above and the applicant's response, RAI 9.2.7 as related to GDC 44, 45, and 46 for the TCCWS is resolved.

The staff reviewed the design of the TCCWS for conformance to 10 CFR 20.1406. DCD Tier 2, Revision 9, Table 12.3-18, which addresses RG 4.21 design objectives and applicable DCD section information, describes provisions related to the TCCWS for the following objectives:

- Minimizing leaks and spills (Design Objective 1)

- Decreasing the spread of contaminant from the source (Design Objective 4)

To meet these two design objectives, the TCCWS utilizes plate and frame type heat exchangers and this design mitigates cross-contamination between TCCWS and the PSWS. The staff agrees that the plate and frame heat exchanger is an improvement in design verses the shell and tube type heat exchangers (known for tube leakage) and cross-contaminating is less of an issue since the plate and frame type utilizes corrugated plates. As discussed in Section 9.2.1 of this report regarding the PSWS, leakage through holes or cracks in the plates is not considered credible based on industry experience with plate type heat exchangers. Accordingly, the staff finds that the design provisions for the TCCWS meet the requirements of 10 CFR 20.1406 and conforms to the guidelines of RG 4.21. Section 12.4 of this report further addresses the ESBWR design in accordance with 10 CFR 20.1406.

9.2.8.4 *Conclusion*

Based on the above discussion, the staff finds the design of the TCCWS acceptable and the requirements of GDC 2 and 4 and 10 CFR 20.1406 satisfied.

9.2.9 Hot Water System

In DCD Tier 2, Revision 9, Section 9.2.9, the applicant states that the hot water system for the ESBWR design has been eliminated and its function replaced with electric (in-duct) heating coils for most building loads and radiant (wall mounted) heating coils for localized heating load. Therefore, the staff's evaluation for the hot water system is deleted. In addition, RAI 9.2-14 was being tracked as an open item in the SER with open items. RAI 9.2-14 was associated with the hot water system but is no longer applicable, given this design change, and is therefore resolved.

9.2.10 Station Water System

9.2.10.1 *Regulatory Criteria*

The staff determined that no current guidance provided in the SRP is directly applicable to the review of the station water system. The staff based its review on portions of the relevant regulatory guidance such as SRP Section 9.2.1, "Station Service Water System," Revision 5. The staff evaluated applicable portions of GDC 2, 4, 44, 45, and 46 and 10 CFR 20.1406 as potential regulatory requirements.

9.2.10.2 *Summary of Technical Information*

DCD Tier 2, Revision 9, Section 9.2.10, describes a conceptual design of the station water system. The station water system provides a supply of water for the following services:

- Makeup water to the circulating water system (CIRC) cooling tower basin
- Makeup water to the PSWS cooling tower basins
- Feedwater to the MWS
- Fill water to the FPS

The station water system consists of the following subsystems:

- Plant cooling tower makeup system
- Pretreated water supply system

The plant cooling tower makeup system provides makeup water to the cooling tower basins for both the PSWS and the CIRC. The supply of water makes up for losses resulting from evaporation, drift, and blowdown from the cooling towers. In addition, the plant cooling tower makeup system provides makeup water to replace water used for PSWS strainer backwash.

The pretreated water supply system filters and chemically pretreats water supplied to the MWS for further treatment for use as demineralized water. The pretreated water supply system also supplies water to the FPS for filling the primary firewater tanks and for maintaining pressure in the yard loop. In addition, the pretreated water supply system provides PSWS cooling tower makeup as an alternate to the plant cooling tower makeup system.

Instruments are provided for monitoring system parameters in the MCR. Pretreated station water storage tank high and low levels, and low suction pressure for each pump taking suction from the storage tank are alarmed to the MCR. Provisions for taking water samples are included.

The COLA FSAR will replace the above CDI for the station water system with site-specific design information.

9.2.10.3 *Staff Evaluation*

In DCD Tier 2, Revision 9, Section 9.2.10.3, the applicant stated that the station water system has no safety design basis and does not perform any safety-related function. Failure of the station water system does not affect any safety-related systems or components.

The applicant states that COLA FSAR will provide the site-specific design information, while the DCD provides the CDI. The staff agrees with the applicant that the nature of the system is site-specific and will review the design of the site-specific design of the station water system in COL applications. The staff may need to evaluate the applicable portions of GDC 2, 4, 44, 45, 46 when the plant-specific design information is available.

In RAI 9.2-16, the staff asked the applicant to identify a COL information item for the site-specific station water system design. In response, the applicant stated that it is unnecessary to assign COL action items to CDI in the DCD, since the need to address this information is specified in RG 1.206. The staff found the applicant's justification for excluding a COL information item to address the CDI to be acceptable. Accordingly, based on the above and the

applicant's response, RAI 9.2-16 is resolved. The staff will review the site-specific design of the station water system in COL applications.

The staff reviewed the design of the station water system for conformance to 10 CFR 20.1406. DCD Tier 2, Revision 9, Table 12.3-18, does not identify any specific station water system design features to address conformance to RG 4.21 design objectives. The staff will review the site-specific design of the station water system in COL applications for conformance to 10 CFR 20.1406 (if necessary) since the ESBWR station water system is considered CDI. Section 12.4 of this report provides the evaluation of ESBWR generic design features for conformance to RG 4.21 and 10 CFR 20.1406.

9.2.10.4 *Conclusion*

Based on the above, the staff finds that the site-specific design of the station water system is not within the scope of the ESBWR design certification application and will be reviewed in connection with COL applications referencing the ESBWR design.

9.3 Process Auxiliaries

9.3.1 Compressed Air System

9.3.1.1 *Regulatory Criteria*

The staff reviewed the ESBWR compressed air system (CAS) in accordance with SRP Section 9.3.1, Revision 2. The staff reviewed DCD Tier 1, Revision 9, Section 2; DCD Tier 2, Revision 9, Section 9.3.1; and various parts of other DCD Tier 2 sections (e.g., Section 19A). The staff's acceptance of the CAS is based on meeting the relevant requirements of the following regulations:

- GDC 1 in part requires that SSCs important to safety shall be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the function to be performed. Where generally recognized codes and standards are used, they shall be identified and evaluated to determine their applicability, adequacy, and sufficiency and shall be supplemented or modified as necessary to ensure a quality product in keeping with the required safety function.

- GDC 2 requires in part that SSCs important to safety shall be designed to withstand the effects of natural phenomena such as earthquakes.

- GDC 5 requires that SSCs important to safety shall not be shared among nuclear power units unless it can be shown that such sharing will not significantly impair their ability to perform their safety functions, including, in the event of an accident in one unit, an orderly shutdown and cooldown of the remaining units.

- 10 CFR 50.63 relates to the ability of a plant to withstand for a specified duration and recover from a SBO.

- 10 CFR 52.47(b)(1), which requires that a design certification application contain the proposed ITAAC that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant

that incorporates the design certification is built and will operate in accordance with the design certification, the provisions of the Atomic Energy Act, and the NRC's regulations.

9.3.1.2 *Summary of Technical Information*

The CAS consists of the IAS, the SAS, the high-pressure nitrogen supply system (HPNSS), and the containment inerting system (CIS). The applicant described the IAS, SAS, HPNSS, and CIS in DCD Tier 2, Revision 9, Sections 9.3.6, 9.3.7, 9.3.8, and 6.2.5.2, respectively.

9.3.1.3 *Staff Evaluation*

During the course of the DCD review, the staff identified areas in which it needed additional information to complete the evaluation of the CAS, and issued RAIs concerning issues that are common and apply to the IAS, SAS, and HPNSS. The following paragraphs describe the staff's RAIs and the applicant's response to each of the RAIs.

RAI 9.3-33

In RAI 9.3-33, the staff stated the following:

> DCD Section 9.3, "Process Auxiliaries," states that the accumulators and valves associated with the main steam isolation, automatic depressurization, and isolation condenser isolation valves are part of the respective systems. However, the DCD sections describing those systems do not include drawings or detailed descriptions regarding the safety-related pressurized gas supplies for operation of those valves. Provide diagrams of safety-related pressurized gas supplies, including separation from the normal nonsafety-related supply of pressurized gas, to all safety-related valve operators, including the following valves: main steam isolation, automatic depressurization, and isolation condenser isolation valves.

In response to RAI 9.3-33, the applicant provided a representative schematic diagram of accumulators that supply air or nitrogen to safety-related valves. In addition to indicating the interface between the safety-related and nonsafety-related components and piping on the schematic diagram, the applicant also stated that safety-related and nonsafety-related separation is at the accumulator check valve.

Based on its review, the staff finds the applicant's response to RAI 9.3-33 acceptable because the schematic drawing clearly depicts the interface between the safety-related and nonsafety-related components and piping. Accordingly, based on the above and the applicant's response, RAI 9.3-33 is resolved.

RAI 9.3-34

In RAI 9.3-34, the staff stated the following:

> DCD Section 9.3 states that the accumulators and valves associated with the main steam isolation, automatic depressurization, and isolation condenser isolation valves are part of the respective systems. However, the DCD sections describing those systems do not include drawings or detailed descriptions regarding the safety-related pressurized gas supplies for operation of those

valves. Clarify the classification of valves, piping, and pressure vessels that provide the pneumatic pressure essential to operation of the following safety-related valves: main steam isolation, automatic depressurization, and isolation condenser isolation valves.

In response to RAI 9.3-34, the applicant referred to the schematic diagram provided in the response to RAI 9.3-33. The staff finds the applicant's response to RAI 9.3-34 acceptable because the schematic diagram clearly depicts the classification of components, valves, and piping that provide the pneumatic pressure essential to operation of the safety-related valves. Accordingly, based on the above and the applicant's response, RAI 9.3-34 is resolved.

RAI 9.3-35

In RAI 9.3-35, the staff stated the following:

> DCD Section 9.3 states that the accumulators and valves associated with the main steam isolation, automatic depressurization, and isolation condenser isolation valves are part of the respective systems. However, the DCD sections describing those systems do not include drawings or detailed descriptions regarding the safety-related pressurized gas supplies for operation of those valves. Describe how the piping, valves and pressure vessels that provide the essential pneumatic pressure for operation of safety-related valves are protected against dynamic effects associated with design basis accidents such that, concurrent with a postulated single active failure, the necessary number of safety-related valves actuate to the correct position.

In response to RAI 9.3-35 regarding how the piping, valves, and pressure vessels that provide the essential pneumatic pressure for operation of safety-related valves are protected against dynamic effects associated with DBAs, the applicant referred to DCD Tier 2, Section 3.6, which addresses the protection provided for safety-related SSCs against dynamic effects associated with DBAs.

In addition, in the responses to RAIs 9.3-33 and 9.3-34, the applicant provided a representative schematic diagram of accumulators that supply air or nitrogen to valves associated with the main steam isolation, automatic depressurization, and isolation condenser isolation functions. The schematic drawing clearly depicts the interface between the safety-related and nonsafety-related pneumatic system components and piping. The safety-related and nonsafety-related separation is at the accumulator check valve. The CAS, with the exception of the inner and outer containment isolation valves and lines in between in the IAS and CIS, is nonsafety-related and has no safety-related function. Failure of the CIS does not compromise any safety-related system or component, and it does not prevent a safe shutdown of the plant.

The staff finds that the RAI response is acceptable since the applicant clarified the safety-related piping, valves, and pressure vessels that provide the essential pneumatic pressure and how they are single-failure-proof. Accordingly, based on the above and the applicant's response, RAI 9.3-35 is resolved.

Section 9.3.6 of this report addresses the staff's evaluation of the IAS. Section 9.3.7 of this report addresses the staff's evaluation of the SAS. Section 9.3.8 of this report addresses the staff's evaluation of the HPNSS. Section 6.2.5.2 of this report addresses the staff's evaluation

of the CIS. Section 8.4.2 of this report addresses the staff's evaluation of those ESBWR design features necessary to cope with an SBO event.

9.3.1.4 *Conclusion*

The staff's conclusions for each of the subsystems of the CAS appear in the respective subsections of this report.

9.3.2 Process and Post-Accident Sampling System

9.3.2.1 *Regulatory Criteria*

The staff reviewed DCD Tier 2, Revision 9, Section 9.3.2, in accordance with SRP Section 9.3.2, Revision 3. The DCD does not describe a post-accident sampling program; however, DCD Tier 2, Revision 9, Table 9.3-1 identifies the sample point parameters, and DCD Tier 2, Revision 9, Sections 7.5.1 and 7.5.2 describe key sample locations for the post-accident monitoring program. In addition, COL Information Item 9.3.2-1-A specifies that the COL applicant needs to develop the post-accident sampling program to monitor the parameters specified in DCD Tier 2, Revision 9, Table 9.3-1. Therefore, this report does not review the postaccident monitoring program. The process sampling system (PSS) is acceptable if the relevant requirements of the following regulations are met:

- 10 CFR 20.1101(b) requires that licensees use, to the extent practicable, procedures and engineering controls based upon sound radiation protection principles to achieve doses that are ALARA.

- GDC 1 requires that SSCs important to safety be designed, fabricated, erected and tested to quality standards commensurate with the importance of the safety functions to be performed.

- GDC 2 requires that SSCs important to safety be designed to withstand the effects of natural phenomena without the loss of the capability to perform their safety functions.

- GDC 13, "Instrumentation and control," requires that instrumentation be provided to monitor variables and systems to ensure adequate safety, including those variables and systems that can affect the fission process, the integrity of the reactor core, and the reactor coolant pressure boundary (RCPB).

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

- GDC 26, "Reactivity control system redundancy and capability," establishes requirements regarding the reliable control of the rate of reactivity changes among other things.

- GDC 60 requires that means be provided to control the release of radioactive materials to the environment.

- GDC 63 requires that systems be provided to monitor the fuel storage and radioactive waste systems to detect conditions that may result in excessive radiation levels.

- GDC 64, “Monitoring radioactivity releases,” requires that means be available for monitoring the containment atmosphere, spaces containing components used for recirculation after a loss-of-coolant accident, effluent discharge paths, and the plant environs for radioactivity that may be released during normal operations, anticipated operational occurrences, and postulated accidents.

- 10 CFR 50.34(f)(2)(xxvi) (TMI Action Plan Item III.D.1.1) requires a program and provisions for leakage control and detection for systems outside containment that contain (or might contain) source term radioactive materials following an accident.

9.3.2.2 *Summary of Technical Information*

The PSS is designed to collect representative water and gaseous samples for analysis contained in the reactor coolant system (RCS) and associated auxiliary system process streams during all normal modes of operation. The proposed design includes permanently installed sample lines, sampling panels with analyzers and associated sampling equipment, provisions for local grab sampling, and permanent shielding to ensure that doses to operators are ALARA during sampling. Provisions are made to ensure that representative samples are obtained from turbulent flow zones to ensure adequate mixing. Continuous sample flows are routed from selected locations to the sampling stations where pressure, temperature, and flow adjustments are made as necessary. Effluents from sample stations are returned to an appropriate process stream or to the radwaste drain headers through a common return line.

The DCD states that the PSS is following the recommendations of SRP Section 9.3.2 and that the PSS is in conformance with the following relevant requirements and criteria:

- 10 CFR Part 20 and 10 CFR 20.1101(b)
- 10 CFR Part 50, Appendix A, GDC 1, 2, 13, 14, 26, 41, 60, 63, and 64
- 10 CFR 50.34(f)(2)(viii) and 10 CFR 50.34(f)(2)(xxvi)

The DCD states that the PSS is in conformance with the following guidelines:

- RG 1.21, “Measuring, Evaluating, and Reporting Radioactive Material in Liquid and Gaseous Effluents and Solid Waste”; RG 1.26; RG 1.29; RG 1.33; RG 1.56, “Maintenance of Water Purity in Boiling Water Reactors (for Comment)”; RG 1.97, “Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants”; and RG 8.8

- NUREG–0737, “Clarification of TMI Action Plan Requirements.”

- American National Standards Institute/Health Physics Society (ANSI/HPS) N13.1, “Sampling and Monitoring Releases of Airborne Radioactive Substances from the Stacks and Ducts of Nuclear Facilities.”

- EPRI Boiling Water Reactor Vessel and Internals Project (BWRVIP)-130, “BWR Water Chemistry Guidelines.”

The design provides the capability to meet the conditions of NEDO-32991-A, “Regulatory Relaxation for BWR Post-Accident Sampling Stations (PASS).”

The PSS can provide information on the following parameters:

- pH
- Iron
- Silica
- Iodine-131
- Sulfate
- Copper
- Sodium
- Chloride
- Isotopics
- Conductivity
- Total anions
- Gross activity
- Dissolved oxygen
- Organic impurities
- Noble gases
- Alpha emitters
- Fission product activity
- Corrosion product activity
- Corrosion product metals
- Gaseous fission products

The PSS does not perform or ensure any safety-related function. However, the system incorporates features that improve operator safety. The sampling stations are closed systems and have chemical fume hoods to preclude the exposure of operating personnel to contamination hazards when taking grab samples. In addition, all sampling lines contain process isolation block valves to minimize leaks in the event of a line break.

9.3.2.3 *Staff Evaluation*

Compliance with GDC 13, 14, 26, 63, and 64 is ensured if the applicant's design is such that the PSS has the capability to sample all normal process systems and principal components, including provisions for obtaining samples from at least the points indicated below. The guidelines of RG 1.21, Regulatory Position C.2, and the EPRI BWR Water Chemistry Guidelines are used to meet the requirements of the relevant GDC. The staff has endorsed the EPRI BWR Water Chemistry Guidelines in its SER for the EPRI Utility Requirements document (NUREG–1242, "NRC Review of Electric Power Research Institute's Advanced Light Water Reactor Utility Requirements Document,").

The intended function of the PSS is to collect and analyze liquid and gaseous samples from the RCS and from associated auxiliary system process streams during all normal modes of operation. The staff reviewed the capability of the PSS to collect and deliver samples of fluids for analysis from systems needed to address GDC 13, 14 26, 63, and 64. According to SRP Section 9.3.2, in order to meet GDC 13, 14, 26, 63, and 64 , the PSS should permit an operator to obtain samples from the reactor coolant, standby liquid control system (SLCS) tank, condensate polishing system, FAPCS, sumps inside containment, main condenser evacuation system, and inlet and outlet of the radwaste tank.

The ESBWR PSS design includes the following sample stations:

- RB sample station

- Local grab sampling stations
- Condensate polishing sample station
- TB sample station
- Condenser sample station
- RW sample station
- Auxiliary boiler building sample station

The RB sample station permits an operator to take continuous samples from the FAPCS. In addition, grab samples can be taken to test for the parameters identified above.

DCD Tier 2, Revision 9, Table 9.3-1, identifies sampled systems and process measurements to be taken. DCD Tier 2, Revision 9, Table 1.9-21, states that RG 1.21 is applicable to the ESBWR without exceptions. The staff reviewed the points and parameters identified for sampling in DCD Tier 2, Table 9.3-1 of the DCD. The staff finds the points and parameters to be consistent with the sample points recommended in SRP Section 9.3.2 and the parameters monitored are appropriate. Local grab sampling points are provided for the following systems:

- RCCWS
- TCCWS
- PSWS
- CWS
- CIRC
- SLCS
- MWS
- CS&TS
- Equipment and floor drain system (EFDS)

The staff notes that local grab sampling points are located throughout the plant to monitor process streams needing intermittent sampling. The grab samples for the SLCS are taken from the standby liquid control tank to measure percent weight sodium pentaborate. However, to meet the requirements of GDC 60 and 63, SRP Section 9.3.2 recommends that samples be taken from the SFP. DCD Tier 2, Revision 9, Table 9.1-1, states that the SFP is located in the FB which hosts no sample station according to DCD Tier 2, Section 9.3.2. The staff requested in RAI 9.3-44 that the applicant identify the process sampling proposed for the SFP and other FB pools, provide the typical process measurements that will be conducted (continuous and grab), and identify where the process samples will be processed.

In response to RAI 9.3-44, the applicant stated that the SFP can be sampled either before or after the FAPCS filter demineralizers. Samples are obtained from the RB sample station and analyzed for the species identified in DCD Tier 2, Table 9.3-1. One FAPCS C/C train is continuously operated to cool and clean the water in the SFP during normal plant operation and during a refueling outage. During this mode of operation, water is drawn from the skimmer surge tanks, pumped through the heat exchanger and water treatment unit to be cooled and cleaned and then returned to the SFP. As the SFP level rises, water spills into the weir and flows back to the skimmer surge tanks. The PSS lines tap off the process downstream of the heat exchangers and again downstream of the filter and demineralizer subsystem. Flow returns to the FAPCS at the suction of the FAPCS pump. Therefore, the SFP can be sampled both before and after the filter and demineralizer subsystem. The sample station for the FAPCS is located in the RB. This central location allows for sampling from pools in the containment, RB, and FB thus minimizing locations of possible spillage and contamination. DCD Tier 2, Revision

9, Table 9.1-1, shows the various pools served by both subsystems of the FAPCS. DCD Tier 2, Revision 9, Table 11.5-5, identifies the SFP as having provisions for being sampled, and DCD Tier 2, Table 9.3-1 identifies the typical process measurements taken from the FAPCS. The staff finds that the response to RAI 9.3-44 is acceptable since the applicant clarified the process sampling for the SFP and other FB pools. Accordingly, based on the above and the applicant's response, RAI 9.3-44 is resolved. Based on this information, the staff finds that the PSS design meets the requirements of GDC 13 with respect to monitoring variables that can affect the fission process, the integrity of the reactor core, and the reactor coolant pressure boundary; GDC 14 with respect to assuring the integrity of the reactor coolant pressure boundary by sampling for chemical species that can affect the reactor coolant pressure boundary; GDC 26 with respect to reliably controlling the rate of reactivity changes by sampling boron concentration; GDC 63 with respect to detecting conditions that may result in excessive radiation levels in the fuel storage and radioactive waste systems; and GDC 64 with respect to monitoring the containment atmosphere and plant environs for radioactivity.

SRP Section 9.3.2 recommends that provisions be made to ensure that representative samples can be obtained from liquid and process streams and tanks. For tanks, provisions should be made to sample the bulk volume of the tank and to avoid sampling from low points or from potential sediment traps. For process stream samples, sample points should be located in turbulent flow zones. SRP Section 9.3.2 also states that provisions should be made to ensure representative samples from gaseous process streams and tanks in accordance with ANSI/HPS Standard N13.1-1999. The guidelines of Regulatory Position C.6 in RG 1.21 are followed to meet these criteria.

DCD Tier 2, Revision 9, Section 9.3.2.2, states that the PSS provides sampling of all principal fluid and gaseous process streams associated with plant operation and that sample connections are located in turbulent flow zones to ensure adequate mixing. Sampling equipment is designed with flushing and blowdown capability to remove sediment deposits and air and gas pockets. Provisions are made to purge sample lines in the sampling stations and, with few exceptions, all flushing fluids are returned to appropriate process streams or sent to the radwaste system. The staff finds these provisions acceptable because they meet the recommendations of Regulatory Position C.6 in RG 1.21.

DCD Tier 2, Revision 9, Section 11.5, describes provisions for sampling liquid and gaseous process and effluent streams and summarizes the scope of radiological analyses for such samples. DCD Tier 2, Revision 9, Tables 11.5-5 to 11.5-8, describes this information. The tables identify plant systems and specify grab or continuous sampling provisions and identify sampling frequencies and types of radiological analyses. The staff finds these provisions acceptable because they are generally consistent with the recommendations of RGs 1.21 and 4.15, "Quality Assurance for Radiological Monitoring Programs (Inception through Normal Operations to License Termination)—Effluent Streams And The Environment," and NUREG–1302, "Offsite Dose Calculation Manual Guidance: Standard Radiological Effluent Controls for Boiling Water Reactors" (Generic Letter 89-01, Supplement No. 1)," in the development of a plant-specific offsite dose calculation manual and standard radiological effluent controls for BWR plants. The COL applicant will address site-specific conformance to the recommendations of RGs 1.21 and 4.15 and NUREG–1302 consistent with COL Information Items 11.5-2-A, and 11.5-3-A. Section 11.5 of this report discusses further these COL information items.

SRP Section 9.3.2 recommends that provisions should be made for purging sampling lines and for reducing plateout in sample lines (e.g., heat tracing). The guidelines of Regulatory Position C.7 in RG 1.21 are followed to meet this criterion.

DCD Tier 2, Revision 9, Section 9.3.2.2, states that heat tracing of sampling lines is provided as necessary to prevent plateout, crystallization, or solidification of sample line contents. The staff finds these provisions acceptable because they meet the recommendations of Regulatory Position C.7 in RG 1.21.

SRP Section 9.3.2 recommends that isolation valves should fail in the closed position, in accordance with the requirements of GDC 60 to control the release of radioactive materials to the environment.

DCD Tier 2, Revision 9, Section 9.3.2.2, states that sampling lines and associated valves and fittings are fabricated from stainless steel. All sampling lines have process isolation block valves located as close as practical to the process taps. These valves can be closed if sample line rupture occurs downstream of the valves. The staff finds these provisions acceptable because they meet the requirements of GDC 60 with respect to controlling the release of radioactive materials to the environment.

SRP Section 9.3.2 recommends that provisions should be made to purge and drain sample streams back to the system of origin or to an appropriate waste treatment system, in accordance with the requirements of 10 CFR 20.1101(b), to keep radiation exposures ALARA. The guidelines of Regulatory Positions 2.d.(2), 2.f.(3), and 2.f.(8) in RG 8.8 are followed to meet this criterion.

DCD Tier 2, Revision 9, Section 9.3.2.2, states that the sample station’s effluents are returned to the appropriate process stream or to the radwaste drain headers through a common return line and that ALARA is considered in station layout and design. The staff finds these provisions acceptable because they meet the requirements of 10 CFR 20.1101(b) to keep radiation exposures at ALARA levels with respect to the sampling systems.

DCD Tier 2, Revision 9, Section 9.3.2.2, states that the sample station’s effluents are returned to the appropriate process stream or to the radwaste drain headers through a common return line. Although the applicant stated that the station layout and design considered the ALARA principle, DCD Tier 2, Revision 9, Section 9.3.2 does not describe how the design of the PSS sample stations incorporate shielding and other design features described in RG 8.8 to minimize personnel doses and contamination, in accordance with 10 CFR 20.1406. It was also unclear whether the applicant had assessed the personnel doses associated with the sampling of radioactive material. For the staff to determine if the applicant had addressed the issues associated with the PSS sample stations, the staff issued RAI 9.3-43. In response to this RAI, the applicant stated that the PSS sampling stations incorporate several of the ALARA design features described in RG 8.8 to minimize personnel exposures to radiation. Sampling stations are located in low radiation areas to minimize operator exposure. Cleaning and flushing is provided at the sample stations and the sample piping is routed to minimize crud traps and hot spots. In order to minimize contamination, in accordance with 10 CFR 20.1406, sampling station work areas and fume hoods are made of stainless steel, and fume hoods draw radioactive gases away from the sample chemist. Epoxy-type wall and floor coverings provide smooth surfaces for ease of decontamination. To limit the extent of contamination in areas where the potential for spills exists, floors are sloped towards drains and curbs are provided to simplify washdown operations. The applicant stated that it had evaluated the personnel doses associated with routine use of the PSS sample stations, and these doses are listed in DCD Tier 2, Revision 9, Table 12.4-2, which lists occupational dose estimates during operation and surveillances. The staff finds that the response is acceptable since the applicant adopted design features to minimize personnel dose and contamination conforms to the guidelines of

RG 8.8 and the requirements of 10 CFR 20.1406. Accordingly, based on the above and the applicant's response, RAI 9.3-43 is resolved.

SRP Section 9.3.2 recommends that passive flow restrictions to limit reactor coolant loss from a rupture of the sample line should be provided in accordance with the requirements of 10 CFR 20.1101(b) to keep radiation exposures ALARA and satisfy the requirements of GDC 60 to control the release of radioactive materials to the environment. The guidelines of Regulatory Position 2.i.(6) in RG 8.8 should be followed to meet this criterion.

DCD Tier 2, Revision 9, Section 9.3.2, states that all sampling lines have the process isolation block valves located as close as practical to the process taps. These valves can be closed if a sample line rupture occurs downstream of the valves. In the event of a loss of cooling water to a sample flow in excess of sample cooler capacity, the sampling system valves are interlocked to prevent high-temperature water flow through the lines. SRVs, vented to the drain headers, are provided in the stations for high-temperature process streams. Continuous samples are taken and monitored continuously. The continuously monitoring equipment transmits signals to the plant computer, and alarms are provided for indicating off-normal operating conditions. The sampling station layout and design also consider ALARA. The staff finds these provisions acceptable because they meet the requirements of GDC 60 to control the release of radioactive materials to the environment.

SRP Section 9.3.2 recommends that, to meet the requirements of GDC 1 and 2, the seismic design and quality group classification of sampling lines, components, and instruments for the PSS should conform to the classification of the system to which the sampling line and components are connected.

DCD Tier 2, Revision 9, Section 9.3.2.2, states that the seismic design and quality group classifications of sample lines and their components conform to the classification of the system to which they are connected, up to and including the block valves. The staff finds that the proposed process sampling system meets the quality standard requirements of GDC 1 and the seismic requirements of GDC 2 by designing the sampling lines and components of the process sampling system to conform to the classification of the system to which each sampling line and component is connected.

9.3.2.4 *Conclusion*

For the reasons set forth above, the staff finds that the design of the process sampling system is acceptable and that the process sampling system meets the relevant requirements of 10 CFR 20.1101(b); GDC 1, 2, 13, 14, 26, 60, 63, and 64; and the requirements of 10 CFR 50.34(f)(2)(xxvi).

9.3.3 Equipment and Floor Drain System

9.3.3.1 *Regulatory Criteria*

The staff reviewed the ESBWR EFDS in accordance with SRP Section 9.3.3," Revision 3. The staff reviewed DCD Tier 1, Revision 9, Section 2; DCD Tier 2, Revision 9, Section 9.3.3,; and various parts of other DCD Tier 2, Revision 9, sections (e.g., Sections 19A). The staff's acceptance of the EFDS is based on meeting the relevant requirements of the following regulations:

- GDC 2, in part, requires that SSCs important to safety shall be designed to withstand the effects of natural phenomena such as earthquakes. Compliance with GDC 2 is based on meeting the guidance of Regulatory Positions C.1 and C.2 of RG 1.29.

- GDC 4, in part, requires that SSCs important to safety shall be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents.

- GDC 60, in part, requires that the nuclear power plant unit design shall include means to control suitably the release of radioactive materials in gaseous and liquid effluents and to handle radioactive wastes produced during normal reactor operation, including anticipated operational occurrences. Sufficient holdup capacity shall be provided for retention of gaseous and liquid effluents containing radioactive materials.

- 10 CFR 52.47(b)(1), requires that a design certification application contain the proposed ITAAC that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design certification is built and will operate in accordance with the design certification, the provisions of the Atomic Energy Act, and the NRC's regulations.

9.3.3.2 *Summary of Technical Information*

The EFDS is a nonsafety-related system that collects and processes the liquid wastes from the equipment and floor drains in various areas during plant operation and outages. The liquid wastes are then transferred to appropriate processing and disposal systems. With the exception of the inner and outer containment isolation valves and lines in between of the drywell sump pump discharge lines, the EFDS is nonsafety-related and serves no safety-related function. Failure of the EFDS does not prevent any safety-related equipment from performing its safety-related functions. Section 6.2.4 of this report addresses the staff's evaluation of the ESBWR design of the containment penetration and associated isolation valves. DCD Tier 2, Revision 9, Table 6.2-43, which provides containment isolation valve information for the EFDS, lists the two penetrations associated with the EFDS.

The EFDS collects liquid wastes from their point of origin and transfers liquid wastes to a suitable processing or disposal system. The EFDS is designed to accommodate the maximum anticipated normal volumes of liquid without overflowing, including such inputs as the anticipated water flow from a fire hose, and other fire suppression water discharges to the area floor drains without impacting the safety function of any safety-related component or system. However, as delineated in DCD Tier 2, Revision 9, Section 3.4.1, the flooding analysis takes no credit for the EFDS system. Section 3.4.1, of this report addresses the staff's evaluation of the postulated flooding events.

To preclude inadvertent transfer of radioactive liquids to non-radioactive systems, the EFDS is divided into two completely separate systems (i.e., there are no cross-connections between the two systems), the clean drain (non-radioactive) system (CDS) and the radioactive waste drain systems (RWDS). Liquid wastes from various floors and equipment drains are drained by gravity to the appropriate sumps and then pumped to the LWMS for processing and disposal. The RWDS is further divided into the following subsystems, so which allow the liquid wastes from various sources to be segregated and processed separately for each specific type of impurity and chemical content:

- Low conductivity waste (LCW) drain subsystem
- High conductivity waste (HCW) drain subsystem
- Detergent drain subsystem
- Chemical waste drain subsystem
- RCCWS drain subsystem

Each of the above subsystems has its own sump, pumps, isolation valves, and instrumentation and piping.

The CDS collects liquid wastes by gravity from the clean non-radioactive equipment and floor drains in sumps and pumps them to an appropriate disposal system. The RWDS subsystems collect liquid wastes from various plant areas by gravity to sumps and pump them to the collection tanks of the LWMS for processing and disposal. Capability is provided to sample the liquids collected in each sump. Section 11.2, of this report addresses the staff's evaluation of the LWMS.

The EFDS design includes provisions for sampling the drain sumps and tanks for radioactive contamination. Contaminated or potentially contaminated liquids are then pumped to the LWMS for processing and disposal. Each sump has two pumps. One pump operates as required and the other is on standby. The lead sump pump starts automatically when the liquid reaches a predetermined level in the sump and stops at a predetermined low level. Both pumps operate simultaneously if one pump cannot accommodate the rate of accumulation of liquids in the sump. The EFDS pumps also can be controlled manually.

The detection of small, unidentified leakage within the drywell is accomplished by monitoring the drywell floor drain HCW and LCW sump pump activity and the drywell sump level changes. Leak detection in other areas is accomplished by monitoring the frequency and duration of sump pump operation. Section 5.2.5, of this report addresses the staff's evaluation of the leakage detection, monitoring, alarm and isolation from various sources within the containment and from areas outside the containment.

9.3.3.3 *Staff Evaluation*

During the course of the DCD review, the staff issued three RAIs regarding drainage of floodwater. In RAIs 9.3-27, 9.3-28, and 9.3-29, the staff requested the applicant to clarify the flood protection measures associated with the EFDS. In responses, the applicant stated and clarified that the floor EFDS was a nonsafety-related system and was not credited for draining floodwater in the flooding analysis. The staff found the results of the flooding analysis to be acceptable, assuming that the floodwater was retained in localized areas or zones. This assumption is conservative in determining the resulting water level of these specific areas. The staff finds that the response is acceptable since the applicant clarified that the ESBWR flooding analysis took no credit for the EFDS. Accordingly, based on the above and the applicant's responses, RAI 9.3-27, 9.3-28, and 9.3-29 are resolved.

The EFDS does not have to comply with Regulatory Position C.1 of RG 1.29 because, with the exception of the inner and outer containment isolation valves and lines in between, the system is nonsafety-related and performs no safety-related function. As stated above, Section 6.2.4 of this report addresses the staff's evaluation of the ESBWR containment isolation penetrations and valve design. As for the nonsafety-related EFDS meeting the guidance of Regulatory Position C.2 of RG 1.29, the EFDS is designed to ensure that failure of the EFDS neither compromises any safety-related system or component nor prevents a safe shutdown.

Therefore, the staff finds that the EFDS meets the guidance of Regulatory Position C.2 of RG 1.29 for ensuring that failure of the EFDS neither compromises any safety-related system or component nor prevents a safe shutdown. Accordingly, the staff finds that the EFDS meets the requirements of GDC 2.

The EFDS, with the exception of the inner and outer containment isolation valves and lines in between, is a nonsafety-related system and is not credited in any safety analysis such as the flooding analysis. Its failure does not lead to the failure of any SSC. Accordingly, the staff finds that the EFDS meets the requirements of GDC 4.

As stated above, to preclude inadvertent transfer of radioactive liquids to non-radioactive systems, the EFDS consists of completely separate systems (i.e., no cross connections between the system); the non-radioactive CDS and the potentially radioactive RWDS. Potentially radioactive drainage is collected in floor and equipment drain sumps in various areas and discharged to the LWMS for processing and disposal. The EFDS is designed to accommodate the maximum anticipated normal volumes of liquid without overflowing, including such inputs as the anticipated water flow from a fire hose and other fire suppression water discharges to the area floor drains without impacting the safety function of any safety-related component or system. Also, the EFDS design includes provisions for sampling the drain sumps and tanks for radioactive contamination. Drainage from sources that are not potentially radioactive is discharged to the clean waste system or the LWMS, as appropriate. Thus, the staff finds that the system design meets the pertinent requirements of GDC 60.

The EFDS, with the exception of the inner and outer containment isolation valves and lines in between, is nonsafety-related, is not credited in the flooding analysis or any other safety analysis, and is not required to achieve or maintain safe shutdown of the plant. Furthermore, the ESBWR design does not use the EFDS to provide defense-in-depth capabilities for any safety function. Therefore, the EFDS is not considered to be a candidate for RTNSS, because it does not meet any of the five criteria described in SECY-94-084.

The EFDS has ITAAC entries in DCD Tier 1. DCD Tier 1, Revision 9, Section 2.16.4, and Table 2.16.4-1 provide the design descriptions and ITAAC for the EFDS. The staff finds that these ITAAC commit to verify that the EFDS is constructed and installed as described in ESBWR DCD Tier 2. Therefore, the staff finds that the EFDS complies with the requirements of 10 CFR 52.47(b)(1).

The EFDS is designed to permit periodic inspection and testing of important components, such as valves, motor operators, and piping, to verify their integrity and capability. In addition, the EFDS functionality is demonstrated by continuous use during normal plant operation.

9.3.3.4 *Conclusion*

The staff finds that the design of the EFDS is acceptable and meets the relevant requirements of GDC 2, 4, and 60 and 10 CFR 52.47(b)(1).

9.3.4 Chemical and Volume Control System

This section does not apply to the ESBWR.

9.3.5 Standby Liquid Control System

9.3.5.1 *Regulatory Criteria*

The ESBWR includes an SLCS that provides backup capability for reactivity control independent of the control rod system. The SLCS functions by injecting a boron solution into the reactor to affect shutdown. The staff's review covers the functional capability of the system to deliver the required amount of boron solution into the reactor.

The staff reviewed DCD Tier 1, Revision 9, Section 2.2.4, and DCD Tier 2, Revision 9, Section 9.3.5, for the ESBWR, in accordance with SRP Section 9.3.5, Revision 3. Acceptability of the SLCS design, as described in the applicant's DCD, is based on specific GDC; the provisions of 10 CFR 50.62, "Requirements for Reduction of Risk from Anticipated Transients Without Scram (ATWS) Events for Light-Water-Cooled Nuclear Power Plants," known as the anticipated transient without scram (ATWS) rule; and RGs. The design of the SLCS is acceptable if the integrated design of the system is in accordance with the following criteria:

- GDC 2, as it relates to structures housing the system and the system itself being capable of withstanding the effects of earthquakes, with acceptance based on meeting the guidance of Regulatory Position C-1 in RG 1.29

- GDC 4, as it relates to dynamic effects associated with flow instabilities and loads, such as water hammer

- GDC 5, as it relates to SSCs important to safety not being shared among nuclear power units unless it can be demonstrated that sharing will not impair the shared components' ability to perform the required safety functions

- GDC 26, as it relates to the requirements that (1) two independent reactivity control systems of different design principles be provided and (2) one of the systems shall be capable of holding the reactor subcritical in the cold condition

- GDC 27, "Combined reactivity control systems capability," as it relates to the requirement that the reactivity control systems have a combined capability, in conjunction with poison addition by the emergency core cooling system (ECCS), of reliably controlling reactivity changes under postulated accident conditions

- 10 CFR 50.62(c)(4), as it relates to (1) the SLCS's being capable of reliably injecting a borated water solution into the RPV at a boron concentration, boron enrichment, and flow rate that provides sufficient reactivity control, and (2) the system's having automatic initiation, as required under the rule, to satisfy ATWS risk-reduction requirements

Because the ESBWR does not have recirculation pumps, 10 CFR 50.62(c)(5), which requires that each BWR must have equipment to trip the reactor coolant recirculating pumps automatically under conditions indicative of an ATWS, does not apply to the ESBWR.

Since the SLCS is part of the ECCS, the staff also used SRP Section 6.3 in its review. Section 6.3 of this report also provides the acceptance criteria and the staff's evaluation of the SLCS as part of the ECCS.

9.3.5.2 *Summary of Technical Information*

The SLCS can be initiated manually for its reactor shutdown function, but it is initiated automatically for ATWS events and LOCAs.

The SLCS is needed in the improbable event that sufficient control rods cannot be inserted in the reactor core to accomplish shutdown and cool down in the normal manner. Its function is to shut down the reactor and keep the reactor from going critical again during cool down. The SLCS is also designed to provide makeup water to the RPV during a LOCA event by injecting the boron solution from both accumulators. As a part of the ECCS, the SLCS is designed to flood the core during a LOCA to provide the required core cooling. Section 15.4.5.3.2.1 of this report includes the staff's evaluation of the system's buffering function.

The boron solution is also credited for buffering the suppression pool so that dissolved iodine does not re-evolve into the containment atmosphere. By providing core cooling following a LOCA, the SLCS, in conjunction with the containment limits the release of radioactive materials to the environment.

The SLCS contains two identical and separate trains. Each train provides 50-percent injection capacity. All components of the SLCS in contact with the boron solution are constructed of, or lined with, stainless steel. The SLCS also includes a nonsafety-related, nitrogen charging subsystem that includes a liquid nitrogen tank, vaporizer, and high-pressure pump for initial accumulator charging and makeup for the normal system losses during routine plant operations. Control of the equipment compartment temperature and humidity conditions avoids solute precipitation in the accumulator or injection line, thereby ensuring proper system operation. This system readiness function is nonsafety-related.

The major components of the SLCS that are necessary for the injection of sodium pentaborate solution into the reactor are located within the RB. The nonsafety-related high-pressure cryogenic nitrogen equipment is located outside the RB at grade elevation. The sparger system, which injects boron into the reactor, is located within the reactor vessel.

The SLCS can be initiated manually from the MCR to inject a boron neutron absorber solution into the reactor, if the operator determines that the reactor cannot be shut down or kept shut down using the control rods. DCD Tier 2, Revision 9, Section 9.3.5.2 states that the method of manual initiation will involve multiple, deliberate operator actions to prevent inadvertent boron injection. Procedural controls govern manual initiation of the SLCS.

Because the presence of nitrogen in the RPV could interfere with ICS operation, the SLCS is designed to prevent injection of nitrogen from the accumulators into the RPV. When injection of the boron solution is complete, redundant accumulator level measurement instrumentation using two-out-of-four logic closes the injection line shutoff valve in each SLCS train, preventing the injection of nitrogen into the RPV.

For ATWS events, the failure of control rods to insert in response to a valid trip demand is assumed. The SLCS automatically initiates when the average power range monitor (APRM) is not downscale (greater than or equal to 6 percent) and one of the following conditions persists for at least 3 minutes:

- Reactor dome gauge pressure greater than or equal to 7.76 megapascals (Mpa) (1125 psig)

- Low reactor vessel water level (Level 2)

Sodium pentaborate solution injection ensures a timely accomplishment of hot shutdown.

Subsequent injections as the reactor depressurizes ensure that cold shutdown can be achieved with no further occurrence of reactor critical conditions. Section 15.5 of this report discusses SLCS performance in the evaluation of ATWS events.

9.3.5.3 *Staff Evaluation*

The design of the ESBWR SLCS departs significantly from a conventional BWR SLCS in several aspects, including the following:

- The logic systems of the ESBWR SLCS differ from conventional BWR SLCS logic. The system is also part of emergency core cooling and starts during a LOCA.
- The SLCS tank is outside the primary containment, which in itself is not a design departure, but the tank is not heated.
- Accumulators instead of pumps drive the SLCS injection; hence, the system is a passive system.
- The SLCS injects into the core bypass between the top and bottom of the active fuel region.

The SLCS is a reactivity control system. Its purpose is to inject sodium pentaborate solution into the reactor coolant to provide an independent means of shutting down the reactor. The SLCS can bring the reactor from rated power to cold shutdown any time during core life, should the normal reactivity control system become inoperable. Section 4.6 of this report discusses reactivity control. Based on this description of the system's purpose and on the staff's acceptance of the design, the staff finds that the intent of GDC 26 is met.

An ATWS with MSIV closure challenges the plant with high neutron flux, vessel pressure, and suppression pool temperature. It is therefore considered a bounding event in terms of the challenge it poses to fuel-cladding integrity. In this scenario, hydraulic scram, alternate rod insertion, and FMCRD run-in are assumed to be unavailable. Additionally, the ESBWR design does not include recirculation pumps, which would otherwise be tripped as required by 10 CFR 50.62(c)(5). Therefore, the SLCS is one of the two means for controlling the core reactivity and, hence, the power during the transient. For this reason, the SLCS meets the applicable requirements of 10 CFR 50.62. Section 15.5.4 of this report provides a more detailed discussion of the basis for this conclusion.

The staff reviewed DCD Tier 2, Revision 9, Section 9.3.5, to evaluate compliance with the remaining regulatory criteria discussed in Section 9.3.5.1 of this report. The staff's evaluation is discussed below.

9.3.5.3.1 System Design and Testing

The SLCS is located in a compartment within the seismic Category I, flood- and tornado-protected RB outside the drywell and below the refueling floor. All portions of the SLCS necessary for the injection of sodium pentaborate solution into the reactor are seismic Category

I, Quality Group B (or Quality Group A if they are part of the RCPB). Thus, the SLCS meets the requirements of GDC 2 and the guidelines of Regulatory Position C.1 of RG 1.29.

The DCD contains a simplified process diagram, which the staff reviewed to determine that the design of the SLCS is completed in accordance with the applicable regulatory requirements.

The RB in which the system is located provides protection against externally or internally generated missiles. The nonsafety-related portions of the system are also located in the RB, with the exception of the high-pressure cryogenic nitrogen equipment, which is located at grade level, outside the RB. Furthermore, the staff, in RAI 9.3-5, requested that the applicant explain in detail how the SLCS meets the requirements of GDC 4. RAI 9.3-5 was being tracked as an open item in the SER with open items.

In response, the applicant stated, that its location inside the RB, with its own compartment, protects the SLCS from internally and externally generated missiles. The system piping is routed and analyzed so that an appropriate distance is provided between it and other high energy piping. To prevent or mitigate the dynamic effects water hammer, the injection line is designed with proper venting. The system components are qualified for the range of environmental conditions postulated for their location. The applicant added this clarification to DCD Revision 5. The staff finds that the response is acceptable since the applicant clarified how the SLCS meets the requirements of GDC 4 and made corresponding changes to the DCD. Accordingly, based on the above and the applicant's response, RAI 9.3-5 is resolved.

The ESBWR design is a single-unit station, and the requirements of GDC 5 are not applicable to the single-unit design.

The pyrotechnic charges used in the squib-actuated injection valves are replaced during scheduled plant shutdowns. The removed charges are tested to confirm their end-of-life capability to function as demanded. Shutoff valves and relief valves are periodically tested to ensure operability. This information serves as adequate confirmation that design provisions permit appropriate in-service inspection and functional testing of the system.

The SLCS meets the divisional separation criteria because it is not located in any proximity to the CRD system, and each independent SLCS train is located on an opposite side of the reactor vessel. In RAI 9.3-9, the staff requested the applicant to describe the system design with respect to the capability to detect, collect, and control system leakage, as well as the capability to isolate portions of the system in case of excessive leakage or malfunctions. RAI 9.3-9 was being tracked as an open item in the SER with open items.

In response, the applicant stated that the SLCS leakage can be monitored using the accumulator pressure and level instrumentation, which provides alarms for out-of-tolerance process conditions. Frequent alarms that call for boron or nitrogen makeup indicate the possibility of system leakage, and system inspections are performed. The SLCS collects leakage through drains and sends it to a stainless steel drum for disposal. In the event of a system leakage, or maintenance, the injection line and accumulators are capable of isolation from the reactor and from each other. The various subsystems are capable of isolation from the main system. The applicant incorporated corresponding statements into DCD Revision 5. The staff finds that the response is acceptable since the applicant modified the DCD to describe the system design with respect to the capability to detect, collect, and control system leakage, as well as the capability to isolate portions of the system in case of excessive leakage or

malfunction. Accordingly, based on the above, the applicant's responses, and DCD changes, RAI 9.3-9 is resolved.

9.3.5.3.2 Adequate System Capacity

The system consists of two accumulators pressurized with nitrogen, two redundant squib-actuated injection valves at each accumulator discharge, two AOVs in series at each accumulator discharge, piping, and controls. Accumulator pressure and accumulator solution levels are indicated in the MCR. Each train provides 50-percent system capacity for both reactivity control and emergency core cooling functions.

All safety-related portions of the SLCS are located within the RB. The applicant stated that electrical heating inside the accumulator tank and the injection line is not necessary because the saturation temperature of the solution is less than 15.5 degrees C (60 degrees F) and the equipment room temperature where the tank is located is maintained above that value at all times by the RB HVAC systems when SLCS injection is required to be operable. However, an electric backup heater is provided in each SLCS room to ensure satisfactory environmental conditions in the event that the RB HVAC systems are not available. The PIP A and B buses power the backup heaters to prevent common-mode failures of the heating systems that provide the appropriate environmental conditions for the SLCS. The NRC staff finds that environmental conditions will be maintained adequately to prevent boron precipitation in the SLCS accumulators.

Piping for the SLCS enters the reactor vessel, extends downward outside the core shroud, and penetrates the core shroud at four elevations of the active fuel region below the core midplane. DCD Tier 2, Revision 9, Table 9.3-5, indicates that during ATWS, at a reactor pressure as high as 8.63 MPa (1,250 psig), boron solution discharge from the SLCS occurs at a volumetric rate of 1.8 cubic meters per minute (m^3/min) (475 gpm) during the initial injection. These flow rates are averages for the first 5.40 m^3 (1427 gallons) of boron solution flow for each of two trains. The staff accepted NEDE-31096-P, "Anticipated Transients Without Scram; Response to NRC ATWS Rule 10 CFR 50.62," in a safety evaluation dated October 21, 1986 (microfiche information on this report is available in the ADAMS Legacy Library under Accession No. 8612050358). The topical report provided specific information relevant to determining whether SLCSs are sufficiently capable of meeting the provisions of the ATWS rule.

Specifically, the NRC approved the following relationship:

$$\frac{Q}{86} \times \frac{M_{251}}{M} \times \frac{C}{13} \times \frac{E}{19.8} \geq 1,$$

Where

- Q = expected SLCS flow rate (gpm)
- M = mass of water in the reactor vessel and recirculation system (conventional BWR) at hot rated condition (pounds)
- C = sodium pentaborate solution concentration (weight percent)
- E = boron-10 isotope enrichment (atom percent)

This relationship used the requirements established in 10 CFR 50.62; specifically, that an SLCS must be capable of injecting 0.33 m^3 /min (86 gpm) of a 13 weight percent sodium pentaborate

decahydrate solution at the natural boron-10 isotope abundance into a reactor vessel with a 6.38 m (251-in.) inner diameter, and provided a means to compare differences in injection flow, vessel size, solution concentration, and enrichment to determine alternative SLCS capabilities that meet the intent of the ATWS rule.

The applicant demonstrated how the ESBWR SLCS satisfies the above relationship in response to RAI 14.3-196 S01, where the injection flow is 1.25 m^3/min (330 gpm), the concentration is 12.5 weight percent, and with a natural abundance of boron-10 (nonenriched). The mass of the water inside the ESBWR reactor vessel with a 7.06 m (278 in.) diameter, based on the fluid control volumes in DCD Tier 2, Figure 5.1-1, is 374,500 kilograms (kg) (823,800 pounds). The mass of the water in a 6.38 m (251 in.) BWR/6 vessel, is 279,200 kg (614,300 pounds). Consequently, the design meets the requirements of the ATWS rule by a factor of 2.75. If 94 percent enriched boron-10 is used instead of the natural boron, the design will meet the requirements of the ATWS rule by a factor of 13.1.

Noting that the NRC previously approved the relationship given above for BWR/4, 5, and 6 designs, the staff also independently analyzed the SLCS shutdown capability with a conservatively developed ESBWR fuel lattice model using the Monte Carlo N-Particle Transport Code System (MCNP5). This model conservatively determined that 256 parts per million (ppm) of boron-10 (i.e., 266 ppm of 96-percent enriched sodium pentaborate), uniformly present in the ESBWR core, would bring the reactor to a cold-shutdown condition. This compares to the licensee's SLCS capability of 1,100 ppm with factor of 4.13.

In RAI 9.3-11, the NRC staff asked for an indication of the time required for the SLCS to bring the ESBWR to a hot-shutdown condition. The applicant noted that an analysis of the SLCS during a limiting ATWS scenario using the TRACG computational software indicated that the time required is 384 seconds. This information is subject to NRC approval of the application of the TRACG code for ESBWR ATWS analysis, as discussed in Chapter 21 of this report. RAI 9.3-11 was being tracked as an open item in the SER with open items.

NEDE-33083P, Supplement 2, Revision 2, "TRACG Application for ESBWR Anticipated Transient without Scram Analyses," documents the applicant's boron mixing and transport models. NEDE-33083P, Supplement 2, Revision 2, also documents the applicant's comparison of its boron mixing model to CFD analyses and experimental data, which shows that the overall TRACG boron mixing and transport models result in a lower reactivity worth and are thus conservative. The staff performed CFD confirmatory calculations and reached similar conclusions. The staff's safety evaluation for NEDE-33083P, Supplement 2, Revision 2, provides additional discussion. A conservative reactivity worth produces slower reduction in power and thus conservative shutdown time. Thus, TRACG provides a conservative means of determining the time for the SLCS to bring the ESBWR to a hot-shutdown condition; therefore, the results in the RAI response are acceptable. Accordingly, based on the above and the applicant's response and in view of the approval TRACG for ATWS scenarios in Safety Evaluation for NEDE-33083P, Supplement 2, Revision 2, RAI 9.3-11 is resolved.

Likewise, in RAI 9.3-12, the NRC staff requested clarification of the RPV pressures discussed during ATWS scenarios. Specifically, the staff requested clarification of pressures discussed in the DCD and the relation of peak pressure to SLCS injection requirements. The applicant provided the necessary clarification based on pressures calculated by TRACG. This information is subject to NRC approval of TRACG for ESBWR ATWS analysis, as discussed in Chapter 21 of this report. RAI 9.3-12 was being tracked as an open item in the SER with open items. In the safety evaluation for NEDE-33083P, Supplement 2, Revision 2, the staff found the TRACG

modeling with regard to SLCS injection to be conservative. This conservatism involves ignoring heat transfer into the nitrogen accumulator, which would increase its pressure, and using a bounding reactor pressure during standby liquid control (SLC) injection. Accordingly, based on the above and the applicant's response and in view of the approval of TRACG for ATWS scenarios in the safety evaluation for NEDE-33083P, Supplement 2, Revision 2, RAI 9.3-12 is resolved.

The above evaluations demonstrate that the applicant has designed the SLCS with sufficient capability to ensure that the following two safety design bases are met:

1. Provide diverse backup capability for reactor shutdown, independent of normal reactor shutdown provisions, and have full capacity for reducing core reactivity between the steady-state rated operating condition of the reactor with voids and the reactor cold-shutdown condition, including shutdown margin, to ensure complete shutdown from the most reactive conditions at any time in core life.

2. Have full capacity for reducing core reactivity between the steady-state rated operating condition of the reactor with voids and the reactor cold-shutdown condition, including shutdown margin, to ensure complete shutdown from the most reactive conditions at any time in core life.

9.3.5.3.3 Standby Liquid Control System Power Supply, Instrumentation, and Initiation

Each accumulator and its associated valves are powered from a redundant emergency power supply. The redundant injection valves are arranged in parallel so that failure of a single valve will not prevent adequate amounts of sodium pentaborate solution from entering the reactor vessel to cause shutdown. Thus, active components are designed with sufficient redundancy to meet the single-failure criterion.

The safety functions of the SLCS receive power from the safety-related 120-volt alternating current electrical systems. The NRC staff finds this acceptable.

The SLCS is automatically initiated after receiving an ATWS signal, or it can be actuated manually in the control room. Since the SLCS system is started automatically as required by the ATWS rule, the SLCS system meets, in part, the requirements of 10 CFR 50.62. (Section 15.5.4 of this report provides additional discussion.)

The ATWS initiation signals for SLCS automatic start include high RPV pressure or low RPV water level and the APRM not downscale for 3 minutes. This 3-minute delay is provided to allow completion of FMCRD run-in, which will take about 3 minutes. When the SLCS is initiated automatically to inject the boron into the reactor, the four injection valves and the two accumulators will begin discharging simultaneously. The reactor water cleanup isolation valves are closed automatically to prevent a loss of the sodium pentaborate solution from the vessel.

The SLCS can be manually initiated from the MCR if the operator determines that SLCS injection is required to affect a reactor shutdown. Manual initiation is implemented through the ATWS/SLC logic processor. The method of manual initiation will involve multiple, deliberate operator actions to prevent inadvertent boron injection. Procedural controls govern manual initiation of the SLC system.

9.3.5.3.4 Boron Mixing

Adequate boron mixing is required for the SLCS to perform its design function of bringing the reactor from rated power to a cold-shutdown condition without exceeding acceptable fuel design limits. The applicant indicated that adequate boron mixing is ensured by the high injection velocity at which the boron solution enters the core shroud through the SLCS injection spargers, which provide two injection jets at each of four radial positions and four elevations in the lower half of the core, and the natural circulation patterns within the core. To support its conclusions, the applicant included, in the DCD, plots that were generated using TRACG of average core boron concentration versus time for SLCS initiation during ATWS events.

NEDE-33083P, Supplement 2, Revision 2, provides additional information about the applicant's analysis of boron injection into the reactor vessel using TRACG. The report includes information about the SLCS configuration and geometry, as well as the applicant's analysis of SLCS injection behavior. The safety evaluation for NEDE-33083P, Supplement 2, Revision 2, discusses the staff's review of NEDE-33083P, Supplement 2, Revision 2.

The staff identified several phenomena that could challenge the capability of the core's natural circulation patterns to disperse boron uniformly. First, the SLCS injects into the core bypass region within the core shroud. It is expected that the presence of fuel channels and, in the middle of the cycle, some control rods will inhibit planar flow. Second, this core has an unconventionally large diameter, which not only poses another challenge to the passive means of boron mixing, but also means that the core is less neutronically coupled than conventional BWRs. Third, restrictions imposed by two-phase flow will inhibit core upflow and thus further limit boron transport in the core. Additional challenges to axial mixing include the presence of chimneys on top of the core, which would prevent the boron from traveling downward into the core via density-driven flow mechanisms, and the possibility of flow reversal in the event of an MSIV closure.

To correct for local mixing nonuniformities, the applicant designed the SLCS to provide 25 percent more boron than required to bring the reactor to cold shutdown. The injection capability of the SLCS was also increased an additional 15 percent to account for potential dilution by the RWCU/SDC. In RAI 9.3-25, the NRC staff requested information about the technical bases underlying the boron concentration conservatisms applied to the SLCS design. The applicant indicated that these conservatisms are based on and greater than those applied to current BWR operating plants. RAI 9.3-25 was being tracked as an open item in the SER with open items.

It may be noted that during the ATWS/MSIV closure scenario, the applicant took credit for high-pressure CRD system flow. This is acceptable to the staff even though the CRD system is not safety grade. CRD flow provides an active means of recirculating small amounts of water through the core and preventing flow stratification in the lower vessel head.

The staff also requested, in RAI 9.3-25, that the applicant provide additional information about local boron concentration at various regions within the core during the evolution of the ATWS/MSIV closure scenario. The applicant provided a response to this request in response to RAI 21.6-42. The staff reviewed the response to RAI 21.6-42 within the context of its review of the application of the TRACG code for ATWS analyses, as discussed in Chapter 21 of this report. Therefore, RAI 9.3-25 was being tracked as an open item in the SER with open items. In response to RAI 9.3-25, the applicant clarified that the ESBWR boron concentration margin is a typically used value and is supported by the TRACG boron mixing and transport model. In the

safety evaluation for NEDE-33083P, Supplement 2, Revision 2, the staff found the TRACG boron mixing and transport model and its prediction of reactivity worth to be conservative. The local boron concentrations and RAI 21.6-42 are associated with the modeling of the reactor vessel bypass region. In the safety evaluation for NEDE-33083P, Supplement 2, Revision 2, the staff found the TRACG modeling of the reactor vessel bypass region adequate to model the boron mixing effect in ESBWR ATWS events. Based on the applicant's response and the approval of TRACG for ATWS scenarios in safety evaluation for NEDE-33083P, Supplement 2, Revision 2, RAIs 9.3-25 and 21.6-42 are resolved.

DCD Revision 2 did not describe the boron injection path to the core. In RAI 9.3-6, the staff requested the applicant to discuss flow pattern (injection geometry) and movement of injected boron solution through the bypass region. The staff asked the applicant to provide a diagram showing spargers in the core bypass region and the header, feeder pipes, nozzles, discharge ports, and jets. The staff also asked the applicant to describe the positions of the injection points relative to the active length of the core. The applicant provided the requested diagram of the sparger in response to RAI 21.6-53. The applicant also clarified the description of the core bypass sparger used for the boron injection in DCD Tier 2, Revision 4, Table 9.3-4. RAI 9.3-6 was being tracked as an open item in the SER with open items.

The staff accepted the SLCS boron injection path in the context of the staff's review of the application of the TRACG code for ATWS analyses, as discussed in Chapter 21 of this report. The staff finds that the response to RAI 9.3-6 is acceptable since the core bypass sparger is described in DCD Tier 2, Revision 4, and is consistent with the sparger parameters modeled in TRACG. Based on the applicant's response, RAI 9.3-6 is resolved.

9.3.5.3.5 Standby Liquid Control System Emergency Core Cooling System Function

A DPV opening signal initiates the SLCS. This logic is in place to increase the water volume available for injection in the event of a LOCA. If both SLCS trains were activated, a total of approximately 15.6 m^3 (4,121 gal) of borated water would be injected into the core. This would result in the addition of enough borated water to increase the level in the vessel approximately by 0.5 m (1.6 ft).

Since the SLCS is part of the ECCS, the guidelines of GDC 2 (seismic design), GDC 5 (sharing SSCs), GDC 17, "Electric power systems," GDC 27 (capability to cool the core), GDC 35, "Emergency core cooling," GDC 36, "Inspection of emergency core cooling system," and GDC 37, "Testing of emergency core cooling system," are applicable. Section 6.3.1.3 of this report includes the evaluation of the SLCS with regard to these GDC.

9.3.5.3.6 Inspections, Tests, Analyses, and Acceptance Criteria

The staff reviewed the SLCS information in DCD Tier 1, Revision 9, Section 2.2.4. In RAI 9.3-15, the staff requested that the applicant add an ITAAC Table 2.2.4-2, to verify that the initial SLC injection flow rate is consistent with the assumptions in the safety analysis. RAI 9.3-15 was being tracked as an open item in the SER with open items. In response, the applicant stated that, instead of using an injection flow rate, the ITAAC specifies a set of injection volumes and maximum injection times. DCD Tier 1, Revision 5, Table 2.2.4-6, Item 7 specifies that the first 5.4 m^3 (190 ft^3) of solution injects in less than 196 seconds and the first and second 5.4 m^3 (190 ft^3) of solution injects in less than 519 seconds. The staff finds that the response is acceptable since specifying an inspection volume and maximum injection time is equivalent to specifying an average injection flow rate. The staff also confirmed that the criteria in DCD Tier

1, Revision 5, Table 2.2.4-6, Item 7 are consistent with the SLCS design information in DCD Tier 2, Revision 5, Table 9.3-5. Accordingly, based on the above and the applicant's response, RAI 9.3-15 is resolved.

9.3.5.4 *Conclusion*

The NRC staff has reviewed the applicant's information related to the SLCS. For the reasons set forth above, the staff finds that the applicant has adequately demonstrated that the SLCS has the capability for reactor shutdown and core makeup. The staff finds that the SLCS meets the requirements of GDC 2, 4, 5, 26, and 27 and 10 CFR 50.62.

9.3.6 Instrument Air System

9.3.6.1 *Regulatory Criteria*

The staff reviewed the ESBWR IAS in accordance with SRP Section 9.3.1, Revision 3. The staff reviewed DCD Tier 1, Revision 9, Section 2; DCD Tier 2, Revision 9, Section 9.3.6; and various parts of other DCD Tier 2, Revision 9, sections (e.g., Sections 19A). The staff's acceptance of the IAS is based on the design's conformance with the following regulations:

- GDC 1, in part, requires that SSCs important to safety shall be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the function to be performed. Where generally recognized codes and standards are used, they shall be identified and evaluated to determine their applicability, adequacy, and sufficiency and shall be supplemented or modified as necessary to ensure a quality product in keeping with the required safety function.

- GDC 2, in part, requires that SSCs important to safety shall be designed to withstand the effects of natural phenomena such as earthquakes. Compliance with GDC 2 is based on meeting the guidance of Regulatory Positions C.1 and C.2 of RG 1.29.

- GDC 5 requires that SSCs important to safety shall not be shared among nuclear power units unless it can be shown that such sharing will not significantly impair their ability to perform their safety functions, including, in the event of an accident in one unit, an orderly shutdown and cooldown of the remaining units.

- 10 CFR 52.47(b)(1) requires that a design certification application contain the proposed ITAAC that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design certification is built and will operate in accordance with the design certification, the provisions of the Atomic Energy Act, and the NRC's regulations.

9.3.6.2 *Summary of Technical Information*

The IAS is a nonsafety-related system and has no safety design basis. Its function is to provide dry, oil free, filtered compressed air to pneumatically operated valve operators, instrumentation, equipment and components. The pneumatically operated devices having safety-related or RTNSS functions, either have safety-related accumulators or are fail-safe and do not rely on any of the compressed air systems to perform these functions. The IAS is designed to ensure that failure of the IAS does not compromise any safety-related system or component nor does it prevent a safe shutdown.

The IAS makes use of the SAS compressors and receives pre-filtered, oil free, compressed air from the SAS. The IAS consists of two identical 100-percent capacity filtration/dryer trains in parallel, one normally operating and the other in standby. The primary components of each IAS filtration/dryer train are filtering/drying unit, air receiver, and instrumentation, valves and piping. Pre-filtered oil free compressed air from the SAS passes through IAS air filtering/drying units and air receivers before being distributed to the instrument air piping system. A cross-tie between the distribution headers of the SAS and IAS is provided to bypass the IAS filtering/drying units and the air receivers. In the unlikely event that both filtration/dryer trains would fail at the same time, the bypass line is capable of supplying service air directly to the IAS header.

Both IAS filtration/dryer trains are connected to a common header which distributes instrument air to the RW, TB, RB, CB, and FB. The IAS has piping connections outside containment to the HPNSS to serve as a manual backup to the HPNSS, and supplies compressed air to the HPNSS loads inside containment via the HPNSS piping during containment de-inerting operations (i.e. shutdown) and refueling operations.

IAS operational tests, including pre-operational testing as described in DCD Tier 2, Revision 9, Section 14.2.8, in accordance with RG 1.68.3, "Preoperational Testing of Instrument and Control Air Systems," are performed periodically for components to ensure system capability and integrity. Air filters are periodically inspected for cleanliness, and the desiccant in the air dryers is periodically sampled to verify its useful life. Periodic testing of air quality is performed to ensure compliance with American National Standard Institute/Instrument Society of American (ANSI/ISA) 7.0.01, "Quality Standard for Instrument Air." In addition, individual components will be tested for proper "failure" (open, close, or as is) to both instantaneous (pipe break) and slow (plugging or freezing) simulated air losses.

Components of the IAS are designed to meet the ASME Code, Section VIII, Division 1, ASME Power Piping Code B31.1, or ASME Process Piping Code B31.3, as applicable.

9.3.6.3 *Staff Evaluation*

For IAS design, the staff in SRP Section 9.3.1, Revision 2, endorsed the use of ANSI/ISA-S7.3-R1981, "Quality Standard for Instrument air," which is superseded by ANSI/ISA 7.0.01 that establishes the following design guidelines for IAS:

- System design including components such as filters, compressors, air treatment systems, air receivers, drain traps, aftercoolers and moisture separators, pressure regulators, pressure-relief devices, and valves and piping

- Air quality standard including pressure dew point, particle size, lubricant content and contaminants

- Air supply pressure

- Initial start-up test and periodic tests to verify system performance and the above cited air quality

- Continuous monitoring for dew point

The IAS is a nonsafety-related system and it is not considered as a candidate for RTNSS. However, the IAS meets the requirements of GDC 1 as it pertains to instrument air quality standards by meeting ANSI/ISA 7.0.01 and the guidance of RG 1.68.3 related to pre-operational testing of IAS. In addition, the components of the IAS are designed to meet ASME Boiler and Pressure Vessel Code, Section VIII, Division 1, ASME Power Piping Code B31.1, or ASME Process Piping Code B31.3, as applicable. Therefore, the staff finds that the IAS meets the relevant requirements of GDC 1.

Section 14.2 of this report addresses the staff's evaluation of the operational tests including pre-operational testing performed for IAS components to ensure system capability and integrity.

Regulatory Position C.1 of RG 1.29 does not apply to the IAS because the system is a nonsafety-related system and performs no safety-related function. As for the guidance of Regulatory Position C.2 of RG 1.29, the IAS is designed to ensure that failure of the IAS does not compromise any safety-related system or component nor does it prevent a safe shutdown. Pneumatically operated devices are designed for a fail-safe mode on loss of instrument air and do not need a continuous air supply under emergency or abnormal conditions. Therefore, the staff finds that the IAS meets the relevant requirements of GDC 2 because it meets the guidance of Regulatory Position C.2 of RG 1.29 for ensuring that failure of the IAS does not compromise any safety-related system or component nor does such failure prevent a safe shutdown.

The ESBWR design is a single-unit station, and the requirements of GDC 5 are not applicable to the single-unit design.

In RAIs 9.3-33, 9.3-34, and 9.3-35, the staff requested the applicant to clarify common design aspects of the CAS, which contains the IAS, SAS, and HPNSS. In responses, the applicant clarified for the common design aspects of IAS, SAS, and HPNSS. The applicant also clarified that the safety-related components, such as valves and accumulators, are in safety-related actuation systems, not in the compressed air systems. The staff finds these clarifications acceptable. Section 9.3.1 of this report discusses further the evaluation and resolution of RAIs 9.3-33, 9.3-34, and 9.3-35. Accordingly, based on the above and the applicant's responses, RAIs 9.3-33, 9.3-34, and 9.3-35 are resolved.

In addition, an issue concerning impacts of moisture and contamination of the instrument air resulting from the bypass via the cross-tie of lower quality/contaminated SAS was raised during the Advisory Committee on Reactor Safeguards (ACRS) subcommittee meeting on November 15, 2007. This issue was raised again during the ACRS full committee meeting. Consequently, in RAI 9.3-41, the staff requested that the applicant demonstrate how failures of the instrument and controls and pneumatic components resulting from the bypass via the cross-tie of lower quality/contaminated IAS would be prevented.

In response to RAI 9.3-41, the applicant stated:

- Any of the SAS compressors is capable of meeting 100 percent demand of the IAS and each of the dryer trains is sized for 100 percent of the instrument air system demand. If the operating dryer train were to fail, the other dryer train would be placed in service. In the unlikely event that both dryer trains failed at the same time, the bypass line is capable of supplying service air directly to the instrument air header.

- The bypass line is meant to be an emergency backup supply used only when both dryer trains are not available.

- The quality of the air from the service air compressors is oil free with particles less than 10 microns in size. DCD Tier 2, Revision 6, Table 9.3.6, specifies less than 3 microns in size for air particles in IAS. (ANSI/ISA 7.0.01 defines instrument quality air as having a maximum 40 micron particulate size.)

- Moisture content is monitored by the continuous dew point monitor that will alarm in the control room on high moisture content in the air dryer outlet.

- The IAS is tested periodically in accordance with ANSI/ISA 7.0.01 to assure the quality of the air provided.

Based on its review of the above information, the SAS air quality which exceeds the established quality standard in a maximum 40 micron for instrument air (e.g. particle size less than 10 microns versus a maximum 40 micron specified in ANSI/ISA 7.0.01), and the bypass, which is only utilized in an unlikely event that both IAS dryer trains failed at the same time, the staff concludes that impacts of moisture and contamination to the instrument air resulting from the bypass is minimal. Therefore, the staff finds the applicant's response to RAI 9.3-41 acceptable. Based on the applicant's response, RAI 9.3-41 is resolved. Furthermore, the staff considers the above cited issue raised during ACRS meetings resolved.

The staff's determination that impacts of moisture and contamination to the instrument air resulting from the bypass are minimal is also based on the staff's previous findings/conclusion as described below from the assessment of the Generic Issue 43, "Contamination of Instrument Air Lines," and GL 88-14, "Instrument Air Supply System Problems Affecting Safety-Related Equipment."

In July 1981, the staff initiated Generic Issue 43, in response to an event at Rancho SECO Nuclear Generating Station. (The staff considers Generic Issue 43 resolved with the issuance of GL 88-14 on August 8, 1988.) In December 1987, the staff published NUREG–1275, Volume 2, "Operating Experience Feedback Reported - Air Systems Problems." Subsequently, the staff issued GL 88-14 which requested each licensee/applicant to review NUREG–1275, Volume 2, and to perform a design and operations verification of the IAS to verify the following:

- Actual instrument air quality is consistent with the manufacturer's recommendations for individual components served.

- Maintenance practices, emergency procedures, and training are adequate to ensure that safety-related equipment will function as intended on loss of instrument air.

- The design of the entire IAS including air or other pneumatic accumulators is in accordance with its intended function, including verification by test that air-operated safety-related components will perform as expected in accordance with all design-basis events, including a loss of the normal instrument air system.

In addition, the staff in GL 88-14 also requested each licensee/applicant discuss their program for maintaining proper instrument air quality.

In 2005, the staff assessed the effectiveness of Generic Issue 43 and GL 88-14. In conducting this assessment, the staff reviewed licensee event reports, inspection findings, and summary analyses of operating experience, such as initiating events studies and studies of the reliability of air systems and their components. In October, 2005, the staff published its findings in NUREG–1837, "Regulatory Effectiveness Assessment of Generic Issue 43 and Generic Letter 88-14."

On the basis of its assessment in NUREG–1837, the staff concluded that:

- Licensee and agency activities, such as the Maintenance Rule, GL 88-14, design-basis reconstitution, and others, have significantly improved air system and component performance and, thereby, resulted in improved reactor safety.

- Issuance of GL 88-14 and targeted NRC inspections led to the identification and resolution of air system design issues impacting safety-related systems and components, again resulting in improved reactor safety. As a result, based on data for pressurized-water reactors, major losses of instrument air are now infrequent, and prompt recovery from such losses is typical, which supports the staff's conclusion that reactor safety has improved.

- As evidenced by the ongoing discovery and correction of air system issues, licensee programs and NRC oversight activities provide assurance that the NRC and its licensees are effectively maintaining reactor safety in this area.

ANSI/ISA 7.0.01 covers the staff's concerns cited in GL 88-14. For a plant that is not built or licensed yet such as ESBWR, SRP Section 9.3.1, Revision 2 endorses the use of ANSI/ISA standard 7.0.01 and provides guidance for the design of IAS.

On the basis of (1) the ESBWR IAS design meeting the guidance of ANSI/ISA 7.0.01, (2) the operation of the IAS bypass only in an unlikely event that both IAS dryer trains would fail at the same time, (3) the applicant's responses to the staff's RAIs, and (4) the staff's review of Generic Issue 43, GL 88-14, NUREG–1275, Volume 2, and NUREG–1837, the staff finds the above cited issue raised during ACRS meetings concerning the impact of moisture and contaminants from the SAS on IAS is resolved.

The IAS is a nonsafety-related system, has no safety design basis, is not credited to achieve or maintain safe shutdown of the plant, and is not used to provide defense-in-depth capabilities for any safety function. Also, the IAS is not considered as a candidate for RTNSS because it does not meet any of the five criteria as described in SECY-94-084. Therefore, the IAS does not need an ITAAC entry in DCD Tier 1, and the staff finds that IAS meets the relevant requirements of 10 CFR 52.47(b)(1).

9.3.6.4 *Conclusion*

The staff finds that the design of the IAS is acceptable and meets the relevant requirements of GDC 1 and 2 and 10 CFR 52.47(b)(1).

9.3.7 Service Air System

9.3.7.1 *Regulatory Criteria*

The staff reviewed the ESBWR SAS in accordance with SRP Section 9.3.1, Revision 3. The staff reviewed DCD Tier 1, Revision 9, Section 2; DCD Tier 2, Revision 9, Section 9.3.7; and various parts of other DCD Tier 2, Revision 9, sections (e.g., Sections 19A, 22). The staff based its acceptance of the SAS on the design's conformance with the requirements of the following regulations:

- GDC 1 requires, in part, that SSCs important to safety shall be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the function to be performed. Where generally recognized codes and standards are used, they shall be identified and evaluated to determine their applicability, adequacy, and sufficiency and shall be supplemented or modified as necessary to ensure a quality product in keeping with the required safety function.

- GDC 2 requires, in part, that SSCs important to safety shall be designed to withstand the effects of natural phenomena such as earthquakes. Compliance with GDC 2 is based on meeting the guidance of Regulatory Positions C.1 and C.2 of RG 1.29.

- GDC 5 requires that SSCs important to safety shall not be shared among nuclear power units unless it can be shown that such sharing will not significantly impair their ability to perform their safety functions, including, in the event of an accident in one unit, an orderly shutdown and cooldown of the remaining units.

- 10 CFR 52.47(b)(1) requires that a design certification application contain the proposed ITAAC that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design certification is built and will operate in accordance with the design certification, the provisions of the Atomic Energy Act, and the NRC's regulations.

9.3.7.2 *Summary of Technical Information*

The SAS is a nonsafety-related system that provides filtered compressed air for general plant use via service air outlets located outside of the containment and to the IAS. With the exception of the inner and outer containment isolation valves and the pipe of the SAS supply air line which penetrates the containment, in between the two valves, the SAS is not safety-related and serves no safety-related function. Failure of the SAS does not prevent any safety-related equipment from performing its safety-related functions.

The SAS consists of four air compressors capable of supplying two identical trains in parallel. The primary components of the SAS are air intake filter/silencers, air compressors, after-coolers, moisture separators, air receivers, valves, and instrumentation and piping. These components meet the ASME Code, Sections III and VIII, Division 1, ASME Power Piping Code B31.1, and ASME Process Piping Code B31.3, as applicable.

During normal operation, operators select one air compressor for continuous operation, while the other serves as standby and starts automatically if the continuously operating air compressor cannot meet system demand. The operating air compressor that takes suction through an air intake filter/silencer automatically loads or unloads in response to the SAS

demand as determined by pressure changes in the air receivers. Both SAS trains are connected to a common header that distributes air to the RW, TB, and RB. One SAS supply air line which penetrates the containment is provided with redundant manually operated containment isolation valves. These containment isolation valves are in the closed positions during normal plant operation and remain closed following a LOCA.

9.3.7.3 *Staff Evaluation*

The SAS, with the exception of the inner and outer containment isolation valves and the line in between, is not a safety-related system and it is not considered as a candidate for RTNSS. The SAS components meet ASME Code, Sections III and VIII, Division 1, ASME Power Piping Code B31.1, and ASME Process Piping Code B31.3, as applicable. Therefore, the staff finds that the SAS meets the relevant requirements of GDC 1.

With the exception of the inner and outer containment isolation valves and the line in between them, the SAS need not comply with Regulatory Position C.1 of RG 1.29 because it is nonsafety-related and performs no safety-related function. Section 6.2.4 of this report addresses the staff's evaluation ESBWR design of the containment penetration and associated isolation valves. DCD Tier 2, Revision 9, Table 6.2-44, which provides containment isolation valve information for the SAS, describes the one penetration associated with the SAS. As for the guidance of the Regulatory Position C.2 of RG 1.29, the SAS is designed to ensure that failure of the SAS does not compromise any safety-related system or component nor does it prevent a safe shutdown. Therefore, the staff finds that the SAS meets the guidance of Regulatory Position C.2 of RG 1.29 for ensuring that failure of the SAS does not compromise any safety-related system or component nor does it prevent a safe shutdown. Therefore, the staff finds that the SAS meets the relevant requirements of GDC 2.

The ESBWR design is a single-unit station, therefore, the requirements of GDC 5 are not applicable to the SAS.

In RAIs 9.3-33, 9.3-34, and 9.3-35, the staff requested the applicant to clarify common design aspects of the CAS, which contains the IAS, SAS, and HPNSS. In responses, the applicant clarified for the common design aspects of IAS, SAS, and HPNSS. The applicant also clarified that the safety-related components, such as valves and accumulators, are in safety-related actuation systems, not in the compressed air systems. The staff finds these clarifications acceptable. Section 9.3.1 of this report discusses further the evaluation and resolution of RAIs 9.3-33, 9.3-34, and 9.3-35. Accordingly, based on the above and the applicant's responses, RAIs 9.3-33, 9.3-34, and 9.3-35 are resolved.

DCD Tier 1 has one ITAAC entry for the SAS. DCD Tier 1, Revision 9, Section 2.12.8 and Table 2.12.8-1, provide the design descriptions and ITAAC regarding the containment penetration and isolation valves for the SAS. Therefore, the staff finds that SAS complies with the requirements of 10 CFR 52.47(b)(1).

In ESBWR DCD Tier 2, Revision 9, Section 9.3.7, the applicant states that (1) the system operability is demonstrated by use during normal plant operation, (2) system components are shop inspected and tested, (3) system operational tests for components normally closed to airflow are performed periodically to ensure system capability and integrity, and (4) filters are periodically inspected for cleanliness. DCD Tier 2, Revision 9, Section 14.2.8.1.19, addresses the periodic inspection and testing requirements for the IA and SAS.

9.3.7.4 *Conclusion*

The staff finds that the design of the SAS is acceptable and meets the relevant requirements of GDC 1 and 2 and 10 CFR 52.47(b)(1).

9.3.8 High-Pressure Nitrogen Supply System

9.3.8.1 *Regulatory Criteria*

The staff reviewed the ESBWR HPNSS in accordance with SRP Section 9.3.1,Revision 3. The staff reviewed DCD Tier 1, Revision 9, Section 2; DCD Tier 2, Revision 9, Section 9.3.8; and various parts of other DCD Tier 2, Revision 9, sections (e.g., Sections 19A). The staff based its acceptance of the HPNSS on the design's conformance with the requirements of the following regulations:

- GDC 1 requires, in part, that SSCs important to safety shall be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the function to be performed. Where generally recognized codes and standards are used, they shall be identified and evaluated to determine their applicability, adequacy, and sufficiency and shall be supplemented or modified as necessary to ensure a quality product in keeping with the required safety function.

- GDC 2 requires, in part, that SSCs important to safety shall be designed to withstand the effects of natural phenomena such as earthquakes. Compliance with GDC 2 is based on meeting the guidance of Regulatory Positions C.1 and C.2 of RG 1.29.

- GDC 5 requires that SSCs important to safety shall not be shared among nuclear power units unless it can be shown that such sharing will not significantly impair their ability to perform their safety functions, including, in the event of an accident in one unit, an orderly shutdown and cooldown of the remaining units.

- 10 CFR 52.47(b)(1), which requires that a design certification application contain the proposed ITAAC that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design certification is built and will operate in accordance with the design certification, the provisions of the Atomic Energy Act, and the NRC's regulations.

9.3.8.2 *Summary of Technical Information*

With the exception of the inner and outer containment isolation valves and the pipes in between of the HPNSS supply lines that penetrate the containment, the HPNSS is not safety-related and serves no safety-related function. Failure of the HPNSS does not prevent any safety-related equipment from performing its safety-related functions.

The HPNSS function is to distribute nitrogen gas from the CIS to the nuclear boiler system (NBS) automatic depressurization subsystem (ADS) SRV accumulators, the ICS steam and condensate line isolation valve accumulators, and other pneumatically operated valves inside containment. The CIS nitrogen supply line for the HPNSS branches outside the containment into two HPNSS distribution lines that penetrate the containment. One branch line supplies the low-pressure nitrogen loads (i.e., instruments, and pneumatically operated valves) while the other branch supplies the high-pressure nitrogen loads (i.e., NBS ADS SRV accumulators and

the ICS piping isolation valve accumulators). Redundant containment isolation valves are provided where the HPNSS supply lines penetrate the containment. A means is provided for the HPNSS to switch over automatically from CIS to backup nitrogen storage bottles during low CIS supply pressure.

The nonsafety-related piping and valves of the HPNSS meet the ASME Piping Code B31.1. The safety-related portions of valves and piping that provide containment isolation functions meet ASME Code Section III, Division 1, NC requirements for Class 2 components. Pneumatically operated components are designed for a fail-safe mode and do not require continuous air/nitrogen supply under emergency or abnormal conditions. Failure of the HPNSS does not prevent any safety-related equipment from performing its safety-related functions.

9.3.8.3 *Staff Evaluation*

With the exception of the inner and outer containment isolation valves and the pipes in between, the HPNSS is not a safety-related system and it is not considered a candidate for RTNSS. The nonsafety-related piping and valves of the HPNSS meet ASME Power Piping Code B31.1. The safety-related portions of valves and piping that provide containment isolation functions meet ASME Section III, Division 1, NC requirements for Class 2 components. Therefore, the staff finds that the HPNSS meets the relevant requirements of GDC 1.

With the exception of the inner and outer containment isolation valves and pipes in between them, the HPNSS need not comply with Regulatory Position C.1 of RG 1.29 because it is nonsafety-related and performs no safety-related function. Section 6.2.4 of this report addresses the staff's evaluation of the containment penetration and isolation valves for the HPNSS supply lines. As for the guidance of the Regulatory Position C.2 of RG 1.29, the HPNSS is designed to ensure that failure of the HPNSS does not compromise any safety-related system or component nor does it prevent a safe shutdown. Therefore, the staff finds that the HPNSS meets the guidance of Regulatory Position C.2 of RG 1.29 for ensuring that failure of the HPNSS does not compromise any safety-related system or component nor does it prevent a safe shutdown. Therefore, the staff finds that the HPNSS complies with GDC 2.

The ESBWR design is a single-unit station; therefore, the requirements of GDC 5 are not applicable to the HPNSS.

In RAIs 9.3-33, 9.3-34, and 9.3-35, the staff requested the applicant to clarify common design aspects of the CAS, which comprises the IAS, SAS, and HPNSS. In responses, the applicant clarified for the common design aspects of IAS, SAS, and HPNSS. The applicant also clarified that the safety-related components, such as valves and accumulators, are in safety-related actuation systems, not in the compressed air systems. The staff finds these clarifications acceptable. Section 9.3.1 of this report discusses further the evaluation and resolution of RAIs 9.3-33, 9.3-34, and 9.3-35. Accordingly, based on the above and the applicant's responses, RAIs 9.3-33, 9.3-34, and 9.3-35 are resolved.

In addition, the staff issued RAI 14.3-91 regarding ITAAC for the HPNSS.

RAI 14.3-91

RAI 14.3-91 stated the following:

> DCD Tier 1, Table 2.4.1 1, Item 12, lists a test and the associated acceptance criteria for the capacity of the accumulators for the isolation condenser isolation valves. However, DCD Section 5.4.6 does not clearly describe the basis for the specified capacity, and DCD Tier 1, Table 2.1.2-2, does not include similar ITAAC regarding the design capability of the compressed gas accumulators for the MSIV and the safety relief valves. Provide specific ITAAC regarding the capability of each safety-related portion of the compressed gas systems to perform its safety function and the design basis for the capability.

In response to RAI 14.3-91, the applicant addressed the compressed gas accumulators for the MSIV and SRVs but did not address the compressed gas systems. In response to RAI 19.1.0-2 regarding RTNSS, the applicant identified in Table 1, that the HPNSS is a safety-related system credited in the PRA sensitivity study; however, the applicant had neither revised DCD Tier 2, Revision 3, Section 9.3.8, to classify the HPNSS as a safety-related system nor included it in Table 3 as RTNSS. Subsequently, the staff issued supplemental RAI 22.5-3.

RAI 22.5-3

RAI 22.5-3 stated the following:

> In MFN 07-066 (response to RAI 19.1.0-2), Enclosure 1, Table 1, the High Pressure Nitrogen Supply System (HPNSS) is identified as a safety system credited in the PRA sensitivity study. However, in DCD Tier 2, Revision 3, Section 9.3.8 and Section 19A.6.1.2.1 identify HPNSS as a nonsafety-related system. Please clarify the safety/non-safety designation of HPNSS and describe any regulatory treatment of nonsafety system (RTNSS) related functions and interfaces.

The applicant stated the following in response to RAI 22.5-3:

> The HPNSS is a non-safety-related system. HPNSS provides nitrogen to the safety/relief valve and main steam isolation valve accumulators to store the necessary gas volume and pressure to ensure that the safety-related functions can be performed. This function was originally modeled in the ESBWR PRA as an HPNSS basic event, and was set to "True" (that is, failed), in accordance with the focused PRA methodology. The function of charging the accumulators is not an active function and is not a postaccident function. Therefore, other than provision for safety-related containment penetrations and isolation valves, HPNSS does not provide a RTNSS function and will not have ITAAC regarding the capability of each safety-related portion of the compressed gas systems to perform its safety function and the design basis for the capability. Revision 3 of DCD Tier 2 Section 19 was corrected to reflect the fact that HPNSS does not meet RTNSS criteria.

The staff finds that the responses to RAIs 22.5-3 and 14.3-91 are acceptable since the applicant clarified, in the response to RAI 22.5-3 that the safety-related accumulators and not the nonsafety-related compressed gas systems support the active functions of the SRVs and

MSIVs. The staff finds this rationale also supports the applicant's position that the HPNSS does not provide an RTNSS function. Accordingly, based on the above and the applicant's responses, RAIs 14.3-91 and 22.5-3 are resolved.

DCD Tier 1, Revision 6, Subsection 2.15.1, provides the design descriptions and ITAAC regarding the containment penetration and isolation valves for the HPNSS. Therefore, the staff finds that HPNSS complies with the requirements of 10 CFR 52.47(b)(1).

DCD Tier 2, Revision 9, Section 14.2.8.1.20, addresses the operational tests including pre-operational testing performed for HPNSS components to ensure system capability and integrity.

9.3.8.4 *Conclusion*

The staff finds that the design of the HPNSS is acceptable and meets the relevant requirements of GDC 1 and 2 and 10 CFR 52.47(b)(1).

9.3.9 Hydrogen Water Chemistry System

9.3.9.1 *Regulatory Criteria*

There are no regulatory requirements for the hydrogen water chemistry system (HWCS). For the ESBWR, it is a nonsafety-related system that could be used by the COL holder to reduce the likelihood of corrosion failures that would adversely affect plant availability." The SRP, through March 2007, does not include a section specifically addressing the HWCS. The staff reviewed the HWCS to ensure that no safety implications are associated with the HWCS as described in the DCD.

9.3.9.2 *Summary of Technical Information*

DCD Tier 2, Revision 9, Section 9.3.9, contains information on the HWCS. The HWCS is composed of hydrogen and oxygen supply systems to inject hydrogen in the feedwater and oxygen in the offgas to convert residual hydrogen to water. The standard plant design includes the capability to incorporate an HWCS, but the system itself is not part of the ESBWR standard plant design. That is, the HWCS is an optional system to be specified by the COL applicants. The HWCS does not perform any safety-related functions.

9.3.9.3 *Staff Evaluation*

The design of the HWCS makes provisions to allow for the installation of a system to add hydrogen to the feedwater at the suction of the feedwater pumps. The system includes monitoring systems to track the effectiveness of the HWCS. DCD Tier 2, Revision 9, Section 9.3.9.6, identifies two COL information items related to the HWCS. COL Information Item 9.3.9-1-A states that the COL applicant will determine whether an HWCS is to be implemented. COL Information Item 9.3.9-2-A states that the COL applicant will provide the hydrogen and oxygen storage facility design and appropriate supply system if it elects to install an HWCS. The staff finds COL Information Items 9.3.9-1-A and 9.3.9-2-A acceptable since the use of hydrogen and oxygen supply systems is site dependent.

The HWCS is nonsafety-related. However, given the potential for hydrogen deflagration or detonation, it is required to be safe and reliable, consistent with the requirements for using hydrogen gas. The applicant stated that the HWCS uses the guidelines in the EPRI Report NP-

4947-SR, "BWR Hydrogen Water Chemistry Guidelines," 1987 Revision. This report describes the methods used to operate the HWCS.

In RAI 9.3-1, the staff asked the applicant to clarify whether the means for storing and handling hydrogen comply with EPRI Report NP-5283-SR-A, "Guidelines for Permanent BWR Hydrogen Water Chemistry Installations." In response, the applicant stated that the HWCS is an option for the COL applicant or holder, if the plant shows a need for the HWCS. The applicant stated that any HWCS installation would have to meet the guidelines in EPRI Report NP-5283-SR-A. The report provides guidance to store and handle hydrogen at nuclear power facilities. The staff has approved EPRI Report NP-5283-SR-A in its SER for the Licensing Topical Report, "Guideline for Permanent BWR Hydrogen Water Chemistry Installations," July, 1987.

The staff did not find the response acceptable because it is not clear whether the COL applicant or the COL holder would be responsible for the above COL information item. In RAI 9.3-37, the staff requested clarification from the applicant. In response, the applicant modified the DCD to state that the COL applicant is responsible for determining whether to install an HWCS. The staff finds that the applicant's responses are acceptable since EPRI Report NP-5283-SR-A is an approved approach and the applicant clarified that the COL applicant is responsible for the HWCS. Accordingly, based on the above and the applicant's responses, RAIs 9.3-1 and 9.3-37 are resolved. The staff confirmed that the applicant included the identified changes in DCD Revision 5.

9.3.9.4 *Conclusion*

The staff concludes that no safety implications are associated with the HWCS as described in the DCD. The staff finds that the EPRI guidelines describe a satisfactory means for storing and handling hydrogen for the ESBWR design. The HWCS is an optional system that, if specified by the COL applicant, will inject hydrogen in the feedwater at the suction of the feedwater pumps. The COL applicant shall specify, and the NRC staff shall review, any safety implications of an HWCS as necessary.

9.3.10 Oxygen Injection System

9.3.10.1 *Regulatory Criteria*

There are no regulatory requirements for the oxygen injection system (OIS). It is a nonsafety-related system that is used to add oxygen to the condensate and feedwater system to reduce corrosion and suppress corrosion product release. The SRP, through March 2007, does not include guidance for the staff to review this nonsafety-related system. The staff reviewed the OIS to ensure that no safety implications are associated with the OIS, as described in the DCD, and to determine whether this system follows the guidelines in EPRI Report NP-5283-SR-A.

9.3.10.2 *Summary of Technical Information*

DCD Tier 2, Revision 9, Section 9.3.10 provides information on the OIS. The OIS is designed to add oxygen to the condensate and feedwater system in order to reduce corrosion and suppress corrosion product release. Industry experience has shown that the most beneficial oxygen concentration is between 30 to 200 parts per billion (ppb). The OIS does not perform any safety-related functions.

9.3.10.3 *Staff Evaluation*

The OIS is designed to add sufficient oxygen (30 to 200 ppb) to reduce corrosion and the release of corrosion products in the condensate and feedwater system. EPRI Report NP-5283-SR-A provides guidelines for the design, operation, maintenance, surveillance, and testing of the oxygen storage facility. In RAI 9.1-38, the staff requested that the applicant clarify whether the means for storing and handling oxygen comply with EPRI Report NP-5283-SR-A. In response, the applicant stated that the OIS is part of the ESBWR standard plant design and is not determined by the COL applicant. Implementation of the HWCS changes the demand for oxygen as well as the storage requirements. DCD Tier 2, Revision 9, Section 9.3.10.6, identifies one COL information item related to the OIS. COL Information Item 9.3.10-1-A states that the COL applicant will provide a description of the oxygen storage facility. If the HWCS is implemented, the hydrogen and oxygen storage facilities will comply with the guidelines of EPRI Report NP-5283-SR-A. The staff finds COL Information Item 9.3.10-1-A acceptable since the use of an oxygen storage facility is depends whether an HWCS is used, which is site dependent.

However, the staff did not find the applicant's response acceptable because it was unclear whether the OIS would need to meet the guidelines of EPRI Report NP-5283-SR-A if the HWCS were not implemented. In RAI 9.3-38 S01, the staff requested that the applicant clarify which document contains the requirements for the design, operation, maintenance, surveillance, and testing of the oxygen storage facility and discuss how the ESBWR meets those requirements, if the OIS does not need to meet the guidelines of EPRI Report NP-5283-SR-A. RAI 9.3-38 was being tracked as an open item in the SER with open items. In response, the applicant revised the DCD to state that the OIS uses the guidelines for gaseous oxygen injection systems in EPRI Report NP-5283-SR-A, 1987 Revision. The staff finds that the response is acceptable since the staff finds that the EPRI guidelines describe a satisfactory means for storing and handling oxygen for the ESBWR design. EPRI Report NP-5283-SR-A provides guidance to store and handle oxygen at nuclear power facilities. The staff has approved EPRI Report NP-5283-SR-A in its SER for the Licensing Topical Report, "Guideline for Permanent BWR Hydrogen Water Chemistry Installations," July, 1987. Accordingly, based on the above and the applicant's response, RAI 9.1-38 is resolved. The staff confirmed that the applicant incorporated the identified changes into DCD Revision 5.

9.3.10.4 *Conclusion*

The staff concludes that no safety implications are associated with the OIS, as described in the DCD. The staff finds that the EPRI guidelines presented describe a satisfactory means for storing and handling oxygen for the ESBWR design. The COL applicant shall specify, and the NRC staff shall review, any safety implications of oxygen storage facilities as necessary.

9.3.11 Zinc Injection

9.3.11.1 *Regulatory Criteria*

There are no regulatory requirements for the zinc injection system (ZIS). The ZIS is a nonsafety-related system that is used optionally, as identified by the COL applicant, to control the buildup of radiation in corrosion films on primary system piping and components. The SRP, through March 2007, does not include a section specifically addressing the ZIS. The staff reviewed the ZIS to ensure that no safety implications are associated with the ZIS, as described in the DCD.

9.3.11.2 *Summary of Technical Information*

DCD Tier 2, Revision 9, Section 9.3.11, contains information on the ZIS. The ZIS is a nonsafety-related system that is used optionally by the COL holder to control the buildup of radiation in corrosion films on primary system piping and components. The standard plant design includes the capability to incorporate a ZIS, but the system itself is not part of the ESBWR standard plant design. The ZIS does not perform any safety-related functions.

9.3.11.3 *Staff Evaluation*

The control of buildup of radiation in reactor systems is a concern in BWR plants. Laboratory testing and plant experience have shown that the presence of trace amounts of soluble zinc in reactor water reduces cobalt-60 buildup in the corrosion films on primary system piping and components.

The applicant has made provisions to permit installation of a system for adding a zinc solution to the feedwater. The applicant stated that the COL applicant or holder shall determine whether a ZIS is required based on the site-specific water quality requirements. In RAI 9.3-39, the staff requested that the applicant clarify whether the decision to implement the ZIS is the responsibility of the COL applicant or the COL holder. In response to RAI 9.3-39, the applicant stated that the COL applicant determines whether a ZIS is warranted based on plant configuration and material selection. Additionally, the COL applicant is required to include the necessary information for system description, tests, and inspections if a ZIS is implemented. DCD Tier 2, Revision 9, Section 9.3.11.6, includes these issues as two COL information items. COL Information Item 9.3.11-1-A states that the COL applicant shall determine if implementation of a ZIS is required at startup based on plant configuration and material selection. COL Information Item 9.3.11-2-A states that if a ZIS is to be installed, the COL applicant shall include necessary information for system description, test, and inspection. The staff finds COL Information Items 9.3.11-1-A and 9.3.11-2-A acceptable since the use of a ZIS is site dependent. The staff finds that the response to RAI 9.3-39 is acceptable since the applicant clarified that the COL applicant is responsible for the use of a ZIS. Accordingly, based on the above and the applicant's response, RAI 9.3-39 is resolved. The staff concludes that there are no safety implications associated with the ZIS as described in the DCD.

9.3.11.4 *Conclusion*

Based on the above discussion, the staff concludes that no safety implications are associated with the ZIS as described in the DCD. The ZIS is an optional system, and the COL applicant will provide the system description, tests, and inspections, if implemented.

9.3.12 Auxiliary Boiler System

9.3.12.1 *Regulatory Criteria*

The auxiliary boiler system (ABS) is a nonsafety-related system and has no safety design basis. The SRP, through the March 2007 revision, does not include a section specifically addressing the auxiliary boiler/steam system. However, the staff reviewed DCD Tier 2, Revision 9, Section 9.3.12, against the requirements of the following GDC to ensure that failure of the ABS as a result of a pipe break or malfunction of the system cannot adversely affect any safety-related systems or components:

- GDC 2, in part, requires that SSCs important to safety shall be designed to withstand the effects of natural phenomena such as earthquakes.

Compliance with GDC 2 is based on meeting the guidance of Regulatory Positions C.1 and C.2 of RG 1.29.

9.3.12.2 *Summary of Technical Information*

The primary ABS components which are located in the auxiliary boiler building contain the following:

- One 100-percent-capacity fire tube auxiliary boiler composed of two 50-percent-capacity fuel oil boilers

- Two complete firing systems, including fuel-oil burners and fans

- Two 100-percent-capacity fuel oil transfer pumps

- Three 50-percent-capacity auxiliary boiler feedwater pumps

- One 100-percent-capacity deaerator with integral storage tank

- One 100-percent-capacity auxiliary boiler blowdown flash tank

- One 100-percent-capacity steam separator

- Instrumentation and controls

During plant startup and shutdown, as well as at normal operation (if required), the ABS provides the necessary nonradioactive steam for the following:

- Steam jet air ejectors
- Turbine gland sealing system
- Feedwater system for preheating during plant startup
- Preoperational testing of off-gas system equipment
- Evaporation of liquid nitrogen for inerting of the containment

The auxiliary boilers boil demineralized water to produce steam during plant startup, shutdown, and offline operation when main steam is unavailable. ABS fuel oil transfer pumps transfer fuel oil from the SDG fuel oil storage tank to the auxiliary boilers. The ABS fuel oil transfer pump suction lines are connected to the SDG fuel oil storage tank at the level which is necessary to maintain the minimum fuel oil inventory for the SDG system. The makeup water system provides makeup feedwater to the ABS.

9.3.12.3 *Staff Evaluation*

The staff identified that DCD Revision 3 did not contain the information needed to determine that the failure of the ABS resulting from a pipe break or malfunction of the system would not adversely affect safety-related systems or components. In RAI 9.3-40, the staff requested the applicant to identify whether the ABS would interface directly with any nuclear process systems, the location of the auxiliary boiler, and whether the ABS lines would pass through areas in which

safety-related equipment is located. RAI 9.3-40 was being tracked as an open item in the SER with open items.

In responses to RAI 9.3-40, the applicant stated the following:

- The ABS does not interface directly with nuclear process systems.
- The auxiliary boiler is located outside the TB, adjacent to the RW.
- ABS piping is routed in the TB.
- Safety-related RPS sensors are located in the TB.

However, the applicant did not specifically address the impact of a failure of the ABS on the safety-related sensors. In RAI 9.3-40 S01, the staff requested the applicant to confirm whether failure of the ABS system as a result of a pipe break or malfunction of the system would adversely affect safety-related systems or associated components and instrumentation.

In response to RAI 9.3-40 S01, the applicant provided a list of safety-related sensors mounted on or potentially mounted near nonsafety-related piping and structures in the TB. The TB included in the ESBWR standard plant design is nonsafety-related. However, the TB structure is designed to prevent a failure of the structure that would impair the ability of nearby safety-related SSCs, including safety-related sensors, from performing their functions. In addition, the potential adverse effect is mitigated by the fail-safe design of the sensors and their respective control systems to provide safety system protection.

The staff finds that the RAI response is acceptable because the applicant demonstrated that failure of the system as a result of a pipe break or malfunction of the system would not adversely affect safety-related systems or components. Accordingly, based on the above and the applicant's response, RAI 9.3-40 is resolved.

Regulatory Position C.1 of RG 1.29 does not apply to the ABS because the system is a nonsafety-related system and performs no safety-related function. As for the guidance of Regulatory Position C.2 of RG 1.29, the ABS is designed to ensure that failure of the ABS does not compromise any safety-related system or component or prevent a safe shutdown. Therefore, the staff finds that the ABS meets the relevant requirements of GDC 2 because it meets the guidance of Regulatory Position C.2 of RG 1.29 for ensuring that failure of the ABS does not compromise any safety-related system or component or prevent a safe shutdown.

The ABS is nonsafety-related, and is not relied upon to achieve or maintain safe shutdown of the plant. Furthermore, the ESBWR design does not use the ABS to provide defense-in-depth capabilities for any safety function. In addition, the ABS is not considered a candidate for RTNSS system, because it does not meet any of the five criteria described in SECY-94-084.

9.3.12.4 *Conclusion*

The staff finds that the design of the ABS is acceptable and meets the relevant requirements of GDC 2.

9.4 Air Conditioning, Heating, Cooling, and Ventilation Systems

9.4.1 Control Building Heating, Ventilation, and Air Conditioning System

9.4.1.1 *Regulatory Criteria*

The staff reviewed DCD Tier 2, Revision 9, Section 9.4.1, in accordance with SRP Section 9.4.1, Revision 3. The staff's acceptance of the control building heating, ventilation, and air conditioning system (CBVS) is based on compliance with the following requirements:

- GDC 2, as it relates to SSCs 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

- GDC 4, as it relates to SSCs important to safety being designed to accommodate the effects of and to be compatible with the environmental conditions associated with postulated accidents

- GDC 5, as it relates to ensuring that sharing among nuclear power units of SSCs important to safety will not significantly impair the ability to perform safety functions, including, in the event of an accident in one unit, an orderly shutdown and cooldown of the remaining unit or units

- GDC 19, as it relates to maintaining the nuclear power unit in a safe condition under accident conditions and providing adequate radiation protection

- GDC 60, as it relates to the nuclear power unit design, including the means to control suitably the release of radioactive materials in gaseous and liquid effluents and to handle radioactive solid wastes produced during normal reactor operation, including anticipated operational occurrences

- 10 CFR 50.63 as it relates to necessary support systems providing sufficient capacity and capability to ensure the capability to cope with an SBO event

- 10 CFR 20.1406, as it relates to the minimization of contamination

9.4.1.2 *Summary of Technical Information*

The CBVS serves all areas of the CB during normal operation. The CBVS maintains space design temperatures, air quality, and pressurization. It provides a controlled environment for personnel safety and comfort and for the proper operation and integrity of equipment located in the CB. The CBVS consist of two systems: the control room habitability area HVAC subsystem (CRHAVS) and the control building general area HVAC subsystem (CBGAVS).

The CRHAVS serves the MCR and associated support areas that comprise the control room habitability area (CRHA). The CRHA envelope can be isolated and protected during emergency modes of operation. When ac power is available, the CRHAVS provides HVAC functions for the CRHA via two nonsafety-related redundant fresh air supply fans and two redundant nonsafety-related internal floor mounted AHUs. Radiological protection is provided from a redundant set of safety-related emergency Filter Units (EFUs).

When ac power is not available, the CRHA is cooled passively via heat transfer to the CRHA passive heat sink, and radiological protection continues to be provided from the safety-related EFUs, which are powered from the safety-related 1E battery power source. The safety-related portions of the CRHAVS include the EFUs and their associated fans; ductwork; instrumentation and controls; the CRHA boundary envelope; and the CRHA isolation dampers and associated ductwork. All remaining CRHAVS equipment is nonsafety-related. The CRHA isolation dampers automatically close to isolate the CRHA envelope, and an EFU is automatically actuated in the event of a loss of normal ac power or during a radiological event.

The CRHAVS provides the following safety-related design basis functions:

- Monitor the CRHA air supply for radioactive particulate, iodine concentrations, or both.

- Isolate the normal CRHA air supply and restroom exhaust, and start an EFU fan.

- Align the air supply through an EFU upon a high radiation detection signal in the CRHA normal air supply or upon an extended loss of ac power.

The portions of the CRHAVS which penetrate the CRHA envelope are safety-related and designed as seismic Category I to provide isolation of the CRHA envelope from the outside and surrounding areas in the event of a DBA. The EFU portion of the subsystem is safety-related and designed and supported as seismic Category I, including the air intakes, ductwork, dampers, fans, and instrumentation and controls. The remaining CRHAVS functions are nonsafety-related. The penetrations contain safety-related isolation dampers or valves that fail closed upon a loss of control signal, power, or instrument air.

An EFU is automatically actuated upon radiological isolation of the CRHA envelope or an extended loss of ac power. If the initial EFU fails to start or is otherwise unavailable, the second standby EFU automatically actuates.

The CBGAVS serves the area outside the CRHA. The CBGAVS is nonsafety-related. The subsystem is made up of two subsets, Set A and Set B, each of which contain a single AHU enclosure with two redundant 100-percent capacity supply fans, internal coils and filters, and associated return/exhaust fans and ductwork.

The AHU subsystems are recirculation type AHUs that recirculate most of the ventilation air and combine it with a smaller quantity of fresh outside air. Set A serves its respective HVAC equipment room, the A N-DCIS room, and the Division 1 and 4 safety-related distributed control information system (Q-DCIS) rooms. Set B serves its respective HVAC equipment room, the B N-DCIS Room, the Division 2 and 3 Q-DCIS rooms, and the corridor area around the CRHA.

CBVS equipment and ductwork whose failure could affect the operability of safety-related systems or components are designed as seismic Category II. The remaining portion of the system is nonsafety-related and nonseismic.

The following CRHA components are safety-related and Seismic Category I:

- CRHA boundary envelope, including structures, doors, and components (including variable orifice relief device)

- EFUs, including HEPA and carbon filters and related system components

- Ductwork from the CRHA boundary envelope up to and including the CRHA isolation dampers

- Tornado dampers, which are provided on EFU air intake openings and are designed to withstand the full negative pressure drop

- Tornado and tornado missile protection provided on all CRHA ventilation penetrations for outside air intake and exhaust openings

- Tornado and tornado missile protection provided on the CBVS outside air intake and return/exhaust openings

The CBVS provides a safety-related means to passively maintain habitable conditions in the CRHA following a DBA (radiological event concurrent with a loss of normal ac power). Radiation detected in the CRHA outside air inlet causes the following actions:

- The normally closed isolation dampers downstream of the operating EFU fan open.
- The normal outside air inlet and restroom exhaust dampers close.
- An EFU fan automatically starts.

The CRHA is isolated during loss of normal ac power conditions and a safety-related EFU provides pressurization and breathing quality air. An EFU is powered from the safety-related battery supply for 72 hours. For longer-term operation (post-72 hours) either of two ancillary diesel generators (ADGs) can power either EFU fan system.

The EFU delivery and discharge system is optimized to ensure that adequate fresh air is delivered and mixed in the CRHA. This is accomplished by using multiple supply registers, which distribute the incoming supply air within the control room air volume and a remote exhaust (variable orifice relief device) to prevent any short cycling. The EFU operation results in turning over the control room volume approximately seven to nine times per day.

This diffusion design (mixing and displacement), in conjunction with the known convective air currents (from heat loads and sinks) and personnel movement, ensures that occupied zone temperature is within acceptable limits. Buildup of contaminants (e.g., carbon dioxide) is minimal, and a freshness of air is maintained.

The CBVS provides the capability to maintain the integrity of the CRHA with redundant safety-related isolation dampers in all ductwork penetrating the CRHA envelope. The active safety-related components (CRHA isolation dampers and EFUs), which ensure habitability in the CRHA envelope, are redundant. Two trains of safety-related EFUs, including high-efficiency particulate air (HEPA) and carbon filters, serve the CRHA envelope. Redundant fans are provided for each EFU to allow continued operability during maintenance of electrical power supplies. Therefore, a single active failure cannot result in a loss of the system design function.

During normal modes of operation and emergency modes with electrical power available, the CRHA is maintained within the temperature and relative humidity (RH) ranges noted in Table 9.4-1 by the nonsafety-related CRHAVS recirculation AHU. During emergency operation, with a loss of normal ac power, a nonsafety-related CRHA recirculation AHU, powered from the nonsafety-related uninterruptible ac power supply (UPS) system, maintains the CRHA within the normal operating temperature range for two hours. This allows the continued operation of certain high heat producing MCR N-DCIS electric loads.

Anytime during a loss of normal ac power, once either ADG is available, the power for either recirculation AHU fan with an auxiliary cooling unit can be provided via the ancillary diesel-powered generator. Thus, a recirculation AHU can operate indefinitely during a CRHA isolation event. If the recirculation AHUs are not available during the loss of normal ac power, safety-related temperature sensors with two-out-of-four logic automatically trip the power to selected N-DCIS components in the MCR, thus removing the heat load caused by these sources. In the event the duration of the loss of normal ac power duration extends beyond two hours, the CRHA heat sink passively cools the reduced CRHA heat load. The CRHA heats sinks consist of the CRHA walls, floor, ceiling, and interior walls; and CRHA access corridors; adjacent Q-DCIS and N-DCIS equipment rooms and electrical chases; and CRHA HVAC equipment rooms and HVAC chases. The CRHA heat sinks limit the CRHA temperature to a maximum temperature value of 33.9 degrees C (93 degrees F) for 72 hours. For the full duration of the DBA, the EFU maintains the safety-related habitability of the CRHA by supplying filtered air for breathing and pressurization to minimize inleakage. During the initial 72 hours, the EFU relies on safety-related batteries. In the post-72 hour period, the EFU relies on RTNSS power supplies.

The auxiliary cooling units provide full-capacity cooling and ventilation for the CRHA, 72 hours after an accident. The auxiliary cooling units are air cooled chillers located in the CB mechanical equipment room, outside of the CRHA, with remote condensers. The auxiliary cooling system provides chilled water to the cooling coils in both the CRHAVS recirculation AHUs and the CBGAVS supply AHUs, located in the MCR and mechanical equipment rooms respectively. This includes auxiliary cooling unit chilled water recirculation pumps, independent of the normal CWS.

The MCR operator starts the auxiliary cooling system in an accident scenario (post-72-hour) when the ADG provides ac power. Interlocked motor operated isolation valves will close off the chilled water supply from the normal CWS and open the supply from the auxiliary cooling units. After the valves are in the proper lineup, the auxiliary cooling system starts. All valves are located outside the CRHA. The valves are provided with power from a system designated as an RTNSS system. This power is available 72 hours after onset of an accident. The CRHA recirculation AHUs, CB general area supply AHUs, and supporting auxiliary cooling units also use power from a system designated as an RTNSS system to remove heat in support of post-72-hour MCR habitability.

The CBVS has RTNSS functions, as described in DCD Tier 2, Revision 9, Appendix 19A, which provides the level of oversight and additional requirements to meet the RTNSS functions. Performance of RTNSS functions is assured by applying the defense-in-depth principles of redundancy and physical separation to ensure adequate reliability and availability. In addition, augmented design standards are applied, as described in DCD Tier 2, Revision 9, Section 19A.8.3.

The CBVS has the following functions:

- Provide a controlled environment for personnel comfort and safety. Sufficient outside air is provided to meet the standards for acceptable indoor air quality (Section 6 of American Society of Heating, Refrigeration and Air Conditioning Engineers (ASHRAE) 62.1-2007, "Ventilation for Acceptable Indoor Air Quality") DCD Table 9.4.1 depicts the area design temperature and humidity design parameters.

- Provide a controlled environment for the proper operation and integrity of equipment in the CB during normal, startup, and shutdown operations.

- Maintain higher than atmospheric (positive) pressure to minimize the infiltration of outside air. Construction materials and processes ensure that the CB structure maintains low leakage or leak tight conditions above and below grade. The CRHA envelope penetrations are sealed and access doors are designed with self-closing devices that close and latch the doors following use. Double door airlocks in the CRHA envelope allow access and egress during emergencies when the CRHA is isolated and an EFU is operating.

- Reduce the potential spread of airborne contamination by maintaining airflow from areas of lower potential for contamination to areas of greater potential for contamination. The CRHA is maintained at a higher pressure than surrounding areas except during the isolation and smoke exhaust modes.

- Detect and limit the introduction of airborne hazardous materials (radioactivity or smoke) into the CRHA.

- Provide the capability to exhaust smoke, heat, and gaseous combustion products from inside the CB to the outside atmosphere in the event of a fire. Construction processes ensure that materials of construction are non-combustible and heat and flame resistant wherever possible. Materials that produce toxic or noxious vapors when subjected to a fire are avoided.

- Use smoke control and removal functions that are in accordance with NFPA guidelines, as described in DCD Tier 2, Revision 9, Section 9.5.1.11.

- The design is such that failure of nonsafety-related equipment does not compromise or otherwise damage safety-related equipment.

The CBVS subsystems, the CRHAVS and the CBGAVS, are recirculating ventilation systems that provide filtered, conditioned air to serve all areas of the CB.

The EFUs provide breathing air and pressurization to the CRHA when the CRHA envelope is isolated from a loss of ac power or high airborne radioactivity. The CBVS maintains space design temperatures and air quality. Outside air is normally supplied to augment the return air to maintain the CB under a slightly positive pressure. The CBGAVS return/exhaust fans normally direct most of the system airflow back to the system return flow, with a portion of the flow exhausted to the atmosphere. The CBVS provides a controlled environment for personnel safety and comfort and for the proper operation and integrity of equipment located in the CB.

CBVS equipment, including fans, AHUs, EFUs, and the CRHA are located within the CB seismic Category I structural areas.

The CRHAVS is configured as a recirculation system, which contains the entire supply and return AHU air flow inside the CRHA and incorporates a common supply duct for introducing outside air to the CRHA. The normal and EFU outside air intake flows are adjusted as necessary to maintain a minimum flow and, in conjunction with a controlled leak path, maintain a 31 Pascal gauge (PaG) (1/8" w.g.) minimum positive pressure in the CRHA. Backflow prevention through the controlled leak path, the variable orifice relief device, is not necessary since the CRHA is at a positive pressure during normal and emergency operation.

The intake design and location are in accordance with RG 1.194, "Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power

Plants." Intake design, location and control also include considerations that minimize the introduction of radiological material, toxic gases, hazardous chemicals, smoke, dust, and other foreign material. Ductwork, housings, and access openings, as well as other design features, are constructed in such a manner as to minimize in leakage of potentially contaminated air into the CRHAVS air stream.

During normal operation, air is conditioned and distributed by an AHU and particulates are removed from the air by medium efficiency filters. Heat is transferred between the air and the heating and cooling coils inside the AHU. Moisture is added to the air stream, if necessary, to maintain minimum humidity levels in the CRHA by the automatically controlled humidifier. The heating and cooling processes inherently remove moisture from the air stream and maintain the humidity below the maximum specified level. The supply AHU distributes conditioned air beneath the CRHA raised floor to the CRHA rooms via registers in the raised floor. The AHU intake is ducted to a location above the suspended ceiling and return air is returned to the AHU via registers in the suspended ceiling.

The CRHA recirculation AHUs provide cooling to the CRHA whenever offsite or onsite ac power is available. The nonsafety-related ac UPS System provides power for the CRHA recirculation AHUs. Each recirculation AHU is equipped with an auxiliary cooling unit with a cooling coil in the AHU. The recirculation AHU fans and associated auxiliary cooling units are battery powered during the first 2 hours of a loss of normal ac power event from the nonsafety-related battery supply. Anytime during a loss of normal ac power event, once either ADG is available, the power for either recirculation AHU fan with auxiliary cooling unit can be provided via the ADG. Thus, a recirculation AHU can operate indefinitely during a CRHA isolation event. If the recirculation AHUs are not available during the loss of normal ac power event, safety-related temperature sensors with two-out-of-four logic automatically trip the power to selected N-DCIS components in the MCR, thus removing the heat load caused by these sources.

Each EFU consists of a medium efficiency filter (40 percent minimum), a HEPA filter (99.97 percent) a carbon adsorption filter (99 percent credited efficiency), and a post-filter downstream of the carbon filter (95 percent). The EFUs operate only during a radiological emergency or a loss of normal ac power and are able to function while powered from an offsite ac source, an onsite ac source, or an onsite safety-related dc source.

The EFUs are monitored by instrumentation that detects a loss of airflow and detects radiation downstream of the EFU filters. Upon such detection, the operating EFU is isolated and the standby EFU is automatically placed in service.

Each EFU provides sufficient quality air to maintain positive pressure in the CRHA when the CRHA envelope is isolated. An EFU is automatically actuated when the CRHA envelope is isolated during a loss of ac power or because of high airborne radioactivity. Controls to manually isolate the CRHA envelope and to manually actuate the EFUs are also provided.

The CBGAVS serves non-divisional equipment rooms, corridors, and other miscellaneous rooms in the CB general areas. Set A serves Division 1 and 4 areas. Set B serves Division 2 and 3 areas. Each set is configured as a recirculation system that incorporates a common supply and return duct system for the distribution of conditioned air. During normal operation, air travels through the AHU stages. Particulates are removed from the air by low and high efficiency filters. Heat is transferred between the air and the heating and cooling coils. The outside air intake and exhaust are adjusted to maintain a slightly positive pressure in the CB general areas.

9.4.1.3 *Staff Evaluation*

The staff's review focused on compliance with regulatory requirements for this system. The staff also reviewed the RTNSS functions for the CBVS, as stated in DCD Tier 2, Revision 9, Appendix 19A, against guidance for selection and identification of such systems in accordance with RG 1.206, Section C.IV.9. The staff used additional guidance documents to evaluate the CBVS passive cooling features as described below.

GDC 2 requires that SSCs important to safety be designed to withstand the effects of natural phenomena, such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches without loss of capability to perform their safety functions. The staff evaluated whether the CRHAVS meets the requirements of GDC 2. The CRHA envelope comprises seismic Category I structures and components that are protected from postulated tornados, hurricanes, tsunamis, seiches, and seismic events. The CRHAVS components are designated as seismic Category II, with the exception of the safety-related CRHA envelope, isolation dampers, the EFUs and associated fans, dampers, ductwork, and instrumentation and controls, which are seismic Category I. The CB structure is a seismic Category I structure. The remaining portion of the CBVS is the CBGAVS, which serves the area outside the CRHA and is nonsafety-related. GDC 2 does not apply to the CBGAVS since this system and its components are not considered important to safety.

In RAI 6.4-23, the staff requested that the applicant revise the DCD to clarify the function, seismic, and safety classification of the variable orifice device, which is used to maintain the pressurization of the CRHA. In response, the applicant revised DCD Tier 2, Sections 6.4.2, 6.4.4, 6.4.7, 9.4.11, and 9.4.1.2. The applicant revised DCD Tier 1, Table 2.16.2-3, to include the CRHA variable orifice relief device as a safety-related, seismic Category 1 component. The staff finds the proposed DCD changes acceptable since they clearly identify the function, seismic, and safety classification of the variable orifice device. Based on the above and the applicant's response, RAI 6.4-23 is resolved.

In RAI 9.4-37, the staff requested that the applicant clarify whether portions of the CBVS penetrating the CRHA should be classified as safety-related since they provide isolation of the CRHA envelope from the outside and surrounding areas in the event of a DBA. RAI 9.4-37 was being tracked as an open in the SER with open items. In response, the applicant clarified that all components that provide isolation of the CRHA envelope are safety-related. The applicant also modified the list of safety-related CRHA components in DCD Revision 4. The staff finds that the response, along with the changes in DCD Revision 4, is acceptable since the applicant identified appropriate safety-related components. Based on the above, the applicant's response, and the DCD revision, RAI 9.4-37 is resolved.

Based on the above, the staff finds that the CBVS meets the requirements of GDC 2.

GDC 4 requires that SSCs important to safety are designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including LOCAs. The staff evaluated whether the CRHAVS meets the requirements of GDC 4. The safety-related CRHA envelope, isolation dampers, EFUs and associated fans, dampers, ductwork, and instrumentation and controls are designed to be protected from all postulated environmental and dynamic effects. The remaining portion of the CBVS is the CBGAVS, which serves the area outside the CRHA and is nonsafety-related. GDC 4 does not apply to the CBGAVS since this system and its components are not considered important to safety. The safety and nonsafety-related portions of the CBVS are

located in the CB which is a seismic Category I structure. The safety and nonsafety-related portions of the CBVS are located in mild environment. The staff finds the design of the safety-related portions of the CBVS satisfies GDC 4 regarding potential dynamic effects, such as pipe whip, jet impingement and missile impacts caused by equipment failure or events outside the plant. The CBVS is designed such that failure of nonsafety-related equipment does not compromise or otherwise damage safety-related equipment. Based on the above, the staff finds that the CBVS meets the requirements of GDC 4.

GDC 5 requires that SSCs important to safety shall not be shared among nuclear power units unless it can be shown that such sharing will not significantly impair their ability to perform their safety functions, including, in the event of an accident in one unit, an orderly shutdown and cooldown of the remaining units. The ESBWR design is a single-unit station, and the requirements of GDC 5 are not applicable to the single-unit design.

GDC 19 requires that a control room be provided from which actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions, including LOCAs. It also requires that adequate radiation protection be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident. Implicit in GDC 19 is that the environmental conditions (such as temperature, humidity, and oxygenation) will be acceptable for personnel and equipment to function.

In the ESBWR design, the CRHA is designed to perform its safety-related functions for 72 hours without ac power. Therefore, the staff evaluated the CRHA in accordance with the requirements of 10 CFR 50.63 concurrent with GDC 19. As it relates to the CRHAVS, 10 CFR 50.63 involves providing assurance that necessary operator actions can be performed and that necessary control room-area equipment will be functional under the expected environmental conditions during and following an SBO, thereby ensuring that the core will be cooled and appropriate containment integrity will be maintained. Regulatory Position C.3.2.4 of RG 1.155, “Station Blackout,” provides guidelines regarding evaluating habitability and environmental conditions during an SBO.

In RAIs 9.4-5, 9.4-5 S01, and 9.4-5 S02, the staff asked the applicant to provide a list of codes and standards used in the design of the ESBWR air conditioning, heating, cooling, and ventilation systems. RAI 9.4-5 was being tracked as an open item in the SER with open items. In response to RAI 9.4-5 S02, the applicant clarified that the applicable codes and standards are discussed in the relevant sections describing the EBWR HVAC systems. The applicant also provided a table in the RAI response showing where relevant standards are discussed throughout the DCD. The staff finds that the response is acceptable since the staff confirmed that the applicable standards were discussed with the relevant systems. Based on the above and the applicant’s response, RAI 9.4-5 is resolved.

The staff reviewed the CRHAVS for radiation protection and for the establishment of acceptable environmental conditions. Radiation protection is provided by isolation, by use of a safety-related EFU, and by pressurization of the control room to minimize unfiltered in leakage.

Normal Operation

The staff reviewed temperature control, air supply distribution, and air mixing for normal operations and finds that the ESBWR CRHA design provides sufficient conditioned air with adequate recirculation by the nonsafety-related supply fans and the RTNSS qualified AHUs with the associated heating and cooling coils. Humidity control is also provided in the recirculation AHU. The system is powered by the station ac system. The applicant states in DCD Tier 2, Revision 9, Table 9.4-1, that the CRHA normal operating temperature will be no greater than 21.1 degrees C (74 degrees F). This maximum operating temperature is within the guidance for the normal temperature range for the control room, as stated in Section 8.2.2.1 of the EPRI Utility Requirements Document (URD), which is endorsed by NUREG 1242, and is therefore acceptable.

Post-Accident with No Loss of ac Power Supply

Since the RTNSS qualified AHUs remain operational whenever offsite or onsite ac power is available, the staff finds that temperature control for post-accident operation is adequate for such accidents. In the case of a loss of normal ac power supply, for the first 2 hours after the loss of the normal ac power supply, the CRHA isolation dampers automatically close and an EFU is automatically started. The nonsafety-related ac UPS system provides power for the CRHA Recirculation AHUs. Each recirculation AHU is equipped with an auxiliary cooling unit with a cooling coil in the AHU. During this period, the power for either recirculation AHU can be provided via an ADG. Since the RTNSS qualified recirculation AHUs remain operational, the staff finds that temperature control for post-accident operation is adequate for accidents in which RTNSS power sources are available or in which normal ac power is restored within 2 hours.

Operation 0-72 hours Post-Accident -Loss of ac Power Supply- Radiation Protection

The staff reviewed the design of the CRHAVS to ensure that adequate radiation protection is provided to permit access and occupancy of the control room in the MCR, in accordance with GDC 19, during the first 72 hours after the onset of an accident that assumes the loss of nonsafety-related ac power for the entire 72-hour period. SRP Sections 6.4 and 9.4.1 identify that these requirements may be addressed by CRHA isolation, an emergency standby atmosphere filtration system that conforms to the guidelines of RG 1.52, and control room in-leakage that is testable in conformance with RG 1.197, “Demonstrating Control Room Envelope Integrity at Nuclear Power Reactors.”

As described in the DCD, the CRHAVS performs the safety-related functions to isolate the CRHA, start an EFU fan, and align the air supply through an EFU upon a high radiation detection signal in the CRHA normal air supply or upon an extended loss of ac power.

CRHA envelope isolation is achieved by closure of redundant isolation dampers on the smoke purge exhaust, toilet exhaust, and normal supply air penetrations. The isolation dampers are seismically qualified and safety-related. The dampers close upon high radiation signals. The dampers also close on loss of power, loss of air, or control signal failures. The portions of the CRHAVS that penetrate the CRHA envelope are safety-related and designed as seismic Category I to provide isolation of the CRHA envelope from the outside and surrounding areas in the event of a DBA. Because the CRHAVS isolation is achieved by means of safety-related equipment, the staff finds the isolation of the CRHA acceptable.

As stated in DCD Tier 2, Revision 9, Section 9.4.1, the CRHAVS EFUs meet the guidance of RG 1.52 as it relates to the design, inspection and testing criteria for the post-accident-engineered safety feature atmosphere cleanup system air filtration and adsorption units. The staff identified that in the technical specifications, DCD Tier 2, Revision 3, Chapter 16, Section 5.5.13, the applicant specified an in-place aerosol leak test criterion of less than 1.0 percent for the charcoal adsorber. Section 6.4 of RG 1.52, Revision 3, issued June 2001, includes a criterion of less than 0.05 percent. In RAI 9.4-35, the staff requested that the applicant correct the criteria in the DCD or justify the exception to the guidance of RG 1.52. In addition, the applicant specified a laboratory methyl iodide penetration test criterion for the carbon adsorber of 1.0 percent. The allowed penetrations in RG 1.52 are 2.5 percent for a 5-centimeter (cm) (2-in.) bed filter and 0.5 percent for a 10 cm (4-in.) bed filter. In RAI 9.4-36, the staff requested that the applicant explain the basis for the laboratory test criteria used to support the 99-percent credited efficiency and provide, in the DCD, the thickness of the charcoal bed. RAIs 9.4-35 and 9.4-36 were being tracked as open items in the SER with open items.

In response to RAIs 9.4-35 and 9.4-36, the applicant revised DCD Tier 2, Chapter 16, Section 5.5.13, to include the in-place aerosol leak test criterion of less than 0.05 percent and the laboratory methyl iodide penetration test acceptance criterion of less than 0.5 percent penetration. The applicant specified the thickness of the charcoal beds to be greater than or equal to 10 cm (4 in.), as specified by RG 1.52. The staff finds that the responses are acceptable since they resulted in changes to bring the DCD into conformance with RG 1.52. Accordingly, based on the above, the applicant's response and DCD changes, RAIs 9.4-35 and 9.4-36 are resolved.

Based on the above, the staff finds that the ESBWR emergency standby atmosphere filtration system (the EFUs), conforms to the guidelines of RG 1.52.

DCD Tier 2, Section 6.4.7, states that the testing of the integrity of the CRHA envelope is performed in accordance with RG 1.197. In RAI 6.4-22, the staff requested that the applicant specify and justify the value for the CRHA access and egress leakage limit or clarify in the DCD that an ESBWR-COL applicant would provide such information. The staff requested this information since the applicant proposed taking credit for near-zero or zero inleakage for CRHA access and egress. The staff request was also based on SRP Section 6.4 and RG 1.197 guidance, which identifies that the acceptance criteria for CRHA unfiltered in leakage during leak testing of the CRHA envelope may not be greater than the amount of unfiltered leakage assumed in the dose consequence analysis minus the amount of unfiltered inleakage allocated for CRHA access and egress.

In response to the RAI, the applicant revised DCD Tier 2, Section 6.4.7, to specify 2.3 l/s (5 cfm) instead of "near-zero" as the amount of unfiltered inleakage allocated for CRHA access and egress. The applicant also clarified DCD Tier 2, Chapter 16, Section 5.5.12, to indicate that the quantitative limit of unfiltered air inleakage test will be the inleakage flow assumed in the design basis analyses of DBA consequences less the amount designated for ingress and egress. The staff finds that the specified unfiltered inleakage allocation of 2.3 l/s (5 cfm) as proposed in the RAI response is reasonable as discussed below. In addition, the change to DCD Tier 2, Chapter 16, Section 5.5.12, clearly allocates the allowed inleakage and is therefore acceptable to the staff. Based on the applicant's response and DCD changes, RAI 6.4-22 is resolved.

SRP Section 6.4 and RG 1.197 guidelines state that the staff considers 4.6 l/s (10 cfm) to be a reasonable estimate for ingress and egress for control rooms without vestibules and that lower values could be considered with additional design features. The ESBWR CRHA design

includes double-vestibule type door air locks for access and egress during emergencies. The access doors are designed with self-closing devices, which close and latch the doors automatically. DCD Tier 2, Section 9.4.1, states that during a radiological event or upon loss of normal ac power, an EFU maintains a positive pressure in the CRHA to minimize infiltration of airborne contamination. The interlocked double-vestibule type doors maintain the positive pressure, thereby minimizing infiltration when a door is opened. Based on the above design features, the staff finds that 2.3 l/s (5 cfm) is a reasonable value to allocate to the access and egress portion of the unfiltered inleakage.

With the clarification of the access and egress inleakage, the staff finds that the test acceptance criterion for CRHA unfiltered inleakage conforms to the RG 1.197 guidance.

Based on conformance with the guidance in RG 1.52 and RG 1.197 for design of the safety-related EFU and by provisions for isolation and pressurization of the control room to minimize unfiltered inleakage, the staff finds that the ESBWR CBVS meets the radiation protection requirements of GDC-19.

Operation 0-72 Hours Post-Accident – Loss of ac Power Supply- Evaluation of CRHA Temperature and Air Quality at 0-72 Hours – Introduction

The ESBWR CBVS incorporates a design feature of reliance on passive safety systems to provide cooling of the CRHA via absorption of heat in the CB concrete to maintain temperature control for 72 hours after the onset of those accidents in which all safety-related ac power is lost. In addition to the regulatory criteria cited in Section 9.4.1.1 of this report, the staff used additional guidance from the following documents to evaluate the adequacy of the unique features of the ESBWR CRHA for such accidents:

- NUREG–1242, as it applies to control room envelope atmosphere temperature limits
- ASHRAE Standard 62.1/2007, as it applies to CRHA indoor air quality standards and acceptance criteria
- NUREG–0700, "Human-System Interface Design Review Guidelines," as it applies to the use of the wet bulb globe temperature index in evaluation of heat stress conditions
- Staff requirements memoranda on SECY 94-084, June 30, 1994, and SECY 95-132, June 28, 1995, as they apply to RTNSS to address uncertainties as a defense-in-depth method

The applicant proposed air quality and temperature and humidity limits based on or derived from these standards. The staff reviewed the proposed standards and acceptance criteria and finds them acceptable for use evaluating of the ESBWR passive control room design as explained below.

Evaluation Methodology

The applicant has proposed an analytical approach, detailed in NEDE-33536, "Control Building and Reactor Building Environmental Temperature Analysis" (hereafter referred to as the CB Environmental Temperature Analysis) as a means to demonstrate the passive heat removal mechanism. The analysis evaluates heat transfer by use of the CONTAIN 2.0 computer code.

In order to determine if this approach was valid the staff reviewed industry literature[4] and current practice of use of this code in containment analysis. In addition, NRC staff developed a first principle model (FPM) as an additional tool to assess the CONTAIN analysis of the ESBWR control room habitability submitted by the applicant as a part of the licensing basis. The objective of the FPM is to independently simulate the effect of the cyclic outdoor air dry-bulb temperature (DBT) and humidity on the ESBWR control room DBT and humidity over the post-accident 72-hour period, when filtered outdoor air is supplied after the failure of the nonsafety-related portions of the HVAC system. Based on the similarity of the output obtained from the applicant's CONTAIN analysis and the staff's independent FPM analysis, and in light of current industry practice of using CONTAIN in other applications, the staff finds that the applicant's use of an analytical approach as a method to demonstrate the passive heat removal mechanism and to demonstrate that CRHA bulk temperature will not exceed design basis limits is reasonable for the ESBWR CRHA. The staff evaluation of the analysis itself is set forth below.

Evaluation Input Assumptions: Outside Environmental Conditions

Since the ESBWR is a passive plant, the CRHA passive safety features need to be evaluated under DBA conditions to ensure that they can perform their safety-related functions without nonsafety-related ac power for 72 hours.

DCD Tier 2, Revision 9, Section 6.4.4 defines the CRHAVS DBA conditions, which include a LOCA with a LOOP. This DBA also takes no credit for the operation of the nonsafety-related ac UPS System or the ADG. It assumes that ac power from nonsafety sources is not restored until 72 hours after the accident. The DBA was evaluated at the two summer conditions and one winter condition identified in DCD Tier 2, Revision 9, Table 2.0-1, for ambient design temperature. The applicant modeled the site parameters in CONTAIN 2.0 as the following: (1) the 0 percent exceedance summer design condition of 47 degrees C (117 degrees F) dry bulb with a mean coincident 26.7 degrees C (80 degrees F) wet bulb (modeled as 47 degrees C (117 degrees F) with a 20-percent RH), (2) the 0 percent exceedance summer design non-coincident condition of 31.1 degrees C (88 degrees F) wet bulb (modeled as 33.3 degrees C (92 degrees F) with a 85-percent RH), and (3) the winter design condition of -40 degrees C (-40 degrees F) (modeled as -40 degrees C [-40 degrees F]). The staff finds the modeling of the site parameters in CONTAIN 2.0 consistent with DCD Tier 2, Revision 9, Table 2.0-1.

Since the applicant has chosen the most limiting (0 percent exceedance) site parameters for the ESBWR design as set forth in DCD Tier 2, Chapter 2, Table 2.0-1, as input assumptions, the staff finds the outside environmental DBA conditions chosen for evaluation of the performance of the CRHAVS passive cooling features reasonable.

Evaluation Input Assumptions: CRHA Heat Loads and Heat Sinks

The staff reviewed the input parameters used in the applicant's CB environmental temperature analysis, such as heat sink wall thickness and surface area, against values for the same parameter when described elsewhere in the DCD. When input parameters depend on site-specific information, realistic or conservative parameters are used, such as the assumed as-built thermophysical properties of CB concrete, orientation of the CB for highest solar radiation, a 15-percent margin in the assumed sensible heat load, an assumed CRHA failure 8 hours before the postulated accident (resulting in increased CRHA air and heat sink temperatures at

4 Yilmaz T.P. & Paschal W.B., "An Analytical Approach to Transient Room Temperature Analysis, *Nuclear Technology*, 114:135-140, April 1996.

the start of the analysis). The applicant assumed the highest normal operating temperature allowed in the ESBWR TS as the initial heat sink temperature. In addition, the applicant used higher heat sink temperatures for walls in contact with the ground than would be expected.

In RAI 9.4-32, the staff requested that the applicant clarify the need to provide cooling to nonsafety-related heat loads in the CRHA following an accident. RAI 9.4-32 was being tracked as an open item in the SER with open items. The applicant clarified that, as stated in DCD Tier 2, Section 9.4.1.2, CRHA nonsafety-related heat loads are automatically de-energized when the CRHA AHUs are not available. No operator action is needed to isolate the nonsafety-related heat loads as safety-related temperature sensors automatically trip the N-DCIS components. The staff finds that the response is acceptable since the nonsafety heat loads are de-energized when CRHA AHUs are not available, and applicant's CB environmental temperature analysis does not need to consider the performance of the nonsafety CRHA AHUs. Based on the above and the applicant's response, RAI 9.4-32 is resolved.

In RAI 9.4-57, the staff requested that the applicant describe how the design basis assumptions on the passive heat sink features and heat loads, such as CRHA occupancy, will be controlled throughout the life of the plant. In response to the RAI, the applicant revised DCD Tier 2, Section 6.4.7, to specify that design changes to the CRHA will ensure that key design assumptions, such as heat sink and heat source assumptions, remain valid. The applicant indicated that DCD Tier 2, Section 17.4, ensures that relevant aspects of plant operation are maintained. COL Information Item 6.4.1-A directs the COL applicants to develop procedures to control such parameters for the CRHA. The staff finds that the response is acceptable since the DCD revisions provide a means to ensure that CRHA heat sink features remain bounded by the design basis assumptions. Accordingly, based on the above, the applicant's response and DCD changes, RAI 9.5-57 is resolved.

Based on review of the submitted analysis, the staff finds that the applicant's input assumptions are either based on information described elsewhere in the DCD or use realistic or conservative assumptions for CRHA heat loads and heat sinks. Therefore, these assumptions are acceptable.

Proposed CRHA Air Quality Acceptance Criteria

The staff reviewed the CRHAVS capability to maintain adequate carbon dioxide concentration in the CRHA. DCD Tier 2, Revision 9, Section 6.4.1.1, states that the emergency habitability system is designed to maintain the ASHRAE fresh air standards for up to 21 MCR occupants (ASHRAE Standard 62.1/2007). The EFU System is designed to maintain carbon dioxide concentration in the CRHA at less than 5,000 ppm, which is the upper limit carbon dioxide defined by ASHRAE. The staff considers the CRHA similar to an office environment where light work is performed. NRC guidelines for human system interfaces (NUREG–0700) cite this reference in its guidelines for workplace design. Since the ESBWR CRHA is designed to meet this major industry standard for indoor air quality which includes criteria for carbon dioxide concentration, the staff finds the proposed CRHA air quality acceptance criteria acceptable. Evaluation of the ESBWR CRHA design to meet this acceptance criterion is discussed below.

Proposed ESBWR CRHA Minimum Temperature Acceptance Criteria

The staff evaluated the proposed ESBWR CRHA minimum temperature criteria. For the first 72 hours following onset of an accident, the CRHA is heated by safety-related CRHA

equipment, and the CRHA is passively heated through exterior walls, floor, ceiling, and interior walls.

DCD Tier 1, Revision 9, Table 2.16.2-4, Design Commitment 4, states that the CRHAVS heat sink passively maintains the temperature of the CRHA within an acceptable range for the first 72 hours following a DBA. The acceptance criteria is that the minimum bulk average CRHA temperature will not be below 12.8 degrees C (55 degrees F) upon a loss of active cooling for 72 hours, given winter post-DBA conditions.

The staff reviewed this criterion against NUREG–0700, Section 12.1.2.1-1, which provides guidance for control room environment temperature winter range. The staff also reviewed NUREG–0700, Section 12.2.5.2-3, which provides guidance for the effects of cold on performance. While the proposed acceptance criterion is below the 20 degrees C (68 degrees F) minimum value for the comfort zone for winter, it is not below the thresholds in NUREG–0700, Table 12.9, for temperatures above which no cold effects occur for tasks such as tracking and having effects of cold on the hands. NRC guidance in NUREG–0700 indicates that a temperature of 12.8 degrees C (55 degrees F) would not significantly affect operator performance. Therefore the staff finds the CRHA minimum temperature acceptance criterion acceptable. The staff evaluation of the ESBWR CRHA design regarding maximum temperature acceptance criterion is below.

Proposed ESBWR CRHA Maximum Temperature Acceptance Criteria

The staff evaluated the proposed ESBWR CRHA maximum temperature criterion. For the first 72 hours following onset of such an accident, safety-related CRHA equipment is passively cooled through the walls, floor, ceiling and interior walls. DCD Tier 2, Revision 9, Table 9.4-1, states that the CRHA is designed such that the maximum CRHA temperature is limited to a value of 33.9 degrees C (93 degrees F). DCD Tier 1, Revision 9, Table 2.16.2-4, Design Commitment 4, states that the CRHAVS heat sink passively maintains the temperature of the CRHA within an acceptable range for the first 72 hours following a DBA. The acceptance criteria is that the CRHA maximum bulk average air temperature will not exceed 33.9 degrees C (93 degrees F) on a loss of active cooling for 72 hours given DBA conditions.

Section 8.2.2.1 of Chapter 9 of the EPRI URD, which is endorsed by the staff in NUREG–1242, states that provisions will be made to limit the average room temperature rise to a maximum of 8.3 degrees C (15 degrees F) at the end of the postulated 72-hour accident for a control room that has a normal temperature range maintained at 22.8 - 25.6 degrees C (73 - 78 degrees F).

Based on the applicant’s chosen maximum normal design temperature of 23.3 degrees C (74 degrees F), the ESBWR design maximum accident CRHA temperature of 33.9 degrees C (93 degrees F) results in a temperature rise of 10.6 degrees C (19 degrees F). This exceeds the temperature rise limit guidance in the EPRI URD and NUREG–1242; however, it would be within the 93 degrees F maximum temperature allowed by the URD for a control room with a normal temperature maximum value of 25.6 degrees C (78 degrees F). Therefore, the staff finds the proposed CRHA maximum temperature acceptance criterion acceptable because it is in accordance with the EPRI URD and NUREG–1242. As described below, the staff has considered the impact of this maximum control room temperature criterion on equipment and operator performance.

Evaluation of Impact of CRHA Temperature Acceptance Criteria on CRHA Equipment

The staff evaluated whether the maximum CRHA temperature acceptance criterion value of 33.9 degrees C (93 degrees F), as stated in DCD Tier 2, Revision 9, Table 9.4-1, supports the mild environment equipment qualification.

In RAI 9.4-34, the staff requested that the applicant clarify if the design considers the reduced airflow and locally increased temperature inside electrical cabinets during the period of passive cooling and if those temperatures pose a challenge to equipment operation. RAI 9.4-34 was being tracked as an open item in the SER with open items.

In the related RAI 3.11-28, the staff requested that the applicant provide additional details on how the service temperature of electrical equipment, including computer-based instrumentation and control systems, will be determined for the ESBWR. In particular, the staff asked the applicant to provide details on this process for equipment that is expected to be located inside electrical cabinets and panels in the RB and CB. The staff also asked the applicant to explain how the detailed design and testing of electrical equipment, including enclosures, would be carried out such that the key assumptions of environmental bounding temperatures in these areas remain conservative.

In response to RAIs 9.4-34 and 3.11-28, the applicant revised DCD Tier 2, Sections 3.11.1.3, 3.11.4.3, and 3.11.3.1, to more fully explain the temperature qualification process. The applicant clarified the definition of "Equipment" in DCD Tier 2, Section 3.11.1.3, to indicate that computer-based instrumentation and control equipment is defined by the equipment plus its surrounding enclosure. The applicant clarified DCD Tier 2, Section 3.11.4.3, to indicate that system testing of computer-based instrumentation and control equipment within its cabinet or enclosure is preferred.

In DCD Tier 2, Section 3.11.3.1, the applicant stated that the CRHA environmental qualification equipment is to be tested at temperatures that are 10 degrees C (18 degrees F) higher than the maximum temperature to which the equipment is exposed for the worst case abnormal operating occurrence, with the equipment at maximum loading. In response to RAI 3.11-37, the applicant clarified that the ESBWR complies with RG 1.209, "Guidelines for Environmental Qualification of Safety-Related Computer-Based Instrumentation and Control Systems in Nuclear Power Plants," which endorses EPRI Topical Report (TR) 107330, "Generic Requirements Specification for Qualifying Commercially Available PLC for Safety-Related Applications in Nuclear Power Plants." The ESBWR follows the TR guidance on an acceptable method for addressing mild-environment qualification of Programmable Logic Controllers (PLCs). The environmental temperature limit in EPRI TR-107330 is 60 degrees C (140 degrees F) plus 2.7 degrees C (5 degrees F) margin for a total temperature of 62.7 degrees C (145 degrees F) for abnormal operating occurrences in a mild environment. This far exceeds the maximum mild environment temperature of 33.9 degrees C (93 degrees F) proposed for this zone.

In addition, DCD Tier 2, Revision 9, Section 3.11.3.2, states that the qualification parameters will include margins to account for normal variations in commercial production of equipment and reasonable errors in defining satisfactory performance and that the environmental conditions shown in the Appendix 3H tables do not show such margins. The staff noted that, in DCD Tier 2, Section 3.11.3.2, the applicant noted that the program margin would be in accordance with the guidance of IEEE Standard (Std) 323, "IEEE Standard for Qualifying Class IE Equipment for Nuclear Power Generating Stations." IEEE Std 323 recommends that a peak temperature

margin of +8 degrees C (+14 degrees F) be applied during the temperature qualification process. Since the applicant is conducting type testing with a margin of 10 degrees C (18 degrees F), the staff finds that the proposed margin exceeds the IEEE Std 323 guidelines.

Thus, since CRHA computer-based instrumentation and control equipment is to be type-tested at 60 degrees C (140 degrees F), with margin, there is significant margin to equipment failure if the actual local temperatures exceed the calculated maximum average CRHA bulk temperature of 33.9 degrees C (93 degrees F) by several degrees. Based on the margin in the assumed normal operating temperature used in the CB environmental temperature analysis, and the conservatism inherent in equipment type-testing, the staff finds that local temperatures are not likely to challenge component operability before ac power is restored 72 hours from the onset of the accident. The staff finds that, independent of operator actions or offsite support, the CBVS design maintains satisfactory environmental conditions for equipment to function for the first 72 hours after the onset of an accident that assumes that all ac power is lost for this period. The staff also finds that the maximum CRHA temperature value of 33.9 degrees C (93 degrees F) supports mild environment equipment qualification temperature conditions in the CRHA. Based on the applicant's responses, RAIs 9.4-34 and 3.11-28 are resolved.

Impact of CRHA Temperature Acceptance Criteria on CRHA Personnel

The staff evaluated whether the maximum CRHA temperature acceptance criterion value of 33.9 degrees C (93 degrees F), as stated in DCD Tier 2, Revision 9, Table 9.4-1, supports satisfactory human performance.

The staff considered the impact of operators operating in an elevated temperature environment. As shown in DCD Tier 2, Revision 9, Figure 3H-2, the applicant's CB environmental temperature analysis indicates that the CRHA dry bulb temperature would reach 30 degrees C (86 degrees F) in approximately 12 hours. After 12 hours, the temperature rate of change is much lower, reaching a CRHA bulk temperature of 33.5 degrees C (92.5 degrees F) at 72 hours. Humidity may also increase from moisture contained in the supply air. Based on NRC and industry standards, the staff noted that human performance is most frequently assessed based on the wet-bulb globe temperature index (WBGT).

In RAI 6.4-24 and it supplements, the staff requested that the applicant justify use of psychrometric wet bulb temperature as a valid index to assess heat stress in the ESBWR CRHA, or alternatively, to amend the DCD to provide a heat stress acceptance criterion and index that is in accordance with NRC guidance. The staff also requested that the applicant demonstrate that such a criterion can be met for the ESBWR environmental footprint. The staff requested that the applicant clarify associated ITAAC for this criterion.

In response to RAI 6.4-24, the applicant revised the DCD to state that the WBGT index would be the design basis means by which a heat stress acceptance criterion would be measured. The applicant stated that the CRHA is designed such that a WBGT of 32.2 degrees C (90 degrees F) would not be exceeded at the end of 72 hours of passive cooling.

The staff reviewed the proposed DCD revisions and acceptance criterion against NRC and industry guidance and finds that the applicant's chosen WBGT index acceptance criterion for heat stress at the end of 72 hours of passive cooling would not need compensatory actions, such as stay times, to be implemented. Specifically, NUREG–0700, Section 12.2.5.1, which provides guidelines for addressing heat stress, identifies that no limits in stay times are applicable below a WBGT of 32.2 degrees C (90 degrees F) for low-metabolic work with normal

work clothes, which is typical of work in the control room. Therefore, the staff finds that the ESBWR CRHA temperature and humidity at the end of 72 hours of passive cooling is acceptable in terms of human performance. Accordingly, based on the above, the applicant's response and DCD changes, RAI 6.4-24 is resolved.

Control Building Environmental Temperature Analysis for the ESBWR

The staff reviewed the means by which the applicant analyzed the CRHAVS heat sink to ensure that the heat sink passively maintains the temperature in the CRHA within the design basis for the first 72 hours following a DBA. To verify this design feature is by means of a CB environmental temperature analysis using heat sink dimensions, thermal properties, exposed surface areas, and the heat loads specified in DCD Tier 2, Revision 9, Table 3H-14. The analysis evaluates heat transfer by use of the CONTAIN 2.0 computer code. As previously discussed, the staff reviewed the use of this code for this application, the analysis input assumptions, and the limiting site parameters for the ESBWR design and finds them to be acceptable.

Temperature Evaluation-Summer Case

The staff evaluated the applicant's submitted CB environmental temperature analysis, the purpose of which is to demonstrate that the final CRHA bulk average temperature does not exceed the proposed acceptance criteria. DCD Tier 2, Revision 9, Section 3H.3.2, describes the applicant's CB environmental temperature analysis. DCD Tier 2, Revision 9, Tables 3H-14 and 3H-15, respectively identify the input assumptions and the results of the CB environmental temperature analysis. NEDE-33536P, which is a Tier 2* reference in DCD Tier 2, Revision 9, Section 3H, provides a detailed description of the CB environmental temperature analysis. The results indicate that the maximum bulk average temperature reached in the CRHA during the 0-72 hour period is less than 33.9 degrees C (93 degrees F), which satisfies the applicant's acceptance criteria.

In RAIs 9.4-33 and 9.4-33 S01, the staff requested that the applicant provide a detailed heat transfer study of the passive heat removal mechanisms, including the analytical assumptions. RAI 9.4-33 was being tracked as an open item in the SER with open items. In response to RAI 9.4-33, the applicant provided the CB environmental temperature analysis assumptions for the control room design and outside environmental conditions for a single node model of the CRHA that demonstrates the mechanism by which heat is removed (i.e., the absorption of heat by thermal mass of concrete). The staff noted some conservative parameters in the analysis, as compared to that specified in the DCD. Based on sensitivity studies conducted by the staff, the most significant of these is the applicant's conservative use of thermophysical properties of lighter concrete than specified in the DCD: 1922.2 kg m^3 (120 lb ft^3) versus 2394.8 kg m^3 (149.5 lb ft^3). In addition the applicant applied a 2000 W (1.7 x 10^6 Calories per hour) margin to the expected sensible heat load in the CRHA.

The staff then consolidated a number of concerns regarding the CB environmental temperature analysis into a new RAI. In RAI 9.4-55 the staff requested that the applicant incorporate the CB environmental temperature analysis in the DCD.

In response to RAI 9.4-55, the applicant submitted the analysis, NEDE-33536P, as DCD Tier 2, Reference 3H.4-8, and indicated in DCD Tier 2, Section 3H.3.2.1 and Table 1.6-1, that this report is Tier 2* information.

The staff finds that the response to RAI 9.4-55 addresses the concerns for RAIs 9.4-33 and 9.4-55 since NEDE-33536P provides a methodology to show that both the baseline CB described in the DCD and the as-built CB will meet the CRHA maximum temperature criteria. The applicant revised the DCD to incorporate a specific analysis methodology to analyze the as-built design and this methodology was reviewed and considered acceptable for this application. Based on the above, the applicant's responses, and DCD changes, RAIs 9.4-33 and 9.4-55 are resolved in terms of incorporating the CB environmental temperature analysis in the DCD. RAIs 9.4-33 and 9.4-55 are discussed below regarding ITAAC.

The staff has reviewed the results of the applicant's CB environmental temperature analysis, as described in NEDE-33536P, as a basis for designing the MCR HVAC systems as stated in Chapter 9, Section 8.2.2.1, of the EPRI URD and SRP Section 9.4.1. The staff reviewed of the applicant's calculation and performed confirmatory calculations using the same methodology and input assumptions. The staff obtained similar results.

Therefore, the staff finds that the applicant's analysis adequately demonstrates that the bulk average CRHA temperature will meet the acceptance criteria value in the 0-72 hour period after an accident in which nonsafety-related ac power is not available. Furthermore, the ESBWR CRHA meets the design guidance for maximum control room temperature stated in the EPRI URD and NUREG–1242.

Temperature Evaluation Winter Case

The staff also reviewed the impact of low temperature air at the winter design condition temperature of -40 degrees C (-40 degrees F), on control room operators. The applicant provided an analysis that indicated that the CRHA bulk temperature will not be below 16 degrees C (61 degrees F).

The applicant evaluated the minimum CRHA temperature using ECOSIMPRO software which was developed and owned by its consultant. The applicant benchmarked the ECOSIMPRO software against the CONTAIN software for the summer design case. The ECOSIMPRO code also assumes a single node for the CRHA. The ECOSIMPRO results showed a minimum bulk temperature in the CRHA of 16 degrees C (61 degrees F) at 72 hours. The staff reviewed the applicant's results and performed confirmatory calculations using a first principles methodology with similar input assumptions. The staff obtained results similar to those of the applicant. Based on the analysis results, the staff concludes that the CB passive heat sinks would likely limit the CRHA occupied zone bulk temperature above this design basis temperature value for 72 hours, assuming no ac power sources are available for that period.

Control Room Habitability Area Air Movement and Air Quality Evaluation

The staff evaluated whether the CBVS provides sufficient control room air quality and air movement. The applicant states that during the loss of ac power condition, the safety-related EFU fan flow, in conjunction with natural convection induced by safety-related passive design features and primarily driven by temperature differences within the CRHA and buoyancy forces, provide adequate air circulation. In addition, the applicant noted that air movement is also promoted by normal personnel movement reasonably expected to occur. Since the CB environmental temperature analysis does not quantify air movement caused by personnel movement and forced convection currents, the staff relied primarily on safety-related EFU fan flow to review the design for adequate air circulation.

The applicant chose to model the CRHA as a single node in the CB environmental temperature analysis. As a single-node model, it cannot simulate the convective mixing mechanism that would also be expected to supplement the forced air movement provided by the EFU fan. The CB environmental temperature analysis also does not include pressure changes in the CRHA from temperature differences between the supply and exhaust air during EFU operation. In RAI 9.4-29 and its supplements, the staff requested the applicant to clarify the basis for the EFU flow rate and to provide information on how the EFU delivers air to the CRHA and promotes mixing to support the design basis analyses. RAI 9.4-29 was being tracked as an open item in the SER with open items.

In responses to RAI 9.4-29, the applicant clarified that the EFU flow rate is consistent with ASHRAE standards for 21 people. To illustrate that air movement is also expected to occur because of convection flows between the CRHA heat sources and heat sinks, the applicant also provided the results of an analysis of a multi-node GOTHIC model. The results demonstrated stratification of temperature in the CRHA and convective mixing. The applicant included CRHA airflow design details obtained from this analysis in DCD Tier 2, Section 6.4, including a description and illustration of the airflow expected in the CRHA occupied zone. Based on review of the design of the EFU air distribution system, the EFU design provision for 7 to 9 air changes per day in the CRHA, and the description of CRHA air distribution in the DCD, the staff finds that mixing would occur and would promote satisfactory air quality and temperature conditions in the CRHA. Because the applicant's chosen singe node modeling methodology assumes the conservative convective heat transfer coefficient of natural convection, and does not credit heat transfer via forced air movement, the CB environmental temperature analysis need not model forced convection, and the added DCD design description of features for mixing and distribution of the EFU supplied inlet air are sufficient to provide assurance that air quality will be within ASHRAE Standard 62.1 guidelines. Accordingly, based on the above, the RAI responses and the proposed DCD changes, RAI 9.4-29 is resolved.

In RAI 9.4-49, the staff requested that the applicant provide additional information on the applicability of ASHRAE Standard 62 to a tightly closed facility, such as the ESBWR MCR, and determine whether there are long-term indoor air quality effects on habitability that need to be addressed. RAI 9.4-49 was being tracked as an open item in the SER with open items. In response, the applicant clarified that the ESBWR MCR is not a tightly closed facility since it has a controlled leak path to balance the air supply provided by the EFU. The applicant also stated that the controlled leakage path is positioned to draw air from the operator breathable zone such that carbon dioxide and odors will be removed. The applicant further stated that pre-operational testing, as described in DCD Tier 2, Section 6.4.7, and surveillances as described in the TS (DCD Tier 2, Chapter 16, Section 5.5.13), will verify that the minimum air flow rate to the CRHA is supplied. The staff finds that the RAI response is acceptable since the TS require that the system be capable of supplying sufficient fresh air to the MCR. In addition, the results of the multi-node analysis, discussed above with RAI 9.4-29, show the effectiveness of the controlled leak path to produce the movement of air through the breathable zone. Because the design includes a forced air supply from a safety-related EFU, and the CRHA exhausts via the CRHA controlled leakage path, the staff finds that there are adequate design features to ensure that ASHRAE Standard 62 air quality standards will be met. Accordingly, based on the above, the applicant's responses, and DCD changes, RAI 9.4-49 is resolved.

Because safety-related EFUs are designed and tested to supply air to the CRHA at the ASHRAE supply rate, which is sufficient for a conservative number of personnel in the MCR, and are supported by the CRHA design features to promote air mixing, as described in the

DCD, the staff finds that the CBVS meets GDC 19 as it applies to control room air quality and air movement.

Evaluation of Control Building Environmental Temperature Analysis for the ESBWR Summary

Based on its review of the analytical basis for maximum and minimum temperatures, the staff concludes that the CB environmental temperature analysis adequately predicts maximum CRHA occupied zone bulk temperature within the applicant's acceptance criteria. The CB environmental temperature analysis adequately demonstrates a mechanism of thermal absorption of heat in the CRHA. As described below in the discussion of the ITAAC, verification of the analysis with as built design and site environmental parameters for both the summer and winter cases provides reasonable assurance that assumptions in the analysis remain valid. The applicant's maximum and minimum temperature acceptance criteria are adequate to ensure that the CRHA would have an acceptable environment for personnel and equipment in a postulated accident. Therefore, the staff finds that the passive cooling design and associated acceptance criteria are acceptable, and the ESBWR CBVS meets GDC 19 as it applies to control room temperature and air quality.

Though not credited by the applicant or the staff to support compliance with GDC 19, the staff notes that ADGs provide a defense-in-depth function for the CBVS. The nonsafety-related ancillary diesels have the RTNSS function to provide ac power for active systems to cool the CRHA after 72 hours. DCD Tier 2, Section 8.3.1.1 states that the ADGs automatically start upon sensing undervoltage on their respective buses. Therefore, the staff notes that the availability of the ADGs in practice serves to minimize uncertainties in the performance of the safety-related passive CRHA design features.

Post-Accident beyond 72 Hours

The staff reviewed the ability of the CBVS design to maintain satisfactory environmental conditions in the MCR, in accordance with GDC 19 during the long term (post-72-hours).

DCD Tier 2, Revision 9, Section 9.4.1.1, describes the auxiliary cooling units, which provide full capacity cooling and ventilation of the CRHA during the post-72-hour period.

In RAI 9.4-31, the staff requested that the applicant clarify the power source for the EFU during the post-72-hour period. In response to the RAI, the applicant modified the design such that the EFUs rely on ADGs, which are RTNSS power supplies. As described below, the staff has reviewed and found acceptable the RTNSS systems associated with the CRHAVS. Accordingly, based on the above and the applicant's response, RAI 9.4-31 is resolved.

The CRHA recirculation AHUs, CBGAVS supply AHUs, and supporting auxiliary cooling units use offsite power or RTNSS power supplies to support MCR habitability after 72 hours. As described below, the staff finds the use of RTNSS power sources and their regulatory treatment acceptable.

In RAI 9.4-50, the staff requested that the applicant to label the AHUs listed in DCD Tier 2, Revision 3, Table 9.4.2, as recirculation AHUs to avoid confusion and to ensure that consistent terminology is used in the text, tables and figures of the DCD. RAI 9.4-50 was being tracked as an open item in the SER with open items. In response, the applicant identified that it had renamed the AHUs renamed "recirculation AHUs" in DCD Tier 2, Revision 4. The staff finds

that the response is acceptable since appropriate AHU were renamed in DCD Tier 2, Revision 4, Sections 9.4 and 9.4.1. Based on the above, the applicant's responses, and DCD changes, RAI 9.4-50 is resolved.

Since the RTNSS qualified AHUs are likely to be available for the post-72 hour period after the onset of a DBA, the staff finds that temperature control for postaccident operation is adequate for such accidents. Each recirculation AHU is equipped with an auxiliary cooling unit with a cooling coil in the AHU. During this period the power for either recirculation AHU can be provided via an ADG. For accidents in which RTNSS power sources are available or in which normal ac power is restored, and thus the RTNSS qualified recirculation AHUs are operational, the staff finds that temperature control for post-accident operation is adequate for the conditions, since the active heating and cooling capacities of the AHUs are far greater than the passive design features described above.

Control Room Habitability in the Event of a Toxic Gas Release

RG 1.78, "Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release," provides guidelines for evaluating the habitability of a nuclear power plant control room during a postulated hazardous chemical release. DCD Tier 2, Revision 9, Section 6.4.9, addresses these guidelines by including COL Information Item 6.4-2-A, which states that the COL applicant will determine the protective measures to be instituted to ensure adequate protection for control room operators as recommended under RG 1.78. These protective measures include features to (1) provide capability to detect releases of toxic or hazardous materials, (2) isolate the control room if there is a release, (3) make the control room sufficiently leak tight, or (4) provide equipment and procedures for ensuring the use of breathing apparatus by the control room operators. The staff finds this acceptable as it relates to the CRHAVS since the COL information item includes provisions to determine protective measures relating to isolating the control room or making the control room sufficiently leak tight. Accordingly, the staff finds that the applicant has adequately addressed the guidelines of RG 1.78 regarding the CRHAVS.

Conformance to 10 CFR 50.63

As discussed above, the CRHA includes passive cooling features to maintain the CRHA environmental conditions within limits necessary for operator actions and within the equipment qualification of control room area equipment for 72 hours without ac electric power. The CRHAVS includes safety-related EFUs, powered by safety-related batteries for 72 hours, to provide filtered fresh air and acceptable environmental conditions, in conjunction with the CRHA passive cooling features. Therefore, the staff finds that the CBVS, in conjunction with the CRHA, adequately addresses the requirements of 10 CFR 50.63 with respect to MCR habitability in that necessary support systems provide sufficient capacity and capability for coping with an SBO, and that the guidance of RG 1.155, including Regulatory Position C.3.2.4, is met.

Based on the above discussions, the staff finds that the CBVS, in conjunction with the CRHA, provides adequate protection to permit access to and occupancy of the control room under accident conditions. In addition, the CBVS, in conjunction with the CRHA, provides acceptable environmental conditions (such as temperature, humidity, and air quality) for personnel and equipment to function. Accordingly, the staff finds that the CBVS meets the requirements of GDC 19 and 10 CFR 50.63.

GDC 60, 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 produced during normal reactor operation, including anticipated operational occurrences.

The CB does not contain any portion of the nuclear steam supply process or other equipment that can act as a source of radioactive material; therefore, the CB has no postulated sources of radioactive materials in either particulate or gaseous form. Therefore, the CBVS, including the CRHAVS, meets the requirements of GDC 60.

Proposed Inspection, Tests, Analyses, and Acceptance Criteria and Proposed Surveillance Requirements

The staff reviewed the proposed ITAAC for the CRHAVS and associated passive design features. The applicant has proposed ITAAC in DCD Tier 1, Revision 9, Table 2.16.2-4, Design Commitments 4i, 4ii and 4iii whereby the as-built CRHAVS heat sink will be analyzed to ensure that the as-built heat sink will passively maintain the temperature in the CRHA within the design basis for the first 72 hours following a DBA. The means to verify these design commitments is a CB environmental temperature analysis using the as built heat sink dimensions, thermal properties, exposed surface areas, as built heat sink thermal properties, and as-built heat loads to confirm the results of the control room design basis CB environmental temperature analysis. The staff finds that satisfactory performance of these ITAAC would ensure that the as-built heat sink will passively maintain the temperature in the CRHA within the design basis for the first 72 hours following a DBA.

In RAI 6.4-24 and it supplements, the staff requested that the applicant justify use of the psychrometric wet bulb temperature as a valid index to assess heat stress in the ESBWR CRHA, or alternatively, to amend the DCD to provide a heat stress acceptance criterion and index that is in accordance with NRC guidance. The staff also requested that the applicant demonstrate that such a criterion can be met for the ESBWR environmental footprint. The staff requested that the applicant clarify associated ITAAC for this criterion.

In response to RAI 6.4-24, the applicant revised DCD Tier 1, Table 2.16.2-4, to include ITAAC 4iii, which requires a licensee to demonstrate that the heat stress acceptance criterion is met via an analysis updated with as built design information.

The staff reviewed the proposed DCD revisions and, because the applicant has included ITAAC to verify the as-built design calculated heat stress condition in the CRHA after 72 hours of passive cooling, RAI 6.4-24 is resolved.

Regarding ITAAC, in RAIs 9.4-33 and 9.4-33 S01, the staff requested that the applicant provide a detailed heat transfer study of the passive heat removal mechanisms, including the analytical assumptions. RAI 9.4-33 was being tracked as an open item in the SER with open items. In response to RAI 9.4-33, the applicant provided the CB environmental temperature analysis assumptions for control room design and outside environmental conditions for a single node model of the CRHA that demonstrates the mechanism by which heat is removed (i.e., the absorption of heat by thermal mass of concrete). The applicant also revised DCD Tier 1, Table 2.16.2-4, to create ITAAC 4i, 4ii, and 4iii to verify such assumptions with as-built information. However, the applicant did not provide a sufficient CB heat up analysis in the DCD.

The staff then consolidated a number of concerns regarding the CB environmental temperature analysis into a new RAI. In RAI 9.4-55, the staff requested the applicant to incorporate the CB environmental temperature analysis in the DCD and revise the ITAAC to specifically refer to this analysis.

In response to RAI 9.4-55, the applicant submitted the analysis, NEDE-33536P, as DCD Tier 2, Reference 3H.4-8, and indicated in DCD Tier 2, Section 3H.3.2.1 and Table 1.6-1, that this report is Tier 2* information and revised DCD Tier 1, Table 2.16.2-4, to clearly link ITAAC 4i, 4ii, and 4iii to NEDE-33536P. The staff finds that the response to RAI 9.4-55 addresses the concerns for RAIs 9.4-33 and 9.4-55 since NEDE-33536P provides a methodology to show that both the baseline CB described in the DCD and the as-built CB, via ITAAC, meet the CRHA maximum temperature criteria. Based on the above, the applicant's responses, and DCD changes, RAIs 9.4-33 and 9.4-55 are resolved regarding ITAAC.

The staff reviewed the ITAAC for the EFU design. DCD Tier 1, Revision 9, Table 2.16.2-6, ITAAC 1 through 12i, provide ITAAC to confirm EFU design assumptions including those for unfiltered air inleakage to the MCR. In particular, DCD Tier 1, Table 2.16.2-6, ITAAC 5.b, provides ITAAC to confirm that CRHA inleakage does not exceed the unfiltered inleakage assumed by control room operator dose analysis. The method of testing (ASTM E741, "Standard Test Method for Determining Air Change in a Single Zone by Means of a Tracer Gas Dilution") included in DCD Tier 1, Table 2.16.2-6, conforms to the integrated test guidance in RG 1.197.

In RAI 15.4-30, based on DCD Revision 3, the staff requested that the applicant include the assumed control room unfiltered air inleakage rate in (1) DCD Tier 1, Section 2.16.2, as an ITAAC item, and (2) the TS in DCD Tier 2, Chapter 16, Section 3.7.2, as a surveillance requirement, in accordance with guidance provided in Technical Specification Task Force (TSTF)-448, "Control Room Habitability," dated July 1, 2003. RAI 15.4-30 was being tracked as an open item in the SER with open items. In response, the applicant pointed out several changes made in DCD Revision 4 to address the staff concerns, including (1) adding DCD Tier 1, Table 2.16.2-6, ITAAC 5.b, addressed above, and (2) modifying the TS in DCD Tier 2, Chapter 16, Section 3.7.2. The staff finds that the response is acceptable, since the applicant made the staff's requested changes, and the staff confirmed that DCD Revision 4 included these changes. Based on the above, the applicant's responses and DCD changes, RAI 15.4-30 is resolved.

Based on the above the staff finds the proposed ITAAC for the CRHAVS, associated passive design features, and the EFUs acceptable.

Regulatory Treatment of Nonsafety Systems

DCD Tier 2, Revision 9, Section 19A.8.4.10, states that component cooling will be performed by the HVAC systems in the RB, EB, FB, CB, and parts of the TB. In RAI 9.4-39, Part D, the staff requested that the applicant identify which parts of the CBVS are classified as RTNSS and which components rely on cooling in the post-72-hour period after an accident. RAI 9.4-39, Part D was being tracked as an open item in the SER with open items. In response, the applicant revised DCD Tier 2, Section 9.4.1.1, to state that the CBVS has RTNSS functions as described in DCD Tier 2, Appendix 19A, with the associated RTNSS design. The applicant added DCD Tier 1, Section 2.16.2.2, Item 10, and Table 2.16.2-4, Item 10, to provide additional ITAAC for RTNSS functions.

DCD Tier 2, Revision 9, Table 19A-2, lists the CRHAVS subsystem of the CBVS as a system that performs functions which fall under SECY-94-084 Criterion B (SSC functions relied upon to resolve long-term safety (beyond 72 hours) and to address associated seismic capabilities). The CRHAVS subsystem of the CBVS provides long-term control room habitability. To support post-accident monitoring beyond 72 hours, it is necessary to provide component cooling for the Q-DCIS cabinets in the CRHA. In DCD Tier 2, Revision 9, Section 19A.3.1.3, the applicant states that the CRHA must have adequate temperature controls during an accident to support operator actions, as well as adequate radiation protection to permit access to and occupancy of the control room under accident conditions for the duration of the accident. In DCD Tier 2, Revision 9, Section 19.A.3.1.4, the applicant states that post-accident monitoring safety functions include control room cooling to remove heat generated by personnel and monitoring equipment. The applicant has chosen to apply regulatory oversight via availability treatment for the system in the ACM. In DCD Tier 2, Revision 9, Section 19.A.8.4.14, the applicant stated that this treatment includes the ancillary ac power that supplies backup power to the control room AHUs. As stated in DCD Tier 2, Revision 9, Section 19A.8.2, all RTNSS systems shall be in the scope of the Design Reliability Assurance Program, which will be incorporated into the Maintenance Rule. DCD Tier 2, Revision 9, Section 19.A.8.3, states that RTNSS Criterion B systems, such as the CRHAVS, have augmented design standards.

The staff reviewed the applicant’s proposed regulatory treatment of the system and system description and finds that, as described in DCD Tier 2, Revision 9, Section 19A, Sections 19A.8.4.4 and 19A.8.4.14, the CRHAVS AHUs, auxiliary heating and cooling units and ADGs and support systems would be subject to regulatory oversight via the ACM. The staff reviewed the proposed regulatory treatment, design standards, and system design basis information in DCD Tier 2 against the criteria for such systems as stated in RG 1.206, Section C.IV.9, and SECY-95-132 and finds that the proposed regulatory treatment of the CBVS for RTNSS conforms to this guidance and is therefore acceptable. The staff reviewed proposed ITAAC for RTNSS functions in DCD Tier 1, Revision 9, Tables 2.16-2-4 and 2.16-2-6, and finds that the proposed ITAAC provide assurance that the identified RTNSS systems will be installed, inspected, and tested in accordance with the design requirements. Accordingly, based on the above and the RAI response, RAI 9.4-39, Part D is resolved.

Minimization of Contamination

In consideration of 10 CFR 20.1406, the staff reviewed the CBVS design to determine how the 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. DCD Tier 2, Revision 9, Table 12.3-18, describes the provisions regarding RG 4.21 design objectives related to CBVS for the following:

- Decreasing the spread of contaminant from the source (Design Objective 4)

- Minimize the generation and volume of radioactive waste both during operation and during decommissioning, by minimizing the volume of components and structures that become contaminated during plant operation. (Design Objective 6)

The CBVS subsystem maintains the MCR at a slightly positive pressure with respect to the outside environment to minimize the infiltration of air. The CBVS detects and limits the introduction of airborne hazardous materials into the control room.

The CBVS meets GDC 60 because the CBVS has no source of radioactive materials in either particulate or gaseous form.

The staff finds that these design provisions for the CBVS meet the requirement of 10 CFR 20.1406 and are consistent with the guidelines of RG 4.21 since the MCR positive pressure will minimize radioactive contamination of the MCR. Section 12.4 of this report further addresses the ESBWR design in accordance with 10 CFR 20.1406.

9.4.1.4 *Conclusion*

Based on the above discussion, the staff finds that the ESBWR CBVS design conforms to the requirements GDC 2, 4, 19, and 60; 10 CFR 50.63; and 10 CFR 20.1406. Because the ESBWR design is a single unit, GDC 5 is not applicable. Conformance with the guidelines of RG 1.78 is addressed by COL Information Item 6.4-2-A.

9.4.2 Fuel Building HVAC System

9.4.2.1 *Regulatory Criteria*

The staff reviewed the ESBWR DCD Tier 2, Revision 9, Section 9.4.2 in accordance with SRP Section 9.4.2, Revision 3. The staff's acceptance of the fuel building HVAC system (FBVS) is based on compliance with the following requirements:

- GDC 2, regarding the capability to withstand earthquakes
- GDC 5, regarding sharing systems and components important to safety
- GDC 60, regarding the capability of the system to suitably control release of gaseous radioactive effluents to the environment
- GDC 61, regarding the capability of the system to provide appropriate containment, confinement, and filtering to limit releases of airborne radioactivity to the environment from the fuel storage facility under normal and postulated accident conditions
- 10 CFR 20.1406, regarding minimizing contamination

9.4.2.2 *Summary of Technical Information*

The FBVS is nonsafety-related except for the isolation dampers and ducting penetrating the FB boundary. The FB boundary is automatically isolated in the event of a fuel handling accident or other radiological accidents. With the above exception, the FBVS performs no safety-related functions.

The FBVS serve the following areas of the FB:

- General areas
- SFP
- Equipment areas

The FBVS is nonsafety-related except for the isolation dampers and ducting penetrating the FB boundary. The FB boundary is automatically isolated in the event of a fuel handling accident or

other radiological accidents. With the above exception, the FBVS performs no safety-related functions.

The FBVS has RTNSS functions as described in DCD Tier 2, Revision 9, Appendix 19A, which provides the level of oversight needed to ensure adequate reliability to meet the RTNSS functions. Performance of RTNSS functions is assured by applying the defense-in-depth principles of redundancy and physical separation to ensure adequate reliability and availability, as described in DCD Tier 2, Revision 9, Section 19A.8.3.

The FBVS maintains space design temperatures, quality of air, and pressurization in the FB. The system consists of two subsystems: the FB General Area HVAC subsystem (FBGAVS) and the FB Fuel Pool Area HVAC subsystem (FBFPVS). The FBGAVS serves the general areas of the FB. The FBFPVS serves the SFP and equipment areas of the FB. Recirculation AHUs provide supplementary cooling for selected rooms in the FB.

The FBGAVS is a once-through air conditioning and ventilation system with an AHU, redundant exhaust fans and FB boundary isolation dampers. The AHU includes filters, heating elements, cooling coils, and redundant AHU supply fans. Outside air is filtered and heated or cooled before being distributed by the AHU. A common supply duct system is incorporated to distribute conditioned air to the general areas of the FB. The exhaust fan discharges the air to the outside atmosphere through the monitored RB/FB vent stack where the exhaust air is monitored for radioactivity. The exhaust air may be manually diverted to the FB HVAC purge exhaust filter unit. Electric unit heaters provide supplementary heating as necessary. A recirculation AHU provides supplementary cooling for the FMCRD room. The CWS provides cooling water for the FBGAVS AHUs. The IAS provides instrument air for the pneumatic actuators.

The FBGAVS AHUs and exhaust fans are located in the FB HVAC Equipment Area. The FMCRD maintenance room recirculation AHU is located in the FB. The FBGAVS provides cooling for FAPCS pump motors, rooms, and/or electrical/instrument panels.

The FBFPVS is a once-through air conditioning and ventilation system with an AHU and redundant exhaust fans. The AHU includes filters, heating elements, cooling coils, and redundant AHU supply fans. Outside air is filtered, heated or cooled, and distributed across the SFP surface and to the equipment areas. Air is exhausted from the SFP area, through redundant FB boundary isolation dampers, to the outside atmosphere through the RB/FB vent stack. During high radiation conditions, the exhaust air may be manually diverted to the FB HVAC purge exhaust filter unit. The exhaust fans are also used for smoke removal. Electric unit heaters provide supplementary heating as necessary. The CWS provides cooling water for the FBFPVS AHUs. Instrument air is provided for the pneumatic actuators. The FBFPVS AHUs and exhaust fans are located in the FB HVAC equipment area.

During high radiation conditions, the FB boundary isolation dampers close automatically and the supply AHU and exhaust fan shut down automatically in both subsystems.

During normal operation, both the FBGAVS and FBFPVS are fully operable. Each subsystem operates with one supply AHU and one exhaust fan in service. The redundant supply fan (in each AHU) and exhaust fan are maintained in standby. In the event of low airflow in an exhaust duct, the standby exhaust fan starts. Simultaneously, because of a loss of negative pressure in the area, the AHU supply fan serving the area stops. The AHU supply fan restarts upon reestablishment of the required negative pressure. In the event of a fan failure, the failed fan automatically shuts down and the standby fan automatically starts.

Upon detection of high radiation, the process radiation monitoring system provides a signal that trips the FBGAVS and FBFPVS. Each subsystem's supply AHU and exhaust fan shuts down and their associated dampers close. Exhaust air from either subsystem may be manually diverted to the FB HVAC purge exhaust filter unit. It is then exhausted to the RB/FB vent stack by the FB HVAC purge exhaust filter unit exhaust fan. Normal ventilation for the area is resumed once the area is decontaminated or the source of radioactivity is removed.

The FMCRD room AHU fan is started and stopped locally. A room thermostat modulates the chilled water valve in response to the room temperature. An individual local thermostat controls each electric unit heater.

9.4.2.3 *Staff Evaluation*

The staff review focused on compliance with GDC for this system which has a safety-related isolation function. The remainder of the system is classified as nonsafety-related. The staff review focused on the safety-related function of the FBVS to isolate the fuel handling building in the event of a radiological accident. The safety-related components are the isolation dampers and the adjoining ducts. The staff has also reviewed the RTNSS functions for the FBVS, as stated in DCD Tier 2, Revision 9, Appendix 19A, against guidance for the selection and identification of such systems stated in RG 1.206, Section C.IV.9.

GDC 2 requires that SSCs important to safety shall be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches without loss of capability to perform their safety functions. DCD Tier 2, Revision 9, Section 9.4.2 states that the safety-related portions of the FBVS are designed to comply with the guidance of RG 1.29 Regulatory Position C.1, which specifies a seismic Category I design. The remainder of the system is classified as nonsafety-related and is designed to seismic Category II in accordance with RG 1.29 Regulatory Position C.2 to ensure that the failure of nonsafety-related portions of the system cannot affect the safety-related components. In addition, DCD Tier 2, Revision 9, Section 9.4.2, states that the FB is a seismic Category I structure except for the penthouse that houses HVAC equipment, which is seismic Category II. All FBVS components are designed as seismic Category II with the exception of the safety-related isolation dampers and associated controls. The FBVS maintains its structural integrity after an SSE. The staff finds that because the FBVS conforms to the guidance of RG 1.29 with respect to seismic categorization, the design conforms to the guidelines of SRP Section 9.4.2.

In RAI 9.4-52, the staff requested the applicant to identify any components in the FB that could be affected by increases in temperature, such as those that could occur during an SBO. RAI 9.4-52 was being tracked as an open item in the SER with open items. In response, the applicant stated that no components in the FB would be affected by increased temperature during an SBO. The staff reviewed the RAI response and the safety-related components that are located in the FB as stated in DCD Tier 2, Revision 9, Table 3.2-1. All electrical components in this table are also listed in DCD Tier 2, Revision 9, Table 3.11-1 as environmentally qualified for harsh environments. Since the environmental conditions during an SBO are not anticipated to exceed the harsh environment conditions, the staff finds that the applicant's response is acceptable. Based on the above, the applicant's responses, and equipment qualification program, RAI 9.4-52 is resolved. Accordingly, the staff finds that the FBVS complies with the requirements of GDC 2.

GDC 5 requires that SSCs important to safety shall not be shared among nuclear power units unless it can be shown that such sharing will not significantly impair their ability to perform their

safety functions, including, in the event of an accident in one unit, an orderly shutdown and cooldown of the remaining units. The ESBWR design is a single-unit station, and the requirements of GDC 5 are not applicable to the single-unit design.

GDC 60 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 produced during normal reactor operation, including anticipated operational occurrences. DCD Tier 2, Revision 9, Section 9.4.2, states that the FBVS design includes redundant safety-related isolation dampers, ducts, and associated instrumentation which contain the release within the fuel handling building. The design includes the capability of directing the system exhaust air to the FB HVAC purge exhaust filter unit during periods of high radioactivity. The FB HVAC purge exhaust filter unit is not a safety-related system and is tested in accordance with Regulatory Guide 1.140. The staff finds that the FBVS design features conform to RG 1.140, "Design, Inspection, and Testing Criteria for Air Filtration and Adsorption Units of Normal Atmosphere Cleanup Systems in Light-Water-Cooled Nuclear Power Plants," and therefore conform to the guidelines of SRP Section 9.4.2. Accordingly, the staff finds that the FBVS complies with the requirements of GDC 60.

GDC 61 requires that the fuel storage and handling, radioactive waste, and other systems which may contain radioactivity shall be designed to assure adequate safety under normal and postulated accident conditions. DCD Tier 2, Revision 9, Section 9.4.2, states that the FBVS provides containment of radioactive releases in the FB as stated in RG 1.13, by safety-related dampers and provides the capability of processing the release through the FB HVAC purge exhaust filter units. As previously noted, the FB HVAC purge exhaust filter unit is not a safety-related system and is tested in accordance with RG 1.140. The staff finds that the FBVS design features conform to RGs 1.13 and 1.140, and therefore conform to the guidelines of SRP Section 9.4.2.

In RAI 9.4-51, the staff requested that the applicant clarify the role of the safety-related FB boundary isolation dampers in containing radioactive release in a postulated fuel handling accident. RAI 9.4-51 was being tracked as an open item in the SER with open items. In response to RAI 9.4-51, and as described in DCD Tier 2, Section 9.4.2.2, upon detection of a high radiation condition, the process radiation monitoring system provides a signal that trips the FBGAVS and FBFPVS. Each subsystem's supply AHU and exhaust fan shuts down and the associated dampers close. In DCD Tier 2, Revision 9, Section 15.4.1.5, the applicant states that no credit is taken for Control Room EFU mitigation, and the RB or fuel handling building integrity is not assumed for such accident. The staff finds that the applicant's response is acceptable since it clarifies that the FBVS isolation dampers close upon a high radiation condition and that credit is not taken for the FB during a fuel handling accident. Based on the above and the applicant's response, RAI 9.4-51 is resolved.

In RAI 9.4-38, the staff requested that the applicant identify any impact on the FB ventilation system as a result of pool boiling. The staff also asked the applicant to identify whether releases during pool boiling mandate routing the FB ventilation system to the RB HVAC purge exhaust filter unit for cleanup. RAI 9.4-38 was being tracked as an open item in the SER with open items. The applicant responded that the FBVS operation would not be impacted by fuel boiling. As stated in DCD Tier 2, Revision 9, Section 9.4.4.3, the FBVS has no function during an accident other than the FB boundary isolation function. After an accident, the FB purge exhaust filter unit (charcoal filter trains) can be employed to clean up the FB. The staff finds that the applicant's response is acceptable since the FBVS is a nonsafety-related system, except for the isolation functions, and the ESBWR design provides a means to clean up the FB following

the design basis boiling of the SFP. Based on the above and the applicant's response, RAI 9.4-38 is resolved. Based on the above, the staff finds that the FBVS complies with the requirements of GDC 61.

DCD Tier 2, Revision 9, Section 19A.8.4.10, states that component cooling will be performed by the HVAC systems in the RB, EB, FB, CB, and parts of the TB. In RAI 9.4-39, Part C, the staff requested that the applicant identify which parts of the FBVS are classified as RTNSS and which components need cooling in the post-72-hour period after an accident. RAI 9.4-39, Part C, was being tracked as an open item in the SER with open items. In response, the applicant revised DCD Tier 2, Section 9.4.2.1, to state that the FBVS has RTNSS functions, as described in DCD Tier 2, Appendix 19A, with the associated RTNSS design requirements. The applicant added DCD Tier 1, Section 2.16.2.5, Item 5, and Table 2.16.2-9, Item 5, to provide additional ITAAC for RTNSS functions associated with post-72-hour cooling for the FAPCS pump motors and N-DCIS components.

DCD Tier 2, Revision 9, Table 19A-2, lists the FBVS as a system that performs functions that fall under SECY-94-084 Criterion C (SSC functions relied upon under power-operating and shutdown conditions to meet the NRC's safety goal guidelines of core damage frequency (CDF) and large release frequency (LRF)). The FBVS is a support system that provides ventilation for the FAPCS and the N-DCIS, which is also a support system for FAPCS. In DCD Tier 2, Revision 9, Section 19A.4.2, the applicant states that the existence of the function provides the CDF and LRF reduction needed to address the PRA uncertainty concerns associated with the performance of passive system components. The applicant has chosen to apply regulatory oversight for the availability of the system through the use of the Maintenance Rule performance monitoring program. As stated in DCD Tier 2, Revision 9, 19A.8.2, all RTNSS systems must be in the scope of the Design Reliability Assurance Program, which will be incorporated into the Maintenance Rule. The staff has reviewed DCD Tier 2, Revision 9, Section 19.A.8.3, and finds that the FBVS is subject to design standards for RTNSS Criterion C systems.

The staff has reviewed the applicant's proposed regulatory treatment of the system and system description and finds that, as described in DCD Tier 2, Revision 9, Section 19A.8.4.9, the room cooler portions of the FBVS would be subject to regulatory oversight via the Maintenance Rule.

The staff has reviewed the proposed regulatory treatment, design standards, and the system design basis information in DCD Tier 2 against the criteria for such systems, as stated in RG 1.206, Section C.IV.9, and SECY-95-132 and has finds that the proposed regulatory treatment of the FBVS for RTNSS conforms to this guidance and is therefore acceptable. The staff has reviewed proposed ITAAC for RTNSS functions in DCD Tier 1, Revision 9, Table 2.16-2-9, and finds that the proposed ITAAC provide assurance that the identified RTNSS systems will be installed, inspected, and tested in accordance with the design. Based on the above, the applicant's responses, and DCD changes, RAI 9.4-39, Part C is resolved.

In RAI 9.4-5, RAI 9.4-5 S01, and RAI 9.4-5 S02, the staff asked the applicant to provide a list of codes and standards used in the design of the ESBWR HVAC systems. RAI 9.4-5 was being tracked as an open in the SER with open items. In response to RAI 9.4-5 S02, the applicant clarified that the applicable codes and standards are discussed in the relevant sections describing the ESBWR HVAC systems. The applicant also provided a table in the RAI response showing where relevant standards are discussed throughout the DCD. The staff finds that the response is acceptable since the staff confirmed the applicable standards were discussed with the relevant systems. Based on the above and the applicant's response, RAI 9.4-5 is resolved.

In consideration of 10 CFR 20.1406, the staff reviewed the FBVS design to determine how the 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.

DCD Tier 2, Revision 9, Table 12.3-18, describes the provisions regarding RG 4.21 design objectives related to FBVS for the following:

- Minimizing leaks and spills (Design Objective 1)
- Decreasing the spread of contaminant from the source (Design Objective 4)
- Facilitating decommissioning by designing the facility to facilitate the removal of equipment or components that may require removal (Design Objective 5)
- Minimizing the generation and volume of radioactive waste both during operation and during decommissioning, by minimizing the volume of components and structures that become contaminated during plant operation. (Design Objective 6)

The FBVS maintains a negative pressure in the building to minimize the exfiltration of potentially contaminated air. The FBVS is provided with access doors for AHUs fans, filter section, and duct-mounted dampers to allow for maintenance as applicable.

Upon detection of high radiation, the process radiation monitoring system provides a signal that trips the FBGAVS and FBFPVS. Each subsystem's supply AHU and exhaust fan shuts down and their associated dampers close. Exhaust air from either subsystem may be manually diverted to the FB HVAC purge exhaust filter unit. It is then exhausted to the RB/FB vent stack by the FB HVAC purge exhaust filter unit exhaust fan. Normal ventilation for the area is resumed once the area is decontaminated or the source of radioactivity is removed.

The FBVS meets GDC 60 by suitably controlling the release of gaseous radioactive effluents to the environment. The FBVS complies with the requirements of GDC 60, as it relates to the system's capability to suitably control release of gaseous radioactive effluents to the environment. The design includes the capability of directing the system exhaust air to the FB HVAC Purge Exhaust Filtration Unit. The FB HVAC purge exhaust filtration units are designed, tested, and maintained in accordance with RG 1.140.

The staff finds that the design provisions for the FBVS meet the requirements of 10 CFR 20.1406 and conforms to the guidelines of RG 4.21. Section 12.4 of this report further addresses the ESBWR design in accordance with 10 CFR 20.1406.

9.4.2.4 *Conclusion*

Based on the above discussions, the staff finds that the ESBWR FBVS design conforms to the requirements GDC 2, 60, and 61, as well as 10 CFR 20.1406. Because the ESBWR design is a single unit, GDC 5 is not applicable.

9.4.3 Radwaste Building Heating, Ventilation and Air Conditioning System

9.4.3.1 *Regulatory Criteria*

The staff reviewed the ESBWR DCD Tier 2, Revision 9, Section 9.4.3 in accordance with SRP Section 9.4.3, Revision 3. The staff's acceptance of the RW Heating, Ventilation and Air Conditioning System (RWVS) is based on compliance with the following requirements:

- GDC 2, regarding the capability to withstand earthquakes

- GDC 5, regarding sharing systems and components important to safety

- GDC 60, regarding the capability of the system to suitably control release of gaseous radioactive effluents to the environment

- 10 CFR 20.1406, regarding the minimization of contamination

9.4.3.2 *Summary of Technical Information*

The RWVS provides a controlled environment for personnel comfort and for proper operation and integrity of equipment.

The RWVS does not have a safety-related function. Operational failure of any single unit of the RWVS does not prevent safety-related equipment from performing its safety-related function. The entire system is classified as nonsafety-related. The nonsafety-related RWVS consists of two subsystems: the RW Control Room HVAC subsystem (RWCRVS) and the RW General Area HVAC subsystem (RWGAVS).

The RWCRVS maintains the RW control room (RWCR) area temperature and maintains the control room areas at a slightly positive pressure relative to adjacent areas to minimize infiltration of air. Redundant components are provided to increase system reliability, availability, and maintainability.

The RWCRVS is a recirculating air conditioning system to provide filtered, heated or cooled, and humidified air to the RWCR area to maintain the required design ambient conditions and pressurization. The RWCR consists of two 100-percent capacity AHUs and a common outside air intake louver. Each AHU contains filters, a humidifier, a chilled water cooling coil, a heating coil, and a supply fan. Conditioned air is supplied to the control room, the electrical equipment room, the elevator machine room, and the HVAC equipment room areas through ducts, dampers, and registers.

The RWCRVS is capable of once-through operation for smoke removal using two 50 percent capacity exhaust fans.

The RWGAVS is a once-through air conditioning and ventilation system that provides filtered and heated or cooled air to the RW general area (RWGA). The RWGAVS supply consists of one AHU with two 100-percent capacity supply fans, in parallel, connected to a supply distribution ductwork system and an outside air intake louver. Each AHU contains filters, cooling and heating coils, two redundant supply fans, and isolation dampers. The RWGAVS exhaust consists of three 50-percent capacity AFUs, each with prefilters and HEPA filters, a 50-percent capacity exhaust fan, and a check valve/backdraft damper. Exhaust capacity is greater

than the supply capacity in order to maintain the minimum RWGA negative pressure. Each AFU is connected to a common exhaust collection duct and a common exhaust duct discharging to the RW vent stack. The RWGAVS exhaust subsystem is capable of once-through operation for smoke removal. The AFUs are bypassed in this mode.

9.4.3.3 *Staff Evaluation*

The staff review focused on compliance with GDC for this important but nonsafety-related system.

GDC 2 requires that SSCs important to safety shall be designed to withstand the effects of natural phenomena, such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches, without loss of capability to perform their safety functions. DCD Tier 2, Revision 9, Section 9.4.3, states that the RWVS is classified as nonsafety and is designed to seismic Category II in accordance with RG 1.29 Regulatory Position C.2, to ensure that the failure of nonsafety-related portions of the system cannot affect safety-related components. DCD Tier 2, Revision 9, Section 3.7.2.8.2 states that the RW is designed in accordance with RG 1.143 Classification RW-IIa, which includes guidelines for the design of the RWVS. The staff finds that because the RWVS conforms to the guidance of RG 1.29 with respect to seismic categorization and RG 1.143, the design conforms to the guidelines of SRP Section 9.4.3. Accordingly, the staff finds that the RWVS complies with the requirements of GDC 2.

GDC 5 requires that SSCs important to safety shall not be shared among nuclear power units unless it can be shown that such sharing will not significantly impair their ability to perform their safety functions, including, in the event of an accident in one unit, an orderly shutdown and cooldown of the remaining units. The ESBWR design is a single-unit station, and the requirements of GDC 5 are not applicable to the single-unit design.

GDC 60 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 produced during normal reactor operation, including anticipated operational occurrences. DCD Tier 2, Revision 9, Section 9.4.3, states that the RWVS design includes redundant isolation dampers, ducts, and associated instrumentation to contain the contamination within the RW. The design includes the capability of directing the system exhaust air to the RWGA exhaust filtration units. RWGA exhaust filtration units are designed, tested, and maintained in accordance with RG 1.140. The staff finds that the RWVS design features conform to RG 1.140 and therefore conforms to the guidelines of SRP Section 9.4.2. Accordingly, the staff finds that the RWVS complies with the requirements of GDC 60.

In RAI 9.4-5, 9.4-5 S01, and 9.4-5 S02, the staff asked the applicant to provide a list of codes and standards used in the design of the ESBWR HVAC systems. RAI 9.4-5 was being tracked as an open item in the SER with open items. In response to RAI 9.4-5 S02, the applicant clarified that the applicable codes and the standards are discussed in the relevant sections describing the EBWR HVAC systems. The applicant also provided a table in the RAI response showing where relevant standards are discussed throughout the DCD. The staff finds that the response is acceptable since the staff confirmed that the applicable standards were discussed with the relevant systems. Based on the above and the applicant's response, RAI 9.4-5 is resolved.

In consideration of 10 CFR 20.1406, the staff reviewed the RWVS design to determine how the 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.

DCD Tier 2, Revision 9, Table 12.3-18, describes the provisions regarding RG 4.21 design objectives related to RWVS for the following:

- Minimizing leaks and spills (Design Objective 1)
- Having leak detection methods and early detection of leaks to avoid release of contamination from undetected leaks and to minimize contamination of the environment (Design Objective 3)
- Decreasing the spread of contaminant from the source (Design Objective 4)
- Minimizing the generation and volume of radioactive waste both during operation and during decommissioning, by minimizing the volume of components and structures that become contaminated during plant operation. (Design Objective 6)

The RWVS meets GDC 60 by suitably controlling the release of gaseous radioactive effluents to the environment. The RWVS complies with the requirements of GDC 60 as to the system's capability to suitably control release of gaseous radioactive effluents to the environment. The design includes the capability of directing the system exhaust air to the RWVS filtration units. The RWVS filtration units are designed, tested, and maintained in accordance with RG 1.140.

The RWCRVS maintains the RWCR areas at a slightly positive pressure relative to adjacent areas to minimize infiltration of air. The RWGAVS maintains the RWGA at a slight negative pressure relative to adjacent areas and outside atmosphere to prevent the exfiltration of air to adjacent areas. Adequate exhaust from the trailer bays is provided to maintain inflow of air from the outside when the truck doors are open. The RWGAVS is comprised of supply and exhaust subsystems to maintain direction of air flow from personnel occupancy areas towards areas of increasing potential contamination. Exhaust hoods are provided at locations where under normal operation, contaminants could escape to the surrounding areas. The RWGAVS provides the capability to exhaust air from the radwaste processing systems.

All exhaust air from the RWGA is discharged to the RW vent stack. Redundant components are provided as necessary to increase system reliability, availability and maintainability. The RWGAVS exhaust air is monitored for radiation prior to discharge to atmosphere.

The staff finds that these design provisions for the RWVS meets the requirement of 10 CFR 20.1406 and conforms to the guidelines of RG 4.21. Section 12.4 of this report further addresses the ESBWR design in accordance with 10 CFR 20.1406.

9.4.3.4 *Conclusion*

Based on the above discussion, the staff finds that the ESBWR RWVS design conforms to the requirements GDC 2 and 60, as well as 10 CFR 20.1406. Because the ESBWR design is a single unit, GDC 5 is not applicable.

9.4.4 Turbine Building Heating, Ventilation and Air Conditioning System

9.4.4.1 *Regulatory Criteria*

The staff reviewed the ESBWR DCD Tier 2, Revision 9, Section 9.4.4 in accordance with SRP Section 9.4.4, Revision 3. The staff's acceptance of the TB HVAC system (TBVS) is based on the applicant's compliance with the following requirements:

- GDC 2, as it relates to the capability to withstand earthquakes

- GDC 5, as it relates to sharing systems and components important to safety

- GDC 60, as it relates to the capability of the system to suitably control release of gaseous radioactive effluents to the environment

- 10 CFR 20.1406, as it relates to minimization of contamination

9.4.4.2 *Summary of Technical Information*

The TBVS includes the TB supply air fans and associated filter trains, and the TB exhaust fans and associated filter trains and the various fan-coil units for local area heating and cooling within the TB. The TBVS does not have a safety-related function. Operational failure of any single unit of the TBVS does not prevent safety-related equipment from performing its safety-related function. The entire system is classified as nonsafety-related.

The TBVS has RTNSS functions as described in DCD Tier 2, Revision 9, Appendix 19A, which provides the level of oversight and additional requirements to meet the RTNSS functions. Performance of RTNSS functions is assured by applying the defense-in-depth principles of redundancy and physical separation to ensure adequate reliability and availability as described in DCD Tier 2, Revision 9, Section 19A.8.3.

The TBVS is designed to minimize exfiltration by maintaining a slightly negative pressure in the TB by exhausting more air than is supplied to the TB. The TBVS is designed to provide for local air recirculation and cooling in high heat load areas using local unit coolers. A minimum of 50-percent standby cooling capacity is provided in areas where a loss of cooling could cause degraded equipment performance. TB ventilation systems and subsystems required for normal plant operation are provided with redundant fans with automatic start logic.

Exhaust air from potentially high airborne contamination TB areas or component vents is collected, filtered, and discharged to the atmosphere through the TB Compartment Exhaust (TBCE) system. Exhaust air from other (low potential airborne contamination) TB areas and component vents is exhausted to the atmosphere through the TB exhaust (TBE) system. TBE air is directed to the TB vent stack where it is monitored for radiation prior to being discharged to the atmosphere.

The TBVS equipment is located in the TB. The chiller rooms, located in the TB, meet ASHRAE-15, "Safety Standard for Refrigeration Systems." They are equipped with a dedicated purge system and leak detectors with alarms.

The nonsafety-related TBVS consists of the following subsystems and components:

- TB air supply (TBAS) subsystem
- TBE subsystem
- TBCE subsystem
- TB lube oil area exhaust (TBLOE) subsystem
- TB decontamination room exhaust (TBDRE) subsystem
- TBVS unit coolers and unit heaters

The TBAS consists of outside air intake louvers, dampers, filters, heating coils, chilled water cooling coils, and three 50-percent capacity supply fans. Two of the three fans are normally operating to supply filtered, temperature-controlled air to all levels of the TB. The third fan is a standby unit that starts automatically upon failure of either operating fan. Each supply fan is provided with pneumatically operated isolation damper. The TBAS uses 100 percent outside air during normal plant operation.

The TBE fans exhaust air from the building clean and low potential contamination areas. The air is exhausted through the monitored vent stack. The TBE subsystem is provided with three 50-percent capacity fans. Two fans are normally in operation and one is in automatic standby. All three TBE fans can be operated simultaneously to provide maximum smoke removal, if necessary.

Each TBE fan is provided with variable speed drives and isolation dampers. A flow controller automatically adjusts the frequency of the operating fans to vary the system airflow rate. Failure of one operating exhaust fan automatically starts the standby fan. The TBVS exhaust fans are interlocked with the TBAS fans.

The TBCE subsystem consists of two 100-percent capacity exhaust fans, one filter unit and associated controls. One fan is normally in operation with the other one in automatic standby. The subsystem includes a 100-percent capacity filter bypass duct for purging smoke in the event of a fire. The air exhausted from the TB high potential airborne contamination compartments and equipment vents is passed through a filter before it is released to the atmosphere through the TB vent stack, except during smoke removal.

The TBCE subsystem has radiation detectors in the exhaust duct to monitor the air for radioactivity prior to its being discharged to the TB vent stack. The two exhaust fans are provided with variable frequency drives and isolation dampers. An airflow controller automatically adjusts the speed of the operating fan to vary the system exhaust flow rate. In the automatic mode, loss of flow from the operating fan starts the standby fan.

The TBLOE subsystem includes two 100-percent capacity fans, isolation dampers, low efficiency filters, and exhaust ductwork. The TBLOE fans discharge the exhaust air directly to TBE Subsystem. One of the two fans is operated to continuously exhaust at a constant volumetric flow rate from the turbine lube oil tank room. A bypass duct is provided around the lube oil exhaust fans for purging high temperature combustion products and limiting room pressurization in the event of a fire in one of the rooms.

The TBDRE subsystem consists of one air filtration unit (AFU), which includes one 100-percent capacity exhaust fan, filters (high efficiency and HEPA), an isolation damper and associated controls. The air exhausted from the TBDRE, once filtered, is exhausted by the TBE subsystem and is finally released to the atmosphere through the TB vent stack.

Localized AHUs and unit heaters are provided as required in various locations within the TB. The AHUs are supplied with chilled water from the BOPCWS and the unit heaters are electric resistance type heaters. The system provides redundant AHUs to allow operation of associated equipment with an AHU out of service, or to maintain cooling upon the failure of one AHU. The main steam tunnel is provided with two 100-percent redundant recirculation AHUs. Temperature controls for the AHUs and unit heaters are located in the unit inlet air path or are installed locally. The cooling coils of the RCCWS, NICWS, selected electrical equipment rooms and IAS and SAS rooms are fed from the corresponding NICWS train.

The TBVS is designed to operate during all modes of normal power plant operation, including start-up and shutdown. The TBVS fans are started manually and operate automatically thereafter. Standby fans start automatically if one of the running fans trip due to low flow or equipment trip.

Upon detection of smoke in the TB, the TBAS outside air supply fans and the TBE subsystem exhaust fans stop automatically. During smoke purge operation in the TBCE subsystem, MCR operators bypass the subsystem filters manually. MCR operators normally initiate the smoke purge mode of operation of the TB. Smoke purge is accomplished by starting two supply fans in the TBAS and two exhaust fans in the TBE subsystem as well as the TBCE and TBLOE exhaust fans. This provides 100 percent outside air. All three fans in the TBAS and in the TBE subsystem can be started to provide maximum smoke removal.

Upon a LOPP, at least one of the fans of the TBE subsystem remains available for operation because it is powered from the nonsafety-related SDGs. The local AHUs of the RCCWS, NICWS, and IAS and SAS rooms and selected electrical equipment rooms also remain in operation.

9.4.4.3 *Staff Evaluation*

The staff review focused on compliance with GDC for this important but nonsafety-related system. The staff has also reviewed the RTNSS functions for the TBVS as stated in DCD Tier 2, Revision 9, Appendix 19A, against guidance for selection and identification of such systems stated in RG 1.206, Section C.IV.9.

GDC 2 requires that SSCs important to safety shall be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches without loss of capability to perform their safety functions. DCD Tier 2, Revision 9, Section 9.4.4, states that the TBVS is classified as nonsafety-related and is designed to seismic Category NS in accordance with RG 1.29, Regulatory Position C.2, to ensure that the failure of nonsafety-related portions of the system can not affect safety-related components. The TB is a seismic Category II nonsafety-related structure. The staff finds that because the TBVS conforms to the guidance of RG 1.29 with respect to seismic categorization, the design conforms to the guidelines of SRP Section 9.4.2. Accordingly, the staff finds that the TBVS complies with the requirements of GDC 2.

GDC 5 requires that SSCs important to safety shall not be shared among nuclear power units unless it can be shown that such sharing will not significantly impair their ability to perform their safety functions, including, in the event of an accident in one unit, an orderly shutdown and cooldown of the remaining units. The ESBWR design is a single-unit station, and the requirements of GDC 5 are not applicable to the single-unit design.

GDC 60 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 produced during normal reactor operation, including anticipated operational occurrences. The TBVS complies with the requirements of GDC 60, in the systems capability to suitably control release of gaseous radioactive effluents to the environment. The design includes the capability of directing the system exhaust air to the TBVS filtration units. The TBVS filtration units are designed, tested, and maintained in accordance with RG 1.140.

The TBVS has adequate provision for maintaining a suitable environment for personnel access and equipment by providing recirculation and exhaust capabilities with adequate heating and cooling that are locally controlled as needed. Provisions are in place to control contamination and gaseous discharges through filter systems and exhaust paths that are monitored prior to release to the environment. The system also has the provision to exhaust smoke in the event of a fire consistent with the smoke management features of DCD Tier 2, Revision 9, Section 9.5.1.

The TBVS has adequate instrumentation that alarm in the MCR for adverse radiological conditions and temperature conditions. The TBVS also has adequate differential pressure indicators for filters, air flow indicators, and controls. Provision exists for testing of key parameters and inspection of components to ensure operating conditions and integrity of the system. The staff finds that the TBVS design features conform to RG 1.140 and therefore conforms to the guidelines of SRP Section 9.4.4. Accordingly, the staff finds that the TBVS complies with the requirements of GDC 60.

DCD Tier 2, Revision 9, Section 19A.8.4.10, states that component cooling will be performed by the HVAC systems in the RB, EB, FB, CB, and parts of the TB. In RAI 9.4-39, Part E, the staff requested that the applicant identify which parts of the TBVS are classified as RTNSS systems and which components require cooling in the post-72-hour period after an accident. RAI 9.4-39, Part E was being tracked as an open item in the SER with open items. In response, the applicant revised DCD Tier 2, Section 9.4.4.1, to state that the TBVS has RTNSS functions as described in DCD Tier 2, Appendix 19A. The applicant also revised DCD Tier 1, Section 2.16.2.4, Item 2, and Table 2.16.2-7, Item 2, to add additional ITAAC for RTNSS functions associated with post-72-hour cooling for the DCIS in the TB and room cooling for the NICWS and RCCWS pumps.

DCD Tier 2, Revision 9, Table 19A-2, lists the TBVS as a system that performs functions which falls under SECY-94-084 Criterion C (SSC functions relied upon under power-operating and shutdown conditions to meet the NRC's safety goal guidelines of CDF and LRF). The TBVS is a support system for the FAPCS. It provides equipment and room cooling to support RCCWS, NICWS, and associated N-DCIS support cooling. In DCD Tier 2, Revision 9, Section 19A.4.2, the applicant states that the existence of the function provides the CDF and LRF reduction needed to address the PRA uncertainty concerns associated with the performance of passive system components. The applicant has chosen to apply regulatory oversight by means of availability treatment for the system through the Maintenance Rule performance monitoring program. As stated in DCD Tier 2, Revision 9, Section 19A.8.2, all RTNSS systems shall be in the scope of the Design Reliability Assurance Program, which will be incorporated into the Maintenance Rule. The staff reviewed DCD Tier 2, Revision 9, Section 19.A.8.3 and finds that the TBVS is subject to design standards for RTNSS criterion C systems.

The staff has reviewed the applicant's proposed regulatory treatment of the system and system description and finds that, as described in DCD Tier 2, Revision 9, Section 19A.8.4.9, the room cooler portions of the TBVS would be subject to regulatory oversight via the Maintenance Rule.

The staff reviewed the proposed regulatory treatment design standards and the system design basis information in DCD Tier 2 against the criteria for such systems as stated in RG 1.206, Section C.IV.9, and SECY-95-132 and finds that the proposed regulatory treatment of the TBVS for RTNSS is acceptable, as described above. The staff reviewed proposed ITAAC for RTNSS functions in DCD Tier 1, Revision 9, Table 2.16-2-7, and finds the proposed ITAAC provides assurance that the identified RTNSS systems will be installed, inspected, and tested, in accordance with the design. Based on the above, the applicant's responses and DCD changes, RAI 9.4-39, Part E is resolved.

In RAI 9.4-40, the staff requested that the applicant clarify DCD Tier 2, Figure 9.4-8, to show all five filter units on the figure or show one filter unit with a note saying that it is typical of all five units. Further, the staff asked the applicant to verify the consistency of the nomenclature used in the figure, table, and text. RAI 9.4-40 was being tracked as an open item in the SER with open items. In response, the applicant stated the TBE system design has been changed, and filter units have been removed. In Revision 4 of the DCD, the applicant also modified DCD Tier 2, Section 9.44, Table 9.4-15, and Figure 9.2-8 to be consistent. The staff finds that the response is acceptable since it addressed the inconsistencies in the description of the TBVS. Accordingly, based on the above, the applicant's responses, and DCD changes, RAI 9.4-40 is resolved.

In RAI 9.4-5, 9.4-5 S01, and 9.4-5 S02, the staff asked the applicant to provide a list of codes and standards used in the design of the ESBWR HVAC systems. RAI 9.4-5 was being tracked as an open in the SER with open items. In response to RAI 9.4-5 S02, the applicant clarified that the applicable codes and the standards are discussed in the relevant sections describing the EBWR HVAC systems. The applicant also provided a table in the RAI response showing where relevant standards are discussed throughout the DCD. The staff finds that the response is acceptable since the staff confirmed the applicable standards were discussed with the relevant systems. Based on the above and the applicant's response, RAI 9.4-5 is resolved.

In consideration of 10 CFR 20.1406, the staff reviewed the TBVS design in order to determine how the 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.

DCD Tier 2, Revision 9, Table 12.3-18, describes the provisions regarding RG 4.21 design objectives related to TBVS for the following:

- Minimizing leaks and spills (Design Objective 1)
- Having leak detection methods and early detection of leaks to avoid release of contamination from undetected leaks and to minimize contamination of the environment (Design Objective 3)
- Decreasing the spread of contaminant from the source (Design Objective 4)
- Minimizing the generation and volume of radioactive waste both during operation and during decommissioning, by minimizing the volume of components and structures that become contaminated during plant operation. (Design Objective 6)

The TBVS meets GDC 60 by suitably controlling the release of gaseous radioactive effluents to the environment. The TBVS complies with the requirements of GDC 60 for the system's

capability to suitably control release of gaseous radioactive effluents to the environment. The design includes the capability of directing the system exhaust air to the TBVS filtration units. The TBVS filtration units are designed, tested, and maintained in accordance with RG 1.140.

The TBCE subsystem has radiation detectors in the exhaust duct to monitor the air for radioactivity prior to its being discharged to the TB vent stack.

TBVS cooling coil condensate is collected in drain pans within the AHUs with the drain pan discharge (condensate) routed to a floor drain located within the room. These floor drains connect to the applicable equipment and floor drain subsystem.

The staff finds that these design provisions are adequate to minimize contamination of the environment and minimize the generation of radioactive waste. The provisions for the TBVS meet the requirement of 10 CFR 20.1406 and conform to the guidelines of RG 4.21. Section 12.4 of this report further addresses the ESBWR design in accordance with 10 CFR 20.1406.

9.4.4.4 *Conclusion*

Based on the above discussions, the staff finds that the ESBWR TBVS design conforms to the requirements in GDC 2 and 60, and 10 CFR 20.1406. Because the ESBWR design is a single unit, GDC 5 is not applicable.

9.4.5 Engineered Safety Feature Ventilation System

The EFU portion of the CRHAVS supplies the engineered safety feature for the CRHA radiological protection, as described in DCD Tier 2, Revision 9, Sections 6.4 and 9.4.1. Sections 6.4 and 9.4.1 of this report provide the staff's evaluation of the EFUs.

9.4.6 Reactor Building Heating, Ventilation, and Air Conditioning System

9.4.6.1 *Regulatory Criteria*

The staff reviewed the ESBWR DCD Tier 2, Revision 9, Section 9.4.6 in accordance with SRP Section 9.4.3, Revision 3. For those areas that contain safety-related equipment, the staff reviewed DCD Tier 2, Revision 9, Section 9.4.6 in accordance with SRP Section 9.4.5, Revision 3. The staff's acceptance of the RB HVAC system (RBVS) is based on the applicant's compliance with the following requirements:

- GDC 2, as it relates to the capability to withstand earthquakes
- GDC 5, as it relates to sharing systems and components important to safety
- GDC 60, as it relates to the capability of the system to suitably control release of gaseous radioactive effluents to the environment
- 10 CFR 50.63, as it relates to necessary support systems providing sufficient capacity and capability to ensure the capability for coping with an SBO event
- 10 CFR 20.1406, as it relates to minimization of contamination

9.4.6.2 *Summary of Technical Information*

The RBVS maintains the design temperature, quality of air, and pressurization in the RB spaces. The isolation dampers and ducting penetrating the RB boundary and associated controls that provide the isolation signal are safety-related. The RBVS performs the safety-related function of automatic isolation of the RB boundary during accidents.

The RBVS serves the following areas of the RB:

- The potentially contaminated areas (contaminated area HVAC subsystem [CONAVS])
- The refueling area (refueling and pool area HVAC subsystem [REPAVS])
- The nonradiologically controlled areas (clean area HVAC subsystem [CLAVS])
- Containment during inerting and deinerting operations

The RBVS has the safety-related function of building isolation. The isolation dampers and ducting penetrating the RB boundary and the associated controls that provide the isolation signal are safety-related. The RBVS performs the safety-related function of automatic isolation of the RB boundary (CONAVS and REPAVS subsystems) during accidents. The RBVS has nonsafety-related RB purge exhaust filter units for mitigating and controlling gaseous effluents from the RB. The RBVS has nonsafety-related RB HVAC accident exhaust filter units for use postaccident (greater than 8 hours) to create a negative pressure in the RB contaminated areas and exhaust the filtered air to the RB/FB stack.

The RBVS has RTNSS functions as described in DCD Tier 2, Revision 9, Appendix 19A, which provides the level of oversight and additional requirements to meet the RTNSS functions. Performance of RTNSS functions is assured by applying the defense-in-depth principles of redundancy and physical separation to ensure adequate reliability and availability. In addition, augmented design standards are applied as described in DCD Tier 2, Revision 9, Section 19A.8.3.

The RBVS provides a controlled environment for personnel comfort and safety and for proper operation and integrity of equipment and maintains potentially contaminated areas at a negative pressure to minimize exfiltration of potentially contaminated air. The RBVS maintains clean areas of the building, except for the battery rooms, at a positive pressure to minimize infiltration of outside air and maintains airflow from areas of lower potential for contamination to areas of greater potential for contamination. Redundant active components are provided to increase the reliability, availability, and maintainability of the systems. The RBVS is capable of exhausting smoke, heat, and gaseous combustion products in the event of a fire and prevents smoke and hot gases from migrating into other fire areas by automatically closing smoke dampers upon detection of smoke.

During radiological events, the RBVS shuts down and isolates the RB boundary (CONAVS and REPAVS) to prevent uncontrolled releases to the outside atmosphere.

The RBVS provides the capability to manually divert exhaust air for processing through the RB HVAC on-line purge exhaust filter units.

RB HVAC on-line purge exhaust filter units can be energized to re-circulate the CONAVS area air space. After a LOCA, one RB HVAC accident exhaust filter unit (the redundant one is in standby) can be energized to create a negative pressure by exhausting the air in the CONAVS area.

The RBVS provides pool sweep ventilation air over the refueling area pool surface.

The RBVS maintains the hydrogen concentration levels in the battery rooms below 2 percent by volume in accordance with RG 1.128, “Installation Design and Installation of Vented Lead-Acid Storage Batteries for Nuclear Power Plants,” and maintains battery room temperatures within a range to maximize output and equipment life.

The RBVS replaces the containment inerted atmosphere with conditioned air during a refueling operation.

The RBVS provides local recirculation AHUs for cooling of the hydraulic control unit (HCU) area. The RBVS maintains SLC accumulator room environmental conditions within temperature-limits, including employing two backup heaters per room. PIP A and PIP B buses provide power for these heaters.

The RBVS provides cooling for CRD and RWCU pump motors, rooms, and electrical and instrument panels and is designed to limit the room and equipment to within their temperature environmental qualification when the building is isolated. The motor cooler heat sink is the RCCWS, while chilled water or direct expansion units are provided for electrical cabinet cooling.

The RBVS consists of three subsystems. The RB CONAVS serves the potentially contaminated areas of the RB. The REPAVS serves the refueling area of the RB. The RB CLAVS serves the clean (non-radiological controlled) areas of the RB. The CONAVS is a two train, once-through ventilation system with each train consisting of an AHU, redundant exhaust fans, and building isolation dampers. It includes a containment purge exhaust fan, recirculation AHUs, and unit heaters. The AHU includes filters, heating and cooling coils, and redundant supply fans. Outside air is filtered and heated or cooled before distribution by the AHU in service. The CWS provides cooling for the CONAVS AHUs. The IAS provides instrument air for the pneumatic actuators. A common supply air duct distributes conditioned air to the potentially contaminated areas of the RB.

Air is exhausted from the potentially contaminated areas of the RB by the operating exhaust fan and discharged to the RB/FB vent stack. During containment de-inerting operations, the supply airflow rate of the AHU supply fan is increased. At the same time, the airflow rate of the exhaust fan is increased an equal amount. In the event of a fire, fire dampers close to isolate the fire area. In the event smoke is detected in the air duct, the system is shut down. After the fire is completely extinguished, the exhaust fans are used for smoke removal, with the exhaust air being monitored for radiological contamination. If the air is contaminated, temporary portable filters may be used to exhaust the contaminated air. The building isolation dampers close, and the supply and exhaust fans stop when there is high radiation in the exhaust ducts.

The CONAVS also includes redundant RB HVAC exhaust filter units (accident and online purge filter assemblies) and exhaust fans. During radiological events, exhaust air from contaminated areas may be manually diverted through the RB HVAC online purge exhaust filter units. The RB exhaust filter units are equipped with pre-filters, HEPA filters, high efficiency filters, and carbon filters for mitigating and controlling particulate and gaseous effluents from the RB. The RB HVAC online purge exhaust filter units can be used to re-circulate the CONAVS area and thereby clean up the contaminated environments in the RB. After a LOCA, one RB HVAC accident exhaust filter unit (the redundant one is in standby) can be energized to create a negative pressure by exhausting the air in the CONAVS area. The supply AHU and normal exhaust fans may be shut down during filtered purge exhaust. Recirculation AHUs provide

supplementary cooling for selected rooms. Cooling is provided for CRD and RWCU pump motor coolers from RCCWS and electrical and instrument panels are provided with either chilled water or direct expansion units designed to limit the room and equipment to within their temperature environmental qualification when the building is isolated. Electric unit heaters provide supplementary heating. The CONAVS AHUs are located in the FB HVAC Equipment Area. The CONAVS exhaust fans are located in the RB. The RB HVAC Purge Exhaust Filter Units and exhaust fans are located in the RB. The refueling machine control room recirculating AHU is located in the RB. Electric unit heaters are located in or near the areas they serve.

The REPAVS is a once-through ventilation system consisting of an AHU, redundant exhaust fans, and building isolation dampers. The AHU includes filters, heating and cooling coils, and redundant supply fans. Outside air is filtered and heated or cooled before distribution by the AHU in service.

The conditioned air is distributed to the refueling area and across the pool surface. Exhaust air is ducted to the exhaust fans and exhausted to the outside atmosphere through the RB/FB vent stack. During a radiological event, exhaust air from the refueling area may be manually diverted through the RB HVAC online purge exhaust filter units. The CWS provides cooling water for the REPAVS AHU. The IAS provides instrument air for the pneumatic actuators. In the event of a fire, fire dampers close to isolate the fire area. In the event smoke is detected in the air duct, the system is shut down. After the fire is completely extinguished, the exhaust fans are then used for smoke removal, with the exhaust air being monitored for radiological contamination. If the air is contaminated, temporary portable filters are used to exhaust the contaminated air. The building isolation dampers close, and the supply and exhaust fans stop when there is high radiation in the exhaust ducts.

The REPAVS AHUs are located in the FB HVAC equipment area. The REPAVS exhaust fans are located in the RB. Electric unit heaters are located in or near the areas they serve.

The CLAVS is a two train recirculating ventilation system, with each train consisting of an AHU and redundant return/exhaust fans and smoke exhaust fans.

The AHU includes filters, heating and cooling coils, and redundant supply fans. A mixture of outside and return air is filtered and heated or cooled before distribution by the AHU in service. A common supply and return/exhaust air duct system distributes conditioned air to and from the RB clean areas. Return air not directed back to the AHU is exhausted directly outdoors. An economizer cycle is used, when outside air conditions are suitable, to reduce mechanical cooling operating hours. The economizer cycle provides all outside air, or a mixture of outside air and return air, to RB clean areas. The temperature of the air provided is at or below the supply air design temperature. In the event of a fire, fire dampers close to isolate the fire area. In the event smoke is detected in the air duct, the system is shut down. After the fire is completely extinguished, the CLAVS exhaust fans are then used for smoke removal. The CWS provides cooling for the CLAVS AHU. The IAS provides instrument air for the pneumatic actuators. Electric unit heaters provide supplementary heating. The CLAVS AHU supplies air to the battery rooms. A minimum exhaust air is continuously extracted from battery rooms to keep hydrogen concentration below 2 percent. This extracted air is exhausted from the battery rooms by the battery room exhaust fans, which discharge directly to the RB/FB vent stack. Battery room temperature is maintained within a range to maximize output and equipment life. Battery room hydrogen indication and loss of ventilation alarm functions are provided.

The CLAVS AHUs and return/exhaust fans are located in the FB HVAC equipment area. The electric unit heaters are located in or near the areas they serve.

9.4.6.3 *Staff Evaluation*

The staff review focused on compliance with GDC for this system which has a safety-related isolation function. The remainder of the system is classified as nonsafety. To review the adequacy of the RBVS passive cooling features for those rooms containing safety- related equipment, the staff focused on compliance with 10 CFR 50.63, which requires a demonstration that the plant has the capability to withstand and recover from a SBO. The staff also reviewed the RTNSS functions for the RBVS given in DCD Tier 2, Revision 9, Appendix 19A against guidance for selection and identification of such systems stated in RG 1.206, Section C.IV.9.

GDC 2 requires that SSCs important to safety shall be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches without loss of capability to perform their safety functions. DCD Tier 2, Revision 9, Section 9.4.4, states that the RBVS isolation dampers, associated instrumentation, and ducts are classified as safety-related and are designed to seismic Category I in accordance with RG 1.29, Regulatory Position C.1. The remainder of the system is classified as nonsafety-related and is designed to seismic Category II in accordance with RG 1.29, Regulatory Position C.2, to assure that the failure of nonsafety-related portions of the system cannot affect safety-related components.

In RAIs 9.4-41, 9.4-42, and 9.4-44, the staff asked the applicant to address several inconsistencies in DCD Tier 2, Revision 3, Section 9.4.6, figures and tables regarding the RBVS safety-related isolation dampers. RAIs 9.4-41, 9.4-42, and, 9.4-44, were being tracked as open items in the SER with open items. In response, the applicant made several changes to DCD Tier 2, Section 9.4.6, figures and tables, including (1) revising the REPAVS in DCD Tier 2, Figure 9.4-11, to be consistent with DCD Tier 2, Table 9.4-10, (2) revising DCD Tier 2, Figure 9.4-9 and Table 9.4-9, to include all REPAVS building isolation dampers, (3) revising DCD Tier 2, Figure 9.4-10, to include all CONAVS building isolation dampers, and (4) revising DCD Tier 2, Table 9.4-11, to identify the building isolation dampers as safety-related. The staff finds that the applicant's response is acceptable since the RBVS safety-related isolation dampers are clearly and consistently identified. The staff confirmed that the applicant incorporated the changes into DCD Revision 4. Based on the above, the applicant's responses, and DCD changes, RAIs 9.4-41, 9.4-42, and 9.4-44 are resolved.

In RAI 9.4-43, the staff requested that the applicant include additional information on the ventilation of the battery rooms associated with DCD Tier 2, Revision 3, Table 9.4-9, and potential hydrogen accumulation. RAI 9.4-43 was being tracked as an open item in the SER with open items. In response, the applicant revised DCD Tier 2, Section 9.4.6, to include the indication of battery room hydrogen concentration and an alarm for high battery room hydrogen concentration. In addition, the applicant clarified that batteries generate hydrogen when charging such that power is available to provide ventilation. The batteries do not generate hydrogen when discharging such that ventilation is not needed to exhaust the hydrogen. The staff finds that the response is acceptable since the applicant included statements in the DCD for monitoring and exhausting hydrogen from the safety-related battery rooms. The staff confirmed that the applicant incorporated the changes into DCD Revision 4. Based on the above, the applicant's responses and DCD changes, RAI 9.4-43 is resolved.

In RAI 9.4-45, the staff requested that the applicant (1) make tables and figures of the main steam tunnel AHU, main steam tunnel recirculation AHU and refuelling machine control room recirculation AHU consistent and (2) clarify the location of CONAVS safety-related dampers. RAI 9.4-45 was being tracked as an open item in the SER with open items. In response to the RAI, the applicant revised DCD Tier 2, Figure 9.4-10, to include the main steam tunnel AHUs. The applicant indicated that the refuelling machine control room recirculation AHU was too small to be included in the simplified system diagram. The applicant also clarified the location of the CONAVS. In DCD Revision 5, the applicant relocated the main steam tunnel AHUs from the RBVS to the TBVS. The staff finds that the RAI response is acceptable since the applicant included the appropriate AHUs in DCD Tier 2, Revision 9, Section 9.4, figures and tables and clearly identified the isolation dampers. Accordingly, based on the above, the applicant's responses and DCD changes, RAI 9.4-45 is resolved.

In RAI 9.4-46, the staff requested that the applicant include the building isolation dampers and note whether they are safety-related in DCD Tier 2, Revision 3, Figure 9.4-9. Because the smoke exhaust could be from contaminated areas, the staff also asked the applicant to identify any provision to monitor for radioactive release. RAI 9.4-46 was being tracked as an open item in the SER with open items. In response to RAI 9.4-46, the applicant revised DCD Tier 2, Figure 9.4-9, to show the building isolation dampers and the CLAVS isolation dampers. The applicant clarified that, because only clean areas are serviced by the CLAVS, radiation monitoring is not required. The staff finds that the applicant's response is acceptable since the applicant made appropriate changes to DCD Tier 2, Figure 9.4-9. In addition, because DCD Tier 2, Figure 9.4-9 is for the CLAVS or the clean portion of the RB, the staff agrees that radiation monitoring is not required. Based on the above, the applicant's responses, and DCD changes, RAI 9.4-46 is resolved.

The staff finds that because the RBVS conforms to the guidance of RG 1.29 with respect to seismic categorization, the design conforms to the guidelines of SRP Section 9.4.3. Therefore, the staff finds that the RBVS complies with the requirements of GDC 2.

GDC 5 requires that SSCs important to safety shall not be shared among nuclear power units unless it can be shown that such sharing will not significantly impair their ability to perform their safety functions, including, in the event of an accident in one unit, an orderly shutdown and cooldown of the remaining units. The ESBWR design is a single-unit station, and the requirements of GDC 5 are not applicable to the single-unit design.

GDC 60 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 produced during normal reactor operation, including anticipated operational occurrences. DCD Tier 2, Revision 9, Section 9.4.6, states that the RBVS includes the capability to suitably control release of gaseous radioactive effluents to the environment. During normal operation, the design includes the capability to direct the system exhaust air to the RB HVAC purge exhaust filtration units, which are designed, tested, and maintained in accordance with RG 1.140. Under accident conditions, the RBVS is isolated by safety-related dampers, ducts, and instruments to prevent the release of contamination to the environment through the intake and exhaust pathways.

In RAI 9.4-5, 9.4-5 S01, and 9.4-5 S02, the staff asked the applicant to provide a list of codes and standards used in the design of the ESBWR air conditioning, heating, cooling, and ventilation systems. RAI 9.4-5 was being tracked as an open item in the SER with open items. In response to RAI 9.4-5 S02, the applicant clarified that the applicable codes and standards are

discussed in the relevant sections describing the ESBWR HVAC systems. The applicant also provided a table in the RAI response showing where relevant standards are discussed throughout the DCD. The staff finds that the response is acceptable since the staff confirmed that the applicant discussed the applicable standards with the relevant systems. Based on the above and the applicant's response, RAI 9.4-5 is resolved.

In RAI 9.4-47 and supplemental RAIs, the staff requested that the applicant identify how the CLAVS exhaust air is monitored for radiation, because DCD Tier 2, Revision 3, Figure 9.4-9, shows that the CLAVS exhausts air directly outdoors, and to discuss the impact of post-accident releases. RAI 9.4-47 was being tracked as an open item in the SER with open items. Independent of the RAI process, the applicant implemented a design change and modified DCD Tier 2, Figure 9.4-9, in Revision 5 to direct the CLAVS exhaust through the RB/FB vent stack instead of directly outdoors. Therefore, in response to RAI 9.4-47 S02, the applicant stated that RB/FB vent stack radiation monitors monitor the CLAVS exhaust air in all modes. The applicant also discussed multiple design features, including maintaining the CLAVS at positive pressure relative to the CONAVS, to prevent contamination being transported from the potentially contaminated areas of the RB to the clean areas. The staff finds that the applicant's response is acceptable since the staff finds that the design change of directing the CLAVS exhaust air to the RB/FB vent stack gives reasonable assurance that releases from the CLAVS area of the RB will not exceed those assumed in the accident analysis. Since the CLAVS exhausts through the RB/FB vent, releases attributable to the CLAVS can be detected. Accordingly, based on the above, the applicant's responses, and DCD changes, RAI 9.4-47 is resolved.

In RAI 9.4-53, the staff requested the applicant explain the role of the CONAVS in the post-72-hour period. In response, the applicant clarified that no credit is taken for the operation of the CONAVS to produce negative pressure in the RB and consequently to reduce the exfiltration flow from the RB in the DCD Tier 2, Chapter 15, dose evaluations. The applicant also clarified that none of the postaccident dose evaluations credited use of the Reactor Building HVAC Accident Exhaust Filter Units for mitigating dose consequences. The applicant also identified that the RB HVAC accident exhaust filter units could be operated as defense-in-depth function after 8 hours following a DBA without causing an increase in the DCD Tier 2, Chapter 15, dose evaluations. The applicant revised the ITAAC in DCD Tier 1, Table 2.16-2-2, Items 11 and 12b, to clarify that the filter must meet two separate tests given in RG 1.140 and ASME AG-1, "Code on Nuclear Air and Gas Treatment": the efficiency as tested in the laboratory and the in place bypass leakage test, which is done in the field. The staff finds that the applicant's response is acceptable since the exfiltration flow from the RB in the dose calculations does not depend on the operation of either the CONAVS or the RB HVAC accident exhaust filter units. The ITAAC change is acceptable since it confirms that the RB HVAC accident exhaust filter units meet regulatory guidelines in RG 1.140 for testing nonsafety-related air filtration units. Accordingly, based on the above, the applicant's responses, and DCD changes, RAI 9.4-53 is resolved. The applicant stated that a portion of its response to RAI 9.4-53 was inadvertently omitted from DCD Revision 7 and provided a markup for incorporation into DCD Revision 8. The staff confirmed that the applicant incorporated the omitted change into DCD Tier 1, Revision 8, Table 2.16-2-2, Item 12b.

The staff finds that the RBVS design features conforms to RG 1.140, the therefore conform to the guidelines of SRP Section 9.4.3. Accordingly, the staff finds that the RBVS complies with the requirements of GDC 60.

10 CFR 50.63 requires a demonstration that the plant has the capability to withstand and recover from an SBO (i.e., loss of offsite electric power system concurrent with reactor trip and

unavailability of the onsite emergency ac electric power system). An SBO analysis covering a minimum acceptable duration (either to withstand the event until an alternate ac source and shutdown systems are lined up for operation or to cope with it for its duration, including the associated recovery period) is required. RG 1.155 provides guidance for complying with SBO requirements.

Evaluation of the Reactor Building Temperature within 0–72-Hours—Introduction

DCD Tier 2, Revision 9, Section 9.4.6.2, states that the RBVS is not required to operate during an SBO. DCD Tier 2, Revision 9, Section 9.4.6.3, states that rooms containing safety-related equipment have passive cooling features designed to limit the room temperature to the equipment's environmental qualification temperature. DCD Tier 2, Revision 9, Table 3H-15 lists, the results of the applicant's environmental temperature analysis for the RB.

The staff chose a room (i.e., number 1720) that contains safety-related DCIS equipment and the least amount of margin between the calculated room temperature at the end of the 72 hour period and the equipment qualification temperature for confirmatory assessment. The duration of the coping period is the 72 hour period in which all nonsafety-related ac power is assumed lost. After 72 hours, the RBVS, the CLAVS, and the CONAVS are expected to function. As described below, these subsystems support the RTNSS function of post 72 hour cooling for the DCIS cabinets and their electrical supporting equipment.

The applicant proposed an analytical approach, NEDE-33536P, (the portion associated with the RB is hereafter referred to as the RB environmental temperature analysis) as a means to demonstrate the passive heat removal mechanism. As described in section 9.4.1.3 of this report, based on industry literature[5] and current practice in containment analysis, the staff finds that the applicant's use of an analytical approach as a method to demonstrate the passive heat removal mechanism and to demonstrate that CRHA bulk temperature will not exceed design basis limits is reasonable.

Details on staff actions to review the CB portion of the CB and RB environmental temperature analysis for the ESBWR in NEDE-33536P are described in Section 9.4.1.3 of this report, and are similar to those used to review the RB portion of this report.

Input Assumptions: RB Heat Loads and Heat Sinks

The staff has reviewed input parameters used in the applicant's RB environmental temperature analysis in NEDE-33536P, such as heat sink wall thicknesses and surface areas, against values for the same parameter described elsewhere in the DCD. When input parameters depend on site-specific information, realistic or conservative parameters are used, such as the assumed as-built thermophysical properties of RB concrete and a conservative assumption of internal heat loads that assumes a high-energy line break with an SBO. Internal heat loads assumed are documented in Table G-3 of the RB environmental temperature analysis in NEDE-33536P for each room. The applicant assumed the highest normal operating temperature allowed in each room to be the initial heat sink temperature. In addition, the applicant used higher heat sink temperatures for walls in contact with the ground than would be expected.

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5 Yilmaz T.P. & Paschal W.B., "An Analytical Approach to Transient Room Temperature Analysis, Nuclear Technology, 114:135-140, April 1996.

In RAI 9.4-58 the staff requested that the applicant incorporate the CB and RB environmental temperature analysis for the ESBWR in NEDE-33536P into the DCD and to revise the ITAAC to specifically refer to this analysis.

In response to RAI 9.4-58, the applicant submitted NEDE-33536P (as Tier 2* information) as the design basis CB and RB environmental temperature analysis, and revised DCD Tier 1, Table 2.16.2-2, to add ITAAC 13. ITAAC 13 requires an applicant to demonstrate the passive heat sink performance of the RB. The applicant is to perform the design basis RB environmental temperature analysis using as-built information. The staff finds that the applicant's response is acceptable since the RB environmental temperature analysis uses a similar methodology to the CB environmental temperature analysis, which was evaluated and found acceptable in Section 9.4.1.3 of this report. In addition, the designation of the methodology as Tier 2* ensures that modeling assumptions evaluated by the staff will be retained in the as-built RB environmental temperature analysis. The staff also finds that the ITAAC is clearly linked to the Tier 2* approved methodology. Accordingly, based on the above, the applicant's responses, and DCD changes, RAI 9.4-58 is resolved.

Based on review of the submitted analysis the staff finds that the applicant's input assumptions are either based on information described elsewhere in the DCD or use realistic or conservative assumptions for RB heat loads and heat sinks and are therefore acceptable.

Proposed ESBWR Reactor Building Maximum Temperature Acceptance Criterion

The staff evaluated the proposed ESBWR RB maximum temperature criterion. For the first 72 hours following onset of such an accident, safety-related RB equipment is passively cooled through walls, floor, ceiling, and interior walls. DCD Tier 2, Revision 9, Section 9.4.6.3, states that the RB rooms containing safety-related equipment are designed to limit the room temperature to the equipment's environmental qualification temperature. This temperature is given at 50 degrees C (122 degrees F) as stated in DCD Tier 2, Revision 9, Table 3H-9.

The staff finds the proposed RB maximum temperature acceptance criterion acceptable because it is in accordance with equipment qualification assumptions used to evaluate the performance of associated equipment. As described below, the staff has considered the impact of this RB maximum temperature criterion on equipment performance.

Impact of Reactor Building Temperature Acceptance Criterion on Reactor Building Equipment

The staff evaluated the impact of the maximum RB temperature acceptance criterion value of 50 degrees C (122 degrees F) on RB equipment. In RAI 3.11-28, the staff requested that the applicant provide additional details on how the service temperature of electrical equipment, including computer-based instrumentation and control systems, will be determined for the ESBWR. In particular, the staff asked the applicant to provide details on this process for equipment that is to be located inside electrical cabinets and panels in the RB and the CB. The staff also asked the applicant to explain how the detailed design and testing of electrical equipment, including enclosures, would be carried out such that the key assumptions of environmental bounding temperatures in these areas remain conservative.

In response to RAI 3.11-28, the applicant revised DCD Tier 2, Sections 3.11.1.3, 3.11.4.3, and 3.11.3.1, to more fully explain the temperature qualification process. The applicant clarified the DCD Tier 2, Section 3.11.1.3, definition of equipment to indicate that computer-based

instrumentation and control equipment is defined by the equipment plus its surrounding cabinet or enclosure. The applicant clarified DCD Tier 2, Section 3.11.4.3, to indicate that system testing of computer-based instrumentation and control equipment within its cabinet or enclosure is preferred.

In DCD Tier 2, Revision 9, Section 3.11.3.1, the applicant states that the RB computer-based instrumentation and control equipment is to be type tested at temperatures that are 10 degrees C (18 degrees F) higher than the maximum temperature to which the equipment is exposed for the worst case abnormal operating occurrence, with the equipment at maximum loading. The RB computer-based instrumentation and control equipment is to be qualified at the nominal temperature of 50 degrees C (122 degrees F), as stated in DCD Tier 2, Table 3H-9. In addition, DCD Tier 2, Revision 9, Section 3.11.3.2, states that margins will be included in the qualification parameters to account for normal variations in commercial production of equipment and reasonable errors in defining satisfactory performance, and that the environmental conditions shown in the DCD Tier 2, Appendix 3H tables do not show such margins. The staff noted that, in DCD Tier 2, Section 3.11.3.2, the applicant referenced that the program margin would be in accordance with the guidance of IEEE Standard 323. IEEE Standard 323 recommends that a peak temperature margin of +8 degrees C (+14 degrees F) be applied during the temperature qualification process. Because the applicant is conducting type testing with a margin of 10 degrees C (18 degrees F), the staff finds that the applicant exceeds the IEEE Standard 323 guidelines. The staff finds that applicant's response is acceptable since the applicant modified the DCD to state that computer-based instrumentation and control systems are tested in the enclosures and that a test margin of 10 degrees C (18 degrees F) is applied to equipment in both harsh and mild environments. Based on the above, the applicant's responses and DCD changes, RAI 3.11-28 is resolved for the RB. Based on the maximum RB temperature acceptance criterion value being the equipment qualification temperature and on the applicant's clarification of the qualification process, the staff finds the maximum RB temperature acceptance criterion value acceptable.

Reactor Building Environmental Temperature Analysis for the ESBWR

The staff reviewed the means by which the RBVS heat sink was analyzed to ensure that the heat sink passively maintains the temperature in the RB within the design basis for the first 72 hours following a DBA. Verification of this design feature is through an RB environmental temperature analysis using heat sink dimensions, thermal properties, exposed surface areas, heat sink thermal properties, and the heat loads specified in DCD Tier 2, Revision 9, Table 3H-14. As previously discussed, the staff reviewed these input assumptions and finds them to be acceptable.

The staff reviewed the results of the applicant's RB environmental temperature analysis, as described in NEDE-33536P, as a basis for demonstrating that the RB can be passively cooled during the postulated accident. The staff reviewed the applicant's calculation and performed confirmatory calculations using the same methodology and input assumptions. The staff obtained similar results.

Based on the results of the RB environmental temperature analysis, as described in NEDE-33536P, confirmed by staff calculations (which show that the calculated RB room temperatures remain below equipment qualification temperatures) and its confirmation by ITAAC by the use of as-built information, the staff finds has confidence that RB environment conditions can be maintained below equipment qualification limits for 72 hours without the use of ac power.

Based on the use of passive design features to control RB air temperature, as reviewed above, the staff finds that the RBVS meets the guidance of RG 1.155, including Regulatory Position C.3.2.4, and therefore addresses the requirement of 10 CFR 50.63 in that necessary support systems provide sufficient capacity and capability for coping with an SBO event.

Regulatory Treatment of Nonsafety Systems

DCD Tier 2, Revision 9, Section 19A.8.4.10, states that component cooling will be performed by the HVAC systems in the RB, EB, FB, CB, and parts of the TB. In RAI 9.4-39, Part A, the staff requested that the applicant identify which parts of the RB are classified as RTNSS systems and which components need post-accident cooling. RAI 9.4-39, Part A was being tracked as an open item in the SER with open items. In response, the applicant revised DCD Tier 2, Section 9.4.6.1 to state that the RB HVAC system has RTNSS functions, as described in DCD Tier 2, Appendix 19A. The applicant added associated RTNSS design requirements in DCD Tier 1, Section 2.16.2.1, Items 7, 12a, and 12b, and Table 2.16.2-2, Items 7 and 12b to provide additional ITAAC for RTNSS functions associated with the post-72-hour N-DCIS cooling for FAPCS and RB HVAC accident exhaust filter efficiency.

For RTNSS functions, DCD Tier 2, Table 19A-2, lists the RBVS accident exhaust filters as a system that performs functions that fall under SECY-94-084 criterion E (SSC functions relied upon to prevent significant adverse systems interactions). The RB accident exhaust filters maintain filtering efficiency to ensure that theoretical control room doses are not exceeded for certain beyond design basis LOCAs. As stated in DCD Tier 2, Revision 9, Section 19.A.6.2.2, the applicant judged the failure to provide adequate filtration to be an adverse system interaction. The applicant has chosen to apply regulatory oversight by means of availability treatment for the system using the ACM to provide assurance that the filters will be capable of performing their function. In addition, as stated in DCD Tier 2, Revision 9, Section 19A.8.2, all RTNSS systems shall be in the scope of the Design Reliability Assurance Program, which will be incorporated into the Maintenance Rule. The staff reviewed DCD Tier 2, Revision 9, Section 19.A.8.3, and finds that the RBVS accident exhaust filters would be designed in accordance with standards for RTNSS Criterion E systems.

The staff reviewed the applicant’s proposed regulatory treatment of the system and the system description and finds that, as described in DCD Tier 2, Revision 9, Section 19A, the RBVS accident exhaust filters are RTNSS, and this portion of the RBVS would be subject to regulatory oversight via the ACM and the Maintenance Rule. The staff reviewed the proposed regulatory treatment, design standards, and the system design basis information in DCD Tier 2 against the criteria for such systems as given in RG 1.206, Section C.IV.9, and SECY-95-132 and has finds that the proposed regulatory treatment of the RBVS accident exhaust filters for RTNSS conform to this guidance and is therefore acceptable. The staff reviewed proposed ITAAC for RTNSS functions in DCD Tier 1, Revision 9, Table 2.16-2-2, and finds that the proposed ITAAC provides assurance that the identified RTNSS systems will be installed, inspected, and tested in accordance with the design. Based on the above, the applicant’s responses, and DCD changes, RAI 9.4-39, Part A is resolved.

Minimization of Contamination

In consideration of 10 CFR 20.1406, the staff reviewed the RBVS design to determine how the 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.

DCD Tier 2, Revision 9, Table 12.3-18, describes the provisions regarding RG 4.21 design objectives related to RBVS for the following:

- Having leak detection methods and early detection of leaks to avoid release of contamination from undetected leaks and to minimize contamination of the environment (Design Objective 3)

- Decreasing the spread of contaminant from the source (Design Objective 4)

- Minimizing the generation and volume of radioactive waste both during operation and during decommissioning, by minimizing the volume of components and structures that become contaminated during plant operation. (Design Objective 6)

The RBVS meets GDC 60 by suitably controlling the release of gaseous radioactive effluents to the environment. During normal operation, the design includes the capability of directing the system exhaust air to the RB HVAC purge exhaust filtration units. RB HVAC purge exhaust filtration units are designed, tested, and maintained in accordance with RG 1.140. Under accident conditions, the RBVS is isolated by safety-related dampers, ducts, and instruments to prevent the release of contamination to the environment through the intake and exhaust pathways. Accordingly, the staff finds that the RBVS design features conform to RG 1.140 and the guidelines of SRP Section 9.4.3.

The RBVS CONAVS uses a common supply air duct to distribute air to potentially contaminated areas of the RB. Air is exhausted from potentially contaminated areas to the RB/VB vent stack. The RB purge exhaust filter units are equipped with pre-filters, HEPA filters, high efficiency filters, and carbon filters for mitigating and controlling gaseous effluents from the RB.

The REPAVS subsystem is designed to permit exhaust air from the refueling area to be diverted through the RB HVAC purge exhaust filter units. The building isolation dampers close and the supply and exhaust fans stop when there is high radiation in the exhaust ducts.

The CLAVS uses a common supply air duct to distribute air to clean areas of the RB.

RBVS cooling coil condensate is collected in drain pans within the AHUs with the drain pan discharge (condensate) routed to a floor drain located within the room. These floor drains connect to the applicable equipment and floor drain subsystem.

The staff finds that these design provisions for the RBVS are adequate to minimize contamination of the environment and minimize the generation of radioactive waste. The provisions meet the requirement of 10 CFR 20.1406 and are consistent with guidelines of RG 4.21. Section 12.4 of this report further addresses the ESBWR design in accordance with 10 CFR 20.1406.

9.4.6.4 *Conclusion*

Based on the above discussion, the staff finds that the ESBWR RBVS design conforms to the requirements GDC 2 and 60; 10 CFR 50.63; and 10 CFR 20.1406. Because the ESBWR design is a single unit, GDC 5 is not applicable.

9.4.7 Electrical Building HVAC System

9.4.7.1 *Regulatory Criteria*

The staff reviewed the DCD Tier 2, Revision 9, Section 9.4.7 in accordance with SRP Section 9.4.3, Revision 3. The staff's acceptance of the Electrical Building HVAC System (EBVS) is based on compliance with the following requirements:

- GDC 2, as it relates to the capability to withstand earthquakes
- GDC 5, as it relates to sharing systems and components important to safety
- GDC 19, as it applies to the habitability criteria specified by NUREG–0696, "Functional Criteria for Emergency Response Facilities," issued February 1981, for the TSC
- GDC 60, as it relates to the capability of the system to suitably control release of gaseous radioactive effluents to the environment
- 10 CFR 20.1406, as it relates to minimization of contamination

The SRP acceptance criteria are also based on conformance to the following guidelines:

- NUREG–0696, which provides guidance for establishing the habitability requirements of the TSC

9.4.7.2 *Summary of Technical Information*

The EBVS maintains acceptable temperatures for equipment and personnel comfort and habitability in the EB. It consists of three subsystems: the electric and electronic rooms (EER) HVAC subsystem (EERVS), the TSC HVAC subsystem (TSCVS), and the diesel generators HVAC subsystem (DGVS). The EERVS and DGVS do not perform or ensure any safety-related function, and thus have no safety design basis. The TSCVS performs functions related to emergency response facilities.

The EBVS is classified as nonsafety-related. The EBVS has RTNSS functions as described in DCD Tier 2, Revision 9, Appendix 19A, which provides the level of oversight and additional requirements to meet the RTNSS functions. Performance of RTNSS functions is ensured by applying the defense-in-depth principles of redundancy and physical separation to ensure adequate reliability and availability as described in DCD Tier 2, Revision 9, Section 19A.8.3.

The EERVS provides conditioned air to maintain acceptable temperatures for equipment and personnel comfort and habitability, provides a sufficient quantity of filtered fresh air for personnel, and maintains the hydrogen concentration levels in the nonsafety-related battery rooms below 2 percent by volume in accordance with RG 1.128

The SDGs provide electrical power to the EERVS in case of a LOPP. The EBVS provides the post-72-hour cooing for safety-related electrical distribution and support for electrical power to the FAPCS.

The EERVS provides a controlled environment for the EB switchgear, electronic, and nonsafety-related battery rooms. The EERVS consists of two independent HVAC trains. One train

services the rooms where the train A electric and electronic equipment is located, and the other EERVS train services the rooms where the train B electric and electronic equipment is located. Each EERVS train is a recirculation ventilation system to provide heated or cooled air to the EER. The recirculating system includes an AHU with filter, heating and cooling coils and two redundant fans. Building air is returned or exhausted by two redundant fans. Dedicated exhaust fans are provided for the seven battery rooms.

The TSCVS provides a controlled environment for personnel comfort and safety and for the proper operation and integrity of equipment in the TSC. The TSCVS also maintains the TSC at a slightly positive pressure with respect to the adjacent rooms and outside environment to minimize the infiltration of air. In addition, the TSCVS automatically switches to the recirculation mode if smoke is detected in the outside intake air. In this case, there may be no differential pressure between the TSC and the surrounding areas.

The TSCVS is a recirculating ventilation system to provide filtered conditioned air to the TSC. Two redundant AFUs with supply fans, HEPA filters, and charcoal filters remove radioactive materials when required. The AFUs provide fresh air to the TSC to augment the return air to maintain the TSC under slightly positive pressure. The recirculating AHU system includes redundant AHUs (with fans, air mixing plenum, filters, heating and cooling coils, and humidifier) to provide conditioned air to the TSC through ducts, dampers, and registers. The exhaust system includes redundant fans to direct the air from the kitchen and toilet areas into the atmosphere.

The TSCVS contains nonsafety–related filter units. The TSCVS filter units are defense-in-depth components and provide the function of filtration for the TSC during conditions of abnormal airborne radioactivity when power is available. Because RG 1.140 applies specifically to normal atmosphere cleanup, and because the filter units are not credited engineered safety feature units in accordance with RG 1.52, the codes and standards that dictate the testing of a filtration system designed for habitability are applicable to the TSCVS. The specific tested and credited filtration efficiencies meet or exceed the guidance in RG 1.140.

The TSCVS detects and limits the introduction of airborne hazardous materials (radioactivity or smoke) into the TSC. The TSCVS removes vitiated air from the kitchen and restrooms. Redundant components are included to increase the reliability, availability, and maintainability of the ventilation system. The SDGs provide electrical power to the TSCVS in case of LOPP.

The DGVS does the following:

- Provides ventilation air to maintain acceptable temperatures within the generator rooms for equipment operation and reliability during periods of diesel generator operation.

- Provides adequate heating and ventilation for suitable environmental conditions for maintenance personnel working in the diesel generator room when the generators are not in operation.

- Provides suitable environmental conditions for equipment operation in each diesel generator electrical and electronic equipment area under the various modes of diesel generator operation.

- Prevents the accumulation of combustible vapors and dissipate their concentration in the fuel oil day tank room.

The SDGs provide electrical power to the DGVS in case of a LOPP.

9.4.7.3 *Staff Evaluation*

The staff review focused on compliance with GDC for this system. The system is classified as nonsafety-related. In addition, the staff considered the guidance of NUREG–0696 and Section 4.6.6 of the EPRI URD. The staff has also reviewed the RTNSS functions for the EBVS as given in DCD Tier 2, Revision 9, Appendix 19A, against guidance for the selection and identification of such systems stated in RG 1.206, Section C.IV.9.

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

DCD Tier 2, Revision 9, Section 9.4.7, states that the EBVS complies with RG 1.29, Regulatory Position C.2 for nonsafety-related portions of the system. The EBVS components are designated as seismic Category NS. The EB is nonsafety-related and seismic Category NS. The staff finds that because the EBVS conforms to the guidance of RG 1.29 in respect to seismic categorization, the design conforms to the guidelines of SRP Section 9.4.3. Accordingly, the staff finds that the EBVS complies with the requirements of GDC 2.

GDC 5 requires that SSCs important to safety shall not be shared among nuclear power units unless it can be shown that such sharing will not significantly impair their ability to perform their safety functions, including, in the event of an accident in one unit, an orderly shutdown and cooldown of the remaining units. The ESBWR design is a single-unit station, and the requirements of GDC 5 are not applicable to the single-unit design.

GDC 19 requires that a control room shall be provided from which actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions, including LOCAs. 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 0.05 sieverts (5 rem) whole body, or its equivalent to any part of the body, for the duration of the accident.

DCD Tier 2, Revision 9, Section 9.4.7, states that the ESBWR TSC, located in the EB, is designed to comply with the NUREG–0696 guidance. NUREG–0696 Section 2.6, guidance on TSC habitability, states that the TSC shall have the same radiological habitability as the control room under accident conditions and that TSC personnel shall be protected from radiological hazards, including direct radiation and airborne radioactivity from in-plant sources under accident conditions, to the same degree as control room personnel. NUREG–0696 Section 2.6 also states that applicable criteria are specified in GDC 19 and SRP Section 6.4. Regarding the TSC ventilation system, NUREG–0696 guidance states that the TSC ventilation system shall function in a manner comparable to the control room ventilation system and that, as a minimum, a TSC ventilation system that includes HEPA and charcoal filters is needed.

The TSCVS design includes HEPA and charcoal filters. DCD Tier 2, Revision 9, Section 9.4.7.2, states that the TSCVS filter units will be designed and tested in accordance with RG 1.140. NUREG–0696 references SRP Section 6.4, "which states that RG 1.52 should be referenced as guidance for ventilation system design and for expected performance of the TSC area. Although the ESBWR TSC is designed to be used during abnormal operating occurrences, it is not credited as a post-accident engineered safety feature system. On this

basis, and in view of the above, the staff finds that use of RG 1.140 to meet NUREG–0696 guidance on TSC habitability is acceptable, and that the ESBWR TSC design adequately addresses NUREG–0696 guidance that the TSC ventilation system shall function in a manner comparable to the control room ventilation system.

To ensure radiological protection of TSC personnel, radiation monitoring systems are provided. Existence of these systems is verified via ITAAC in DCD Tier 1, Revision 9, Section 2.3.2 and Table 2.3.1-1.

The TSCVS is supplied by a nonsafety-related power source. As stated in DCD Tier 2, Revision 9, Section 9.4.7.1, the nonsafety-related SDGs provide electrical power to the TSC HVAC subsystem in case of LOPP. Although the supply of ac power to the TSCVS is not identified as an RTNSS function, the staff notes that availability of power to the TSCVS is enhanced by the RTNSS regulatory treatment of the nonsafety-related SDGs in the ACM. If all ac power is lost during an accident, in accordance with NUREG–0696, the TSC plant management function could be performed by the control room while the TSC remains uninhabitable.

In RAIs 9.4-25 and 14.3-61, the staff requested the applicant to clarify its compliance with the recommendations of NUREG–0696 and to provide corresponding ITAAC. RAIs 9.4-25 and 14.3-61 were being tracked as open items in the SER with open items. In responses, the applicant indicated that a discussion of compliance with NUREG–0696 would be included in DCD Tier 2, Revision 3. The applicant revised DCD Tier 1, Section 2.16.2.7, to include TSCVS ITAAC required to confirm that the TSC provides a habitable work environment when nonsafety-related power is available. DCD Tier 1, Revision 9, Section 2.16.2.7, Items 3, 4, and 5, and ITAAC in DCD Tier 1, Revision 9, Table 2.16.2-10 Items 3, 4, and 5, provide assurance that the HEPA filters and charcoal of the TSCVS AFU are installed in accordance with the DCD and that TSCVS AFU maintain the TSC at a slight positive pressure with respect to the surrounding areas. The staff finds that the applicant’s response is acceptable because, as discussed above, DCD Tier 2, Revision 9, adequately addresses conformance with NUREG–0696. In addition, the EBVS ITAAC incorporate the key features of the TSCVS that conform to NUREG–0696. Accordingly, based on the above, the applicant’s responses, and DCD changes, RAIs 9.4-25 and 14.3-61 are resolved.

Based on TSCVS conformance to NUREG–0696 and the corresponding ITAAC, the staff finds that the EBVS complies with the requirements of GDC 19.

GDC 60 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 produced during normal reactor operation, including anticipated operational occurrences. The EB, including the EERVS, TSCVS, and DGVS service areas, does not have any source of radioactive materials in either particulate or gaseous form. Therefore, the staff finds that the EBVS meets the requirements of GDC 60.

DCD Tier 2, Revision 3, Section 9.4.7.1, states that the EERVS provides fresh filtered air. In RAI 9.4-48, the staff requested the applicant to provide in Section 9.4.7 the major components of the EBVS, including subsystems and basic design features such as flow rates. RAI 9.4-48 was being tracked as an open item in the SER with open items. In response, the applicant revised DCD Tier 2, Figure 9.4-12, to show the air inlet louvers more clearly and indicated that DCD Tier 2, Table 9.4-16, lists EBVS subsystem flow rates. The applicant updated DCD Tier 2, Revision 4, to describe the EBVS more clearly. The applicant indicated that major component data are included in DCD Tier 2, Table 9.4-16. The staff finds that the RAI response is

acceptable because the revised DCD Tier 2, Revision 9, Section 9.4.7 and associated tables and figures identify the basic design features and system parameters of the EBVS. Based on the above, the applicant's responses, and DCD changes, RAI 9.4-48 is resolved.

DCD Tier 2, Section 19A.8.4.10, states that component cooling will be performed by the HVAC systems in the RB, EB, FB, CB, and parts of the TB. In RAI 9.4-39, Part B, the staff requested that the applicant identify which parts of the EBVS are classified as RTNSS systems and which components need post-accident cooling. RAI 9.4-39, Part B was being tracked as an open item in the SER with open items. In response, the applicant revised DCD Tier 2, Section 9.4.7.1, to state that the EBVS has RTNSS functions as described in DCD Tier 2, Appendix 19A, with the associated RTNSS design requirements. The applicant added DCD Tier 1, Section 2.16.2.7 and Table 2.16.2-10, to provide additional ITAAC for RTNSS functions associated with post-72-hour cooling for diesel generators and safety-related electrical distribution and with support for electrical power to the FAPCS.

For RTNSS functions, DCD Tier 2, Revision 9, Table 19A-2, lists the EBVS as a system that performs functions that fall under SECY-94-084 criterion C (SSC functions relied upon under power-operating and shutdown conditions to meet the NRC's safety goal guidelines of CDF and LRF). The EBVS is a support system for the FAPCS. It provides equipment and room cooling to support the SDGs and PIP buses. In DCD Tier 2, Revision 9, Section 19A.4.2, the applicant states that the existence of the function provides the CDF and LRF reduction needed to address the PRA uncertainty concerns associated with the performance of passive system components. The applicant has chosen to apply regulatory oversight by means of availability treatment for the system through the Maintenance Rule performance monitoring program. As stated in DCD Tier 2, Revision 9, 19A.8.2, all RTNSS systems shall be in the scope of the Design Reliability Assurance Program, which will be incorporated into the Maintenance Rule. The staff reviewed Section DCD Tier 2, 19.A.8.3, and finds that the EBVS is subject to design standards for RTNSS Criterion C systems.

The staff reviewed the applicant's proposed regulatory treatment of the system and system description and finds that, as described in DCD Tier 2, Revision 9, Section 19A, the EERVS and DGVS portions of the EBVS are RTNSS systems while the TSCVS is not, and the EERVS and DGVS portions of the EBVS would be subject to regulatory oversight via the Maintenance Rule.

The staff reviewed the proposed regulatory treatment, design standards, and the system design basis information in DCD Tier 2 against the criteria for such systems given in RG 1.206, Section C.IV.9, and SECY-95-132. The staff finds that the proposed regulatory treatment of the EBVS for RTNSS conforms to this guidance and is therefore acceptable. The staff reviewed the response to RAI 9.4-39, Part B, and the proposed ITAAC for RTNSS functions in DCD Tier 1, Table 2.16-2-10, and finds that the proposed ITAAC provide assurance that the identified RTNSS systems will be installed, inspected, and tested in accordance with the design. Based on the applicant's response and DCD revision, RAI 9.4-39, Part B is resolved.

In RAIs 9.4-5, 9.4-5 S01, and 9.4-5 S02, the staff asked the applicant to provide a list of codes and standards used in the design of the ESBWR HVAC systems. RAI 9.4-5 was being tracked as an open item in the SER with open items. In response to RAI 9.4-5 S02, the applicant clarified that the applicable codes and standards are discussed in the relevant sections describing the ESBWR HVAC systems. The applicant also provided a table in the RAI response showing where relevant standards are discussed throughout the DCD. The staff finds that the response is acceptable since the staff confirmed that the applicant discussed the

applicable standards with the relevant systems. Based on the above and the applicant's response, RAI 9.4-5 is resolved.

In consideration of 10 CFR 20.1406, the staff reviewed the EBVS design to determine how the 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.

DCD Tier 2, Revision 9, Table 12.3-18, describes the provisions regarding RG 4.21 design objectives related to EBVS for the following:

- Minimizing the generation and volume of radioactive waste both during operation and during decommissioning, by minimizing the volume of components and structures that become contaminated during plant operation. (Design Objective 6)

- Decreasing the spread of contaminant from the source (Design Objective 4)

The EBVS meets GDC 60 because the EERVS, TSCVS, and DGVS have no source of radioactive materials in either particulate or gaseous form. The exhaust systems have no provision for filtration or adsorption because these areas are clean.

The TSCVS subsystem maintains the TSC at a slightly positive pressure with respect to the outside environment to minimize the infiltration of air. The TSCVS detects and limits the introduction of airborne hazardous materials into the TSC.

The staff finds that these design provisions for the EBVS meet the requirement of 10 CFR 20.1406 and conform to the guidance in RG 4.21. Section 12.4 of this report further addresses the ESBWR design in accordance with 10 CFR 20.1406.

9.4.7.4 *Conclusion*

Based on the above discussions, the staff finds that the ESBWR EBVS design conforms to the requirements of GDC 2, 19, and 60 and 10 CFR 20.1406. Because the ESBWR design is a single unit, GDC 5 is not applicable. The staff also finds that the ESBWR EBVS design conforms to the guidelines of NUREG–0696.

9.4.8 Drywell Cooling System

9.4.8.1 *Regulatory Criteria*

The staff reviewed DCD Tier 2, Revision 9, Section 9.4.8, in accordance with SRP Section 9.4.3, Revision 3. The staff's acceptance of the DCS is based on compliance with the following requirements:

- GDC 2, as it relates to the capability to withstand earthquakes

- GDC 5, as it relates to sharing systems and components important to safety

- GDC 60, as it relates to the capability of the system to suitably control release of gaseous radioactive effluents to the environment

- 10 CFR 20.1406, as it relates to minimization of contamination

9.4.8.2 *Summary of Technical Information*

The DCS maintains the thermal environment within the drywell to specified conditions during normal reactor operation, hot standby and refueling using fan cooling units (FCUs). The cooling medium of the FCUs is CWS water. There are separate FCUs for the upper and the lower drywell regions.

The DCS is classified as a nonsafety-related and seismic Category II system. During stable and transient operating conditions through the entire operating range, from startup to full load condition to refueling, the DCS maintains temperature in the upper and the lower drywell spaces within specified limits, accelerates drywell cooldown during the period from hot shutdown to cold shutdown, and aids in complete purging of nitrogen from the drywell during shutdown. The DCS also maintains a habitable environment for plant personnel during plant shutdowns for refueling and maintenance and limits drywell temperature during LOPP.

The DCS is a closed loop recirculating air/nitrogen cooling system with no outside air/nitrogen introduced into the system except during refueling. The system uses direct-drive type FCUs, with variable frequency drives, to deliver cooled air/nitrogen to various areas of the upper and lower drywell. Ducts distribute the cooled, recirculated air/nitrogen through diffusers and nozzles. The FCUs and the fans are redundant.

The cooling coils of the FCUs transfer the drywell heat loads to the CWS. The DCS consists of four FCUs, two located in the upper drywell and two in the lower drywell. During normal plant operating conditions, one fan in each upper drywell FCU is in operation. In this configuration, 50 percent of the upper drywell design heat load is accommodated by each FCU. Each FCU comprises a cooling coil and two fans downstream of the coil. One of the fans operates while the other is on standby. The fan on standby automatically starts upon loss of the lead fan in each FCU. Upon loss of one FCU, both fans in the affected unit are secured, and the fans in the remaining FCU are started or continue to operate. During this upset operation, the cooling capacity of the operating FCU increases to twice its normal capacity with double the airflow; however, the ambient temperature is also increased.

Cooled air/nitrogen leaving the upper FCUs enters a common plenum and is distributed to the various zones in the upper drywell through distribution ducts. Return ducts are also provided. The upper FCUs draw air/nitrogen directly from the upper drywell.

During normal plant operating conditions, one fan in each lower drywell FCU is in operation. In this configuration, each FCU accommodates 50 percent of the lower drywell design heat load. Each FCU comprises a cooling coil and two fans downstream of the coil. One of the fans operates while the other is on standby. The fan on standby automatically starts upon loss of the lead fan in each FCU. Upon loss of one FCU, both fans in the affected FCU are secured, and the fans in the remaining FCU are started or continue to operate. During this upset operation, the cooling capacity of the operating FCU increases to twice its normal capacity with double the airflow; however, the ambient temperature is also increased.

Cooled air/nitrogen is supplied below the RPV and in the RPV support area through supply ducts. Return ducts are also provided. The lower FCUs draw air/nitrogen directly from the lower drywell.

Each FCU has a condensate collection pan. The condensate collected from the FCUs in the upper and the lower drywell is piped to an LD&IS flowmeter to measure the condensation rate contribution to unidentified leakage.

The CWS piping penetrates the containment at two independent locations, redundantly. The system is designed so that both FCUs in the upper drywell and both FCUs in the lower drywell are always operating during normal plant operation, even upon failure of any single FCU motor or fan. Upon failure of one FCU, the two fans of the remaining FCU are in service. One FCU with two fans in operation maintains the drywell temperature below the maximum allowed. The FCU fans and fan motors are designed to be operable during containment integrated leak rate testing.

9.4.8.3 *Staff Evaluation*

The staff review focused on compliance with the GDC for this system. The system is classified as nonsafety-related.

GDC 2 requires that SSCs important to safety shall be designed to withstand the effects of natural phenomena, such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches without loss of capability to perform their safety functions. DCD Tier 2, Revision 9, Section 9.4.8, states that the DCS complies with RG 1.29, Regulatory Position C.2, for nonsafety-related portions of the system. The DCS components are designated as seismic Category II. The staff finds that the DCS design conforms to RG 1.29 with respect to seismic categorization. Therefore the design conforms to the guidelines of SRP Section 9.4.3 for GDC 2. Accordingly, the staff finds that the DCS complies with the requirements of GDC 2.

GDC 5 requires that SSCs important to safety shall not be shared among nuclear power units unless it can be shown that such sharing will not significantly impair their ability to perform their safety functions, including, in the event of an accident in one unit, an orderly shutdown and cooldown of the remaining units. The ESBWR design is a single-unit station, and the requirements of GDC 5 are not applicable to the single-unit design.

GDC 60 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 produced during normal reactor operation, including anticipated operational occurrences. DCD Tier 2, Revision 9, Section 9.4.8, states that the DCS includes the capability to suitably control release of gaseous radioactive effluents to the environment. The FCUs recirculate air/nitrogen inside the upper and lower drywell. The recirculated air/nitrogen is retained in the primary containment structure. The liquid condensate from the fan cooling coils is collected and measured by the LD&IS to determine the condensation rate contribution to the unidentified leakage. The staff finds that the DCS design features conform to the guidelines of SRP Section 9.4.3. Accordingly, the staff finds that the DCS complies with the requirements of GDC 60.

In RAIs 9.4-5, 9.4-5 S01, and 9.4-5 S02, the staff asked the applicant to provide a list of codes and standards used in the design of the ESBWR HVAC systems. RAI 9.4-5 was being tracked as an open item in the SER with open items. In response to RAI 9.4-5 S02, the applicant clarified that the applicable codes and standards are discussed in the relevant sections describing the ESBWR HVAC systems. The applicant also provided a table in the RAI response showing where relevant standards are discussed throughout the DCD. The staff finds that the response is acceptable. The staff confirmed that the applicant applied the appropriate

standards to the HVAC systems. Based on the above and the applicant's response, RAI 9.4-5 is resolved.

In consideration of 10 CFR 20.1406, the staff reviewed the DCS design to determine how the 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.

DCD Tier 2, Revision 9, Table 12.3-18, describes the provisions regarding RG 4.21 design objectives related to DCS for the following:

- Minimizing leaks and spills (Design Objective 1)

The DCS is a closed loop recirculating air/nitrogen cooling system with no outside air/nitrogen introduced into the system except during refueling. During normal operation, the DCS re-circulates air with no connection to any HVAC system outside containment. Only during drywell purge operations is the containment air connected with the CONAVS of the RBVS. During drywell purge operations, the containment purge fan can be used to discharge containment air to the CONAV. The CONAVS has RB HVAC purge exhaust filter units that are designed, tested and maintained in accordance with RG 1.140.

The staff finds that these design provisions for the DCS meet the requirement of 10 CFR 20.1406 and conform to the guidelines of RG 4.21. Section 12.4 of this report further addresses the ESBWR design in accordance with 10 CFR 20.1406.

9.4.8.4 *Conclusion*

Based on the above discussion, the staff finds that the ESBWR DCS design conforms to the requirements of GDC 2 and 60 and 10 CFR 20.1406. Because the ESBWR design is a single unit, GDC 5 is not applicable.

9.4.9 Containment Inerting System

9.4.9.1 *Regulatory Criteria*

No SRP guidelines are directly applicable to the review of the CIS.

9.4.9.2 *Summary of Technical Information*

The CIS is described in DCD Tier 2, Revision 9, Sections 6.2.5.2 and 9.4.9. The CIS does not perform any safety-related function.

The CIS establishes and maintains an inert nitrogen atmosphere within the primary containment during all plant operating modes, except during plant shutdown for refueling or maintenance and during limited periods of time to permit access for inspection during reactor low-power operation. The purpose of the system is to provide an inert containment atmosphere (less than 3 percent oxygen) during normal operation to minimize hydrogen burn inside the containment. The CIS maintains a positive pressure in containment to prevent air in-leakage from the RB.

The CIS comprises a pressurized liquid nitrogen storage tank, a steam-heated main vaporizer for large nitrogen flow, an electric heater for vaporizing makeup flow, two supply injection lines

(a makeup line and an inerting line), two exhaust lines, a bleed line, a containment overpressure protection line, and associated valves, controls, and instrumentation.

The CIS penetrates containment via nitrogen injection lines in the drywell and suppression pool airspace. The CIS includes an exhaust line from the lower drywell on the opposite side of containment from the injection points. For containment overpressure protection during severe accident conditions, the exhaust is from the suppression pool airspace. The exhaust lines connect to the RBVS exhaust before being diverted to the plant stack.

The CIS also provides nitrogen to the HPNSS.

The CIS can be used under post-accident conditions for containment atmosphere dilution by a controlled purge of the containment atmosphere with nitrogen to reduce combustible gas concentrations. The CIS can also be used manually during severe accident conditions for containment overpressure protection. However, these functions are not credited in the safety analysis.

9.4.9.3 *Staff Evaluation and Conclusion*

The CIS is intended to provide an inerted containment in compliance with 10 CFR 50.44(c)(2), "Combustible gas control for nuclear power reactors." Section 6.2.5 of this report addresses the staff's evaluation of the design's compliance with the requirements of 10 CFR 50.44(c)(2).

The CIS provides nitrogen to the HPNSS. Section 9.3.8 of this report addresses the staff's evaluation of the HPNSS.

9.5 Other Auxiliary Systems

9.5.1 Fire Protection Program

9.5.1.1 *Regulatory Criteria*

The staff reviewed the DCD Tier 1, Revision 9, Section 2.16.3, and DCD Tier 2, Revision 9, Section 9.5.1, in accordance with SRP Section 9.5.1, Revision 5. The staff's acceptance of the ESBWR fire protection program (FPP) is based on meeting the relevant requirements of the following regulations:

- 10 CFR 50.48(a)(4), "Fire protection," requires, in part, that each applicant for a design certification under 10 CFR Part 52 must have a description and analysis of the fire protection design features for the standard plant necessary to demonstrate compliance with GDC 3, "Fire Protection."

- GDC 3 requires the following:
  - SSCs important to safety be designed and located to minimize, consistent with other safety requirements, the probability and effect of fires and explosions
  - Noncombustible and heat resistant materials be used wherever practical throughout the unit
  - Fire detection and fighting systems of appropriate capacity and capability be provided and designed to minimize the adverse effects of fires on SSCs

- Fire fighting systems be designed to assure that their rupture or inadvertent operation does not significantly impair the safety capability of these SSCs

- GDC 19 requires the plant design to include a control room that allows plant operators to maintain the plant in a safe condition under normal and accident conditions and to make equipment available at alternate locations outside the control room to achieve and maintain hot shutdown with the potential capability for subsequent cold shutdown of the reactor.

- GDC 23, “Protection system failure modes,” requires that the reactor protection system be designed to fail in a safe state if postulated adverse environments occur, including extreme heat and fire and water discharged from fire suppression systems.

- 10 CFR 52.47(b)(1) requires an application for design certification to contain proposed ITAAC which are necessary and sufficient to provide reasonable assurance that, if inspections, tests, and analyses are performed and acceptance criteria are met, a plant that references the design is built and will operate in accordance with the design certification.

The SRP acceptance criteria are also based on conformance to the following guidelines:

- RG 1.189, Revision 1, “Fire Protection for Nuclear Power Plants,” provides guidance and acceptance criteria for one acceptable approach for an FPP that meets the regulatory requirements described above.

In addition to the regulatory requirements and guidance provided above, SRP Section 9.5.1 provides enhanced fire protection criteria for new reactor designs, as documented in SECY-90-016, “Evolutionary Light Water Reactor (LWR) Certification Issues and Their Relationship to Current Regulatory Requirements,” dated January 12, 1990; SECY-93-087, “Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs,” dated April 2, 1993; and SECY-94-084. SECY-90-016 provides enhanced fire protection criteria for evolutionary LWRs. SECY-93-087 recommends that the enhanced criteria be extended to include passive reactor designs. The Commission approved SECY-90-016 and SECY-93-087 in staff requirements memoranda. SECY-94-084, in part, provides criteria defining safe-shutdown conditions for passive LWR designs.

9.5.1.2 *Summary of Technical Information*

The technical information in this section of the report includes a summary of the applicant’s key fire protection design commitments set forth in the DCD Revision 9. The FPS is the integrated complex of equipment and components that provide early fire detection and suppression to limit the spread of fires. The FPS is part of the overall FPP, which includes the plant design and layout, as well as administrative controls and procedures to prevent or mitigate fires. In accordance with SRP Section 9.5.1 and RG 1.189, the FPP uses the concept of defense-in-depth to achieve the required degree of reactor safety through administrative controls, FPS features, and safe-shutdown capability. The ESBWR FPS does not perform any safety-related function; however, because of nonsafety-related to safety-related interfaces and RTNSS positions, some FPS equipment and structures have elevated seismic and quality classifications.

The FPS can serve a nonsafety-related defense-in-depth function of providing a backup source of makeup water through a piping connection to the FAPCS for the ICS/PCCS pools and the SFP and a backup source for reactor water inventory control following a DBA. If necessary, the

makeup function will begin no later than 72 hours after a LOCA. The minimum total makeup flow rate is 46 m^3/hr (200 gpm) and the fire water storage is sufficient to provide this makeup through at least the 7th day after the accident. This function of the FPS is considered to be RTNSS rather than safety-related because it is not relied upon until at least 72 hours after the LOCA. In addition to meeting the applicable regulatory requirements, the ESBWR FPP and FPS are in accordance with applicable industry standards, including NFPA 804, "Standard for Fire Protection for Advanced Light Water Reactor Electric Generating Plants" (2006 edition), and the IBC.

DCD Tier 2, Revision 9, Section 9.5.1, includes a description of FPP compliance with the IBC. Because the staff's evaluation of the ESBWR FPP is only applicable to U.S. nuclear power plants, a review of IBC compliance is not required and was not performed.

DCD Tier 2, Revision 9, Sections 9.5.1.16 and 9A.7, list the fire protection COL information items.

9.5.1.3 *Staff Evaluation*

The staff reviewed the ESBWR FPP in accordance with SRP Section 9.5.1, Revision 5, and RG 1.189.

Fire Hazards Analysis

DCD Tier 2, Revision 9, Appendix 9A describes the ESBWR fire hazards analysis (FHA). The ESBWR FHA establishes and evaluates distinct fire areas for the RB, FB, CB, TB, RW, EB, yard, pump house, guard house, hot machine shop, service water/water treatment building, cold machine shop, warehouse, training center, service building, auxiliary boiler building, administration building, ancillary diesel building, and the fire pump enclosure. The FHA is based on an assessment of every fire area, using the defense in depth approach from RG 1.189. The aim of defense-in-depth, as described by the applicant in DCD Tier 2, Revision 9, Section 9A.3.1, is to provide a high degree of fire protection by implementing three concepts: (1) preventing potential fires from starting, (2) quickly detecting those fires that occur and promptly controlling and extinguishing fires to limit damage, and (3) providing structural protection (such as fire-rated barriers) for buildings, equipment, and circuits so that a fire that is not promptly extinguished will not prevent safe shutdown, cause loss of life, or result in radioactive release in excess of 10 CFR Part 20 limits. None of the defense-in-depth concepts is complete by itself.

The FHA is based on the existing design and on the currently specified, but not yet purchased, equipment. It is also based on the introduction of transient combustibles to any area of the plant, subject to administrative controls. The analysis assumes combustible transient materials are controlled to comply with the guidance of RG 1.39, "Housekeeping Requirements for Water-Cooled Nuclear Power Plants."

The applicant conservatively determined the combustible loading limit for electrical areas as 1,400 megajoules per square meter (MJ/m^2) (123,600 British thermal units per square foot [BTU/ft^2]), and conservatively calculated the combustible loading limit for all other indoor areas as 700 MJ/m^2 (61,800 BTU /ft^2). The fire loading of electrical cable in trays is based on flame-retardant, cross-linked polyethylene insulation with a maximum calorific value of 29.8 MJ per kg (12,834 BTU per pound-mass [lbm]). The cable trays are assumed to have the maximum (40 percent) design fill; actual cable fills may be lower. The analysis uses 48.8 kg of insulation

per square meter (10 lbm/ft^2) of cable tray. The combustible loading is based on maximum loading.

Rooms that exceed the combustible loading limits stated above rely upon automatic fire suppression. This approach conservatively assumes that all combustible material within a fire area instantaneously releases its net heat content upon ignition of the fire. Because of the considerable separation of components and fire barriers provided in the ESBWR plant layout, a detailed analysis or modeling of fire damage and plume temperatures resulting from any given fire was not considered necessary and was not performed.

The FB, RW, EB, yard, and TB do not contain any safe-shutdown components, a fire in these buildings does not affect the capability of any of the four divisions used to bring the reactor to hot standby and then cold shutdown conditions. The TB has safety-related monitoring devices, but these devices are not credited for safe shutdown.

The applicant has evaluated the capability to achieve and maintain post-fire safe shutdown when offsite power is available and when it is not. For the ESBWR design, loss of offsite power in the event of a fire is more limiting than a fire with offsite power available. In accordance with the guidance in RG 1.189, the applicant assumed a LOOP for the bounding analysis for a fire in the MCR that warrants evacuation.

In RAI 9.5-78, the staff requested that the applicant revise DCD Tier 2, Section 9.5.1.12, to base the FHA on SSCs important to safety rather than safe shutdown, in conformance with RG 1.189. Similarly, in RAI 9.5-82, the staff requested that the applicant revise DCD Tier 2, Section 9.5.1.15.6, to base the program to control combustibles, hazardous materials, and ignition sources on SSCs important to safety rather than safe shutdown, in conformance with RG 1.189. GDC 3 requires that the FPP provide protection for SSCs important to safety. In responses, the applicant clarified that the SSCs that meet the definition of important to safety in RG 1.189 are safety-related in the ESBWR design. The applicant also clarified how the ESBWR design conforms to RG 1.189, Regulatory Position 6.2, for nonsafety equipment necessary to achieve and maintain stable shutdown. The applicant made corresponding changes to DCD Revision 6. The staff finds that the response is acceptable since the applicant implemented the guidelines for “important to safety” in RG 1.189 to meet GDC 3 for the FHA and the program to control combustibles, hazardous materials and ignition sources on SSCs. Based on the above, the applicant’s responses, and DCD changes, RAIs 9.5-78 and 9.5-82 are resolved.

In RAI 9.5-87, the staff requested that the applicant correct an apparent contradiction within DCD Tier 2, Revision 5, Table 9A.5-6, concerning Fire Areas F5201 and F5204. DCD Tier 2, Table 9A.5-6, stated that both fire areas contain safety-related divisional equipment or cables for all four divisions, but the safe-shutdown evaluations stated that a fire in the area affects no safety-related equipment. In response, the applicant agreed to change the wording in the safe-shutdown evaluation in DCD Tier 2, Table 9A.5-6, by removing the comment that a complete burnout of all equipment in these areas affects “no safety-related equipment.” The staff finds that the response is acceptable since the applicant addressed the contradiction in DCD Tier 2, Table 9A.5-6, and clarified the impact on safety-related equipment. Based on the above, the applicant’s responses, and DCD changes, RAI 9.5-87 is resolved.

Based on the above, the staff finds that the ESBWR FHA conforms to the guidelines of SRP Section 9.5.1 and RG 1.189.

Passive Fire Protection, Detection and Suppression Features

DCD Tier 2, Revision 9, Section 9.5.1 describes the materials of construction as they relate to the FPP. Within the safety-related structures, interior walls, partitions, structural components, materials for insulation, and radiation shielding are either noncombustible or have low ratings for fire contribution. The flame spread and smoke development rating of these materials is 25 or less. Surface finishes are specified to have a flame-spread, fuel-contributed, and smoke-evolved index of 25 or less (Class A) as determined by ASTM E84, "Standard Test Method for Surface Burning Characteristics of Building Materials" (NFPA 255, "Standard Method of Test of Surface Burning Characteristics of Building Materials").

Exposed structural steel protecting safety-related areas is fireproofed with material with a fire rating of up to 3 hours as determined from the FHA. The fireproofing of structural steel members, where required by calculation based on combustible loading, is accomplished by application of an Underwriters Laboratories, Inc. (UL)-listed or Factory Mutual (FM)-approved cementitious or ablative material or by a UL-listed or FM-approved boxing design.

Based on the above, the staff that finds the materials of construction conforms to the guidelines of SRP Section 9.5.1 and RG 1.189.

DCD Tier 2, Revision 9, Section 9.5.1.10, describes the ESBWR FPP fire barriers.

Fire barriers of 3-hour fire resistance rating separate the following:

- Safety-related systems from any potential fires in nonsafety-related areas that could affect the ability of safety-related systems to perform their safety function

- Redundant divisions or trains of safety-related systems from each other to prevent damage that could adversely affect a safe-shutdown function from a single fire

- Components within a single safety-related electrical division that present a fire hazard to components in another safety-related division

- Electrical circuits, both safety-related and nonsafety-related, whose fire-induced failure could cause a spurious actuation that could adversely affect a safe-shutdown function

Three-hour-rated fire barriers separate safety-related equipment on a divisional basis, except equipment mounted in the control room or containment, and equipment covered by special cases that are discussed in DCD Tier 2, Revision 9, Section 9A.6.

The fire barriers in safety-related areas of buildings are seismic Category I. Penetrations through fire barriers are sealed or closed to provide fire resistance ratings at least equal to that of the barrier. Only noncombustible materials qualified in accordance with ASTM E-119, "Standard Test Methods for Fire Tests of Building Construction and Materials," are used for construction of fire barriers. Openings in fire barriers or firewalls are equipped with fire doors, frames, and hardware qualified by fire endurance testing to a fire resistance rating, as required by the applicable codes, up to the same fire resistance rating of the fire barrier itself. There are also some doors that provide fire area separation that may not that have been qualified as fire doors by tests but do provide equivalent protection. Typically, these are the doors for the personnel air lock into the reactor containment and the missile/tornado doors at the equipment access entrance to the RB. (The term "doors," when used in the FHA, includes doors, frames,

and hardware.) Elevator doors are 1.5-hour fire rated in 3-hour fire-rated barriers. Access stairwells are enclosed in minimum 2-hour-rated firewalls and equipped with self-closing fire-rated doors. Fire dampers protect ventilation duct openings in fire barriers as required by NFPA 90A, "Standard for Installation of Air-Conditioning and Ventilating Systems."

Electrical cable fire-stops are tested to demonstrate a fire rating equal to the rating of the barrier they penetrate. As a minimum, the penetrations meet the guidance of NUREG–1552, "Fire Barrier Penetration Seals in Nuclear Power Plants," issued July 1996, including Supplement 1, issued January 1999. The documented test results for the acceptable fire-stops will be part of the plant design records.

The COL applicant will provide specific design and certification testing details for fire barriers and electrical raceway fire barrier systems in accordance with the applicable sections of NFPA 251, "Standard Method of Tests of Fire Endurance of Building Construction and Materials"; ASTM E119; and the guidance in RG 1.189. DCD Tier 2, Revision 9, Sections 9.5.1.11 and 9.5.1.16 identify this as COL information item 9.5.1-5-A.

For the reason set forth above, the staff finds that the ESBWR fire barriers conform to the guidelines of SRP Section 9.5.1 and RG 1.189.

DCD Tier 2, Revision 9, Section 9.5.1, describes the ESBWR suppression and detection systems. Equipment arrangements and the combustible loading in each area determine the type of fire suppression provided and the areas protected. The ESBWR design provides automatic sprinkler systems for areas in which either installed combustible loading is large enough to warrant the installation or a significant transient combustible loading is most likely to occur as a result of combustibles introduced by normal maintenance operations. The FHA is based on the introduction of transient combustibles to any area of the plant, subject to administrative controls. Fixed automatic fire suppression systems are installed in areas identified by the FHA as having a high fire hazard rating. Electrical areas that exceed a combustible loading of 1,400 MJ/m^2 (123,600 BTU/ft^2) and all other indoor areas with a combustible loading in excess of 700 MJ/m^2 (61,800 BTU/ft^2) warrant automatic fire suppression.

The plant design provides building standpipes and hose stations in major buildings. The sprinkler systems supply lines and the hose station standpipes supply lines have different connections to the fire water main, which are separated by an isolation valve in the fire main. Therefore, no single failure can impair both systems. Portable fire extinguishers are strategically located throughout the plant in accordance with NFPA 10, "Standard for Portable Fire Extinguishers," except in highly radioactive areas. The plant design also provides an automatic fire detection, alarm, supervisory control, and indication system in selected plant areas, as provided by the FHA. Portable fire detection equipment is for use inside primary containment during maintenance outages when the space is not inerted.

Each fire suppression system automatically actuated by a fire detection system has the control logic and capability for manual actuation available at the local fire alarm panel for the protected area. Remote manual actuation of these suppression systems is also available from the MCR. Dedicated data links transmit command and status information to and from the local fire alarm panels and the main fire alarm panel in the MCR.

DCD Tier 2, Revision 9, Section 9.5.1.2, states that the type of fire suppression is based on the combustible loading and the extent of safe-shutdown equipment within a fire area. GDC 3

requires that the FPP provide protection for SSCs important to safety. Safe-shutdown equipment is a subset of equipment important to safety. In RAI 9.5-74, the staff requested that the applicant change its basis from safe-shutdown equipment to equipment important to safety. In response, the applicant clarified that the SSCs that meet the definition of important to safety in RG 1.189 are designated as safety-related in the ESBWR design. The applicant also clarified how the ESBWR design conforms to RG 1.189, Regulatory Position 6.2, for nonsafety equipment necessary to achieve and maintain stable shutdown. The applicant made corresponding changes to DCD Revision 6. The staff finds that the response is acceptable since the applicant implemented the guidelines for important to safety in RG 1.189 to meet GDC 3 for the suppression systems. Based on the above, the applicant's responses, and DCD changes, RAI 9.5-74 is resolved.

Based on the above, the staff finds that the ESBWR fire suppression measures conform to the guidelines of SRP Section 9.5.1 and RG 1.189.

DCD Tier 2, Revision 9, Section 9.5.1.9 describes the ESBWR detection and alarm systems, which include standpipes and hose stations.

Instrumentation for the fire detection system provides signals for early detection and warning of fires. In accordance with NFPA 72, "National Fire Alarm Code," local fire alarm panels supervise fire and smoke detectors. The local fire alarm panels are in turn connected to the main fire alarm panel via a dedicated data link. Signals transmitted include detector status (normal, alarm, supervisory, and trouble) as well as local fire alarm panel status. Upon receipt of a signal from any of the area fire detectors, alarms and visual indications are activated at the main fire alarm panel in the MCR and at the local fire alarm panel. Instrumentation for fire detection is either FM-approved or UL-listed, where available.

Smoke detectors installed in rooms containing safety-related equipment, except primary containment, and in areas containing significant amounts of combustible materials as determined by the FHA, provide early detection and warning of fires. At least two detectors are installed in any single room containing safety-related equipment. All fire and smoke detection circuits have electrical supervision to detect circuit breaks, ground faults, and power failures. The design of the detector circuits is such that the failure, removal, or replacement of a detector does not affect the performance of the fire detection loop.

In RAI 9.5-77, the staff requested that the applicant revise DCD Tier 2, Section 9.5.1.9, to base the detection and alarm system coverage on equipment important to safety rather than safe shutdown, in conformance with RG 1.189. GDC 3 requires that the FPP provide protection for SSCs important to safety. In response, the applicant clarified that the SSCs that meet the definition of important to safety in RG 1.189 are designated as safety-related in the ESBWR design. The applicant also clarified how the ESBWR design conforms to RG 1.189, Regulatory Position 6.2, for nonsafety equipment necessary to achieve and maintain stable shutdown. The applicant made corresponding changes to DCD Revision 6. The staff finds that the response is acceptable since the applicant implemented the guidelines for important to safety in RG 1.189 to meet GDC 3 for the detection and alarm system coverage. Based on the above, the applicant's responses, and DCD changes, RAI 9.5-77 is resolved.

In RAI 9.5-84, the staff requested that the applicant clarify the location of the manual fire alarm pull boxes for the ancillary diesel building. In response, the applicant clarified that manual fire alarm pull boxes are installed at each building exit of the ancillary diesel building. The applicant revised DCD Tier 2, Section 9A4.10 to state that manual fire alarm pull boxes will be located at

each building exit. The staff finds that the response is acceptable since the applicant clarified the location of the manual fire alarm pull boxes for the ancillary diesel building. Based on the above, the applicant's responses, and DCD changes, RAI 9.5-84 is resolved.

Based on the above, the staff finds that the ESBWR detection and alarm systems conform to the guidelines of SRP Section 9.5.1 and RG 1.189.

DCD Tier 2, Revision 9, Section 9.5.1.4 describes the ESBWR water supply, fire pumps, and fire water piping. Water for the FPS must come from a minimum of two reliable sources. The primary source will be two dedicated, seismic Category I fire water storage tanks. Each source has sufficient capacity to meet the maximum fire water demand of the system for 120 minutes. The secondary source may be a second fire water storage tank, a cooling tower water basin, or a large body of water with the capacity to meet the total water demand for at least 120 minutes. Water sources that are used for multiple purposes ensure that the required quantity of fire water is dedicated for fire protection purposes. The COL applicant will provide information on the final quantity and capacity of secondary fire water storage. DCD Tier 2, Revision 9, Sections 9.5.1.4 and 9.5.1.16, identify this as COL information item 9.5.1-1-A.

The primary seismic Category I fire water storage tanks provide the required emergency makeup water volume for the ICS/PCCS pools and SFP to the FAPCS following the design-basis LOCA. The primary source of fire water has a minimum capacity of 3,900 m^3 (1,030,000 gallons). The secondary source has a minimum capacity of 2,082 m^3 (550,000 gallons) dedicated for fire protection use.

The ESBWR design provides two primary nuclear island fire pumps. The lead primary fire pump is motor driven, and the backup is a seismic Category I diesel-driven fire pump. The backup diesel-driven fire pump provides fire water in the event of failure of the motor-driven fire pump or LOPP. In addition, the ESBWR provides for two nonseismic secondary fire pumps. The lead secondary fire pump is motor driven, and the backup secondary fire pump is diesel driven.

Each of the fire water pumps is rated at 454.2 m^3/h (2,000 gpm) and provides 100 percent of the fire water demand to the worst-case fire within the nuclear island (RB, FB, and CB) or 50 percent of the fire water demand to the worst-case fire within the balance of the plant. The largest fire water demand is 967 m^3/h (4,256 gpm) for a design-basis TB fire, including hose streams. All fire pumps are capable of delivering the flow and pressure required to the location that is farthest from the fire water supply source.

The fuel oil tanks for the diesel-driven fire pumps have a capacity sufficient to allow operation of the diesel engines for approximately 96 hours before refilling, based on the fuel consumption and margin criteria provided in NFPA 24, "Standard for the Installation of Private Fire Service Mains and Their Appurtenances."

Based on the above, the staff finds that the ESBWR water supply and fire pump designs conform to the guidelines of SRP Section 9.5.1 and RG 1.189.

The fire water supply piping consists of a buried, nonseismic yard main loop and a suspended, seismic Category I nuclear island piping loop constructed to the standard of ASME Power Piping Code B31.1. The seismic Category I loop is designed to remain functional following an SSE. The primary fire pumps supply fire water to the seismic Category I loop that supplies fire water within the structures of the nuclear island. The secondary fire pumps supply fire water

directly to the yard main loop, in accordance with NFPA 24. Isolation valves are located between the buried, nonseismic yard piping loop and the suspended, ASME Power Piping Code B31.1, seismic Category I piping loop.

The COL applicant will determine the design characteristics of the yard main loop piping. DCD Tier 2, Revision 9, Section 9.5.1.5, identifies this as COL information item 9A.7-1-A. Locked open sectionalizing postindicator valves installed in the fire yard loop permit isolation of any part of the loop without completely removing the system from service. Fire hydrants located at approximately 76.2 m (250 ft) intervals along the fire main loop provide fire fighting capability, especially near areas or buildings containing combustible materials. The fire hydrants are generally located no closer than 12.2 m (40 ft) from the protected buildings and are safeguarded from vehicular traffic.

Fire suppression system piping in the RB, CB, and FB is designed and installed to withstand an SSE and remain operational. Fire suppression system piping in the TB, RW, and EB is designed and installed to meet the seismic requirements of NFPA 13, “Standard for Installation of Sprinkler Systems.” The COL applicant will provide FPS P&IDs showing complete site-specific system design. DCD Tier 2, Revision 9, Sections 9.5.1.5 and 9.5.1.16, identify this as COL information item 9.5.1-4-A.

Based on the above, the staff finds that the ESBWR firewater piping design conforms to the guidelines of SRP Section 9.5.1 and RG 1.189.

DCD Tier 2, Revision 9, Section 9.5.1, describes the ESBWR manual suppression system, which includes standpipes and hose stations. The wet standpipes and hose stations are designed to NFPA 14, “Standard for Standpipe and Hose Systems,” Class III service. Each hose rack has 30.5 m (100 ft) of 40 mm (1.5 in.) lined fire hose. The water supply pressure maintains a gauge pressure of 448.2 kilopascals gauge (kPaG) (65 psig) at the most hydraulically remote 40 mm (1.5 in.) hose station and 689 kPaG (100 psig) at the most hydraulically remote 65 mm (2.5 in.) hose station. If the gauge pressure at a 40 mm (1.5 in.) hose station exceeds 689 kPaG (100 psig), orifice discs installed in the hose couplings reduce the reaction force at the hose end. For areas containing equipment for safe shutdown, standpipes and hose connections for manual fire fighting remain functional following an SSE to provide at least two working standpipes and two hose stations. The piping system serving these hose stations is analyzed for SSE loading and satisfies ASME Power Piping Code B31.1 requirements.

All rooms within the plant buildings are within the reach of at least one effective hose stream from a Class III hose station. Effective hose streams from two separate hose stations cover each room containing equipment required for safe shutdown that is not protected by a fixed fire suppression system. The need for coverage from two hose stations is also based on the fire hazard present. Hose stations for manual fire fighting inside containment are located outside the containment near access openings to provide complete coverage of the accessible areas inside containment. During normal plant power operation, the containment atmosphere is inerted and cannot sustain a fire.

In RAI 9.5-73, the staff requested that the applicant revise DCD Tier 1, Figure 2.16.3-1, and DCD Tier 2, Figure 9.5.1, to indicate that fire water supply is available at the CB hose stations by opening the hose valve at each station. In response, the applicant revised both figures to clarify that the closed valves represent a typical hose station valve by adding a note to the figures indicating that these valves represent a typical hose station valve. The staff finds that

the response is acceptable since the applicant clarified the hose stations in the CB. Based on the above, the applicant's responses, and DCD changes, RAI 9.5-73 is resolved.

Based on the above, the staff finds that the ESBWR manual fire suppression system design conforms to the guidelines of SRP Section 9.5.1 and RG 1.189.

Protection of Safe-shutdown Capability

DCD Tier 2, Revision 9, Section 9.5.1, describes the ESBWR fire protection for circuits and cables. Safety-related raceway and circuit routing comply with BTP Plant Systems Branch (SPLB) 9.5-1, "Guidelines For Fire Protection For Nuclear Power Plants," (attached to SRP Section 9.5.1, Revision 4) except that they are separated by fire barriers rather than distance outside the MCR and primary containment. Control, power, or instrument cables and equipment of redundant systems used for bringing the reactor to hot shutdown and maintaining safe shutdown are separated from each other by 3-hour-rated fire barriers, except within the MCR and containment and where the equipment of more than one division is required to be located within a single fire area.

Where multiple divisions of cable or equipment are located in the same fire area, the configurations are evaluated and justified as acceptable on an individual basis. The acceptance criterion is that a single fire cannot degrade the performance of more than one division of safe-shutdown equipment controlled from the MCR. All electrical cables (safety-related and nonsafety-related) conform to IEEE Standard 1202-2006, "Standard for Flame-Propagation Testing of Wire and Cable," flame test criteria. The raceway design avoids the use of electrical raceway fire barrier systems for the ESBWR, relying instead on divisional separation by fire area and structural fire barriers. As described below, the staff finds the ESBWR evaluations of locations where multiple divisions of cables or equipment are in the same area to be acceptable.

Cables for local indication are included in the safe-shutdown analysis where failure of the cable could cause failure of functionally associated circuits or where relied upon to provide either diagnostic or process parameter information for recovery.

For specific areas and components where fire barrier separation is not feasible, ESBWR design features provide reasonable assurance that post-fire safe shutdown can be achieved and maintained long term as follows:

- Fire-induced failure of reactor protection system scram circuits is limited to the loss of power to the scram solenoids and can cause a half-scram or scram condition, which is a fail-safe condition.

- Fire-induced failure of the MSIV sensors and cabling in the TB results in automatic closure of the MSIVs.

- Fire-induced failure of main steam line tunnel area radiation monitoring will cause a trip. Leak detection temperature monitors in the main steam line tunnel area will cause an MSIV closure on elevated temperature due to a fire in the area.

- Main steam line automatic DPV actuation solenoids and control circuits are located in the normally inerted containment. The cabling is contained in conduit and physically separated to the extent possible. The area has a low fire loading and is inaccessible during plant operation. A fire inside the solenoid coil compartment of one pilot does not influence the coil

or cable of the redundant pilot. Electrical arcing damage to a cable or solenoid coil cannot result in inadvertent opening of the main valve because shorts, open circuits, or grounds at the solenoid cannot cause the solenoid to energize. Short circuits at this location cannot jeopardize Class 1E power supplies because resistance is sufficient to permit appropriate circuit protection coordination.

- Redundant valves perform the main steam line isolation function. One valve and its control and protection cabling for each main steam line is located outside the primary containment, and one valve with its cabling is located inside the normally inerted drywell. Consequently, a single fire cannot affect the capability to cause a scram or isolate the main steam lines.

- Cabling for electrical circuits located under the reactor vessel is protected from fire by the inerted atmosphere of the containment during operation and by segregating divisions via separate metal conduits. During operation there will be no combustible materials in this area other than the cable insulation inside metal conduit.

- Some areas contain more than one division of instrumentation needed to isolate redundant sets of isolation valves, either for HVAC purposes or for some other purpose warranting redundancy. The divisional safety-related panels in these areas are generally designed and located to serve a single division.

- Multidivisional panels and racks are located in divisional compartments with physical separation between divisions. The incoming cables for each division are in separate conduit and, where possible, the conduit is embedded in concrete.

- Loss or spurious actuation of leak detection instrumentation inside containment as a result of a fire does not affect safe shutdown.

- Spurious operation or failure of the SLCS does not affect safe shutdown.

- Loss of RB operating deck radiation monitors as a result of a fire does not affect plant safety.

- In accordance with an ESBWR design provision, cables for outboard containment isolation valves located in fire areas of a division different from that of the valve are not routed through fire areas containing any circuitry associated with the inboard valve of the isolation pair.

- The postulated MCR fire assumes loss of all component functions within the MCR, and the analysis considers spurious actuations. The safety system and logic control system automatically actuate the safety systems, and operators can control nonsafety-related systems from either of the two RSS panels located in separate fire areas.

- Complete burnout of all safety-related devices and their cables in the TB does not affect the ability to achieve and maintain post-fire safe shutdown.

- Complete burnout of all equipment and cables within any of the four HCU rooms in the RB (each HCU room is a separate fire area) results in loss of one redundant train and one division of safe-shutdown equipment and circuits, as well as loss of redundant Division I and II HCU solenoid circuits. However, if HCUs are unavailable for reactor scram, plant operators can use either the FMCRD portion of the CRD system or the SLCS to scram the

reactor (components and circuits for either are located outside the fire area). For other systems in each HCU room, the remaining three divisions of safe shutdown and redundant train are unaffected by fire and are operable. The automatic logic control scheme (any two-out-of-four redundant signals) remains operable.

In RAI 9.5-71 and its supplements, the staff requested that the applicant describe how the ESBWR design specifically prevents or mitigates spurious actuations that could prevent safe shutdown because of the effects of fire, including smoke, and that the applicant include these design features in the DCD. In various responses, the applicant provided (in DCD Tier 2, Revision 6, Sections 7.1.3.2, 7.1.5.3, and 9.5.1.10) additional description and clarification of the design features that prevent or mitigate spurious actuations. The staff finds that the responses are acceptable because the applicant described specific features that prevent spurious actuations in its discussion of fire barriers and of the ESBWR instrumentation and control systems. Based on the above, the applicant's responses, and DCD changes, RAI 9.5-71 is resolved.

In RAI 9.5-92, the staff requested that the applicant add the wording from its response to RAI 19.1-173 to DCD Tier 2, Section 9A.2.4, indicating that fire induced multiple spurious actuations would be assumed to occur simultaneously or in rapid succession. The staff also requested that the applicant clarify that the final post-fire safe-shutdown circuit analysis for the as-built and as-purchased plant, including circuit routing, will be performed using an approach similar to the one described in the industry guidance document for circuit analysis, Nuclear Energy Institute (NEI) 00-01, "Guidance for Post-Fire Safe Shutdown Circuit Analysis". In response, the applicant added wording to DCD Tier 2, Section 9A.2.4, stating that (1) the post-fire safe-shutdown circuit analysis will assume that any spurious actuations associated with a postulated fire occur simultaneously or in rapid succession, and (2) circuit routing will conform to the methodology provided in Revision 1 of NEI 00-01, "Guidance for Post-Fire Safe Shutdown Circuit Analysis," in accordance with RIS 2005-030, "Clarification of Post-Fire Safe-Shutdown Circuit Regulatory Requirements," dated December 20, 2005. The staff finds that the response is acceptable because the applicant clarified the FHA acceptance criteria in DCD Tier 2, Section 9A.2.4, as requested by the staff. Based on the above, the applicant's responses, and DCD changes, RAI 9.5-92 is resolved.

Based on the above, the staff finds that the ESBWR fire protection for circuits and cables conforms to the guidelines of SRP Section 9.5.1 and RG 1.189.

DCD Tier 2, Revision 9, Section 9.5.1, describes the ESBWR post-fire operator actions. The only operator action credited in the ESBWR post-fire safe-shutdown analysis is manual scram of the reactor before evacuation from the MCR in the event of a fire in the MCR that requires evacuation. According to the applicant's response to RAI 15.5-4, after the operator regains control at the remote shutdown panel, manual action may be necessary to control the ICS to ensure that the maximum cooldown rate does not exceed 55.6 degrees C per hour (100 degrees F per hour). Because the controls at the remote shutdown panel are identical to those in the MCR, the operator can fully control the ICS from the remote shutdown panel as in the MCR. Therefore, operator action is kept to a minimum for ESBWR post-fire safe shutdown, which is in accordance with NRC guidance.

Based on the above, the staff finds that the ESBWR post-fire operator actions conform to the guidelines of SRP Section 9.5.1 and RG 1.189.

As discussed above, the staff finds that the ESBWR design provides adequate protection of safe-shutdown capability in the event of a fire.

Miscellaneous

DCD Tier 2, Revision 9, Section 9.5.1, describes the additional design features that support the ESBWR FPP. Charcoal filters in the off-gas and ventilation systems of the plant have fire protection water spray systems that are not normally connected to the fire water supply system. The water flows to the charcoal by means of fixed piping terminating at the exterior of the equipment assembly with manual shutoff valves. In the event of charcoal ignition, plant operators can connect the piping to the fire water supply system through a standard hose or jumper fitting.

Plant drainage systems are designed to accommodate the maximum anticipated normal volumes of liquid, including such inputs as the anticipated water flow from a fire hose and other fire suppression water discharges to the area floor drains, without overflowing and without impacting the safety function of any safety-related component or system.

Direct current switchgear and inverters are not located in battery rooms where hydrogen may potentially accumulate. The battery rooms contain only batteries and eye wash stations. Failure of the battery room exhaust fans is alarmed in the MCR.

Spill control is provided to contain the contents of any above-grade oil-filled vessel or tank larger than 208.2 liters (l) (55 gallons) and all tanks containing chemicals used in water and wastewater treatment or quality control. In accordance with the guidance in RG 1.189, the ESBWR design provides spill containment and drainage facilities for a given area based on the following:

- The spill of the largest single container of any flammable or combustible liquids in the area
- Where automatic suppression is provided throughout, the credible volume of discharge (as determined by the FHA) for the suppression systems operating for a period of 30 minutes
- Where automatic suppression in not provided throughout, the contents of piping systems and containers that are subject to failure in a fire
- Where the installation is outside, credible environmental factors such as rain and snow
- Where automatic suppression is not provided throughout, a volume based on a manual fire-fighting flow rate of 1,892.5 l/min (500 gpm) for a duration of 30 minutes, unless the FHA demonstrates a different flow rate and duration.

In RAI 9.5-93 and its supplement, the staff requested that the applicant clarify how the ESBWR fire brigade communication systems conform to the guidelines of RG 1.189, Regulatory Position 4.1.7. In response, the applicant stated that the DCD will direct the COL applicants to describe in full the fire brigade communication systems, including portable radio/wireless and fixed emergency communication systems. COL Information Item 9.5.2.5.5-A, “Fire Brigade Radio System,” states, in part “the COL applicant will describe the Fire Brigade Radio System in accordance with RG 1.189, Position 4.1.7.” The staff finds that the response is acceptable because the fire brigade communication systems are site specific and the applicant committed

to conform to the guidelines of RG 1.189, Regulatory Position 4.1.7. Based on the above, the applicant's responses, and DCD changes, RAI 9.5-93 is resolved.

Based on the above, the staff finds that these ESBWR design features conform to the guidelines of SRP Section 9.5.1 and RG 1.189.

Enhanced Fire Protection Criteria

The staff reviewed the ESBWR FPP with the guidelines of SECY-90-016, SECY-93-087, and SECY-94-084, which provide enhanced fire protection criteria for advanced reactor designs.

New reactor designs should ensure that safe shutdown can be achieved, assuming that all equipment in any one fire area will be rendered inoperable by fire and that re-entry into the fire area for repairs and operator actions is not possible. Because of its physical configuration, the control room is excluded from this approach, provided that an independent alternative shutdown capability that is physically and electrically independent of the control room is included in the design. New reactor designs should provide fire protection for redundant shutdown systems in the reactor containment building that will ensure, to the extent practicable, that one shutdown division will be free of fire damage. Additionally, new reactor designs should ensure that smoke, hot gases, or the fire suppressant will not migrate into other fire areas to the extent that they could adversely affect safe-shutdown capabilities, including operator actions. These criteria are specific to plants with active safety-related systems, but within the constraints of the active-to-passive design differences; the ESBWR design meets these criteria. The ESBWR FPP design bases include provisions to maintain the ability to safely shut down the reactor and keep it shut down during all modes of plant operation by providing adequate separation of safety-related equipment.

Fire protection for redundant shutdown systems in the reactor containment building, where it is not practicable to separate redundant trains by physical barriers, is provided by inerting the containment atmosphere during operation to preclude the initiation or propagation of a fire, minimizing exposed combustible materials, and separating redundant safety-related trains by as much distance as possible. An RSS physically and electrically independent of the MCR, ensures safe-shutdown capability in the event of a fire that requires evacuation of the MCR.

Safe shutdown is achieved primarily through the ICS. This is a system employed for both hot standby and long-term core cooling modes. It can operate at full RCS pressure and is thereby able to place the reactor in the long-term cooling mode immediately after reactor shutdown. Operation of the plant in the long-term cooling mode is automatic. The system does not depend on any ac power or other support systems such as cooling water. Operation does not involve any pumps or valve operation once initial alignment is established. The system initiation is based on a two-out-of-four logic. Actuation still occurs with one division failed as a result of a fire.

The ESBWR systems credited to achieve and maintain safe shutdown in the event of a fire are as follows:

- ICS
- GDCS
- ADS
- PCCS
- Associated controls and instrumentation

The FPS is designed to prevent inadvertent operation of fire suppression systems from jeopardizing the capability to achieve safe shutdown and to preclude damage to plant safety-related SSCs in the event of an earthquake. All fire protection detection, alarm, and suppression systems meet the requirements of the appropriate NFPA fire codes, where applicable, to the maximum extent practicable. Based on the above, the staff finds that the ESBWR program meets the enhanced fire protection criteria for advanced reactor designs.

Exceptions to the Standard Review Plan and RG 1.189

DCD Tier 2, Revision 9, Sections 9.5.1.12 and 9A.6, describe specific exceptions and alternatives to the NRC acceptance criteria for FPPs. These sections describe and justify, in detail, each of the plant configurations and designs that deviate from the NRC acceptance criteria for FPPs. As described below, the staff has reviewed each of the exceptions and alternative approaches and their justifications described in DCD Tier 2, Sections 9.5.1.12 and 9A.6 and finds them acceptable.

- Individual electrical cabinets and consoles in the MCR complex will not have installed smoke detectors inside the enclosures as recommended by Section 6.1.2.2 of RG 1.189. In the ESBWR design, the electrical cabinets in the MCR are air-cooled and vent to the MCR, where a smoke detection system is provided throughout the area. The MCR is constantly occupied, and portable extinguishers and manual hose stations are readily available for extinguishing a fire. A fire in any single cabinet or console does not disable the capability to safely shut down the plant. The DCD states that this alternative approach will be used unless the FHA identified it as a significant fire hazard.

- Rooms adjacent to the MCR will not have installed automatic fire suppression systems, as recommended by Section 6.1.2 of RG 1.189. In the ESBWR design, these rooms are a low-risk fire area. They do not contain any high or medium-voltage equipment or cabling. Interior finishing materials are noncombustible or have a flame spread and smoke developed rating of 25 or less. The rooms will have smoke detection capabilities, and the MCR is constantly occupied. Portable extinguishers and manual hose stations are readily available for extinguishing a fire. The DCD states that this alternative approach will be used unless the FHA identified it as a significant fire hazard.

- The area below the raised floor in the MCR will not have installed automatic fire suppression as recommended by Section 6.1.2.1 of RG 1.189. In the ESBWR design the MCR complex and subfloor volume is considered to be a low risk fire area, because of the lack of high- or medium-voltage equipment or cabling. The area below the raised floor will have a smoke detection system throughout. The characteristics of the subfloor cabling are such that the probability of a fire ignition is very low and any fire that occurred would be self-extinguishing. The raised floor consists of noncombustible sectional panels that can be individually removed to provide fire-fighting access to a subfloor fire. The MCR is constantly occupied, and portable extinguishers and manual hose stations are readily available for extinguishing a fire. The DCD states that this alternative approach will be used unless the FHA identified it as a significant fire hazard.

- The SDG indoor fuel oil day tanks will likely exceed the limit recommended by Section 6.1.8 of RG 1.189 for indoor SDG day tanks. However, the SDGs of the ESBWR are nonsafety-related and are not relied upon to maintain safe-shutdown conditions for the 72-hour period following a fire event. In addition, the passive fire protection and active fire suppression provided for these tanks justify exceeding the recommended tank size.

- The main ADG fuel oil tank capacity will exceed the limit recommended by Section 6.1.8 of RG 1.189 for indoor diesel generator day tanks. The capacity of each of the ADG day tanks will not exceed 4,164 l (1,100 gallons); however, the main fuel oil storage tanks for these diesels will exceed this capacity. Neither ADG is necessary to achieve and maintain safe-shutdown conditions for the 72-hour period following an accident or fire event. Each fuel oil storage tank is located in the auxiliary diesel building in a dedicated 3-hour fire rated compartment. There is no equipment important to safety located in the same building as the fuel oil tank rooms. The passive fire protection and active fire suppression provided for these tanks justify exceeding the recommended tank size.

- The water-based automatic fixed suppression systems in each SDG and ADG room are not designed to ensure continued operation of the DGs in the event of system discharge, as recommended by Section 6.1.8 of RG 1.189. The ESBWR design includes two independent and physically separated nonsafety-related SDGs, either of which is capable of providing the full electrical load for the redundant nonsafety-related electrical buses. The ESBWR design also includes two independent and physically separated nonsafety-related ADGs, either of which is capable of providing redundant post-accident power. None of these diesel generators is necessary to achieve and maintain safe-shutdown conditions for the 72-hour period following an accident or fire event. Because the DGs are not safety-related and are not required to maintain safe-shutdown conditions for the 72-hour period following a fire event; and the suppression system is a preaction type; the exception to the recommended automatic fire suppression design is justified.

- ESBWR computer rooms that contain safety-related equipment do not have fixed automatic fire suppression protection, as recommended by Section 6.1.4 of RG 1.189. The computer rooms are considered to be low-risk fire areas because of the lack of high- or medium-voltage equipment or cabling. Interior finishing materials are noncombustible. The rooms will have smoke detection capabilities, and the MCR is constantly occupied. Portable extinguishers and manual hose stations are readily available for extinguishing a fire. Papers within computer rooms are stored in file cabinets, bookcases, or other storage locations except when in use. Outside the MCR complex, safety-related computers are located in divisional rooms separated from each other by 3-hour fire-rated barriers such that a single fire does not affect computer equipment from multiple divisions. The reduced combustible loadings, the manual firefighting capabilities, and divisional separation justify the exception to the fixed automatic fire suppression protection.

- The ESBWR design exceeds the maximum hose length to reach safety-related equipment in containment, as recommended by Section 6.1.1.2 of RG 1.189. Standpipes and hose stations external to containment and portable extinguishers provide protection during refueling and maintenance operations. Hose stations are located such that any location within containment can be reached by two effective hose streams with a maximum of 61 m (200 ft) of hose. The 30.5 m (100 ft) hose coverage recommendation cannot be met in containment for all areas with standpipes located outside containment. While at power, containment is inerted. The use of two hose streams justify exceeding the recommended hose lengths.

In RAIs 9.5-44, 9.5-45, and 9.5-46, and their supplements, the staff requested that the applicant provide COL information items for (1) a post-fire safe-shutdown circuit analysis, (2) the FHA for all areas of the plant that contain SSCs important to safety, and (3) the exceptions and alternative in DCD Tier 2, Sections 9.5.1.12 and 9A.6. RAIs 9.5-44, 9.5-45, and 9.5-46 were being tracked as open items in the SER with open items. In responses, the applicant stated that

the FHA cannot be completed because final cable and piping routing and other design details are not complete. In DCD Revision 6, the applicant revised the COL holder item to COL Information Item 9.5.1-7-A to state that the COL applicant will provide a milestone for confirming the assumptions of the FHA against the as-built conditions and updating the FHA as necessary. The staff finds that the response is acceptable, as augmented by the revised COL Information Item 9.5.1-7-A, because the COL information item addresses the FHA in a comprehensive way such that individual elements do not need to be identified. The COL information item conforms to RG 1.206, Part III, Section C.I.9.5.1, which acknowledges that some information may not be available at the time of the license application. Based on the above, the applicant's responses, and the subsequent DCD changes, RAIs 9.5-44, 9.5-45 and 9.5-46 are resolved.

Inspections, Tests, Analyses, and Acceptance Criteria

The DCD Tier 1, Revision 9, Section 2.16.3, identifies ITAAC to verify the design parameters of the FPS. Among the ITAAC included in the ESBWR design are inspections to verify that the 3-hour fire barriers protecting post-fire safe-shutdown systems and equipment are installed where required, that penetrations through the barriers are closed in accordance with the design of the barrier, that noncombustible materials qualified in accordance with ASTM E119 are used for construction of the fire barriers, and that fire dampers in ventilation duct openings meet NFPA 90A.

In RAI 14.3-396, the staff requested that the applicant in DCD Tier 1, Table 2.16.3-2, commit to verifying that hose station protection will be provided for locations outside containment that contain or could present a hazard to SSCs important to safety rather than safe shutdown in conformance with RG 1.189. GDC 3 requires that the FPP provide protection for SSCs important to safety. In response, the applicant clarified that the SSCs that meet the definition of important to safety in RG 1.189 are designated as safety-related in the ESBWR design. The applicant also clarified how the ESBWR design conforms to RG 1.189, Regulatory Position 6.2 for nonsafety equipment necessary to achieve and maintain stable shutdown. The applicant made corresponding changes to DCD Tier 1, Revision 6, Section 2.16.3. The staff finds that the response is acceptable because the applicant implemented the guidelines for important to safety in RG 1.189 to meet GDC 3 for the DCD Tier 1 verifications of the design. Based on the applicant's response and DCD changes, RAI 14.3-196 is resolved.

The staff reviewed the descriptive and other information provided in DCD Tier 1, Revision 9, Section 2.16.3, and finds that it conforms to the FPS and fire barriers design basis, as described in DCD Tier 2, Revision 9, Section 9.5.1. Accordingly, the staff finds that the FPS ITAAC complies with the requirements of 10 CFR 52.47(b)(1).

COL information items

DCD Tier 2, Revision 9, Sections 9.5.1.16 and 9A.7, list the following fire protection COL information items:

- 9.5.1-1-A, "Secondary Firewater Storage Source" - The COL Applicant will provide the capacity of the secondary firewater source (DCD Tier 2, Section 9.5.1.4).

- 9.5.1-2-A, "Secondary Firewater Capacity" - The COL Applicant shall provide documentation that the secondary fire protection pump circuit design will supply a minimum of 484 m3/hr (2,130 gpm) with sufficient discharge pressure to develop a minimum of 107 psig of line pressure at the TB and yard interface boundary (DCD Tier 2, Section 9.5.1.4).

- 9.5.1-4-A, “Piping and Instrument Diagrams” - The COL Applicant shall provide simplified FPS P&IDs showing complete site-specific systems (DCD Tier 2, Section 9.5.1.5).

- 9.5.1-5-A, “Fire Barriers” - The COL Applicant shall provide specific design and certification testing details for fire barriers and electrical raceway fire barrier systems in accordance with the applicable section of NFPA 251, “Standard Methods of Tests of Fire Resistance of Building Construction and Materials,” ASTM E119, and guidance in RG 1.189 (DCD Tier 2, Section 9.5.1.10).

- 9.5.1-6-A, “Smoke Control” - The COL Applicant shall include in its operating procedure development program procedures for manual smoke control by manual actions of the fire brigade for all plant areas in accordance with NFPA 804 guidelines (DCD Tier 2, Section 9.5.1.11).

- 9.5.1-7-A, “Fire Hazards Analysis (FHA) Compliance Review” - The COL Applicant shall provide a milestone for confirming the assumptions and requirements of the FHA against the as-buit conditions and updating the FHA as necessary (DCD Tier 2, Section 9.5.1.12).

- 9.5.1-8-A, “Fire Protection (FP) Program Description” - The COL Applicant shall provide a milestone for implementation of the applicant’s FPP (DCD Tier 2, Section 9.5.1.15).

- 9.5.1-10-A, “Fire Brigade” - The COL Applicant shall provide a milestone for implementing the provisions for manual fire-fighting capability for all plant areas (DCD Tier 2, Section 9.5.1.15.4).

- 9.5.1-11-A, “Quality Assurance” - The COL Applicant shall provide details of the QA program for the FPP (DCD Tier 2, Section 9.5.1.15.9).

- 9A.7-1-A, “Yard Fire Zone Drawings” - The COL applicant shall include fire zone drawings for those portions of the yard except for that associated with TB and EB equipment (DCD Tier 2, Section 9A.4.7).

- 9A.7-2-A, “Fire Hazards Analysis for Site Specific Areas” - A more detailed evaluation of the service water/water treatment building, service building and the yard area will be added during the COL application for a specific site (DCD Tier 2, Section 9A.4.7).

The COL applicant’s satisfactory completion and description of the action items identified above will provide the staff with sufficient information to assess the acceptability of the FPP for a COL, although the staff retains the discretion to issue RAIs in connection with the COL application. As described in RG 1.206, applicants should include the implementation milestones for programmatic aspects of the FPP in the COL within the license condition on operational program implementation. Accordingly, the staff finds the COL information items acceptable.

9.5.1.4 *Conclusion*

The staff finds that the applicant’s FPP design criteria are acceptable and meet the applicable requirements of 10 CFR Part 50 and 10 CFR Part 52, and conform to Commission policy contained in SECY-90-016, SECY-93-087, and SECY-94-084 (plants with passive safe-shutdown), as well as other applicable acceptance criteria. As described above, the staff finds that the applicant meets the guidelines of the applicable RGs and related industry standards.

The applicant has demonstrated that safe shutdown can be achieved, even assuming that all equipment in any one fire area (excluding the control room and reactor containment) will be rendered inoperable by fire and that re-entry into the fire area for repairs and operator actions is not possible. The applicant's design has provided an independent alternative shutdown capability that is physically and electrically independent of the control room. The applicant's design provides fire protection for redundant shutdown systems in the reactor containment building that will ensure, to the extent practicable, that one shutdown division will be free of fire damage. Additionally, the applicant's design ensures that smoke, hot gases, or fire suppressants will not migrate into other fire areas to an extent that could adversely affect safe-shutdown capabilities, including operator actions.

The applicant has demonstrated that SSCs important to safety are adequately protected from the effects of fires and explosions. The applicant's design uses noncombustible and heat-resistant materials whenever practical and provides fire detection, suppression, and fire fighting capabilities of appropriate capacity and capability to minimize the adverse effects of fire on SSCs important to safety.

The staff finds that ITAAC for the FPP provide reasonable assurance that the implementation of the FPP will be in accordance with the approved design and operational program descriptions, where applicable.

9.5.2 Communications Systems

9.5.2.1 *Regulatory Criteria*

The staff reviewed the communications systems based on the guidance provided in SRP Section 9.5.2, Revision 3. Acceptance criteria are based on meeting the relevant requirements of the following Commission regulations:

- Appendix E, "Emergency Planning and Preparedness for Production and Utilization Facilities," to 10 CFR Part 50, particularly part IV.E(9), as it relates to the provision of at least one onsite and one offsite communications system, each with a backup power source.
- 10 CFR 52.47(a)(8) and 50.34(f)(2)(xxv), as they relate to providing an onsite TSC (Three Mile Island (TMI) Action Plan Item III A.1.2).
- 10 CFR 50.47(a)(8), as it relates to equipment and facilities to support emergency response
- 10 CFR 50.55a
- GDC 1, as it relates to the quality of standards and records
- GDC 2, as it relates to the design basis for protection against natural phenomena
- GDC 3, as it relates to fire protection
- GDC 4, as it relates to environmental and missile design bases
- GDC 19, as it relates to the control room

- 10 CFR 73.45(e)(2)(iii), as it relates to communications subsystems and procedures to provide for notification to authority

- 10 CFR 73.45(g)(4)(i), as it relates to providing communications networks

- 10 CFR 73.46(f), as it relates to fixed site physical protection systems, subsystems, components, and procedures - communications subsystems

- 10 CFR 73.55(i), as it relates to detection and assessment systems

- 10 CFR 73.55(j), as it relates to communications requirements

- 10 CFR 52.47(b)(1), which requires that a design certification application contain the proposed ITAAC that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design certification is built and will operate in accordance with the design certification, the provisions of the Atomic Energy Act, and the NRC regulations

9.5.2.2 *Summary of Technical Information*

DCD Tier 2, Revision 9, Section 9.5.2, describes the ESBWR communications systems. The communications systems provide the means to conveniently and effectively communicate between various plant locations and with offsite locations during normal, maintenance, transient, fire, and accident conditions under maximum potential noise levels. The communications systems allow guards and watchmen on duty to maintain continuous communications with personnel in manned alarm stations, and offsite and onsite agencies as required by 10 CFR Part 73.55. This is accomplished by either private automatic branch exchange (PABX) or wireless communications systems. Communications equipment used with respiratory protection gear are designed and selected in accordance with EPRI Report NP-6559, “Voice Communication System Compatible with Respiratory Protection,” issued November 1989. The communications systems consist of the following systems:

- Plant page/party-line (PA/PL) system
- PABX system
- Plant sound-powered telephone system
- Plant radio system
- Evacuation alarm and remote warning system
- Emergency offsite communications system
- Completely independent communications system for security purposes.

The communications systems above are described in detail in DCD Tier 2, Revision 9, Section 9.5.2.2. Key features that address the regulations and other important notable features are summarized below.

The communications systems power generation design bases are as follows:

- Communications systems are independent of one another; therefore, a failure in one system does not degrade the performance of the other systems.

- The communications systems are in accordance with applicable codes and standards and the equipment is shielded, as necessary, from the adverse effects of electromagnetic interference (EMI) and radio frequency interference (RFI).

- The communications systems are functional during a LOOP.

Plant Page/Party-Line System

The PA/PL is a very flexible hard-wired intra-plant paging system with circuits wired in a ring topology to prevent loss of the system in the event of a single cable failure. This system is a multiple-channel, multiple-system-split-type design that permits simultaneous in-plant use of a page line and four party lines. One circuit of the handset station is connected to a telephone line permitting simultaneous broadcasting from a security telephone line. Each handset station can be used to communicate with any other station or the central station. The system is operated from a battery source with a normal and spare battery charger.

Private Automatic Branch Exchange System

The PABX is the plant multimode telephone system that is connected to the commercial telephone system and a licensee private network. The nodes for this system are located in separate communications rooms. Through this system the plant has normal and emergency offsite communication. Power is provided from plant nonsafety buses made up of independent batteries and chargers for each node. The battery capacity is approximately 8 hours with the loss of the ac power supply.

Plant Sound-Powered Telephone System

The plant sound-powered telephone system is independent of the PABX and the PA/PL systems. This system uses portable sound-power telephones that can plug into local terminal jacks wired back to a main communications patch board. The system allows uninterrupted private communications between the MCR and many plant areas. Different areas in the plant can communicate by linking their circuits at the patch board. The system does not rely on external power supply for operation.

Plant Radio System

The plant radio system is for normal and emergency communications within the plant. This radio system is independent of the PA/PL, PABX, and the sound-powered telephone system. This system consists of antennas distributed throughout the plant with a central re-broadcast transmitter and communications consoles located at selected plant locations including the MCR and the remote shutdown station. The system is designed to permit radio-to-radio and radio-to-console communications within the plant and surrounding plant buildings. Power for the base station and consoles is from the security system power supply that is backed by batteries and a standby generator. The radios are equipped with multiple channels including channels for: Operations, Maintenance, Management, Health Physics, Fire Brigade (optional), Crisis Management (or unassigned), and Emergency. By dialing through the PABX to a radiotelephone interconnect panel calls can be made between the telephone system and this in-plant radio system. The plant radio system has a channel for emergency use.

The Evacuation Alarm and Remote Warning System

The Evacuation Alarm and Remote Warning system consists of two parts. The Evacuation Alarm part consists of siren tone generators, public address speakers, and an outdoor siren to provide warning to personnel of emergency conditions. The remote warning part consists of a message storage device, microphone, remote broadcast speakers, and an output feedback monitoring system. Power is supplied from a nonsafety bus backed by a standby onsite ac power supply system and backed by the station batteries.

Emergency Communication System

The emergency communication system is provided by the public telephone lines and the licensee's network connected to the PABX and radio system. Emergency telephones are color-coded to distinguish them from normal telephones. The emergency communication system provides communication links that are considered site specific and addressed by COL information items. These include: 1) The Emergency Notification System (ENS), which provides a communications link with the NRC in accordance with Inspection & Enforcement (IE) Bulletin 80-15. (COL Information Item 9.5.2.5-1-A); 2) A Health Physics Network, which provides a communications link with NRC health physics personnel (COL Information Item 9.5.2.5-3-A); 3) A Ringdown Phone System, which provides a communications link with local and state agencies (COL Information Item 9.5.2.5-4-A); 4) A Crisis Management Radio System, which provides communications capability in accordance with the NUREG–0654 (COL Information Item 9.5.2.5-3-A); 5) A Fire Brigade Radio System in accordance with RG 1.189, Position 4.1.7 (COL 9 Item.5.2.5-5-A); and 6) A Transmission System Operator Communication Link (COL Information Item 9.5.2.5-2-A).

9.5.2.3 *Staff Evaluation*

The staff reviewed the design of the communications systems in accordance with SRP Section 9.5.2, Revision 3. DCD Tier 2, Revision 9, Table 1.9-9, indicates that the DCD conforms to SRP Section 9.5.2 (Revision 2). An evaluation of each of the regulatory criteria follows.

10 CFR Part 50, Appendix E, IV.E(9), requires that adequate provisions shall be made as described for emergency facilities and equipment, including at least one onsite and one offsite communications systems, each with a backup power source. DCD Tier 2, Revision 9, Section 9.5.2.2, states that the following systems as providing onsite communications: the PA/PL, the PABX telephone system, the plant sound-powered telephone system, and the plant radio systems. DCD Tier 2, Revision 9, Section 9.5.2.2 states that the PABX and plant radio systems provide offsite communications. Diverse nonsafety-related power supplies connected to the plant standby generators power the PA/PL telephone, PABX and plant radio systems.

DCD Tier 2, Revision 9, Section 9.5.2.2, identifies six emergency communications systems covered by the five COL information items:

(1) Emergency Notification System (COL Information Item 9.5.2.5-1-A)

(2) Health Physics Network (COL Information Item 9.5.2.5-3-A)

(3) Ringdown Phone System (COL Information Item 9.5.2.5-4-A)

(4) Crisis Management Radio System (COL Information Item 9.5.2.5-3-A)

(5) Fire Brigade Radio System (COL Information Item 9.5.2.5-5-A)

(6) Transmission System Operator Communication Link (COL Information Item 9.5.2.5-2-A).

The staff finds that these COL information items are for necessary portions of the site-specific communications systems and are sufficient.

The communications system is classified as nonsafety-related. The failure of any communications system does not adversely affect safe-shutdown capability. It is not necessary for plant personnel in safety-related areas of the plant to communicate with the MCR in order to achieve safe shutdown of the plant. There are three independent voice communications systems for emergency facilities and equipment and support onsite and the failure of any or all of their components does not affect any safety-related equipment. Based on the applicant identifying at least one onsite and offsite communications systems with backup power sources, the staff finds that the design meets the requirements of 10 CFR Part 50 Appendix E.IV.E(9).

10 CFR 50.34(f)(2)(xxv) [TMI Action Plan Item III A.1.2] requires an applicant, among other things, to provide an onsite TSC for the facility. SRP Section 9.5.2 states that information regarding TMI Action Plan Item III A.1.2 is acceptable if provisions are made for an onsite Technical Support Center and an onsite operational support center. In DCD Tier 2, Revision 9, Section 13.3, the applicant indicated that the standard plant design complies with all the TSC design criteria. The TSC is provided with reliable voice and data communications with the MCR and emergency operating facility (EOF) and reliable voice communications with the onsite support center (OSC), NRC Operations Centers, and state and local operations centers. Based on the applicant's descriptions of the communications systems for the TSC, OSC, and EOF, the staff finds that the design meets the requirements of 10 CFR 50.34(f)(2)(xxv) with regard to communications systems.

10 CFR 50.47(a)(8) requires adequate equipment and facilities to support emergency response. SRP Section 9.5.2 states that information regarding 10 CFR 50.47(a)(8) will be found acceptable if adequate emergency facilities and equipment to support the response are provided and maintained. DCD Tier 2, Revision 9, Section 9.5.2.2, specifically describes communications systems and equipment that support emergency response including the PA/PL, PABX, sound-powered telephone, evacuation alarm and remote warning system and especially the plant radio system with the emergency channel. DCD Tier 2, Revision 9, Section 13.3, includes the applicant's descriptions of the application of these communications systems for support in the TSC, OSC, and EOF and as part of the Emergency Plan. DCD Tier 2, Revision 9, Chapter 17, describes the applicant's quality assurance program for equipment maintenance and is evaluated in Chapter 17 of this report. Chapter 13 of this report includes the staff's evaluation of emergency planning response. Therefore, based on the above, the staff finds that the design meets the requirements of 10 CFR 50.47(a)(8) with regard to communications systems.

10 CFR 50.55a requires an applicant to address codes and standards. In DCD Tier 2, Revision 9, Table 3.2-1, the communications systems are classified as nonsafety-related systems. In DCD Tier 2, Revision 9, Table 1.9-9, the applicant indicates no departures from the guidance of SRP Section 9.5.2. DCD Tier 2, Revision 9, Table 1.9-20, lists SRP and BTP applicable to the ESBWR and includes SRP Section 9.5.2. Based on the communications descriptions and the information above, the staff finds that classification is acceptable for a nonsafety-related system

and that the design has adequately addressed the requirements of 10 CFR 50.55a with regard to communications systems.

GDC 1 requires that SSCs important to safety shall be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed. Where recognized codes and standards are used, they shall be identified and evaluated to determine their applicability, adequacy, and sufficiency and shall be supplemented or modified as necessary to assure a quality product in keeping with the required safety function. In DCD Tier 2, Revision 9, Table 3.2-1, the communications systems are classified as a nonsafety-related, non-seismic systems where system components are mounted to seismic Category II requirements in safety-related areas. DCD Tier 2, Revision 9, Table 1.9-20, lists SRP and BTP applicable to the ESBWR and includes SRP Section 9.5.2 in effect at the time of filing of the DCD application. Nonsafety-related items are controlled by the QA program described in DCD Tier 2, Revision 9, Chapter 17, in accordance with the functional importance of the item. Based on the communications systems descriptions, the information above, and the documentation in DCD Tier 2, Revision 9, Chapters 3 and 17, the staff finds that the communications systems design satisfies the requirements of GDC 1.

GDC 2 requires that SSCs important to safety shall be designed to withstand the effect of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches without loss of capability to perform their safety functions. DCD Tier 2, Revision 9, Table 3.2-1, classifies the communications systems as nonsafety-related systems. DCD Tier 2, Revision 9, Table 3.2-1, states that the communications systems components are mounted in accordance with seismic Category II requirements in safety-related areas. Chapter 3 of this report includes the evaluation of protection for natural phenomena such as earthquakes, tornadoes, hurricanes, and floods. However, DCD Tier 2, Revision 9, Section 9.5.2.2, states that the PA/PL, PABX, and plant radio systems are physically independent systems powered from diverse nonsafety-related power supplies backed by the standby onsite ac power supply system. They serve as backup to one another in the event of system failures. These three independent voice communications systems are designed and installed to provide assurance that any single event does not cause a complete loss of intra-plant communication. This is accomplished by the use of diverse technology, separate routing of cables, and separate SDG-backed power supplies. Accordingly, based on these design features, the importance of the functions of these systems (see discussion of GDC 19 below), and combined with the protection discussed in Chapter 3 of this report, the staff finds that the communications systems design has sufficient diversity and independence and has adequately addressed the requirements of GDC 2.

GDC 3 requires that SSCs important to safety shall be designed and located to minimize, consistent with other safety requirements, the probability and effect of fires and explosions. Section 9.5.1 of this report evaluates the fire protection features. DCD Tier 2, Revision 9, Table 3.2-1, classifies the communications systems as nonsafety-related systems. However, two-way voice communications are used to support safe shutdown and emergency response in the event of fire. DCD Tier 2, Revision 9, Section 9.5.2.2, states that the plant radio system complies with RG 1.189, Regulatory Position 4.1.7, which states the communications system design should provide effective communications between plant personnel in all vital areas during fire conditions under maximum potential noise levels. DCD Tier 2, Revision 9, Section 9.5.2.2, states that three systems (PA/PL, PABX, and plant radio systems) are physically independent systems powered from diverse nonsafety-related power supplies backed by the standby onsite ac power supply. The three systems serve as a backup to one another in the event of system failure as might be caused by fire. These three independent voice communications systems are designed and installed to provide assurance that any single event does not cause a complete

loss of intra-plant communication. Accordingly, based on these design features combined with the protection discussed in Section 9.5.1 of this report, the staff finds that the communications systems design has sufficient diversity and independence and has adequately addressed he requirements of GDC 3.

GDC 4 requires that SSCs important to safety shall be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including LOCAs. DCD Tier 2, Revision 9, Table 3.2-1, classifies the communications systems as nonsafety-related systems. Chapter 3 of this report includes the evaluation of protection for pipe rupture and flooding, EMI and RFI, and EQ. The plant radio system uses lower power portable radios to ensure there is no EMI with control and instrument circuits and vice versa. DCD Tier 2, Revision 9, Section 9.5.2.2, states that the PA/PL, PABX, and plant radio systems are physically independent systems powered from diverse nonsafety-related power supplies backed by the standby onsite ac power supply system. They serve as backup to one another in the event of system failure. These three independent voice communications systems are designed and installed to provide assurance that any single event does not cause a complete loss of intra-plant communication. This is accomplished by the use of diverse technology, separate routing of cables, and separate SDG-backed power supplies. The communications systems components are mounted in accordance with seismic Category II requirements in safety-related areas. The environmental conditions in safety-related areas are maintained to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including LOCAs. Accordingly, based on these design features combined with the protection discussed in Chapter 3 of this report, the staff finds that the communications systems design has adequately addressed the requirements of GDC 4.

GCD 19 requires that an MCR shall be provided from which actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions, including LOCAs. GDC 19 is not directly applicable to the communications systems. The reactor can be shut down safely without these nonsafety systems. Accordingly, the communications systems need not be credited in evaluating compliance with GDC 19. Nonetheless, the various independent and diverse communications systems located in the MCR and described in DCD Tier 2, Revision 9, Section 9.5.2.2, significantly increase the overall command and control the reactor operators have over the plant by providing the ability to communicate and direct activities with operations, maintenance, health physics, firefighters, security, and rescue teams in the plant. In addition, 10 CFR 73.45(e)(2)(iii) requires that communications systems and procedures provide for notification of an attempted unauthorized or unconfirmed removal of strategic special nuclear material. DCD Tier 2, Revision 9, Section 9.5.2, identifies that the ESBWR has a completely independent communication (radio) system for security purposes. Other communications systems such as the PA/PL and PABX are available as alternate means if necessary. The application of the communications systems described in DCD Tier 2, Revision 9, Section 9.5.2, in support of conformance to 10 CFR 73.45(e)(2)(iii), is evaluated in Section 13.6 of this report under conformance to 10 CFR 73.55 and the security plan.

10 CFR 73.45(g)(4)(i) requires rapid and accurate transmission of security information among onsite forces for routine security operation, assessment of a contingency, and response to a contingency. SRP Section 9.5.2 states information regarding the requirements of 10 CFR 73.45(g)(4)(i) will be found acceptable if communications networks are provided to transmit rapid and accurate security information among onsite forces for routine security operation, assessment of a contingency, and response to a contingency. DCD Tier 2, Revision 9, Section

9.5.2, identifies that the ESBWR has a completely independent communication (radio) system for security purposes. The PA/PL, PABX, and plant radio system are physically independent systems and can serve as backup systems in the event of failure of the security communication (radio) system. The application of these communications for security purposes is described in DCD Tier 2, Revision 9, Section 13.6 and evaluated in Section 13.6 of this report under conformance to 10 CFR 73.55 and the security plan for the reasons given in that section. The staff finds that communications systems have the capability to support the notifications system required by 10 CFR 73.45(g)(4)(i).

10 CFR 73.46(f) requires that the communications systems shall be capable of maintaining continuous communications between each guard, watchman, or armed response individual on duty with the manned alarm stations. SRP Section 9.5.2 states that information regarding the requirements of 10 CFR 73.46(f) will be found acceptable if (1) each guard, watchman, or armed response individual on duty shall be capable of maintaining continuous communications with an individual in each continuously manned alarm station required by 10 CFR 73.46(e)(5), who shall be capable of calling for assistance from other guards, watchmen, and armed response personnel and from law enforcement authorities; (2) each alarm station required by 10 CFR 73.46(e)(5) shall have both conventional telephone service and radio or microwave transmitted two-way voice communication, either directly or through an intermediary, for the capability of communications with the law enforcement authorities; and (3) non-portable communications equipment controlled by the licensee and required by 10 CFR 73.46(f) shall remain operable from independent power sources in the event of the loss of normal power. DCD Tier 2, Revision 9, Section 9.5.2, specifies that the communications systems allow guards and watchmen on duty to maintain continuous communications with personnel in manned alarm stations, and offsite/onsite agencies as required by 10 CFR 73.55. This is accomplished by either PABX or wireless communications systems backed by the PA/PL. As described in DCD Tier 2, Revision 9, Section 9.5.2.2, the PA/PL, PABX, and plant radio system are physically independent systems powered from diverse nonsafety-related power supplies backed by the standby onsite ac power supply system. They serve as backup to one another in the event of system failure. These three independent voice communications systems are designed and installed to provide assurance that any single event does not cause a complete loss of intra-plant communication. This is accomplished by the use of diverse technology, separate routing of cables, and separate standby diesel-generator-backed power supplies. Accordingly, the staff finds that the communications systems design has the capability to support the communications required by 10 CFR 73.46(f).

10 CFR 73.55(e) and 10 CFR 73.55(i), now apply to physical protection of licensed activities in nuclear power reactors. The application of communications systems as supporting systems is described in DCD Tier 2, Revision 9, Section 13.6, and evaluated in Section 13.6 of this report.

10 CFR 73.55(j) requires: (1) the licensee shall establish and maintain continuous communication capability with onsite and offsite resources to ensure effective command and control during both normal and emergency situations; (2) individuals assigned to each alarm station shall be capable of calling for assistance; (3) all on-duty security force personnel shall be capable of maintaining continuous communication with an individual in each alarm station, and vehicle escorts shall maintain continuous communication with security personnel; (4) the following continuous communication capabilities must terminate in both alarm stations required by this section: radio or microwave transmitted two-way voice communication, either directly or through an intermediary, in addition to conventional telephone service between local law enforcement authorities and the site and a system for communications with the control room; (5) non-portable communications equipment must remain operable from independent power

sources in the event of the loss of normal power; and (6) the licensee shall identify site areas where communication could be interrupted or cannot be maintained, and shall establish alternative communication measures or otherwise account for these areas in implementing procedures. DCD Tier 2, Revision 9, Section 9.5.2, identifies that the ESBWR has a completely independent communication (radio) system for security purposes that is capable of maintaining continuous communication capability with onsite and offsite resources to ensure effective command and control during both normal and emergency situations. The emergency communication system has color-coded telephones for offsite communications with the NRC, state officials, state and local emergency centers, local fire departments, and local police authorities. The PA/PL, PABX, and plant radio systems are physically independent systems and can serve as backup systems in the event of failure of the security communication (radio) system. The plant sound-powered telephone provides another diverse system that does not require external power. The PA/PL, PABX, and plant radio systems are physically independent systems powered from diverse nonsafety-related power supplies backed by the standby onsite ac power supply system. These three independent voice communications systems are designed and installed to provide assurance that any single event does not cause a complete loss of intra-plant communication. The application of these communications for security purposes is described in DCD Tier 2, Revision 9, Section 13.6, and evaluated in Section 13.6 of this report. Based on the above and security information from Section 13.3 of this report, the staff finds that the requirements of 10 CFR 73.55(j) are adequately addressed with regard to the communications systems design described in DCD Tier 2, Revision 9, Section 9.5.2.

10 CFR 52.47(b)(1) requires that a design certification application contain the proposed ITAAC that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design certification is built and will operate in accordance with the design certification, the provisions of the Atomic Energy Act, and the NRC's regulations. DCD Tier 1, Revision 9, Section 2.13.7, states that no ITAAC are required for this system. The staff finds that this is acceptable and that a stand-alone ITAAC is not “necessary because: (1) the communications systems are nonsafety-related and do not have any RTNSS functions; (2) a significant portion of the communication systems are indirectly tested in the EP ITAAC; (3) the regulation refers to systems that have been constructed, but much of the wireless (radio) communication equipment used by the fire brigade is commercial off-the -shelf equipment; (4) the applicant has committed to meeting RG 1.189, Regulatory Position 4.1.7, for the plant radio system, which states the communications system design should provide effective communications between plant personnel in all vital areas during fire conditions under maximum potential noise levels; (5) there is redundant radio equipment and the radio equipment will receive significant and continual direct and indirect testing through pre-operational test, startup tests, and routine drills; and (6) much of the other communication equipment as telephones, headsets, public address boxes, etc., are commercial off-the-–shelf pre-tested items. Based on the above, the staff finds that the communications systems design satisfies the requirement of 10 CFR 52.47(b)(1).

9.5.2.4 *Conclusion*

Based on the above, the staff finds that the communications systems design is acceptable and meets the requirements of 10 CFR Part 50, Appendix E, Section IV.E(9); 10 CFR 50.34(f)(2)(xxv); 10 CFR 50.47(a)(8); 10 CFR 50.55a; GDC 1, 2, 3, and 4; 10 CFR 73.45(e)(2)(iii); 10 CFR 73.45(g)(4)(i); 10 CFR 73.46(f); 10 CFR 73.55(j); and 10 CFR 52.47(b)(1)).

9.5.3 Plant Lighting System

9.5.3.1 *Regulatory Criteria*

No GDC or RGs directly apply to the functions of the lighting system. However, the plant lighting system is necessary to support accident mitigation (e.g., FPP) and safety-related maintenance and operating activities, and should have the capability to: (1) provide adequate normal lighting during all plant operating conditions, (2) provide adequate emergency lighting during all other plant operating conditions, including fire, transient and accident conditions and (3) address the effect of the loss of all ac power (i.e., during an SBO) on the emergency lighting system. The lighting system for the ESBWR should be designed in accordance with SRP Section 9.5.3 and with lighting levels recommended in NUREG–0700, which is based on the Illuminating Engineering Society of North America (IESNA) Lighting Handbook.

9.5.3.2 *Summary of Technical Information*

The plant lighting systems furnish the illumination necessary for safe performance of plant operation, security, shutdown, and maintenance activities. Emergency lighting is provided in areas where emergency operations are performed and for the safety of personnel during a power failure. The emergency lighting system maintains the lighting levels for at least 72 hours following a design-basis event, including the loss of all ac power sources. The lighting system illumination ranges for normal illumination are based on the IESNA Lighting Handbook.

The plant lighting system includes normal, standby, emergency, and security lighting. Section 13.6 of this report discusses the security lighting system. The lighting systems are designed in accordance with applicable industry standards for lighting fixtures, cables, grounding, penetrations, conduits, and controls. Lighting fixtures located in the vicinity of safety-related equipment are supported so that they do not adversely impact the equipment when subjected to the seismic loading of an SSE. The lighting circuits of the normal, standby, and emergency lighting subsystems are routed in separate conduits. The design of the lighting system for areas containing rotating equipment includes special provisions to eliminate the risk of stroboscopic effects caused by flicker. The circuits to the individual lighting fixtures (other than the dc self-contained, battery-operated lighting units) are staggered to the extent possible, and separate power sources supply the staggered circuits to ensure that some lighting is retained in each room in the event of a circuit failure. Mercury vapor lamps and mercury switches are not present in fuel handling areas. Additionally, the primary containment, main steam tunnel, and refueling level of the RB use either incandescent lamps or light-emitting diode illuminating devices. The emergency lighting system is tested to ensure the operability of the dc self contained battery-operated lighting units and other major components of the system.

Normal Lighting

The normal lighting system, as supplemented by the standby lighting system, provides standard illumination under normal plant operating, maintenance, and testing conditions. This system provides lighting for all indoor and outdoor areas. The nonsafety-related power generation buses supply power to the normal lighting system. The high-intensity discharge and fluorescent lighting fixtures in this subsystem are powered from 480/277 V ac, three-phase, four-wire, and grounded neutral system distribution panels supplied from normal 480 V ac motor control centers. The incandescent lighting fixtures on refueling platforms are powered from 480/277 V ac, three-phase, four-wire, and grounded neutral system distribution panels. Other incandescent lighting fixtures are powered from dry-type transformers rated at 480-208/120 V

ac, three-phase, four-wire, and grounded, or 480-240/120 V ac, single-phase, three-wire, and grounded.

Standby Lighting

The standby lighting system, in addition to reinforcing the normal lighting system, supplements the emergency lighting system in selected areas of the plant where emergency operations are performed, including the access and egress routes to and from those areas. The standby lighting system is designed to provide a minimum level of illumination to selected areas of the plant to aid in emergencies, during safe shutdown, or in restoring the plant to normal operation. This system consists of fluorescent lighting fixtures powered from 480/277 V ac, three-phase, four-wire, and grounded neutral system distribution panels normally supplied by the PIP nonsafety-related buses. The primary areas served by this system are as follows:

- MCR
- Remote shutdown rooms
- Operational support centers
- Technical support centers
- Auxiliary switchgear rooms
- Safety-related dc equipment rooms
- Stairwells and aisle way
- DCIS equipment rooms
- Diesel generator rooms
- Diesel generator control room

The standby lighting distribution panels also serve as the preferred power supply to the 8-hour emergency lighting units and the stair lighting units. The standby lighting is maintained as long as power is available from the PIP nonsafety-related buses.

Emergency Lighting

The emergency lighting system provides acceptable levels of illumination throughout the station, particularly in areas where emergency operations are performed, such as control rooms, remote shutdown area, battery rooms, and containment, upon loss of the normal lighting system. The emergency lighting system comprises the following:

- MCR and remote shutdown area emergency lighting

- Nonsafety-related dc self-contained battery-operated lighting units for exit lights, emergency lighting units, and stair lighting units

The emergency lighting system components and installation inside and outside the MCR remain functional during design-basis events and in particular withstand the seismic loads of a design-basis earthquake. The standby and emergency lighting fixtures, switches, and associated cables used in the MCR are non-Class 1E.

The MCR and remote shutdown area emergency lighting power is supplied from the safety-related UPS system. Electrical isolation of nonsafety-related emergency lighting circuits from safety-related UPS is accomplished by the use of series isolation devices that are designed to coordinate with upstream 120 V ac distribution panel circuit breakers. Raceways carrying cables to the lighting fixtures, as well as the lighting fixtures for both standby and emergency

lighting inside the MCR, utilize seismic Category I support. Both the standby and emergency lighting fixtures are nonsafety-related. Cables used for emergency lighting in the MCR and the remote shutdown area are nonsafety-related. The MCR emergency lighting complies with human factor requirements by using semi-indirect, low-glare lighting fixtures.

In areas outside the MCR, emergency lighting is provided by 8-hour, self-contained battery pack, sealed-beam lighting units. These units are powered from the nonsafety-related power source and provide illumination for safe ingress and egress of personnel following a loss of normal lighting in areas that are needed for power restoration and recovery to comply with the recommendation of RG 1.189. In addition, these units are used in areas where normal actions are needed for operation of equipment needed during a fire and in stairwells serving as escape or access routes for fire fighting.

The dc self-contained, battery-operated emergency and stair lighting units are powered from the same circuit that powers the normal or standby lighting fixture, whose loss of power then causes the operation of the particular emergency or stair lighting unit.

Emergency exit lighting consists of battery-powered, self-contained “exit” light units. Each unit consists of a 90-minute battery, battery charger, and exit sign and is normally energized by 277 V ac or 120 V ac from the normal lighting system power supply.

Emergency lighting units provide lighting instantaneously and automatically on the failure or interruption of the normal or standby lighting power supply, as applicable. Each emergency lighting unit consists of a battery, a charger, and control and monitoring circuits, enclosed in a self-contained unit. Each emergency lighting unit is capable of supplying emergency lighting through sealed beam lamps locally mounted on the battery pack unit, remotely mounted near the battery pack unit, or a combination thereof for 8 hours without the charger.

The emergency lighting units are designed with a time delay following restoration of ac power. The emergency lighting only turns off after adequate time for the normal or standby lighting to restart. The units are normally energized from the same circuit whose loss of light initiates the operation of the unit.

Panel Lighting

Panel lighting is designed to provide lighting for interior maintenance of the panels as described below.

- Panel lighting consists of lighting fixtures located inside the wide display panel in the MCR. The fixtures are powered from a nonsafety-related power source and are normally off.

- Raceways carrying cables up to the lighting fixtures as well as the lighting fixtures are supported by seismic Category I support.

9.5.3.3 *Staff Evaluation*

DCD Tier 2, Revision 3, Section 9.5.3.3.3.2, states that each emergency lighting unit is capable of supplying sealed beam lamps for 8 hours without the charger. However, there are 2-hour-rated units and 90-minute-rated units in different applications. In RAI 9.5-58, the staff asked the applicant to clarify the discrepancy. In response, the applicant stated that the 90-minute-rated units are used for exit signs only, and the 8-hour-rated units are used in areas outside the MCR.

The applicant clarified that 2-hour-rated units are not used in any area of the plant. The applicant stated that it would revise the first bulleted item under DCD Tier 2, Revision 3, Section 9.5.3.3.3.2, to delete the use of 2-hour-rated units. The staff finds that the response is acceptable because the applicant clarified the use of 8-hour and 90-minute lighting. Based on the above and the applicant's response, RAI 9.5-58 is resolved. RAI 9.5-58 was being tracked as a confirmatory item in the SER with open items. The staff finds that the applicant deleted the 2-hour rated units in DCD Tier 2, Revision 4. Therefore, this confirmatory item is closed.

DCD Tier 2, Revision 3, Section 9.5.3.3.3.2, stated that 2-hour-rated units, as a minimum, are used in other areas of the plant. In RAI 9.5-59, the staff asked the applicant to clarify where the 2-hour-rated units will be used. In response, the applicant clarified that the 2-hour-rated units are not used in any area of the plant and deleted reference to 2-hour rated units in the DCD. The staff finds that the response is acceptable because the applicant clarified the use of 2-hour lighting. Based on the above and the applicant's response, RAI 9.5-59 is resolved.

Based on the review of DCD Tier 2, Revision 3, the staff determined that emergency lighting supplied by the 72-hour Class 1E UPS system is not used in remote shutdown areas. The staff determined that this was unacceptable because the remote shutdown areas have human-system interface comparable to the MCR and therefore the remote shutdown areas should have emergency lighting comparable to the MCR. In RAI 9.5-60, the staff asked the applicant to provide justification for not using emergency lighting supplied by the 72-hour Class 1E UPS system in remote shutdown areas. In response the applicant stated that the 72-hour Class 1E UPS system is used for the safety-related DCIS, instrumentation required for regulatory compliance, and the MCR emergency lighting. Emergency lighting in areas outside the MCR, such as the remote shutdown room, is accomplished by 8-hour, self-contained battery pack, sealed-beam lighting units. These units are nonsafety-related, provide illumination for safe ingress and egress of personnel and shutdown activities, and are powered from diesel-backed buses upon loss of normal ac power. The staff determined that the response was not acceptable.

In RAI 9.5-60 S01, the staff asked the applicant to provide justification for not providing an emergency lighting capacity of 72 hours at the remote shutdown rooms such that the emergency lighting capability in these rooms is equivalent to that in the MCR. The staff also asked the applicant to discuss the emergency lighting in remote shutdown areas in DCD Tier 2, Section 9.5.3.3.3. In response to RAI 9.5-60 S01, the applicant stated that emergency lighting in the remote shutdown area is fed from the safety-related UPS for 72-hours similar to the power supply arrangement for the MCR emergency lighting. In response to RAI 9.5-60 S02, the applicant provided a markup copy of DCD Tier 2, Section 9.5.3.3.3.1, which stated that the control room and remote shutdown area emergency lighting are supplied from the safety-related UPS, as shown in DCD Tier 2, Figure 8.1-4, Sheet 1 of 1. DCD Tier 2, Figure 8.1-4, Sheet 1 of 1, indicated that MCR emergency lighting is supplied from four divisions of the safety-related UPS, while the remote shutdown area emergency lighting is supplied from Divisions 1 and 2 of the UPS.

In RAI 9.5-60 S03, the staff asked the applicant to explain why the emergency lighting from Divisions 1 and 2 is acceptable in the remote shutdown area. In response to RAI 9.5-60 S03, the applicant stated that the RSS panels are each provided with Division 1 and Division 2 lighting and PIP A and PIP B lighting. Other than manual scram and the isolation switches, the only controls or instrumentation on each of the RSS panels are a Division 1 and Division 2 visual display unit (VDU) (for control and monitoring of the respective divisions) and a PIP A and PIP B VDU (for control and monitoring of the PIP RTNSS and BOP functions as power is

available and for monitoring of all divisional information). If Division 1 and Division 2 power from the UPS is not available, then only PIP A and PIP B functionality is retained, which is sufficient to scram the plant and bring it to safe shutdown. Lighting derived from PIP A and PIP B is sufficient to operate the PIP A and PIP B VDUs. If PIP A and PIP B lighting is lost, the PIP A and PIP B VDUs will be lost; however, the Division 1 and Division 2 UPS lighting is sufficient to operate the Division 1 and Division 2 VDUs. Based on above, power supply from Division 3 and Division 4 is not necessary for RSS area lighting because it is provided by the eight hour battery powered lights and nonsafety-related power from the PIP buses.

The staff finds that the RAI responses are acceptable because the applicant clarified the emergency lighting in the remote shutdown areas and the basis for its power supplies. The staff confirmed that the applicant revised DCD Tier 2, Revision 5, Sections 9.5.3.3.3, 9.5.3.3.3.1, and 9.5.3.3.3.2 accordingly. Based on the above, the applicant's responses, and DCD changes, RAI 9.5-60 is resolved.

MCR emergency lighting is supplied from the Class 1E UPS. The lighting fixtures, circuits, and associated cables are non-Class 1E. In RAI 9.5-61, the staff asked the applicant to discuss isolation devices to be used between the Class 1E power supply and non-Class 1E circuits. In response, the applicant stated that the Class 1E power supply and non-Class 1E circuits are isolated through a series of breakers that are coordinated for proper isolation during the design phase of the project. The applicant further replied that DCD Tier 2, Revision 3, Section 9.5.3.3.3.1, is to be revised in its entirety for clarity and to add isolation provisions (e.g., "The safety-related UPS and the MCR emergency lighting circuitry are isolated by a series of circuit breakers that are coordinated for isolation"). However, in response to RAI 9.5-63, the applicant stated that the MCR emergency lighting system is safety-related and classified as Class 1E. In a combined RAI 9.5-61 S01 and RAI 9.5-63 S01, the staff asked the applicant why an isolation device is needed if the MCR emergency lighting system (power supply, cables, switches, fixtures, and so forth) is safety-related and classified as Class 1E. RAIs 9.5-61 and 9.5-63 were being tracked as open items in the SER with open items. In response, the applicant clarified that MCR emergency lighting fixtures are nonsafety-related; hence, separation devices are necessary. In DCD Revision 5, the applicant further clarified that the lighting fixtures, circuits, and associated cables are nonsafety-related. The staff confirmed that the applicant revised DCD Tier 2, Revision 5, Sections 9.5.3.1, 9.5.3.3.3.1 and 9.5.3.4 accordingly. The staff finds that the RAI response, with the additional DCD changes, is acceptable since the applicant clarified the isolation devices for the emergency lighting. Based on the above, the applicant's responses, and DCD changes, RAIs 9.5-61 and 9.5-63 are resolved.

DCD Tier 2, Revision 3, Section 9.5.3.3.3.1, states that the MCR emergency lighting is supplied from four divisions of 72-hour Class 1E UPS. In RAI 9.5-62, the staff asked the applicant to discuss the separation requirement between four divisions of UPS supplies and cables outside the MCR. In response, the applicant stated that the four divisions of 72-hour Class 1E UPS are independent, located in separate rooms, and cannot be interconnected, and that their circuits are routed in dedicated, physically separated raceways. This level of electrical separation prevents the failure or unavailability of a single battery, battery charger, or inverter from adversely affecting a redundant division. The staff finds that the response is acceptable since the applicant clarified the separation between the four divisions of UPS supplies and cables outside the MCR. Based on the above and the applicant's response, RAI 9.5-62 is resolved.

DCD Tier 2, Revision 3, Section 9.5.3.4, states that the MCR emergency lighting system is safety-related and classified as Class 1E. Also, in DCD Tier 2, Revision 3, Section 9.5.3.1, the

applicant states that the MCR emergency lighting system is Class 1E. However, DCD Tier 2, Revision 3, Subsection 9.5.3.3.3.1, states that the standby and emergency lighting fixtures, switches, and associated cables used in the MCR are non-Class 1E. In RAI 9.5-63, the staff asked the applicant to address the discrepancy and verify that the MCR emergency lighting system is safety-related and classified as Class 1E. In response, the applicant stated that the MCR emergency lighting system is safety-related. The power source for the MCR emergency lighting, switches, associated cables, and lighting fixtures is safety-related. Raceways carrying cables to the lighting fixtures, as well as the lighting fixtures for both emergency and standby lighting inside the MCR, use seismic Category I support. In response to RAI 9.5-61, the applicant stated that safety-related UPS and the MCR emergency lighting circuitry are isolated by a series of circuit breakers that are coordinated for isolation. In RAI 9.5-61 S01, and RAI 9.5-63 S01, the staff asked the applicant why an isolation device is needed if the MCR emergency lighting system (power supply, cables, switches, fixtures, and so forth) is safety-related and classified as Class 1E. RAIs 9.5-61 and 9.5-63 were being tracked as open items in the SER with open items. In response, the applicant clarified that MCR emergency lighting fixtures are nonsafety-related; hence, separation devices are necessary. In DCD Revision 5, the applicant further clarified that the lighting fixtures, circuits, and associated cables are nonsafety-related. The staff confirmed that the applicant revised DCD Tier 2, Revision 4, Sections 9.5.3.1, 9.5.3.3.3.1 and 9.5.3.4 accordingly. The staff finds that the RAI response, with the additional DCD changes, is acceptable since the applicant clarified that the power supplies for the emergency lighting up to isolation devices are safety-related and the emergency lighting fixtures, circuits, and associated cables are nonsafety-related. Based on the above, the applicant's responses, and DCD changes, RAIs 9.5-61 and 9.5-63 are resolved.

DCD Tier 2, Revision 3, Section 9.5.3, contains no design description of lighting in the MCR at the safety panels. In RAI 9.5-64, the staff asked the applicant to provide a design description of panel lighting in the MCR or provide a technical basis for not doing so. In response, the applicant stated that the ESBWR MCR is designed using human factors engineering principles. The configuration of the MCR is significantly different than that of a conventional BWR in that it does not have panels located in areas behind the main console. The three panels inside the MCR are the wide display panel, main control console, and the shift supervisor console. The emergency lighting provides a minimum of 108-lux (10-foot-candles) illumination at the consoles in the event of loss of normal lighting. Additionally, the wide display panel has lights that are powered from a nonsafety-related power source and are mounted inside the console. The supports for the lighting fixtures are seismic Category I. The applicant further stated that it will add a new Section 9.5.3.3.3.3, to DCD Tier 2, which will read as follows:

> Panel lighting is designed to provide lighting for interior maintenance of the panels as described below:
>
> - Panel lighting consists of lighting fixtures located inside the wide display panel in the MCR. The fixtures are powered from nonsafety-related power source and are normally off.
> - Raceways carrying cables up to the lighting fixtures as well as the lighting fixtures are supported by seismic Category I support.

The staff finds that the RAI response is acceptable since the applicant proposed to add a design description of panel lighting in the MCR. Based on the above and the applicant's response, RAI 9.5-64 is resolved. RAI 9.5-64 was being tracked as a confirmatory item. The staff confirmed that the applicant added a new Section 9.5.3.3.3.3, to DCD Tier 2, Revision 4, and, therefore, this confirmatory item is closed.

Based on the above, the staff finds that the normal, standby, and emergency lighting systems will provide adequate lighting during normal and emergency plant operating conditions. The emergency lighting system will provide adequate station lighting to all vital areas from onsite power sources during the full spectrum of accident and transient conditions and to the access routes to and from these areas. The staff finds the information provided for the plant lighting system to be sufficient to meet the guidance of SRP Section 9.5.3.

9.5.3.4 *Conclusion*

Based on the above, the staff finds that the design of the lighting system for the ESBWR is in accordance with the lighting levels recommended in NUREG–0700, which is based on the IESNA Lighting Handbook. Therefore, the design is acceptable.

9.5.4 Diesel Generator Fuel Oil Storage and Transfer System

9.5.4.1 *Regulatory Criteria*

The staff reviewed the ESBWR SDG and ADG fuel oil storage and transfer systems (FOSTS) in accordance with SRP Section 9.5.4, Revision 3. The staff reviewed DCD Tier 2, Revision 9, Section 9.5.4. The staff also reviewed DCD Tier 1, Revision 9, Section 2.0, and other DCD Tier 2, Revision 9, sections noted below. The staff's acceptance of the FOSTS is based on the design's conformance with the requirements of the following regulations:

- GDC 2 requires, in part, that SSCs important to safety shall be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes and floods, Compliance with GDC 2 is based on meeting the guidance of Regulatory Position C.1 for the safety-related portions and Regulatory Position C.2 for the nonsafety-related portions of RG 1.29.

- GDC 4 requires, in part, that SSCs important to safety shall be designed to accommodate the effects of, and be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents.

- GDC 5 requires that SSCs important to safety shall not be shared among nuclear power units unless it can be shown that such sharing will not significantly impair their ability to perform their safety functions, including, in the event of an accident in one unit, an orderly shutdown and cooldown of the remaining units.

- GDC 17 requires, in part, that an onsite electric power system and an offsite electric power system shall be provided to permit functioning of SSCs important to safety.

- 10 CFR 52.47(b)(1) requires that a design certification application contains the proposed ITAAC that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design certification is built and will operate in accordance with the design certification, the provisions of the Atomic Energy Act, and the NRC's regulations.

9.5.4.2 *Summary of Technical Information*

There are two redundant onsite seismic Category II SDG units in the ESBWR design for a single unit plant to provide power to nonsafety-related ac loads in the event of a loss of normal

and preferred ac power supplies. Each SDG unit is an independent system complete with its support systems, which are the FOSTS, jacket cooling water system (JCWS), starting air system, lubrication system, and combustion air intake and exhaust system. Each SDG unit is housed in a separate seismic Category II structure. The design provides adequate separation between the two SDG units, including their support systems, so that failure in one SDG does not result in loss of function of the other SDG.

In addition, in ESBWR DCD Revision 5, the applicant revised the design of the onsite ac power supply system by adding two ADGs to provide 480 V ac power to meet the post-72 hour power demand following an extended loss of all other ac power sources. Each ADG unit, which is an independent system provided as a complete skid-mounted package with its integral support systems, is housed in a separated seismic Category II structure. The design provides adequate separation between the two ADG units so that failure in one ADG does not result in loss of function of the other ADG.

SDGs, ADGs, and their support systems are nonsafety-related and non-Class 1E electrical systems and have no safety-related design basis. However, they are relied upon to be available to provide an ac source of power 72 hours after an abnormal event. Therefore, the SDGs, ADGs, and their supporting systems including FOSTS, have RTNSS functions, as supporting systems, to provide power and are included in the plant ACM, which will ensure that they have sufficient reliability and availability to perform their RTNSS functions.

Standby Diesel Generator

The FOSTS for each SDG is designed to supply the day tank with sufficient fuel oil capacity for a minimum of 8 hours of SDG operation at full load and sufficient fuel oil capacity onsite sufficient to support continuous SDG operation for 7 days without replenishing. In addition, the FOSTS has piping connections to supply fuel oil to the ABS, the diesel-engine driven FPS pump day tank, and the ADG fuel oil storage tanks. The piping connections tie into the SDG fuel oil storage tank at an elevated nozzle connection, which ensures that fuel oil inventory stored below this level for the SDG will not be affected by ABS usage, FPS usage, or transfers to the ADG fuel oil storage tanks. This ensures that the diesel fuel oil intended to support 7 days of SDG operation at full load cannot be used for any other purposes. The COL applicant will establish administrative controls to ensure that a minimum fuel oil inventory is maintained on site at all times.

The primary components of each SDG FOSTS are the yard fuel oil storage tank, two fuel oil transfer pumps, fill and recirculating pump, day tank, and associated piping, valves, and instrumentation controls. Transfer pumps supplying fuel oil to the day tank from the yard fuel oil storage tank allow manual operation; however, level sensors on the day tanks normally operate them automatically. A “low” level signal starts the first transfer pump, a “low-low” level signal starts the standby transfer pump, and a “high” level signal stops both pumps. An engine-driven fuel oil pump supplies fuel oil to the diesel engine fuel manifold from the day tank.

Ancillary Diesel Generator

The FOSTS for each ADG consists of a separate fuel oil storage tank, fuel oil day tank, fuel oil transfer pumps, strainers and filters, oil purifier or tank connections for tying in a portable fuel oil purification system, and associated piping, valves, and instrumentation controls. The FOSTS for each ADG is designed to supply sufficient fuel oil onsite for its associated ADG operation for 7 days without replenishing and to be filled by either a tanker truck via a fill station or by

manually controlled transfer from the yard SDG fuel oil storage tanks. The COL applicants will establish administrative controls to ensure that a minimum fuel oil inventory is maintained on site at all times. The system operation for the ADG FOSTS is identical to that described above for SDG FOSTS.

The SDG and ADG FOSTS permit periodic testing and inspection in accordance with the ACM. FOSTS functionality is demonstrated during the regularly scheduled operational tests of the SDGs and ADGs. Also, periodic testing of instruments, controls, sensors, and alarms assures reliable operation.

Routine sample tests are conducted at regular intervals to ensure that the stored fuel oil meets the standards of the ASTM D975, "Standard Specification for Diesel Fuel Oils," and the diesel engine manufacturer. Each fuel oil storage tank is emptied and accumulated sediments are removed every 10 years to conform to Federal and State examination requirements.

For both SDG and ADG, the FOSTS piping and components up to the engine skid connections are designed and constructed in accordance with the ASME Code, Section VIII, Division 1, and ASME Power Piping Code B31.1. Corrosion protection for underground portions of the FOSTS, including piping and fuel oil storage tanks, is determined and provided based on the material of the underground portions.

9.5.4.3 *Staff Evaluation*

The staff reviewed the FOSTS to determine if the design meets the relevant requirements of GDC 2. As stated above, the SDGs, ADGs, and their support systems are nonsafety-related and non-Class 1E electrical systems and have no safety-related design basis. However, the SDGs and ADGs and their supporting systems, including the FOSTS have RTNSS functions, as supporting systems to provide an ac source of power 72 hours after an abnormal event. Based on its review as discussed in Sections 3.4.1, 3.5.1.1, 3.5.1.4, and 3.5.2 of this report, as described below, the staff finds that the SDG FOSTS and ADG FOSTS meet the relevant requirements of GDC 2 as it pertains to Regulatory Position C.2 of RG 1.29. The FOSTS also meet the requirements of GDC 2 as it pertains to Regulatory Position C.1 of Reg. Guide 1.29.

Section 3.4.1 of this report addresses the staff's evaluation of flood protection provided for RTNSS systems. Section 3.5.1.1 of this report addresses the staff's evaluation of protection provided for RTNSS systems from internally generated missiles outside containment. Section 3.5.1.4 of this report addresses the staff's evaluation of protection provided for RTNSS systems from missiles generated by natural phenomena. Section 3.5.2 of this report addresses the staff's evaluation of protection provided for RTNSS systems from externally generated missiles.

The staff reviewed the FOSTS to determine if the design meets the relevant requirements of GDC 4. Based on the staff's evaluation in Section 3.6.1 of this report, the staff finds that the SDGs, the ADGs and their support systems, including the FOSTS are protected against the effects of, and are compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents. Section 3.6.1 of this report addresses the staff's evaluation of the design of structures, shields, and barriers necessary for RTNSS to be protected against dynamic effects of high-energy line breaks. The staff, therefore, finds that the SDG FOSTS and ADG FOSTS meet the requirements of GDC 4.

The ESBWR design is a single-unit station, and the requirements of GDC 5 are not applicable to the single-unit design.

Sections 8.2 and 8.3 of this report address the staff's evaluation of the ESBWR design with respect to the requirements of GDC 17 regarding the provision of an onsite electric power system and an offsite electric power system to permit functioning of SSCs important to safety.

During the review of DCD Revision 0, the staff determined that, during a postulated post-LOCA and a complete loss of ac power supplies, the SDG units were used to supply power for recharging batteries to support post-LOCA monitoring beyond 72 hours. Therefore, the SDGs and their support systems met Criteria B and C, established in SECY-94-084 for the passive ALWR plant design, to determine the systems that are candidates for RTNSS consideration. The staff determined that the SDGs and their supporting systems should have been considered as candidates for RTNSS systems and should have ITAAC entries. Therefore, in RAI 19.1.0-2, the staff requested the applicant to provide documentation or analyses in support of the process used to identify RTNSS systems.

In response to RAI 19.1.0-2, the applicant included the SDG units as RTNSS, Criterion C systems. However, it was not clear to the staff that all of the SDG supporting systems would be considered as RTNSS systems. Subsequently, in RAI 22.5-4, the staff requested the applicant to clarify that all SDG supporting systems were considered as RTNSS systems. In response to RAI 22.5-4, the applicant stated that all SDG supporting systems, including SDG FOSTS were considered as RTNSS systems. In DCD Tier 2, Revision 4, Section 9.5.4, the SDG FOSTS was included and classified as an RTNSS system. The staff finds that the responses to RAIs 19.1.0-2 and 22.5-4 regarding the SDG FOSTS are acceptable since the applicant clarified that all SDG supporting systems, including SDG FOSTS were considered as RTNSS systems. Based on the above, the applicant's responses, and DCD changes, RAIs 19.1.0-2 and 22.5-4 regarding the SDG FOSTS are resolved. RAI 22.5-4 was being tracked as a confirmatory item in the SER with open items. The staff confirmed that the applicant incorporated the changes into DCD Revision 4; therefore, the confirmatory item is closed.

The staff determined that the applicant should identify ITAAC for the SDG supporting systems since they are considered to be RTNSS systems. In RAI 14.3-151, the staff requested the applicant to include ITAAC for all of the SDG supporting systems. In response to RAI 14.3-151, the applicant stated that it would not include ITAAC for SDG supporting systems because they were nonsafety-related systems. The applicant's response was not acceptable to the staff because the SDG supporting systems had been reclassified as RTNSS systems. In RAI 14.3-151 S01, the staff again requested the applicant to include ITAAC for all SDG supporting systems. RAI 14.3-151 was being tracked as an open item in the SER with open items. Subsequently, in DCD Tier 1, Revision 4, Table 2.13.4-2, the applicant included ITAAC for only two SDG supporting systems, the SDGFOSTS and the SDG starting air system (SDGSAS). The staff further issued RAI 14.3-177, to request the applicant to include ITAAC for all SDG supporting systems. Finally, in the response to RAI 14.3-177, the applicant committed to revising DCD Revision 5 to include an ITAAC entry for each of the SDG supporting systems. The staff has finds that the response to RAI 14.3-177 is acceptable since the applicant added ITAAC for all SDG supporting systems, which also addresses the concerns of RAI 14.3-151. Based on the above, the applicant's responses, and DCD changes, RAIs 14.3-151 and 14.3-177 regarding the SDG FOSTS are resolved. The staff confirmed that the applicant incorporated the changes into DCD revision 5.

In DCD Tier 1, Revision 9, Section 2.13.4, and Table 2.13.4-2, the applicant provided the design descriptions and ITAAC for the SDGs, ADGs, and their supporting systems, including the FOSTSs. The staff finds that these ITAAC commit to verifying that the SDGs, ADGs, and their supporting systems, including the FOSTSs, are constructed and installed as described in DCD

Tier 2, Revision 9. Therefore, the staff finds that SDGs, ADGs, and their supporting systems, including, the FOSTSs, comply with the requirements of 10 CFR 52.47(b)(1).

The quality of the fuel oil for the SDGs and ADGs is addressed by the applicant committing to meet the fuel oil standards of ASTM D975 and the engine manufacturer. The staff finds this acceptable because the fuel quality standards will be based on the manufacturer's recommendations and on the same industry standard referenced by the staff in RG 1.137, "Fuel Oil Systems for Standby Diesel Generators," for safety-related diesel generators. With respect to fuel testing, the applicant stated in response to RAI 9.5-69 that periodic testing of the fuel will be part of the operating and maintenance procedures developed by COL applicants under COL Information Item 13.5-2-A. In Tier 2, Revision 9, Section 13.5.2, the applicant states that RTNSS systems are included in the scope of the operating and maintenance procedures. This is acceptable because it requires COL applicants to address fuel testing and inspection procedures that will be available for NRC review or inspection.

COL Information Item 9.5.4-2-A addresses corrosion protection of the underground portion of the storage tank and piping for the SDGs and ADGs. As stated in DCD Tier 2, Revision 9, Section 9.5.4.2, corrosion protection for any underground portions of the fuel oil system will be determined based on the material and its corrosion susceptibility. COL Information Item 9.5.4-2-A instructs COL applicants to describe the material and corrosion protection for the underground portion of the system including underground fuel oil storage tanks. In the response to RAI 9.5-69, the applicant stated that if portions of fuel oil storage tanks are underground they will have to comply with federal, state, and local laws for underground petroleum storage tanks, which include corrosion protection. The staff finds the provisions for corrosion protection acceptable because they ensure that corrosion protection will be included in site-specific designs and submitted in COL applications to the NRC for review.

DCD Tier 2, Revision 9, Section 14.2.8.1.37, provides the initial testing provisions associated with this review item. The staff reviewed DCD Tier 2, Revision 9, Section 9.5.4, against the guidance in SRP Section 14.3.7, and finds that no additional ITAAC are needed in connection with this section.

DCD Tier 2, Revision 9, Chapter 16, does not have any TS requirements associated with this review item. The staff reviewed DCD Tier 2, Revision 9, Section 9.5.4, against 10 CFR 50.36, and agrees that no TS are needed in connection with this section.

DCD Tier 2, Revision 9, Section 9.5.4.6 and Table 1.10-1, include the following:

- COL Information Item 9.5.4-1-A, "Fuel Oil Capacity," specifies that the COL applicant will establish procedural controls to ensure a minimum fuel oil capacity is maintained onsite at all times for both SDGs and ADGs.

- COL Information Item 9.5.4-2-A, "Protection of Underground Piping," specifies that COL applicants shall describe the material and corrosion protection for the underground portion of the FOSTS, which includes underground fuel oil storage tanks. If portions of fuel oil storage tanks are underground they will have to comply with federal, state, and local laws for underground petroleum storage tanks, which include corrosion protection.

The staff finds COL Action Item 9.5.4-1-A acceptable because it will ensure that a minimum fuel oil capacity maintained onsite at all times for SDGs and ADGs. The staff finds COL Action Item

9.5.4-2-A acceptable because it ensures that corrosion protection will be included in site-specific designs and submitted in COL applications to the NRC for review.

Section 13.5, of this report addresses the staff's evaluation of plant operating procedures including procedural controls to ensure a minimum fuel oil capacity maintained onsite for SDGs and ADGs. Section 22.0, of this report addresses the staff's evaluation regarding conformance of RTNSS systems with the guidelines of SECY-94-084.

9.5.4.4 *Conclusion*

The staff finds that the FOSTSs for SDGs and ADGs meet the guidelines of SRP Section 9.5.4, Revision 3. Based on the above, the staff finds that the FOSTS for SDGs and ADGs design is acceptable and meets the relevant requirements of GDC 2, 4, and 17, and of 10 CFR 52.47(b)(1).

9.5.5 Diesel Generator Jacket Cooling Water System

9.5.5.1 *Regulatory Criteria*

The staff reviewed the ESBWR SDG and ADG JCWS in accordance with SRP Section 9.5.5, Revision 3. The staff reviewed DCD Tier 2, Revision 9, Section 9.5.5. The staff also reviewed DCD Tier 1, Revision 9, Section 2.0, and other DCD Tier 2, Revision 9, sections noted below. The staff's acceptance of the JCWS is based on the design meeting the relevant requirements of the following regulations:

- GDC 2 requires, in part, that SSCs important to safety shall be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, and floods. Compliance with GDC 2 is based on meeting the guidance of Regulatory Position C.1 for the safety-related portions and Regulatory Position C.2 for the nonsafety-related portions of RG 1.29.

- GDC 4 requires in part that SSCs important to safety shall be designed to accommodate the effects of, and be compatible with, the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents.

- GDC 5 requires that SSCs important to safety shall not be shared among nuclear power units unless it can be shown that such sharing will not significantly impair their ability to perform their safety functions including, in the event of an accident in one unit, an orderly shutdown and cooldown of the remaining units.

- GDC 17 requires, in part, that an onsite electric power system and an offsite electric power system shall be provided to permit functioning of SSCs important to safety.

- GDC 44 requires, in part, that a system shall be provided to transfer heat from SSCs important to safety to an UHS.

- GDC 45 requires that the cooling water system shall be designed to permit appropriate periodic inspection of important components, such as heat exchangers and piping, to ensure the integrity and capability of the system.

- GDC 46 requires, in part, that the cooling water system shall be designed to permit appropriate pressure and functional testing.

- 10 CFR 52.47(b)(1) requires that a design certification DC application contains the proposed ITAAC that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design certification is built and will operate in accordance with the design certification, the provisions of the Atomic Energy Act, and the NRC's regulations.

9.5.5.2 *Summary of Technical Information*

There are two redundant onsite seismic Category II SDG units in the ESBWR design for a single unit plant to provide power to nonsafety-related ac loads in the event of a loss of normal and preferred ac power supplies. Each SDG unit is an independent system complete with its support systems, which are the FOSTS, JCWS, starting air system, lubrication system, and combustion air intake and exhaust system. Each SDG unit is housed in a separate seismic Category II structure. The design provides adequate separation between the two SDG units, including their support systems, so that failure in one SDG does not result in loss of function of the other SDG.

In addition, in DCD Revision 5, the applicant revised the design of the onsite ac power supply system by adding two ADGs to provide 480 V ac power to power the post-72 hour power loads following an extended loss of all other ac power sources. Each ADG unit, which is an independent system provided as a complete skid-mounted package with its integral support systems, is housed in a separate seismic Category II structure. The design provides adequate separation between the two ADG units so that failure in one ADG does not result in loss of function of the other ADG.

SDGs, ADGs, and their support systems are nonsafety-related, and non-Class 1E electrical systems and have no safety-related design basis. However, they are relied upon to be available to provide an ac power 72 hours after an abnormal event. Therefore, the SDGs, ADGs, and their supporting systems, including the JCWS, have RTNSS functions as supporting systems to provide power and are included in the plant ACM to ensure that they have sufficient reliability and availability to perform their RTNSS functions.

Standby Diesel Generator

Each of the two SDG units has its own independent, integrally mounted JCWS designed to maintain SDG operating temperature at full load. A self-contained closed-loop system circulates cooling water to the diesel engine, lube oil cooler, and various engine components to maintain system operating temperature. The jacket cooling water is cooled by a heat exchanger that rejects heat to the RCCWS. The JCWS includes a keep-warm circuit consisting of a temperature-controlled electric heater and a small motor-driven water circulating pump that maintains the jacket water in a warm standby condition to facilitate rapid starting.

The functionality of the SDG JCWS is tested and inspected in accordance with the ACM during scheduled operational testing of the overall engine. Instrumentation is provided to monitor cooling water temperatures, pressure, and head tank level. Instruments receive periodic calibration and inspection to verify their accuracy. During standby periods, the keep-warm feature of the engine water jacket cooling closed-loop system is checked at scheduled intervals to ensure that the water jackets are warm. The cooling water in the engine water jacket cooling

closed-loop system is sampled and analyzed at regular intervals and is treated, as necessary, to maintain the desired quality.

Ancillary Diesel Generator

As stated in the above, each of the two ADG units is provided as a complete skid-mounted package. Therefore, a separate JCWS beyond the cooling system provided integrally with the ADGs is not necessary.

9.5.5.3 *Staff Evaluation*

The staff reviewed the JCWS to determine if the design meets the relevant requirements of GDC 2. As stated above, the SDGs, ADGs, and their support systems are nonsafety-related and non-Class 1E electrical systems and have no safety-related design basis. However, the SDGs, ADGs, and their supporting systems, including the SDG JCWS, have RTNSS functions as supporting systems to provide an ac power 72 hours after an abnormal event. Based on its review as discussed in Sections 3.4.1, 3.5.1.1, 3.5.1.4, and 3.5.2 of this report, as described below, the staff finds that the SDG JCWS meets the relevant requirements of GDC 2 as it pertains to Regulatory Position C.2 of RG 1.29. The SDG JCWS also meets the relevant requirements of GDC 2 as it pertains to Regulatory Position C.1 of RG 1.29.

Section 3.4.1 of this report addresses the staff's evaluation of flood protection provided for RTNSS systems. Section 3.5.1.1 of this report addresses the staff's evaluation of protection provided for RTNSS systems from internally generated missiles outside containment. Section 3.5.1.4 of this report addresses the staff's evaluation of protection provided for RTNSS systems from missiles generated by natural phenomena. Section 3.5.2 of this report addresses the staff's evaluation of protection provided for RTNSS systems from externally generated missiles.

The staff reviewed the JCWS to determine if the design meets the relevant requirements of GDC 4. Based on its review as discussed in Section 3.6.1 of this report, the staff finds that the SDGs, ADGs, and their support systems, including the SDG JCWS are protected against the effects of, and are compatible with, the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents. The staff, therefore, finds that the SDG JCWS meets the requirements of GDC 4.

Section 3.6.1 of this report addresses the staff's evaluation of the design of structures, shields, and barriers for RTNSS systems, including the SDG JCWS, to be protected against dynamic effects of high-energy line breaks.

The ESBWR design is a single-unit station, and the requirements of GDC 5 are not applicable to the single-unit design.

Sections 8.2 and 8.3 of this report address the staff's evaluation of the ESBWR design with respect to the requirements of GDC 17 regarding the provision of an onsite electric power system and an offsite electric power system to permit functioning of SSCs important to safety.

The staff reviewed the JCWS to determine if the design meets the relevant requirements of GDC 44, 45, and 46. As stated in the above, each of the two SDG units has its own independent integrally mounted JCWS designed to maintain SDG operating temperature at full load. A self-contained, closed-loop system circulates cooling water to the diesel engine, lube oil cooler, and various engine components to maintain system operating temperature. The jacket

cooling water is cooled by a heat exchanger that rejects heat to the RCCWS. Heat removed from the RCCWS is rejected to the normal power heat sink or to the AHS.

Based on its review, the staff finds that the SDG JCWS meets the relevant requirements of GDC 44, 45 and 46, because the SDG JCWS is designed with the following considerations:

- Capability of transferring heat loads from SSCs to a heat sink under normal and accident conditions

- Component redundancy so the system remains functional assuming a single active failure coincident with a LOOP

- Capability to isolate components or piping so system function is not compromised

- Design provisions to permit inspection and operational testing of components and equipment

In addition, the SDGs and their supporting systems are included in the plant ACM which will ensure that they have sufficient reliability and availability to perform their RTNSS functions.

During the review of DCD Revision 0, the staff determined that during a postulated post-LOCA and a complete loss of ac power supplies, the SDG units were used to supply power for recharging batteries to support post-LOCA monitoring beyond 72 hours. Therefore, the SDGs and their support systems met Criteria B and C, established in SECY-94-084 for the passive ALWR plant design, to determine the systems that are candidates for RTNSS consideration. The staff determined that the SDGs and their supporting systems should be considered as candidates for RTNSS systems and should have ITAAC entries. Therefore, in RAI 19.1.0-2, the staff requested the applicant to provide documentation or analyses in support of the process used to identify RTNSS systems.

In response, to RAI 19.1.0-2 the applicant included the SDG units as RTNSS, Criterion C, systems. However, it was not clear to the staff that all of the SDG supporting systems would be considered as RTNSS systems. Subsequently, in RAI 22.5-4, the staff requested the applicant to clarify that all SDG supporting systems were considered as RTNSS systems. In response to RAI 22.5-4, the applicant stated that all SDG supporting systems, including SDG JCWS were considered as RTNSS systems. In DCD Tier 2, Revision 4, Section 9.5.5, the SDG JCWS was included and classified as an RTNSS system. The staff finds that the responses to RAIs 19.1.0-2 and 22.5-4 regarding the SDG JCWS are acceptable since the applicant clarified that all SDG supporting systems, including SDG JCWS were considered as RTNSS systems. Based on the above, the applicant's responses, and DCD changes, RAIs 19.1.0-2 and 22.5-4 regarding the SDG JCWS are resolved. RAI 22.5-4 was being tracked as a confirmatory item in the SER with open items. The staff confirmed that the applicant incorporated the changes into DCD Revision 4; therefore, the confirmatory item is closed.

The staff determined that the applicant should identify ITAAC for the SDG supporting systems since they are considered to be RTNSS systems. In RAI 14.3-151, the staff requested the applicant to include ITAAC entries for all of the SDG supporting systems. In response, to RAI 14.3-151 the applicant stated that it would not include ITAAC entries for SDG supporting systems because they were nonsafety-related systems. The applicant's response was not acceptable to the staff because SDG supporting systems had been reclassified as RTNSS systems and should have ITAAC entries. In RAI 14.3-151 S01, the staff again requested the

applicant to include ITAAC for all SDG supporting systems. RAI 14.3-151 was being tracked as an open item in the SER with open items. Subsequently, in DCD Tier 1, Revision 4, Table 2.13.4-2, the applicant included ITAAC for only two SDG supporting systems, the SDG FOSTS and the SDGSAS. The staff further issued RAI 14.3-177, to request the applicant to include ITAAC entries for all SDG supporting systems. Finally, in the response to RAI 14.3-177, the applicant committed to revising DCD Revision 5 to include an ITAAC entry for each of the SDG supporting systems. The staff finds that the response RAI 14.3-177 is acceptable since the applicant added ITAAC for all SDG supporting systems, which also addresses the concerns of RAI 14.3-151. Based on the above, the applicant's responses, and DCD changes, RAIs 14.3-151 and 14.3-177 regarding the SDG JCWS are resolved. The staff confirmed that the applicant incorporated the changes into DCD revision 5.

In DCD Tier 1, Revision 9, Section 2.13.4 and Table 2.13.4-2, the applicant provides the design descriptions and ITAAC for the SDGs, ADGs, and their supporting systems, including the SDG JCWS. The staff finds that these ITAAC commit to verifying that the SDG and ADG units and their supporting systems, including the SDG JCWS, are constructed and installed as described in DCD Tier 2, Revision 9. Therefore, the staff finds that the SDGs, ADGs, and their supporting systems, including the SDG JCWS, comply with the requirements of 10 CFR 52.47(b)(1).

DCD Tier 2, Revision 9, Section 14.2.8.1.37, provides the initial testing provisions associated with this review item. The staff reviewed DCD Tier 2, Revision 9, Section 9.5.5, against the guidance in SRP Section 14.3.7, and finds that no additional ITAAC are needed in connection with this section.

DCD Tier 2, Revision 9, Chapter 16, does not have any TS requirements associated with this review item. The staff reviewed DCD Tier 2, Revision 9, Section 9.5.5, against 10 CFR 50.36 and agrees that no TS are needed in connection with this section.

DCD Tier 2, Revision 9, Section 9.5.5.6 and Table 1.10-1, do not have COL information items for this section. The staff agrees that no COL information items are needed in connection with this section.

Section 22.0 of this report addresses the staff's evaluation regarding conformance of RTNSS systems with the guidelines of SECY-94-084.

9.5.5.4 *Conclusion*

Based on the above, the staff finds that the JCWS for the SDG design is acceptable and meets the relevant requirements of GDC 2, 4, 17, 44, 45, and 46 and 10 CFR 52.47(b)(1).

9.5.6 Diesel Generator Starting Air System

9.5.6.1 *Regulatory Criteria*

The staff reviewed the ESBWR SDGSAS in accordance with SRP Section 9.5.6, Revision 3. The staff reviewed DCD Tier 2, Revision 9, Section 9.5.6. The staff also reviewed DCD Tier 1, Revision 9, Section 2.0, and other DCD Tier 2, Revision 9, sections noted below. The staff's acceptance of the SDGSAS is based on the design's conformance with the requirements of the following regulations:

- GDC 2 requires in part that SSCs important to safety shall be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes and floods, Compliance with GDC 2 is based on meeting the guidance of Regulatory Position C.1 for the safety-related portions and Regulatory Position C.2 for the nonsafety-related portions of RG 1.29.

- GDC 4 requires in part that SSCs important to safety shall be designed to accommodate the effects of, and be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents.

- GDC 5 requires that SSCs important to safety shall not be shared among nuclear power units unless it can be shown that such sharing will not significantly impair their ability to perform their safety functions, including, in the event of an accident in one unit, an orderly shutdown and cooldown of the remaining units.

- GDC 17 requires in part that an onsite electric power system and an offsite electric power system shall be provided to permit functioning of SSCs important to safety.

- 10 CFR 52.47(b)(1) requires that a design certification application contains the proposed ITAAC that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design certification is built and will operate in accordance with the design certification, the provisions of the Atomic Energy Act, and the NRC's regulations.

9.5.6.2 *Summary of Technical Information*

There are two redundant onsite seismic Category II SDG units in the ESBWR design for a single unit plant to provide power to nonsafety-related ac loads in the event of a loss of normal and preferred ac power supplies. Each SDG unit is an independent system complete with its support systems, which are the FOSTS, JCWS, starting air system, lubrication system, and combustion air intake and exhaust system. Each SDG unit is housed in a separate seismic Category II structure. The design provides adequate separation between the two SDG units, including their support systems, so that failure in one SDG does not result in loss of function of the other SDG.

In addition, in DCD Revision 5, the applicant revised the design of the onsite ac power supply system by adding two ADGs to provide 480 V ac power to the post-72 hour power loads following an extended loss of all other ac power sources. Each ADG unit, which is an independent system provided as a complete skid-mounted package with its integral support systems, is housed in a separate seismic Category II structure. The design provides adequate separation between the two ADG units so that failure in one ADG does not result in loss of function of the other ADG.

SDGs, ADGs, and their support systems are nonsafety-related, and non-Class 1E electrical systems and have no safety-related design basis. However, they are relied upon to be available to provide an ac power 72 hours after an abnormal event. Therefore, the SDGs, ADGs, and their supporting systems, including SDGSAS, have RTNSS functions, as supporting systems to provide power and are included in the plant ACM, which will ensure that they have sufficient reliability and availability to perform their RTNSS functions.

Standby Diesel Generator

Each of the two SDG units is provided with its own dedicated SDGSAS which consists of two redundant 100-percent capacity air compressors, an air receiver, a 100-percent capacity air dryer, associated piping, and valves.

Periodic tests and inspections are performed in accordance with the ACM on the following:

- Air receiver pressure control switches
- Low pressure alarm signal for low receiver pressure
- Engine air start valves and the admission line vent valve
- Pressure gages on the receivers
- Air receivers to clear accumulated moisture using the blowdown connection as necessary
- Air quality – oil, particulates, and dew point

Ancillary Diesel Generator

Each of the two ADG units is provided as a complete skid-mounted package. The ADGs are started via an electrical system provided integrally with the ADGs. Thus, a starting air system is not required for the ADGs.

9.5.6.3 *Staff Evaluation*

The staff reviewed the SDGSAS to determine if the design meets the relevant requirements of GDC 2. As stated above, the SDGs, ADGs, and their support systems are nonsafety-related and non-Class 1E electrical systems, and have no safety-related design basis. However, the SDGs, ADGs, and their supporting systems, including the SDGSAS, have RTNSS functions, as supporting systems to provide ac power 72 hours after an abnormal event. Based on its review as discussed in Section 3.4.1, 3.5.1.1, 3.5.1.4, and 3.5.2 of this report, as described below, the staff finds that the SDGSAS meets the relevant requirements of GDC 2 as it pertains to Regulatory Position C.2 of RG 1.29. The SDGSAS also meet the relevant requirements of GDC 2 as it pertains to Regulatory Position C.1 of RG 1.29.

Section 3.4.1 of this report addresses the staff's evaluation of flood protection provided for RTNSS systems. Section 3.5.1.1 of this report addresses the staff's evaluation of protection provided for RTNSS systems from internally generated missiles outside containment. Section 3.5.1.4 of this report addresses the staff's evaluation of protection provided for RTNSS systems from missiles generated by natural phenomena. Section 3.5.2 of this report addresses the staff's evaluation of protection provided for RTNSS systems from externally generated missiles.

The staff reviewed the SDGSAS to determine if the design meets the relevant requirements of GDC 4. Based on its review as discussed in Section 3.6.1 of this report, the staff finds that the SDGs, ADGs, and their support systems, including the SDGSAS, are protected against the effects of, and are compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents. Section 3.6.1 of this report addresses the staff's evaluation of the design of structures, shields, and barriers necessary for RTNSS to be protected against dynamic effects of high-energy line breaks. The staff, therefore, finds that the SDGSAS meet the requirements of GDC 4.

The ESBWR design is a single-unit station, and the requirements of GDC 5 are not applicable to the single-unit design.

Sections 8.2 and 8.3 of this report address the staff's evaluation of the ESBWR design in accordance with the requirements of GDC 17 regarding the provision of an onsite electric power system and an offsite electric power system to permit functioning of SSCs important to safety.

During the review of the DCD revision 0, the staff determined that during a postulated post-LOCA and a complete loss of ac power supplies, the SDG units were used to supply power for recharging batteries to support post-LOCA monitoring beyond 72 hours. Therefore, the SDGs and their support systems met Criteria B and C established in SECY-94-084 for the passive ALWR plant design to determine the systems that are candidates for RTNSS consideration. The staff determined that the SDGs and their supporting systems should be considered as candidates for RTNSS systems and should have ITAAC entries. Therefore in RAI 19.1.0-2, the staff requested the applicant to provide documentation or analyses in support of the process used to identify RTNSS systems.

In response to RAI 19.1.0-2, the applicant included the SDG units as RTNSS, Criterion C, systems. However, it was not clear to the staff that all the SDG supporting systems would be considered as RTNSS systems. In RAI 22.5-4, the staff requested the applicant to clarify that all SDG supporting systems were considered as RTNSS systems. In response, to RAI 22.5-4, the applicant stated that all SDG supporting systems, including SDGSAS were considered as RTNSS systems. In DCD Tier 2, Revision 4, Section 9.5.6, the SDGSAS was included and classified as an RTNSS system. The staff finds that the responses to RAIs 19.1.0-2 and 22.5-4 regarding the SDGSAS are acceptable since the applicant clarified that all SDG supporting systems, including SDGSAS were considered as RTNSS systems. Based on the applicant's response and DCD changes, RAIs 19.1.0-2 and 22.5-4 regarding the SDGSAS are resolved. RAI 22.5-4 was being tracked as a confirmatory item in the SER with open items. The staff confirmed that the applicant incorporated the changes into DCD Revision 4 and the confirmatory item is closed.

The staff determined that the applicant should identify ITAAC for the SDG supporting systems since are considered to be RTNSS systems. In RAI 14.3-151, the staff requested request the applicant to include ITAAC for all of the SDG supporting systems. In response to RAI 14.3-151, the applicant stated that it would not include ITAAC for SDG supporting systems because they were nonsafety-related systems. The applicant's response was not acceptable to the staff because the SDG supporting systems had been reclassified as RTNSS systems. In RAI 14.3-151 S01, the staff again requested the applicant to include ITAAC for all SDG supporting systems. RAI 14.3-151 was being tracked as an open item in the SER with open items. Subsequently, in DCD Tier 1, Revision 4, Table 2.13.4-2, the applicant included ITAAC for only two SDG supporting systems, the SDGFOSTS and the SDGSAS. The staff further issued RAI 14.3-177, to request the applicant to include ITAAC for all SDG supporting systems. Finally, in the response to RAI 14.3-177, the applicant committed to revising DCD Revision 5 to include an ITAAC entry for each of the SDG supporting systems. The staff finds that the response to RAI 14.3-177 is acceptable since the applicant added ITAAC for all SDG supporting systems, which also addresses the concerns of RAI 14.3-151. Based on the above, the applicant's responses, and DCD changes, RAIs 14.3-151 and 14.3-177 regarding the SDGSAS are resolved. The staff confirmed that the applicant incorporated the changes into DCD revision 5.

In DCD Tier 1, Revision 9, Section 2.13.4 and Table 2.13.4-2, the applicant provided the design descriptions and ITAAC for the SDGs, ADGs, and their supporting systems, including the SDGSAS. The staff finds that these ITAAC commit to verifying that the SDGs, ADGs, and their supporting systems, including the SDGSAS, are constructed and installed as described in DCD

Tier 2, Revision 9,. Therefore, the staff finds that SDGs and ADGs, and their supporting systems, including the SDGSAS, comply with the requirements of 10 CFR 52.47(b)(1).

DCD Tier 2, Revision 9,, Section 14.2.8.1.37, provides the initial testing provisions associated with this review item. The staff reviewed DCD Tier 2, Revision 9,, Section 9.5.6, against the guidance in SRP Section 14.3.7 and finds that no additional ITAAC are needed in connection with this section.

DCD Tier 2, Revision 9,, Chapter 16, does not have any TS requirements associated with this review item. The staff reviewed DCD Tier 2, Revision 9,, Section 9.5.6, against 10 CFR 50.36 and agrees that no TS are needed in connection with this section.

DCD Tier 2, Revision 9,, Section 9.5.6.6 and Table 1.10-1, do not have COL information items for this section. The staff agrees that no COL information items are needed in connection with this section.

Section 22.0 of this report addresses the staff's evaluation regarding conformance of RTNSS systems with the guidelines of the SECY-94-084.

9.5.6.4 *Conclusion*

Based on the above, the staff finds that the SDGSAS design is acceptable and meets the relevant requirements of GDC 2, 4, and 17, and 10 CFR 52.47(b)(1).

9.5.7 Diesel Generator Lubrication System

9.5.7.1 *Regulatory Criteria*

The staff reviewed the ESBWR SDG lubrication system (SDGLS) in accordance with SRP Section 9.5.7, Revision 3. The staff reviewed DCD Tier 2, Revision 9, Section 9.5.7. The staff also reviewed DCD Tier 1, Revision 9, Section 2.0, and other DCD Tier 2, Revision 9, sections noted below. The staff's acceptance of the SDGLS is based on the design's conformance with the requirements of the following regulations:

- GDC 2 requires, in part, that SSCs important to safety shall be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, and floods. Compliance with GDC 2 is based on meeting the guidance of Regulatory Position C.1 for the safety-related portions and Regulatory Position C.2 for the nonsafety-related portions of RG 1.29.

- GDC 4 requires, in part, that SSCs important to safety shall be designed to accommodate the effects of, and be compatible with, the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents.

- GDC 5 requires that SSCs important to safety not be shared among nuclear power units unless it can be shown that such sharing will not significantly impair their ability to perform their safety functions including, in the event of an accident in one unit, an orderly shutdown and cooldown of the remaining units.

- GDC 17 requires, in part, that an onsite electric power system and an offsite electric power system shall be provided to permit functioning of SSCs important to safety.

- In 10 CFR 52.47(b)(1), the NRC requires that a design certification application contain the proposed ITAAC that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design certification is built and will operate in accordance with the design certification, the provisions of the Atomic Energy Act, and the NRC's regulations.

9.5.7.2 *Summary of Technical Information*

There are two redundant onsite seismic Category II SDG units in the ESBWR design for a single-unit plant to provide power to nonsafety-related ac loads in the event of a loss of normal and preferred ac power supplies. Each SDG unit is an independent system complete with its support systems, which are the FOSTS, JCWS, starting air system, lubrication system, and combustion air intake and exhaust system. Each SDG unit is housed in a separated seismic Category II structure. The design provides adequate separation between the two SDG units, including their support systems, so that failure in one SDG does not result in loss of function of the other SDG.

In addition, in ESBWR DCD Revision 5, the applicant revised the design of the onsite ac power supply system by adding two ADGs to provide 480 Vac power to power the post-72 hour power loads following an extended loss of all other ac power sources. Each ADG unit, which is an independent system provided as a complete skid-mounted package with its integral support systems, is housed in a separate seismic Category II structure. The design provides adequate separation between the two ADG units so that failure in one ADG does not result in loss of function of the other ADG.

SDGs, ADGs, and their support systems are nonsafety-related and non-Class 1E electrical systems, and have no safety-related design basis. However, they are relied upon to be available to provide ac power 72 hours after an abnormal event. Therefore, the SDGs, ADGs, and their supporting systems, including SDGLS, have RTNSS functions, as supporting systems to provide power and are included in the plant ACM, which will ensure that they have sufficient reliability and availability to perform their RTNSS functions.

Standby Diesel Generator

Each of the two SDGs is equipped with its own dedicated lubrication system, which includes a lube oil sump tank, circulating pump, filter elements, and a cooler. The subsystems, including lubrication system, associated with each SDG engine are independent and separated from the subsystems associated with the other SDG engine. Their failures do not lead to the failure of any SSCs important to safety.

The functionality of the SDGLS is tested and inspected in accordance with the ACM during scheduled operational testing of the overall engine. Instrumentation is provided to monitor lube oil temperature, pressure, and sump level, ensuring proper operation of the system. During standby periods, the keep-warm system is checked at scheduled intervals to ensure that the oil is warm. The lube oil is periodically sampled and analyzed to ensure quality.

Ancillary Diesel Generator

Each of the two ADGs is provided as a complete skid-mounted package. A separate lubrication system, beyond that provided integrally with the ADGs, is not required.

9.5.7.3 *Staff Evaluation*

The staff reviewed the SDGLS to determine if the design meets the relevant requirements of GDC 2. As stated in the above, the SDGs, ADGs, and their support systems are nonsafety-related and non-Class 1E electrical systems, and have no safety-related design basis. However, the SDGs, ADGs, and their supporting systems including the SDGLS have RTNSS functions, as supporting systems to provide ac power 72 hours after an abnormal event. Based on its review as discussed in Sections 3.4.1, 3.5.1.1, 3.5.1.4, and 3.5.2 of this report, as described below, the staff finds that the SDGLS meets the relevant requirements of GDC 2 as it pertains to Regulatory Position C.2 of RG 1.29. The SDGLS also meet the relevant requirements of GDC 2 as it pertains to Regulatory Position C.1 of RG 1.29.

Section 3.4.1 of this report addresses the staff's evaluation of flood protection provided for RTNSS systems. Section 3.5.1.1 of this report addresses the staff's evaluation of protection provided for RTNSS systems from internally generated missiles outside containment. Section 3.5.1.4 of this report addresses the staff's evaluation of protection provided for RTNSS systems from missiles generated by natural phenomena. Section 3.5.2 of this report addresses the staff's evaluation of protection provided for RTNSS systems from externally generated missiles.

The staff reviewed the SDGLS to determine if the design meets the relevant requirements of GDC 4. Based the staff's evaluation in Section 3.6.1 of this report, the staff finds that the SDGs, the ADGs, and their support systems, including the SDGLS, are protected against the effects of, and are compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents. Section 3.6.1 of this report addresses the staff's evaluation of the design of structures, shields, and barriers necessary for RTNSS to be protected against dynamic effects of high-energy line breaks. The staff, therefore, finds that the SDGLS meets the requirements of GDC 4.

The ESBWR design is a single-unit station, and the requirements of GDC 5 are not applicable to the single-unit design.

Sections 8.2 and 8.3 of this report address the staff's evaluation of the ESBWR design with respect to the requirements of GDC 17 regarding the provision of an onsite electric power system and an offsite electric power system to permit functioning of SSCs important to safety.

During the review of DCD Revision 0, the staff determined that, during a postulated post-LOCA and a complete loss of ac power supplies, the SDG units were used to supply power for recharging batteries to support post-LOCA monitoring beyond 72 hours. Therefore, the SDGs and their support systems met Criteria B and C established in SECY-94-084 for the passive ALWR plant design to determine the systems that are candidates for RTNSS consideration. The staff finds that the SDGs and their supporting systems should be considered as candidates for RTNSS systems and should have ITAAC entries. Therefore, in RAI 19.1.0-2, the staff requested the applicant to provide documentation or analyses in support of the process used to identify RTNSS systems.

In response to RAI 19.1.0-2, the applicant included the SDG units as RTNSS, Criterion C, systems. However, it was not clear to the staff that all of the SDG supporting systems would be considered as RTNSS systems. Subsequently, in RAI 22.5-4, the staff requested the applicant to clarify that all SDG supporting systems were considered as RTNSS systems. In response to RAI 22.5-4, the applicant stated that all SDG supporting systems, including SDGLS, were considered as RTNSS systems. In DCD Tier 2, Revision 4, Section 9.5.7, the SDGLS was

included and classified as an RTNSS system. The staff finds that the responses to RAIs 19.1.0-2 and 22.5-4 regarding the SDGLS are acceptable since the applicant clarified that all SDG supporting systems, including SDGLS, were considered as RTNSS systems. Based on the above, the applicant's responses, and DCD changes, RAIs 19.1.0-2 and 22.5-4 regarding the SDGLS are resolved. RAI 22.5-4 was being tracked as a confirmatory item in the SER with open items. The staff confirmed that the applicant incorporated the changes into DCD Revision 4; therefore, the confirmatory item is closed.

The staff determined that the applicant should identify ITAAC for the SDG supporting systems since are considered to be RTNSS systems. In RAI 14.3-151, the staff requested the applicant to include ITAAC for all of the SDG supporting systems. In response to RAI 14.3-151, the applicant stated that it would not include ITAAC for SDG supporting systems because they were nonsafety-related systems. The applicant's response was not acceptable to the staff because the SDG supporting systems had been reclassified as RTNSS systems. In RAI 14.3-151 S01, the staff again requested the applicant to include ITAAC for all SDG supporting systems. RAI 14.3-151 was being tracked as an open item in the SER with open items. Subsequently, in DCD Tier 1, Revision 4, Table 2.13.4-2, the applicant included ITAAC for only two SDG supporting systems, the SDGFOSTS and the SDGSAS. The staff further issued RAI 14.3-177, to request the applicant to include ITAAC for all SDG supporting systems. Finally, in response to RAI 14.3-177, the applicant committed to revising DCD Revision 5 to include an ITAAC entry for each of the SDG supporting systems. The staff finds that the response to RAI 14.3-177 is acceptable since the applicant added ITAAC for all SDG supporting systems, which also addresses the concerns of RAI 14.3-151. Based on the above, the applicant's responses, and DCD changes, RAIs 14.3-151 and 14.3-177 regarding the SDGLS are resolved. The staff confirmed that the applicant incorporated the changes into DCD revision 5.

In DCD Tier 1, Revision 9, Section 2.13.4, and Table 2.13.4-2, the applicant provides the design descriptions and ITAAC for the SDGs, ADGs, and their supporting systems, including the SDGLS. The staff finds that these ITAAC commit to verifying that the SDGs, ADGs, and their supporting systems, including the SDGLS, are constructed and installed as described in DCD Tier 2, Revision 9,. Therefore, the staff finds that SDGs, ADGs, and their supporting systems, including the SDGLS, comply with the requirements of 10 CFR 52.47(b)(1).

DCD Tier 2, Revision 9, Section 14.2.8.1.37, provides the initial testing provisions associated with this review item. The staff reviewed DCD Tier 2, Revision 9, Section 9.5.7, against the guidance in SRP Section 14.3.7 and finds that no additional ITAAC are needed in connection with this section.

DCD Tier 2, Revision 9, Chapter 16, does not have any TS requirements associated with this review item. The staff reviewed DCD Tier 2, Revision 9, Section 9.5.7, against 10 CFR 50.36 and agrees that no TS are needed in connection with this section.

DCD Tier 2, Revision 9, Section 9.5.7.6 and Table 1.10-1, do not have COL information items for this section. The staff agrees that no COL information items are needed in connection with this section.

Section 22.0 of this report addresses the staff's evaluation of RTNSS systems in conformance with the guidance of SECY-94-084.

9.5.7.4 *Conclusion*

Based on the above, the staff finds that the lubrication systems for the SDGs and ADGs designs are acceptable and meet the relevant requirements of GDC 2, 4, and 17, and 10 CFR 52.47(b)(1).

9.5.8 Diesel Generator Combustion Air Intake and Exhaust System

9.5.8.1 *Regulatory Criteria*

The staff reviewed the ESBWR diesel generator combustion air intake and exhaust system (DGCAIES) in accordance with SRP Section 9.5.8, Revision 3. The staff reviewed DCD Tier 2, Revision 9, Section 9.5.8. The staff also reviewed DCD Tier 1, Revision 9, Section 2.0, and other DCD Tier 2 sections noted below. The staff's acceptance of the DGCAIES is based on the design meeting the relevant requirements of the following regulations:

- GDC 2 requires, in part, that SSCs important to safety shall be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, and floods. Compliance with GDC 2 is based on meeting the guidance of Regulatory Position C.1 for the safety-related portions and Regulatory Position C.2 for the nonsafety-related portions of RG 1.29.

- GDC 4 requires, in part, that SSCs important to safety shall be designed to accommodate the effects of, and be compatible with, the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents.

- GDC 5 requires that SSCs important to safety not be shared among nuclear power units unless it can be shown that such sharing will not significantly impair their ability to perform their safety functions including, in the event of an accident in one unit, an orderly shutdown and cooldown of the remaining units.

- GDC 17 requires in part that an onsite electric power system and an offsite electric power system shall be provided to permit functioning of SSCs important to safety.

- In 10 CFR 52.47(b)(1), the NRC requires that a design certification application contain the proposed ITAAC that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design certification is built and will operate in accordance with the design certification, the provisions of the Atomic Energy Act, and the NRC's regulations.

9.5.8.2 *Summary of Technical Information*

There are two redundant onsite seismic Category II SDG units in the ESBWR design for a single unit plant to provide power to nonsafety-related ac loads in the event of a loss of normal and preferred ac power supplies. Each SDG unit is an independent system complete with its support systems, which are the FOSTS, JCWS, starting air system, lubrication system, and combustion air intake and exhaust system. Each SDG unit is housed in a separate seismic Category II structure. The design provides adequate separation between the two SDG units, including their support systems, so that failure in one SDG does not result in loss of function of the other SDG.

In addition, in DCD Revision 5, the applicant revised the design of the onsite ac power supply system by adding two ADGs to provide 480 V ac power to meet the post-72 hour power loads following an extended loss of all other ac power sources. Each ADG unit, which is an independent system provided as a complete skid-mounted package with its integral support systems, is housed in a separated seismic Category II structure. The design provides adequate separation between the two ADG units so that failure in one ADG does not result in loss of function of the other ADG.

The SDGs, ADGs and their support systems are nonsafety-related and non-Class 1E electrical systems, and have no safety-related design basis. However, they are relied upon to be available to provide ac source of power 72 hours after an abnormal event. Therefore, the SDGs, ADGs and their supporting systems, including the DGCAIES, have RTNSS functions as supporting systems to provide power and are included in the plant ACM, which will ensure that they have sufficient reliability and availability to perform their RTNSS functions.

Standby Diesel Generator

Each of the two SDGs is equipped with its own dedicated DGCAIES which is designed to supply combustion air to the SDG engine and to exhaust combustion products out of the SDG to the atmosphere. It includes intake and exhaust silencers to quiet engine operation. The subsystems, including DGCAIES, associated with each SDG engine are independent and separated from the subsystems associated with the other SDG engine. Their failures do not lead to the failure of any SSCs important to safety.

Visual inspection of the DGCAIES is performed concurrently with regularly scheduled SDG testing and inspection, which are performed in accordance with the ACM. Inspection of the integrity of the ducting and joints, filter condition, and intake and exhaust silencer condition is also included in SDG maintenance procedures.

Ancillary Diesel Generator

Each of the two ADGs is provided as a complete skid-mounted package. A separate combustion air intake and exhaust system beyond that provided integrally with the ADGs is not required.

9.5.8.3 *Staff Evaluation*

The staff reviewed the DGCAIES to determine if the design meets the relevant requirements of GDC 2. As stated in the above, the SDGs, ADGs, and their support systems are nonsafety-related and non-Class 1E electrical systems, and have no safety-related design basis. However, the SDGs, ADGs, and their supporting systems, including the DGCAIES, have RTNSS functions as supporting systems to provide ac power 72 hours after an abnormal event. Based on its review as discussed in Sections 3.4.1, 3.5.1.1, 3.5.1.4, and 3.5.2 of this report, as described below, the staff finds that the DGCAIES meets the relevant requirements of GDC 2 as it pertains to Regulatory Position C.2 of RG 1.29. The DGCAIES also meet the relevant requirements of GDC 2 as it pertains to Regulatory Position C.1 of RG 1.29.

Section 3.4.1 of this report addresses the staff’s evaluation of flood protection provided for RTNSS systems. Section 3.5.1.1 of this report addresses the staff’s evaluation of protection provided for RTNSS systems from internally generated missiles outside containment. Section 3.5.1.4 of this report addresses the staff’s evaluation of protection provided for RTNSS systems

from missiles generated by natural phenomena. Section 3.5.2 of this report addresses the staff's evaluation of protection provided for RTNSS systems from externally generated missiles.

The staff reviewed the DGCAIES to determine if the design meets the relevant requirements of GDC 4. Based on the staff's evaluation in Section 3.6.1 of this report, the staff finds that the SDGs, the ADGs and their support systems, including the DGCAIES, are protected against the effects of, and are compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents. Section 3.6.1 of this report addresses the staff's evaluation of the design of structures, shields, and barriers necessary for RTNSS to be protected against dynamic effects of high-energy line breaks. The staff, therefore, finds that the DGCAIES meets the requirements of GDC 4.

The ESBWR design is a single-unit station, and the requirements of GDC 5 are not applicable to the single-unit design.

Sections 8.2 and 8.3 of this report address the staff's evaluation of the ESBWR design with respect to the requirements of GDC 17 regarding the provision of an onsite electric power system and an offsite electric power system to permit functioning of SSCs important to safety.

During the review of DCD Revision 0, the staff determined that during a postulated post-LOCA and a complete loss of ac power supplies, the SDG units were used to supply power for recharging batteries to support post-LOCA monitoring beyond 72 hours. Therefore, the SDGs and their support systems met Criteria B and C established in SECY-94-084 for the passive ALWR plant design to determine the systems that are candidates for RTNSS consideration. The staff determined that they should be considered as candidates for RTNSS systems and should have ITAAC entries. Therefore, in RAI 19.1.0-2, the staff requested the applicant to provide documentation or analyses in support of the process used to identify RTNSS systems.

In response to RAI 19.1.0-2, the applicant included the SDG units as RTNSS, Criterion C, systems. However, it was not clear to the staff that all of the SDG supporting systems would be considered as RTNSS systems. In RAI 22.5-4, the staff requested the applicant to clarify that all SDG supporting systems were considered as RTNSS systems. In response to RAI 22.5-4, the applicant stated that all SDG supporting systems, including the DGCAIES, were considered as RTNSS systems. In DCD Tier 2, Revision 4, Section 9.5.8, the DGCAIES was included and classified as an RTNSS system. The staff finds that the responses to RAIs 19.1.0-2 and 22.5-4 regarding the DGCAIES are acceptable since the applicant clarified that all SDG supporting systems, including the DGCAIES, were considered as RTNSS systems. Based on the above, the applicant's responses, and DCD changes, RAIs 19.1.0-2 and 22.5-4 regarding the DGCAIES are resolved. RAI 22.5-4 was being tracked as a confirmatory item in the SER with open items. The staff confirmed that the applicant incorporated the changes into DCD Revision 4; therefore, the confirmatory item is closed.

The staff determined that the applicant should identify ITAAC for the SDG supporting systems since are considered to be RTNSS systems. In RAI 14.3-151, the staff requested request the applicant to include ITAAC for all of the SDG supporting systems. In response to RAI 14.3-151, the applicant stated that it would not include ITAAC for SDG supporting systems because they were nonsafety-related systems. The applicant's response was not acceptable to the staff because the SDG supporting systems had been reclassified as RTNSS systems. In RAI 14.3-151 S01, the staff again requested the applicant to include ITAAC for all SDG supporting systems. RAI 14.3-151 was being tracked as an open item in the SER with open items. Subsequently, in DCD Tier 1, Revision 4, Table 2.13.4-2, the applicant included ITAAC for only

two SDG supporting systems, the SDGFOSTS and the SDGSAS. The staff further issued RAI 14.3-177, to request the applicant to include ITAAC for all SDG supporting systems. Finally, in response to RAI 14.3-177, the applicant committed to revising DCD Revision 5 to include an ITAAC entry for each of the SDG supporting systems. The staff finds that the response to RAI 14.3-177 is acceptable since the applicant added ITAAC for all SDG supporting systems, which also addresses the concerns of RAI 14.3-151. Based on the above, the applicant's responses, and DCD changes, RAIs 14.3-151 and 14.3-177 regarding the DGCAIES are resolved. The staff confirmed that the applicant incorporated the changes into DCD revision 5.

In DCD Tier 1, Revision 9, Section 2.13.4 and Table 2.13.4-2, the applicant provides the design descriptions and ITAAC for the SDGs, ADGs, and their supporting systems. The staff finds that these ITAAC commit to verifying that the SDGs, ADGs, and their supporting systems, including the DGCAIES, are constructed and installed as described in DCD Tier 2, Revision 9. Therefore, the staff finds that SDGs, ADGs, and their supporting systems, including the DGCAIES, comply with the requirements of 10 CFR 52.47(b)(1).

DCD Tier 2, Revision 9, Section 14.2.8.1.37, provides the initial testing provisions associated with this review item. The staff reviewed DCD Tier 2, Revision 9, Section 9.5.8, against the guidance in SRP Section 14.3.7 and finds that no additional ITAAC are needed in connection with this section.

DCD Tier 2, Revision 9, Chapter 16, does not have any TS requirements associated with this review item. The staff reviewed DCD Tier 2, Revision 9, Section 9.5.8, against 10 CFR 50.36 and agrees that no TS are needed in connection with this section.

DCD Tier 2, Revision 9, Section 9.5.8.6 and Table 1.10-1, do not have COL information items for this section. The staff agrees that no COL information items are needed in connection with this section.

Section 22.0 of this report addresses the staff's evaluation of RTNSS systems in conformance with the guidance of SECY-94-084.

9.5.8.4 *Conclusion*

Based on the above, the staff finds that the combustion air intake and exhaust systems for the SDG and ADG design is acceptable and meets the relevant requirements of GDC 2, 4, and 17 and 10 CFR 52.47(b)(1).

10.0 STEAM AND POWER CONVERSION SYSTEM

10.1 Summary Description

The design control document (DCD) Tier 2, Revision 9, Chapter 10, for the economic simplified boiling-water reactor (ESBWR) describes the steam and power conversion system. The components of this system are designed to produce electric power using the steam generated by the reactor, condense the steam into water, and return the water to the reactor as heated feedwater (FW), with a major portion of its gaseous, dissolved, and particulate impurities removed to maintain reactor water quality.

The steam and power conversion system includes the turbine main steam system (TMSS), main turbine generator, main condenser, condenser air removal system, turbine gland seal system (TGSS), turbine bypass system (TBS), condensate purification system, condensate and feedwater system (CFS), and circulating water system. The majority of the steam and power conversion system piping and components are located in the turbine building.

10.2 Turbine Generator

10.2.1 Regulatory Criteria

The U.S. Nuclear Regulatory Commission (NRC) evaluates the turbine generator design in accordance with the guidance provided in Sections 10.2 and 10.2.3 of NUREG–0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, (LWR Edition)," March 2007, (SRP). The design of the turbine generator is acceptable if it satisfies the applicable requirements specified in Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, "Domestic Licensing of Production and Utilization Facilities." Specifically, the design must meet the requirements of General Design Criterion (GDC) 4, "Environmental and dynamic effects design bases," in Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50, as it relates to the protection of structures, systems, and components (SSCs) important to safety from the effects of turbine missiles and ruptures of the low-pressure turbine exhaust hood connection joints to the main condenser. The staff also applies the guidance provided in SRP Section 3.5.1.3 relative to periodic inspection, testing, and maintenance of turbine steam admission and extraction nonreturn valves.

10.2.2 Turbine Generator Arrangement and Operational Considerations

This section evaluates the location, arrangement, and orientation of the main turbine, as well as operational considerations that prevent the main turbine from exceeding 120 percent of rated speed. The staff reviews this section in accordance with the guidance in SRP Section 10.2, using the regulatory criteria discussed in Section 10.2.1 above.

10.2.2.1 *Summary of Technical Information*

DCD Tier 2, Revision 9, Section 10.2, describes the turbine generator. The turbine building contains the main turbine for the ESBWR, which consists of one high-pressure and three low-pressure turbine elements. In DCD Tier 2, Revision 9, Figures 10.3-1, 10.3-2, 10.4-6a, and 10.4-7a show the relative locations of the turbine and associated steam admission valves (i.e., stop, control, intermediate stop, and intercept valves) and extraction steam nonreturn

valves, while DCD Tier 2, Revision 9, Figure 3.5-2, shows the turbine orientation with respect to other SSCs.

The turbine generator control system (TGCS) uses a digital monitoring and control system that, in coordination with the turbine steam bypass and pressure control system (SB&PCS), controls the turbine speed, load, and steam flow for startup, normal operation, and transient conditions. During normal plant operation, the TGCS adjusts the positions of the turbine control and intercept valves to regulate reactor pressure, while the frequency of the electrical grid maintains the turbine speed. The stop valves and intermediate stop valves normally remain in their full open positions while the plant is operating. The TGCS is designed to accommodate a loss of generator load without initiating a turbine trip; it includes a primary (normal) turbine overspeed trip function and an emergency turbine overspeed trip function.

The TGCS provides redundancy by using separate turbine steam admission and extraction steam nonreturn isolation valves, speed sensors, circuitry, controllers, trip solenoid valves, hydraulic dump lines, and air dump valves for isolating steam flow to the main turbine. Diversity primarily extends to the design of the overspeed trip controllers and use of different valve types for performing the stop valve and control/intercept valve functions (although they are all hydraulically controlled).

10.2.2.2 *Staff Evaluation*

To satisfy GDC 4, and as discussed in Section 3.5.1.3 of this report, the main turbine should have a low probability of rotor failure to minimize the likelihood that turbine missiles will affect SSCs important to safety. As the turbine speed increases above its design limit of 120 percent of rated speed, the probability of rotor failure increases to the point where rotor failure ultimately occurs at its destructive overspeed limit (160 percent to 190 percent of rated speed). Therefore, the evaluation in Section 3.5.1.3 relies on the TGCS to ensure that turbine overspeed conditions that exceed 120 percent of rated speed are very unlikely. The staff’s evaluation in this section confirms that the TGCS is adequate in this regard and that SSCs important to safety are adequately protected from turbine missiles. The staff’s evaluation also confirms that steam released from a rupture of the connection joints between the low-pressure turbine elements and condenser will not adversely affect SSCs that are important to safety. The staff based its evaluation of the turbine generator on the information provided in Revision 9 of the DCD.

During its initial review of the turbine generator for the ESBWR, the NRC staff found that it needed additional information, primarily to address TGCS diversity considerations and the vulnerability of SSCs important to safety to turbine missiles and to provide a more complete description of the TGCS with respect to redundancy, single failure, and reliability considerations. The applicant provided additional information in response to the NRC’s request for additional information (RAI) 10.2-18 (including Supplements 1, 2, and 3) to address these considerations and included corresponding changes in Revision 7 of the DCD to reflect this additional information.

10.2.2.2.1 Design Considerations

A. Turbine Arrangement and Orientation

The staff reviewed the information referred to in Section 10.2.2 above and confirmed that it adequately described the turbine generator system, located in the turbine building. The turbine

stop, control, intermediate stop, intercept, and nonreturn valve arrangements are typical of other designs approved by the staff.

The arrangement of the main turbine should be such that a failure of the low-pressure turbine exhaust hood-to-condenser joint will not adversely affect essential SSCs. DCD Tier 2, Revision 9, Section 10.2.4, indicates that there are no essential systems or components (as defined in Branch Technical Position, SPLB 3-1) in the turbine area. Branch Technical Position SPLB 3-1 defines essential systems and components as “Systems and components required to shut down the reactor and mitigate the consequences of a postulated piping failure, without offsite power.” Because the ESBWR is a passive reactor, both non-safety-related SSCs that are designated as regulatory treatment of non-safety systems (RTNSS) Category B (as described in DCD Tier 2, Revision 9, Section 19.A) and safety-related SSCs that are necessary to shut down the reactor and mitigate the consequences of pipe failures are essential. As discussed in DCD Tier 2, Revision 9, Section 10.2.4, safety-related SSCs are not adversely affected by a failure of the low-pressure turbine exhaust hood-to-condenser joint. Based on the response to RAI 10.2-18, S02, the staff also confirmed that the turbine building contains no RTNSS Category B SSCs. Therefore, a failure of the low-pressure turbine exhaust hood-to-condenser joint will not adversely affect essential SSCs, and the NRC considers the turbine arrangement for the ESBWR to be acceptable in this regard.

Turbine orientation is an important consideration for the staff’s evaluation in Sections 3.5.1.3 and 3.5.2 of this report. As shown in DCD Tier 2, Revision 9, Figure 3.5-2, and discussed in DCD Tier 2, Revision 9, Sections 3.5.1.1, 10.2.1, and 10.2.4, the main turbine is oriented so that SSCs important to safety will not be adversely affected by turbine missiles generated within the low-trajectory turbine missile strike zone. As discussed in response to RAI 10.2-18, S01 and S02, and reflected in the DCD description, SSCs important to safety in this regard include those that are safety-related; those referred to in Regulatory Guide 1.115, “Protection Against Low-Trajectory Turbine Missiles,” Revision 1, and listed in Appendix A to Regulatory Guide 1.117, “Tornado Design Classification,” issued April 1978; and those designated as RTNSS Category B in DCD Tier 2, Revision 9, Section 19A, of the DCD. Note that, because the ESBWR is a passive plant, SSCs that are classified as RTNSS Category B are relied upon for post-72 hour heat removal following an accident and are therefore important to safety, requiring protection from the effects of turbine missiles.

In response to RAI 10.2-18 S01, the applicant indicated that the only SSCs important to safety that could be affected by low-trajectory turbine missiles include the condenser pressure transmitters, turbine bypass valve position sensors, and cabling connections to the reactor protection system. As discussed in the response, these items are fail–safe, and consequently the staff agrees that they will not be adversely affected by low-trajectory turbine missiles (i.e., the designated safety functions will not be compromised). Likewise, because the TGCS is also designed to be fail-safe, the staff finds that the TGCS turbine trip function will not be adversely affected by low-trajectory turbine missiles. Therefore, orientation of the main turbine is favorable with regard to SSCs that are important to safety, and review considerations in this and other parts of this report that pertain to favorably oriented turbines are applicable.

B. Turbine Speed Control and Overspeed Protection

The TGCS, in conjunction with the SB&PCS, is designed to maintain turbine speed and reactor pressure during normal plant operating and transient conditions, including during a loss of generator load. Abnormal conditions are annunciated in the control room.

Turbine speed is precluded from exceeding 120 percent of rated speed by the normal speed control function of the TGCS, along with its primary (normal) and emergency turbine overspeed trip functions. A turbine trip actuation by either the primary or emergency turbine overspeed trip circuit will dump hydraulic fluid from the actuators for the steam admission valves and dump air from the actuators for the extraction steam nonreturn isolation valves, causing them to close and isolate steam flow to the turbine. These trip functions are both electronic, and they are fully redundant and independent of each other, except that the turbine steam admission valves and extraction steam nonreturn isolation valves (including the air release flowpath and air dump valves) are shared. However, the shared components provide sufficient redundancy so that a single active failure will not compromise the turbine trip function. In addition to the control and turbine trip functions, the following TGCS design considerations are also important and pertinent to the staff's review:

- Operators can manually trip the main turbine from the control room and locally at the turbine. The manual trip circuits are hardwired and independent of the software interfaces for the primary and emergency trip functions.

- In addition to overspeed conditions, a turbine trip is also initiated to protect the main turbine from abnormal conditions, such as excessive turbine shaft vibration or low lube oil pressure.

- Spring-assisted nonreturn isolation valves are provided in those extraction steam lines that have sufficient energy to cause turbine speed to exceed 120 percent of rated speed. Redundancy is provided by the nonreturn valves, in that failure of a single nonreturn valve to close will not cause the turbine speed to exceed 120 percent of rated speed.

- The TGCS is fail-safe, in that most abnormal conditions will either cause the affected circuit to make up the logic for one of two signals that are needed to trip the turbine, or the condition will result in a turbine trip (such as a loss of hydraulic oil pressure).

Based on the above considerations, except for diversity (which is discussed in the following two paragraphs), the TGCS satisfies the review guidance specified by SRP Section 10.2, Paragraph III.2.A, and is acceptable in this regard.

As indicated in DCD Tier 2, Revision 9, Table 1.9-10, under SRP Section 10.2, and discussed in DCD Tier 2, Revision 9, Section 10.2.2.4, turbine overspeed trip protection for the ESBWR does not include a mechanical turbine overspeed trip device. Consequently, the level of diversity that is provided for turbine overspeed protection is somewhat reduced from that called for by SRP Section 10.2. Diversity minimizes the potential for common-cause and common-mode failures and tends to improve reliability. In RAI 10.2-18 S01, the staff asked the applicant to justify deviating from the criteria specified in SRP Section 10.2, in that the design did not include a mechanical overspeed trip device. This was identified as Open Item 10.2-18 in the safety evaluation report (SER) with open items.

Design attributes of a TGCS that do not include a completely separate and independent mechanical overspeed trip device that could cause the TGCS to be more susceptible to common-cause or common-mode failures include commonalities that exist between the primary and emergency trip functions of the digital control system; use of a single hydraulic oil or air drain or discharge flowpath for closing the turbine steam admission valves and extraction steam nonreturn valves (as applicable); and use of active components of the same design for both the primary and emergency trip functions. In evaluating the acceptability of the TGCS in this regard, the staff found the following considerations to be pertinent:

- The primary and emergency electronic overspeed trip functions are diverse to a large extent because they use unique hardware and logic design and implementation, as explained in the response to RAI 10.2-18, S03 (Item A.2).

- Because the TGCS is electronic, it includes extensive diagnostic routines that continually monitor it for abnormal conditions. Any problems that are identified are typically alarmed in the control room and result in a 1-of-3 trip (a 2-of-3 trip initiates a turbine trip).

- As discussed in DCD Tier 2, Revision 9, Sections 9.3.6, 9.3.7, and 10.2.2.4, the applicant has incorporated design provisions to address hydraulic oil and air system problems that can lead to common-cause and common-mode degradation mechanisms. This is important to ensure that the accumulation of impurities and corrosion products in hydraulic control and air systems (as applicable) does not prevent turbine steam admission valves and extraction steam nonreturn valves from closing.

- The hydraulic control system for the turbine steam admission valves provides multiple hydraulic oil return and (drain) paths, in addition to any needed vent paths; consequently, flow blockage in one hydraulic return or drain line will not prevent automatic isolation of main steam flow to the turbine following a turbine trip demand.

- While a single (shared) air discharge flowpath is used from the extraction steam nonreturn valve actuators to the parallel air dump solenoid valves, it is unlikely that this single flowpath will become plugged by impurities and corrosion products if it is installed properly and is not crimped to restrict air flow. Air quality problems tend to adversely affect air system components and components that are served by air systems, while air discharge flowpaths are typically unaffected. Design provisions for minimizing air quality problems (referred to in the third bullet above) make it even less likely that this single (shared) air discharge flowpath will become obstructed to the point where it adversely affects closure of the extraction steam nonreturn valves.

- While the turbine steam admission valves are hydraulic, they are diverse in other respects, consistent with designs previously approved by the staff.

- To some extent, both the primary and emergency turbine overspeed trip functions use the same types of components. For example, trip solenoid valves are the same, and the spring-assisted extraction steam nonreturn isolation valves are all of the same design. However, because all active components of the TGCS will be periodically inspected, tested, and maintained over time, in accordance with Combined License (COL) Information Item 10.2-1-A, incipient problems should be readily identified and corrected before they become vulnerable to common-cause or common-mode failures.

- Mechanical overspeed trip devices have periodically experienced problems, such as mechanical binding and spring failure that sometimes caused these devices to be unreliable. Unlike electronic overspeed trip circuits, the functionality of mechanical trip devices cannot be monitored continuously. Also, testing a mechanical trip device requires the normal speed control to be bypassed and turbine speed to be increased to the actual turbine overspeed trip setpoint. Should the mechanical trip device fail to function and the turbine continue to overspeed while a mechanical overspeed trip device is being tested, and if the electric overspeed trip device should also fail to function, operator action must be relied upon to manually trip the turbine. Electronic turbine overspeed trip circuits can be tested by

inserting test signals, without bypassing the turbine normal speed control circuit and increasing the turbine speed to the trip setpoint.

- The applicant evaluated the reliability of the TGCS for the ESBWR and determined that the proposed design is more reliable, by an order of magnitude, than previous designs that include mechanical overspeed trip protection.

Based on the above considerations, the staff found that using two independent electronic trip circuits to ensure that turbine speed will not exceed 120 percent of rated speed (as described in DCD Tier 2, Revision 9, Section 10.2.2.4) will not reduce the reliability of turbine overspeed protection below that provided by one electric trip circuit and a mechanical trip device. Also, the ability to continuously monitor the functional status of the TGCS and to perform turbine trip testing without bypassing the normal turbine speed control circuit and subjecting the turbine to overspeed conditions is a substantial improvement over designs that include a mechanical overspeed trip device. Therefore, the diversity provided by the TGCS, in conjunction with the other considerations referred to above, is sufficient to provide reliable overspeed trip protection for the main turbine. The staff considers the proposed deviation from SRP Section 10.2 to be acceptable and Open Item 10.2-18 is satisfactorily resolved.

Upon a loss of load condition, the normal speed control function of the TGCS is designed to limit turbine overspeed to at least 1 percent below the primary and emergency turbine overspeed trip setpoints of approximately 110 percent of rated speed by closing the control and intercept valves. The turbine trip setpoints for the primary and emergency overspeed trip functions are established so that, upon failure of the normal turbine speed control function, turbine speed is prevented from exceeding the design overspeed limit of 120 percent of rated speed by closing the turbine steam admission valves and extraction steam nonreturn isolation valves. The primary and emergency overspeed trip circuits use separate and independent sets of turbine rotor speed sensors, and the control signals from the emergency trip circuit are isolated from and independent of the control signals generated by the primary trip circuit. The turbine trip setpoints and corresponding basis are consistent with the review guidance specified by SRP Section 10.2, Paragraphs III.2.B, C, and D, and the staff considers them to be acceptable in this regard.

RAI 10.2-18 requested that the applicant provide the approximate percentages of rated turbine speed for turbine trip actuation. In response, the applicant indicated that COL applicants will establish turbine trip set points and bases, as specified by a COL information item that will be established for this purpose. The staff identified these (i.e., trip set points and bases) as separate confirmatory items in the safety evaluation report with open items. Based on a review of Revision 7 of the DCD, the staff confirmed that establishing the turbine trip set points and bases are factors in the turbine missile probability analysis that need to be addressed by COL applicants, as discussed in DCD Tier 2, Revision 7, Section 10.2.3.8, and specified by COL Information Item 10.2-2-A. Therefore, these confirmatory items are resolved.

C. Turbine Steam Admission and Extraction Steam Non-Return Isolation Valves

The primary and emergency turbine overspeed trip circuits actuate to close the turbine stop, control, intermediate stop, intercept, and spring-assisted extraction steam nonreturn isolation valves to prevent the turbine from exceeding its design overspeed limit of 120 percent of rated speed. The turbine stop and intermediate stop valves are diverse and redundant from their respective control and intercept valves, although both types are hydraulic. The valve arrangements are typical of designs previously approved by the staff. The turbine steam

admission valves and flowpaths are sized so that three of the four flowpaths can accommodate at least 85 percent of the rated steam flow to satisfy transient analysis considerations. As shown in Figures 10.2-1, 10.2-2, and 10.2-3, the applicant has established minimum allowed closure time limits for the turbine stop and control valves to satisfy reactor performance and transient analysis considerations. COL applicants will establish the maximum valve closure times for the turbine steam admission valves, as specified by COL Information Item 10.2-1-A. As discussed in DCD Tier 2, Revision 9, Section 10.2.2.2.6, the spring-assisted extraction steam nonreturn isolation valves will close within 2 seconds of tripping the air relay dump valves. The staff finds that the applicant has adequately addressed the considerations referred to in SRP Section III.3, and the design ensures that no single valve failure can disable or otherwise compromise the overspeed control function of the TGCS. Therefore, the staff considers the design to be acceptable in this regard.

D. Capability to Perform On-LineTesting

The staff reviewed the description of the TGCS to confirm that essential components can be tested while the turbine generator is operating. This capability is important to ensure that incipient problems are readily identified and corrected before they can lead to more serious common-cause or common-mode failure vulnerabilities, or otherwise compromise single-failure protection. As discussed in DCD Tier 2, Revision 9, Section Section 10.2.2.4, the primary and emergency turbine overspeed trip circuits and components can be tested while the turbine is operating. Also, as discussed in DCD Tier 2, Revision 9, Section Section 10.2.2.7, provisions are provided for testing the turbine steam admission valves, the spring-assisted extraction steam nonreturn isolation valves, and the air solenoid dump valves. Therefore, the design of the TGCS includes the capability to test those components that are essential to turbine overspeed protection while the plant is operating, and the staff considers the design to be acceptable in this regard.

E. Inservice Inspection, Testing, and Maintenance of Valves Essential for Turbine Overspeed Protection

Turbine overspeed protection relies upon the ability of the steam admission and extraction steam nonreturn isolation valves to function properly over the life of the plant. Consequently, programs for performing inservice inspection (ISI), testing, and maintenance of these valves need to be established as specified by COL information Item 10.2-1-A. DCD Tier 2, Revision 9, Sections 10.2.2.7 and 10.2.3.7, indicate that the following inspections will be performed in this regard:

- All of the turbine steam admission valves will be disassembled and visually inspected once during the first three refueling outages. Subsequent inspections will be performed as necessary to support the assumptions in the turbine missile probability analysis.

- The turbine steam admission valves will be exercised at least once within each calendar quarter, or as required, to support the assumptions in the turbine missile probability analysis and thus confirm acceptable performance.

- A seat tightness test for the turbine steam admission valves may be performed as required to confirm that valve leakage rates do not exceed assumptions for preventing the turbine speed from exceeding 120 percent of rated speed.

- Inspections of spring-assisted nonreturn valves will be inspected in accordance with vendor recommendations and will include seat-to-disk contact, binding, and other problems that could result in unacceptable performance.

- All valves of a functional type or size will be inspected for any unusual or detrimental condition that is identified during the inspection of any single valve of that size or type.

The staff noted that the information provided in DCD Tier 2, Sections 10.2.2.7 and 10.2.3.7 did not fully and adequately address assumptions with respect to valve performance and reliability considerations and the acceptance criteria specified by SRP Section 3.5.1.3. For example, the following review considerations from SRP Section 3.5.1.3 are pertinent in this regard:

- The frequency specified by the DCD for exercising valves once a quarter is not consistent with the SRP guidance which specifies a frequency of weekly.

- The provision specified in the DCD for dismantling and inspecting all steam admission valves at least once during the first three refueling outages will not necessarily satisfy the SRP guidance of completing this action for at least one valve of each type at intervals of approximately 3 years.

- To the extent that abnormal conditions or component failures do not insert a turbine trip signal for the affected circuit, allowed outage times need to be specified consistent with the guidance provided in the SRP.

However, irrespective of the information provided in DCD Tier 2, Revision 9, Sections 10.2.2.7 and 10.2.3.7, COL Information Item 10.2-1-A specifies that COL applicants need to provide a description of the turbine maintenance and inspection program necessary to establish inspection, testing, and maintenance provisions for the turbine valves (i.e., steam admission and extraction nonreturn valves) and control system sufficient to address performance and reliability considerations, including the criteria identified in Section II of SRP Section 3.5.1.3. Consequently, the staff's evaluation of COL Information Item 10.2-1-A will ensure that all aspects of the staff's review criteria and other considerations specified by this COL information item are adequately addressed. Therefore, the information provided in DCD Tier 2, Revision 9, Sections 10.2.2.7 and 10.2.3.7, as supplemented by COL Information Item 10.2-1-A, is acceptable.

10.2.2.2.2 Inspections, Tests, Analyses, and Acceptance Criteria

Applicants for standard plant design approval must provide inspections, tests, analyses, and acceptance criteria (ITAAC), in accordance with 10 CFR 52.47(b)(1) requirements. DCD Tier 1, Revision 9, Sections 2.11.1 and 2.11.4, provide design certification information and ITAAC for the TMSS and main turbine, respectively. Section 14.3.7 of this report evaluates DCD Tier 1 information for balance-of-plant SSCs, and the evaluation of DCD Tier 1 information in this section is an extension of the evaluation provided in Section 14.3.7. This evaluation pertains to plant systems aspects of the proposed DCD Tier 1 information for the main turbine.

The staff reviewed the DCD Tier 1 information in the sections referred to above to confirm that it included the appropriate DCD Tier 1 requirements were specified for the TGCS. In particular, the staff confirmed that functional arrangement drawings include the turbine steam admission valves and spring-assisted extraction steam nonreturn isolation valves; safety-related SSCs and additional SSCs that are designated RTNSS Category B (shown on DCD Tier 1, Revision 9,

Table 2.11.4-1) are not adversely affected by low-trajectory turbine missiles or by a rupture of the low-pressure turbine exhaust hood connection joint to the condenser; closure times of the turbine stop and control valves are limited, consistent with DCD Tier 2, Revision 9, Figures 10.2-1 through 10.2-3; the turbine is able to accommodate 85 percent of the rated steam flow through three control valves; and ISI and testing requirements for the turbine steam admission valves and spring-assisted extraction steam nonreturn isolation valves satisfy the specifications, considerations, and assumptions identified in the turbine missile probability analysis. Therefore, the staff considers the DCD Tier 1 information and ITAAC for the TGCS to be acceptable in this regard.

10.2.2.2.3 Initial Test Program

Section 14.2 of this report contains the evaluation of the initial test program for the ESBWR and is an extension of the evaluation provided in Section 14.2. The following initial test program specifications pertain to the turbine generator design considerations evaluated in Section 10.2.2.2.1 of this report:

- 14.2.8.1.53 Main Turbine Control System Preoperational Test
- 14.2.8.1.57 Extraction Steam System Preoperational Test
- 14.2.8.1.59 Main Turbine and Auxiliaries Preoperational Test
- 14.2.8.2.14 Plant Automation and Control Test
- 14.2.8.2.20 Turbine Valve Performance Test
- 14.2.8.2.27 Turbine Trip and Generator Load Rejection Test
- 14.2.8.2.33 Steam and Power Conversion System Performance Test

The staff reviewed the information provided in the test specifications referred to above to confirm that it adequately addressed the turbine generator design and performance considerations. In particular, the staff confirmed that the test program verifies proper performance and integrated operation of the main turbine and TGCS for normal operating and transient conditions. Therefore, the staff considers the initial test program to be acceptable in this regard.

10.2.2.3 *Combined License Information Items*

The applicant has established the following COL information items that are pertinent to the staff's evaluation of the turbine generator, as discussed in Section 10.2.2.2.1 of this report:

- COL information Item 10.2-1-A, as it relates to DCD Tier 2, Revision 9, Sections 10.2.2.7 and 10.2.3.6, and the need for COL applicants to provide a description of the turbine maintenance and inspection program necessary establish inspection, testing, and maintenance provisions for the turbine valves (i.e., steam admission and extraction nonreturn valves) and control system sufficient to address performance and reliability considerations, to satisfy the turbine missile probability analysis, including the criteria identified in Section II of SRP Section 3.5.1.3, and to address any valve and control system maintenance, inspections, and tests that are needed

- COL information Item 10.2-2-A, as it relates to DCD Tier 2, Revision 9, Section 10.2.3.8, which indicates (among other things) that the turbine missile probability analysis report should include a description of the minimum required ISI and testing program for valves essential to overspeed protection, as well as a description of inservice tests, inspections,

and maintenance that are necessary for the turbine and valve assemblies to support considerations in the turbine missile probability analysis

The above COL information items identify additional information that needs to be included in the final safety analysis report to describe plant-specific inspections, tests, and maintenance provisions that will be implemented to ensure the reliable performance of the TGCS over the life of the plant. The staff finds that these COL information items are appropriate and sufficient for this purpose, and no additional COL information items are necessary for this area of review.

10.2.2.4 *Conclusions*

The information provided in Revision 9 of the DCD related to the turbine generator and discussed above in the evaluation section is sufficient to address the review considerations identified in Sections 10.2.1 and 10.2.2. The design of the turbine generator is acceptable and satisfies GDC 4 requirements with respect to the protection of SSCs important to safety from the effects of (1) turbine missiles, and (2) ruptures of the connection joint between the low-pressure turbine exhaust hood and the main condenser. The applicant has met this requirement, based on the following considerations:

- The turbine is favorably oriented such that SSCs important to safety will not be adversely affected by low-trajectory turbine missiles.

- The design of the TGCS will control the speed of the turbine under all operating conditions and will ensure that turbine speed will not exceed 120 percent of rated speed following a load rejection while operating at full power. Although TGCS does not include a mechanical overspeed trip device, SRP considerations are satisfied by implementing design and programmatic measures to ensure highly reliable performance.

- SSCs important to safety that are located in the turbine building are fail-safe such that a rupture of the connection joint between the low-pressure turbine exhaust hood and the condenser will have no adverse affect.

The staff also reviewed ITAAC, initial test program specifications, and COL information items that pertain to the turbine generator, as discussed in Sections 10.2.2.2.2, 10.2.2.2.3, and 10.2.2.3, respectively. The staff confirmed that the applicant has established ITAAC to address important turbine generator design considerations; the initial test program verifies proper performance and integrated operation of the main turbine and TGCS for normal operating and transient conditions; and COL information items that were established are necessary and sufficient for identifying additional plant-specific information that COL applicants must provide.

In conclusion, the staff considers the information in the DCD concerning the turbine generator to be sufficient; the turbine generator design conforms to the requirements specified by GDC 4, conforms to 10 CFR 52.47(b)1, and satisfies the review criteria specified in SRP Section 10.2. Therefore, the staff finds that the design of the turbine generator is acceptable.

10.2.3 Turbine Rotor Integrity

GDC 4 of Appendix A to 10 CFR Part 50 requires that SSCs important to safety shall be appropriately protected against environmental and dynamic effects, including the effects of missiles that may result from equipment failure. Because turbine rotors have large masses and rotate at relatively high speeds during normal reactor operation, failure of a rotor may result in

the generation of high-energy missiles, which may affect the proper function of safety systems. To satisfy GDC 4, turbine rotor integrity must be maintained to minimize the probability of turbine rotor failure.

SRP Section 10.2.3, Revision 2, provides guidance to achieve integrity of the turbine rotor. Specifically, SRP Section 10.2.3 provides criteria to ensure that the turbine rotor materials have acceptable fracture toughness and elevated temperature properties to minimize the potential for failure. In addition, these criteria will ensure that the rotor is adequately designed and will be receiving preservice inspections (PSI) and periodic ISIs to monitor potential degradation. The staff used the criteria in SRP Section 10.2.3 to evaluate the integrity of the turbine rotor in Section 10.2.3 of DCD Tier 2, Revision 6.

10.2.3.1 *Summary of Technical Information*

DCD Tier 2, Revision 6, states that turbine rotors are made from vacuum melted or vacuum degassed nickel-chromium-molybdenum-vanadium (Ni-Cr-Mo-V) alloy steel to minimize flaw occurrence and provide adequate fracture toughness. Chemical elements such as sulfur and phosphorus are controlled to low levels. Fracture appearance transition temperatures (FATT) obtained from Charpy energy will be obtained based on industry standards. Nil-ductility transition temperature (NDT) obtained in accordance with industry standards may be used in lieu of FATT. The FATT and Charpy energy of the rotor material are maintained within the acceptable value.

Fracture Toughness

DCD Tier 2, Revision 6, states that suitable material toughness is obtained through the use of selected materials to produce a balance of material strength and toughness to ensure safety while simultaneously providing high reliability, availability and efficiency during operation. Stress calculations include components due to centrifugal loads, interference fit and thermal gradients where applicable. The ratio of material fracture toughness, K_{IC} (as derived from material tests on each major part or rotor), to the maximum tangential stress intensity at speeds from normal to design overspeed, is at least two at minimum operating temperature. Adequate material fracture toughness needed to maintain this ratio is assured by a large historical database of tests.

Turbine operating procedures are employed to preclude brittle fracture at startup by ensuring that metal temperatures are (1) adequately above the FATT, and (2) sufficient to maintain the fracture toughness to tangential stress ratio at or above 2. The turbine operating instruction will specify sufficient warm-up time to ensure that toughness is adequate to prevent brittle fracture during startup.

The operating temperatures of the high-pressure rotors are below the stress rupture range. Therefore, creep-rupture is not a significant failure mechanism.

Turbine Design

The turbine assembly is designed to withstand normal conditions and anticipated transients, including those resulting in turbine trip, without loss of structural integrity. Turbine shaft bearings are designed to retain their structural integrity under normal operating loads and anticipated transients, including those leading to turbine trips. The multitude of natural critical frequencies of the turbine shaft assemblies existing between zero speed and 20-percent

overspeed are controlled in the design and operation so as to cause no distress to the unit during operation. The turbine rotor average tangential stress at design overspeed resulting from centrifugal forces, interference fit, and thermal gradients does not exceed 0.75 of the yield strength of the materials. Turbine components are designed for an overspeed between 106 -109 percent resulting from a loss of load. The turbine rotor design facilitates ISI of all high stress regions.

Pre-service Inspection (PSI)

Forgings undergo 100-percent volumetric (ultrasonic), visual and surface visual examinations, using established acceptance criteria. Subsurface indications will be either removed or evaluated to ensure that they do not grow to a size that would compromise the integrity of the unit during its service life. Specific portions of finished machined rotors are subjected to a magnetic particle test or liquid penetrant examination. Each fully bladed turbine rotor assembly is factory spin-tested at 20-percent overspeed which is approximately 10 percent above the highest anticipated speed resulting from loss of load. PSIs include air leakage tests on the hydrogen cooling system, hydrogen purity tests, generator windings and motors tests, vibration tests on required motor-driven equipment, hydrostatic tests on all coolers, and piping and valve leakage tests.

Inservice Inspection (ISI)

The ISI program for the turbine assembly includes the complete inspection of all normally inaccessible parts, such as couplings, coupling bolts, turbine shafts, low-pressure turbine buckets, and low-pressure and high-pressure rotors. Turbine inspections are performed in sections during the refueling outages so that a total inspection has been completed at least once within the time period recommended by the manufacturer. The turbine inspection consists of visual, magnetic particle and ultrasonic examinations of all accessible surfaces of the rotors, a visual and surface examination of all low-pressure turbine blades, and a 100-percent visual examination of all couplings and coupling bolts. In accordance with COL Information Item 10.2-1-A, COL applicants will provide a description of the plant-specific turbine maintenance and inspection program.

DCD Tier 2, Revision 9, states that the ISI of valves important to overspeed protection includes the following tests and inspections:

1. All main stop valves, control valves, extraction non-return valves, intermediate stop valves, and intercept valves will be tested under load.

2. Main stop valves, control valves, extraction nonreturn valves, intermediate stop valves, and intercept valves will be tested by the COL licensee in accordance with the turbine surveillance test program.

3. Tightness tests of the main stop and control valves are performed at least once per maintenance cycle by checking the coastdown characteristics of the turbine from no load with each set of four valves closed alternately. As alternative methods, warm up steam may be used as an indicator or the turbine speed may be monitored when on the turning gear while opening each set of four main stop and main control valves alternatively

4. All main stop valves, main control valves, intermediate stop valves, and intercept valves will be inspected once during the first three refueling or shutdowns. The COL licensee will

schedule subsequent inspections in accordance with the applicable industry practice. The inspections will look for wear of linkages and stem packings, erosion of valve seats and stems, deposits on stems and other valve parts, and distortions or misalignment.

5. Inspection of all valves of one type should be conducted if any unusual condition is discovered.

10.2.3.2 *Staff Evaluation*

10.2.3.2.1 Turbine Rotor Design

The staff used SRP Section 10.2.3, Rev.2, to review the turbine rotor material selection, turbine design, and inspection requirements in Section 10.2.3 of DCD Tier 2. The goal of the staff's evaluation is to ensure that the turbine rotor integrity is maintained to minimize the probability of turbine missile generation. This evaluation also touched briefly on turbine overspeed controls and turbine valve inspections. Section 10.2.2 of this report discusses the turbine overspeed control in detail.

In RAI 10.2-22, the staff asked the applicant to provide additional information regarding the turbine rotor design, such as diagrams of the turbine rotor, the number of rotor stages, the bucket design, how the buckets are attached to the rotor, and rotor fabrication. In response, the applicant proposed to add a new Section 10.2.3.8, Turbine Missile Probability Analysis, in DCD Tier 2, Revision 4 that includes a requirement to provide turbine rotor design details as part of this analysis. In addition, in DCD Tier 1, Revision 4, Section 2.11.4, the applicant included an ITAAC to discuss the design and structural integrity of the turbine rotor. The applicant also proposed to include a COL information item in DCD Tier 2, Revision 4, Section 10.2.5, that requires the COL applicant to provide a turbine missile probability analysis meeting the requirements specified in proposed Section 10.2.3.8. The applicant included the proposed information in Revision 4 of the DCD. However, in Revision 6 of the DCD the applicant deleted the ITAAC and replaced it with a COL information item that states that the COL applicant will provide an evaluation of the probability of the turbine missile generation using the criteria in accordance with NRC requirements. This is identified as COL Information Item 10.2-2-A in DCD Tier 2, Revision 6, Section 10.2.3.8. The staff finds this acceptable for addressing the turbine rotor design details. The staff considers RAI 10.2-22 resolved.

10.2.3.2.2 Turbine Missile Protection

In DCD Tier 2, Revision 9, Section 3.5.1.1.1.2, the applicant stated that the ESBWR turbine generator placement and orientation meet the guidelines of RG 1.115. RG 1.115 establishes that turbine orientation and placement, shielding, quality assurance (QA) in design and fabrication, inspection and testing programs, and overspeed protection systems are the principal means of safeguarding against turbine missiles. In SRP Section 3.5.1.3, the staff determined that plant designs that have a favorable turbine generator placement and orientation and adhere to the guidelines presented in RG 1.115 will be considered adequately protected against turbine missile hazards, and that exclusion of safety-related SSCs from low-trajectory turbine missile strike zones constitutes adequate protection against low-trajectory turbine missiles. Based on the applicant's conformance to RG 1.115, and favorable turbine generator placement and orientation the design meets the criteria in SRP Section 3.5.1.3; therefore, the staff finds this aspect of the design acceptable.

In NRC Information Notice 94-01, “Turbine Blade Failures Caused by Torsional Excitation from Electrical System Disturbance,” dated January 7, 1994, the staff discussed turbine blade failures of low-pressure turbines, which were attributed to torsional excitation of the turbine generator shaft as a result of an electrical system disturbance. The staff asked the applicant to discuss whether the turbine will be designed to preclude torsional excitation of the shaft. In response, the applicant added a new Section 10.2.3.8 in DCD Tier 2, Revision 4, which includes a requirement to ensure that the turbine design considers the torsional vibration analysis. The staff finds the applicant’s response acceptable because the DCD includes requirements that ensure the turbine design will consider torsional vibration and, thus, precludes torsional excitation of the shaft.

10.2.3.2.3 Turbine Rotor Material Specifications

SRP Section 10.2.3.II.1.A recommends that sulfur and phosphorus in the turbine rotor material be controlled to low levels because high levels of sulfur and phosphorus have a deleterious effect on the toughness of the turbine rotor. In RAI 10.2-2, the staff asked the applicant to provide the percentage of sulfur and phosphorus in the turbine rotor material and discuss whether their chemical contents are considered low level. In response, the applicant proposed a revision to DCD Section 10.2.3.1 to be consistent with SRP Section 10.2.3.II.1.A regarding the amounts of sulfur and phosphorus in the turbine rotor material. However, the applicant does not have the information on the exact percentage of sulfur and phosphorus in the rotor material at this time because the turbine has not been purchased. Thus, the applicant proposed, in a letter dated August 2, 2007, to revise Section 10.2.5 of DCD Tier 2 to require turbine material property data be provided as part of the turbine missile probability analysis discussed in a proposed new Section 10.2.3.8 in Revision 4 of the DCD. The applicant included the proposed information in Revision 6 of the DCD, and on that basis the staff finds that the applicant’s revisions are consistent with SRP Section 10.2.3.II.1.A, and therefore, are acceptable. The staff considers RAI 10.2-2 resolved.

SRP Section 10.2.3.II.1.C recommends that the Charpy V-notch energy at the minimum operating temperature of each low-pressure rotor in the tangential direction be at least 361 Newton-meters (N-m) (60 foot-pounds [ft-lbs]). DCD Tier 2, Revision 1, Section 10.2.3.1, stated that the room temperature Charpy energy is above 271 N-m (45 ft-lbs), which is not consistent with the minimum 361 N-m (60 ft-lb) recommended in SRP Section 10.2.3.II.1.C. In response to RAI 10.2-4, the applicant proposed a revision to Section 10.2.3.1 of DCD Tier 2, to be consistent with SRP Subsection 10.2.3.II.1.C with regard to the recommended 361 N-m (60 ft-lb) Charpy V-notch energy. Based upon that commitment, the staff found that the proposed revision to Section 10.2.3.1 of DCD Tier 2, Revision 1, is consistent with SRP Section 10.2.3.II.1.C and, therefore, is acceptable. However, following the review of changes made in Revision 3 to the DCD, the staff noted an inconsistency in DCD Tier 2, Revision 3, Section 10.2.3.1 (page 10.2-10, third paragraph). Specifically, the DCD stated that the FATT will be no higher than -1.1 degrees Centigrade (C) (+30 degrees Fahrenheit [F]); and that the Charpy V-notch energy at the minimum operating temperature will be at least 271 N-m (45 ft-lbs). In RAI 10.2-23, the staff requested that the applicant justify these two design limits because they are not consistent with SRP Section 10.2.3. II.1 In a response, the applicant stated that material testing has shown that FATT increases (and Charpy V-notch energy decreases) from the outer surface to the deep-seated region of the forging as a result of variation (slowing from outside to center) in the cooling rate during the quenching process. The cooling rate variation causes the FATT (and Charpy V-notch energy) to change rapidly near the surface of the forging and then changes gradually at deeper forging locations. As a result, material acceptance requirements

for FATT and Charpy V-notch greatly depend on the location in the forging where test samples are obtained.

The values for FATT and Charpy V-notch energy (-17 degrees C and 361 N-m [0 degrees F and 60 ft-lbs.], respectively) specified in SRP Section 10.2.3.II.1 are based on material acceptance data taken from specimens at the surface of a shrunk-on wheel (i.e., disc) forgings. In cases where the shrunk-on disc design is utilized, surface specimens are used because deep-seated specimens (specimens taken from near the center of the forging) cannot be obtained during acceptance testing without destroying the wheel forging. FATT test results based on surface measurements are lower (and the Charpy V-notch energy is higher) than test results based on deep-seated forging properties.

The values for FATT and Charpy V-notch energy included in the ESBWR DCD Tier 2, Revision 3, Section 10.2.3.1 pertains to integral rotor forgings. The values are based on material acceptance data obtained from specimens taken from a radial trepan (closer to the center of the forging), beyond the region where FATT changes rapidly with position. This is the location where measurements are made on every ESBWR integral rotor forging. As such, the criteria established in the ESBWR DCD for integral rotor forgings are deep-seated values for FATT and Charpy V-notch energy, as opposed to surface values. A large data set of centerline FATT values and FATT location variation is available from previous integral rotor testing. Evaluation of this data set shows that the FATT and Charpy V-notch limits set forth in the DCD accurately reflect the material capability for single piece rotor forgings, and provide a suitable means to evaluate the bore FATT. Based upon the known stress-related fracture mechanics associated with integral rotors, crack propagation typically originating from the center of the forging, it is more appropriate to evaluate the material characteristics based on these deep-seated values to verify structural integrity. The fact that the bore stresses for integral rotors are lower than those of the shrunk-on wheel design provides an additional margin of safety. The staff finds the applicant's basis acceptable and concurs with its conclusion that the specified fracture toughness criteria (FATT no higher than -1.1 degrees C [+30 degrees F]; and Charpy V-notch of 271 N-m [45 ft-lb] energy at the minimum operating temperature) are acceptable for a large integral turbine rotor because a large data set of centerline FATT values and FATT location variation is available from previous integral turbine rotor testing to support the applicant's conclusions. The staff considers RAIs 10.2-4 and 10.2-23 resolved.

During its review of DCD Tier 2, Revision 3, the staff noted that Section 10.2.3.2 is not consistent with SRP Section 10.2.3.II.2 because it is not clear how fracture toughness properties of the turbine rotor are obtained. SRP Section 10.2.3.II.2 specifies four methods (a, b, c, and d) for obtaining fracture toughness properties for the turbine rotor. In RAI 10.2-24, the staff requested that the applicant describe the method to be used in the DCD. In response, the applicant responded that each integral (single piece) rotor forging receives the following material acceptance tests: (1) tensile test, (2) room temperature Charpy V-notch test, and (3) FATT determination. These tests are conducted in the body of the rotor at a representative radial trepan. When a rotor is bored, these tests are also conducted in the center core material. Previous testing of this nature performed on integral rotors fabricated from the same material has established a database with reliable material characteristic correlations suitable for application on new, unbored rotor forgings. The fracture toughness (K_{Ic}) value is determined using a value of deep-seated FATT based on the measured FATT values from trepan specimens, and a correlation factor obtained from historical integral rotor test data as described above. This is the same methodology that was used to analyze the shrunk-on wheel rotors in the past. This method of verification most closely resembles method (c) in SRP Section 10.2.3, II.2, with the exception that the correlation factors used are derived from the manufacturers' test

data and extensive background on integral forged rotors (in place of the Begley-Logsdon paper, which was published in 1971). The applicant indicated that test data and calculated toughness curve are to be part of the missile analysis report for the turbine that is discussed in a proposed new Section 10.2.3.8 to be included in Revision 4 of the DCD. The applicant also proposed to revise Section 10.2.3.2 in Revision 4 of DCD to document and clarify rotor fracture toughness test requirements. The applicant included the proposed information in Revision 4 of the DCD. The staff finds that the applicant's revised Section 10.2.3.2 of DCD Tier 2, Revision 4 acceptable because the fracture toughness, $K_{Ic,}$ will be determined using deep-seated FATT based on the measured FATT values from trepan specimens, and correlation factor obtained from historical integral rotor test data. The staff considers RAI 10.2-24 resolved.

DCD Tier 2, Revision 1, Section 10.2.3.2 stated that the ratio of material fracture K_{IC}, to the maximum tangential stress at speeds from normal to 115 percent of rated speed is at least 10 millimeter$^{1/2}$ (mm$^{1/2}$) (0.39 in.$^{1/2}$). In RAI 10.2-5, the staff asked the applicant to clarify whether this ratio is obtained at the minimum operating temperature as recommended in SRP Section 10.2.3.II.2. In response, the applicant proposed a revision to DCD Tier 2, Revision 1, Section 10.2.3.2 to specify that the ratio is obtained at minimum operating temperature. The staff finds that the applicant's proposed revision is consistent with SRP Section 10.2.3.II.2, and, therefore, is acceptable. The staff confirmed that Revision 3 of the DCD accurately incorporated the applicant's proposed revision. The staff considers RAI 10.2-5 resolved.

DCD Tier 2, Revision 1, Section 10.2.3.2, stated that stress calculations include components due to centrifugal loads, interference fit, and thermal gradients where applicable. In RAI 10.2-6, the staff asked the applicant to provide a description of the stress calculations. If unavailable, the staff requested that Section 10.2.5.1 of DCD Tier 2, Revision 1, include a commitment to provide such calculations as a COL action item. In response, and as updated, the applicant proposed to revise the COL information item in Section 10.2.5.1 in Revision 4 of DCD Tier 2, to require a COL holder (i.e., the Licensee) to provide an analysis whose requirements are specified in a new proposed Section 10.2.3.8 that includes the material property data, warm-up time, and stress calculations of turbine components when the turbine is purchased and the turbine-specific data are available. Furthermore, in Revision 3 to the DCD, the applicant added an ITAAC in Tier 1, Section 2.11.4, to require stress analysis that includes turbine material property data, rotor and blade design (including loading combinations, assumptions and warm-up time). The staff finds the applicant's approach of requiring the turbine rotor design information and a detailed analysis be submitted as part of ITAAC to be acceptable since the analysis will require the COL Applicant to use as-built, plant-specific turbine rotor data.

Subsequently the ITAAC was revised to verify that the as-built turbine material properties, turbine rotor and blade designs, preservice inspection and testing results, and inservice inspection and testing requirements meet the requirements of the Turbine Missile Probability Analysis. DCD Tier 2, Section 10.2.3.8 was also revised to include COL Information Item 10.2-2-A requiring the COL Applicant to provide an evaluation of the probability of turbine missile generation using criteria in accordance with NRC requirements. The staff finds this acceptable because the COL applicant will provide the turbine missile analysis for staff review prior to issuance of a license and an ITAAC is provided for as-built verification. The staff considers RAI 10.2-6 resolved.

DCD Tier 2, Revision 1, Section 10.2.3.3 stated that operating temperatures of the high-pressure rotors are below the stress rupture range; therefore, creep-rupture is not a significant failure mechanism. To verify the above statement, in RAI 10.2-7, the staff asked the applicant to identify the normal operating temperatures and the maximum possible temperature of the

high pressure rotors, and identify the temperature at the stress rupture range, and discuss how this temperature was obtained. In response, the applicant responded that DCD Tier 2, Revision 1, Figure 10.1-2 shows the turbine main steam (MS) temperature to be approximately 282.6 degrees C (540.6 degrees F). Long term creep rupture begins to occur at about 427 to 482 degrees C (800 to 900 degrees F) in Ni-Cr-Mo-V low alloy steels and increases with increasing temperature. Therefore, stress rupture is not a plausible failure mode because the maximum turbine temperature will be about 291 degrees C (555 degrees F), which is significantly less than 427 degrees C (800 degrees F). The staff agrees with the applicant that at maximum operating temperature of about 291 degrees C (555 degrees F) the turbine will not exceed the temperature at which creep-rupture occurs. Therefore, the staff concludes that creep-rupture is not a concern for the high pressure turbine rotors. The staff considers RAI 10.2-7 resolved.

During its review of DCD Tier 2, Revision 3, Section 10.2.5.1, the staff noted that the DCD stated that a COL Holder is required to provide an evaluation of the probability of turbine missile generation using criteria in accordance with NRC requirements. As discussed in SRP Section 3.5.1.3, the probability of turbine missile generation should be completed before license issuance so that the staff can verify whether the probability of turbine missile generation meets the acceptance Criteria in SRP Section 3.5.1.3. In RAI 10.2-21, the staff requested that the applicant justify the use of “the COL Holder” in lieu of “the COL Applicant” in Section 10.2.5.1. In response, the applicant responded that the Turbine Missile Probability Analysis will not be available until after the as-built turbine material properties and final as-built rotor design details are available and is therefore specified as a COL information item. In addition, DCD Tier 1, Section 2.11.4 discusses external turbine missile probability and requires it to be less than 1×10^{-4} per turbine year. Based on proposed turbine rotor designs that utilize integral forgings, the probability of turbine missile generation is less than 1×10^{-5} for the ESBWR as stated in the DCD Tier 2, Section 10.2.1. This probability is lower than that specified by the guidance in SRP Section 3.5.1.3, Table 3.5.1.3-1, for loading the turbine and bringing the plant (system) on line. This probability is to be confirmed by calculation and/or analysis in the Turbine Missile Probability Analysis in accordance with ITAAC. To clarify the scope of the Turbine Missile Probability Analysis and meet the guidance of RG 1.206,”Combined License Application for Nuclear Power Plants,” in response to RAIs 10.2-22 and 3.5-17, the applicant proposed to add a new Section 10.2.3.8 to Chapter 10 in Revision 4 to the DCD. New Section 10.2.3.8 required the turbine missile probability analysis to include the aspects described in COL Information Items 10.2.5.1, 10.2.5.2, and 10.2.5.3 that appeared in Revision 2 of DCD Chapter 10. The applicant also proposed to revise COL Information Item 10.2.5.1 for the turbine missile probability analysis to reference new Section 10.2.3.8. In Revision 6 to the DCD the COL information item was relabeled as COL Information Item 10.2-2-A. The NRC staff finds that the specified value of less than 1×10^{-5} for the ESBWR probability of turbine missile generation is acceptable because this value is lower than that specified by the guidance in SRP Section 3.5.1.3. The staff considers RAIs 10.2-21, 10.2-22 and 3.5-17 resolved.

DCD Tier 2, Section 10.2.3.5 discussed the preservice inspection of the turbine rotor. In RAI 10.2-12, the staff asked the applicant to clarify the acceptance criteria for indications as a result of rotor inspection. In response, the applicant responded that when a surface indication is detected on the rotor, it will be blended. If a subsurface indication is detected, it will be excavated and plug welded. The turbine owner’s maintenance manual, to be supplied to the owner with receipt of the turbine, will contain the procedures for the rotor surface inspection. All subsurface indications are addressed before to the rotor is accepted and shipped to the owner. In addition, DCD Tier 2, Section 10.2.3.6 discusses requirements to perform visual inspections on all low-pressure turbine rotor, buckets, and coupling bolts. During its review of Revision 3 of the DCD, in RAI 10.2-25, the staff requested that the applicant describe the specific codes and

standards to which the preservice examination (ultrasonic and surface) of the forgings will be adhered as recommended in SRP Section 10.2.3.II.3. In response to the staff's request, the applicant stated that in accordance with standard industry practices, pre-service surface and visual examinations of the finish-machined rotor forgings will be conducted during the PSI phase of the turbine rotor fabrication. As a result, the applicant proposed to revise DCD Tier 2, Section 10.2.3.5, in Revision 4 to state that 100-percent ultrasonic examination and acceptance criteria that are equivalent or more restrictive than the criteria specified for Class 1 components in ASME Code, Section III and V will be performed on the turbine rotor. In addition, surface and visual examination, including any bores, keyways, or drilled holes, are subject to magnetic particle examination, and all flaw indications in keyways and drilled holes are required to be removed. The staff finds the applicants approach acceptable because ultrasonic examination employing restrictive acceptance criteria will be performed on the turbine rotor. The staff considers RAIs 10.2-12 and 10.2-25 to be resolved.

Following the staff's review of changes made to Revision 3 of the DCD, in RAI 10.2-20, the staff requested that the applicant explain why details pertaining to the turbine inservice test and inspection program were deleted from the DCD. In response, the applicant responded that this information was relocated to DCD Tier 1, Section 2.11.4 as ITAAC 4b. ITAAC 4b required that the turbine and turbine valve inservice test and inspection program includes scope, frequency, methods, acceptance criteria, disposition of reportable indications, corrective actions, and technical basis for inspection frequency. In DCD, Revision 6, the applicant deleted the ITAAC 4b and included the information in DCD Tier 2, Revision 6, Section 10.2.3.8 and in COL Information Item 10.2-2-A. Inservice test, inspection, and operating procedures are to be in accordance with industry practice and meet original equipment manufacturer (OEM) requirements for turbine missile probability. The staff finds this acceptable because the information provided in DCD Tier 2, Section 10.2.3.8 and COL Information Item 10.2-2-A will ensure that the turbine test and inservice program will be conducted and that the turbine will meet the OEM requirements for turbine missile probability. The staff considers RAI 10.2-20 resolved.

The staff finds that the ISI of the turbine rotor as discussed in DCD Tier 2, Revision 1, Section 10.2.3.5 is consistent with SRP Section 10.2.3. However, in RAI 10.2-13, the staff asked the applicant to clarify visual and/or surface examination of turbine rotors, buckets, and couplings. In response, the applicant stated that necessary subsurface inspections and repairs will be addressed during turbine manufacturing. Surface inspections will detect possible propagation of surface indications caused by pitting, cracks, erosion, or corrosion. Buckets are not removed from the rotor when performing visual examinations of the rotor and buckets. A surface examination at the rotor/bucket interface (i.e., root) is acceptable to detect new flaws as they propagate from the outside surface toward the inside surface which can be visually detected. The subsequent inspection results of the turbine components are compared to the PSI results to determine whether new degradation has occurred. Any indications are evaluated and dispositioned as a repair or replacement, as required.

The preservice and ISI procedures discussed above are the general and minimum requirements specified by the DCD. The individual turbine manufacturer will provide inspection procedures to the plant owner at the time of turbine delivery. SRP Section 10.2.3.II.5 recommends that the ISI and maintenance program for the turbine assembly comply with the manufacturer's recommendations. DCD Tier 2, Section 10.2.3.5 requires that the turbine ISI be performed within the period recommended by the turbine manufacturer. The staff finds that the turbine preservice and ISI descriptions in the DCD are consistent with SRP Section 10.2.3, and therefore, are acceptable. The staff considers RAI 10.2-13 resolved.

In RAI 10.2-16, the staff asked the applicant to discuss how the environmental conditions, the operational parameters, design features, fabrication, material properties, and maintenance are managed and considered to mitigate potential degradation of the turbine rotor and buckets. In response, the applicant stated that DCD Tier 2, Revision 1, Section 10.2.3, gives the design guidelines that will be followed during the initial turbine design, installation, PSI, ISI, and testing program. These guidelines along with the recommended operational and maintenance parameters will mitigate degradation in the turbine rotor and buckets. DCD Tier 2, Revision 3, Section 10.2.4 further describes that the turbine is designed, constructed and inspected to minimize the possibility of any major turbine component failure. The staff finds that the applicant has demonstrated that the turbine is designed and fabricated to minimize potential degradation. The associated turbine inspection program is designed to monitor the integrity of the turbine components. The staff considers RAI 10.2-16 resolved.

10.2.3.2.4 Conclusion

The staff concludes that the rotor design and material selection in DCD Tier 2, Revision 9, Section 10.2.3 are consistent with SRP Section 10.2.3, and therefore, are acceptable. The staff also concludes that the rotor inspection in DCD Tier 2, Revision 9, Section 10.2.3 is consistent with SRP Section 10.2.3.

10.3 <u>Turbine Main Steam System</u>

10.3.1 Regulatory Criteria

The staff reviewed the design of the TMSS in accordance with Section 10.3 of the SRP, Revision 4, 2007. The design of the TMSS is acceptable if its integrated design meets the requirements of 10 CFR Part 50.

Specifically, acceptability of the TMSS design is based on meeting the following:

- GDC 2, “Design bases for protection against natural phenomena,” with respect to the safety-related portions of the system being capable of withstanding the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, and floods, and the positions of the following:

    RG 1.29, “Seismic Design Classification,” Revision 4, issued March 2007, as related to the seismic design classification of system components, Positions C.1.a, C.1.e, C.1.f, C.2, and C.3

- GDC 4 with respect to the ability of portions of the system important to safety to withstand the effects of external missiles and internally generated missiles, pipe whip, and jet impingement forces associated with pipe breaks

- GDC 5, “Sharing of structures, systems, and components,” with respect to the ability of the shared systems and components important to safety to perform required safety functions

The NRC staff review also considered the guidance of SECY-93-087, “Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs,” dated April 2, 1993, which is applicable to boiling-water reactor (BWR) plants that do not incorporate a main steam isolation valve leakage control system (MSIVLCS) and for which main steamline fission product holdup and retention are credited in the analysis of design-basis accident radiological consequences.

10.3.2 Summary of Technical Information

The function of the TMSS is to transport the steam generated in the reactor to the main turbine system. The TMSS is bounded by, but does not include, the seismic interface restraint, turbine stop valves, and turbine bypass valves. Steam supply lines to other services, up to and including their isolation valves, are also part of the TMSS. The system is designed to deliver steam from the reactor to the turbine generator system for a range of flows and pressures varying from warm-up to rated conditions. It also provides steam to the reheaters, the steam jet air ejectors (SJAEs), the TGSS, the offgas system, and the TBS.

The TMSS is not required to perform or support any safety-related function. However, the supply system is designed to (1) accommodate operational stresses such as internal pressure and dynamic loads without failures, (2) provide a seismically analyzed fission product leakage path to the main condenser, (3) provide suitable accesses to permit inservice testing and inspections, and (4) close the steam auxiliary valve(s) on a main steam isolation valve (MSIV) isolation signal, (5) open drain valve(s) on an MSIV isolation signal to provide the MSIV leakage path to the main condenser, (6) TMSS piping provides a nominal turbine inlet pressure that is consistent with rated turbine heat balance.

The TMSS piping consists of four lines from the seismic interface restraint to the main turbine stop valves. The four main steamlines are connected to a header upstream of the turbine stop valves to permit testing of the turbine stop and control valves during plant operation with a minimum load reduction. Section 5.4 of this report discusses in detail the portions of the MS and FW piping located upstream of the seismic restraints, including the MS isolation system.

10.3.3 Staff Evaluation

In DCD Tier 2, Section 10.3.1.1, the applicant stated that the quality group B portions of the system are analyzed, fabricated, and examined to ASME Code Class 2 requirements, classified as seismic Category II, and subject to pertinent QA requirements of Appendix B, “Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants,” to 10 CFR Part 50. ISI will be performed in accordance with ASME Section XI requirements for Code Class 2 piping. In RAI 10.3-1, the staff asked the applicant to justify the discrepancy with SRP Section 10.3, Criterion III.3.b, and RG 1.29 Position C.1.e, which provide that the subject portions of the TMSS are to be designed to seismic Category I.

In response to RAI 10.3-1, the applicant stated that the portion of the MS piping inside the containment, including the inboard MSIVs, containment penetrations, outboard MSIVs, and piping up to the seismic restraints, is classified as seismic Category I. The applicant also stated that the TMSS piping portion of the MS piping (i.e., downstream of the seismic restraint) is a non-safety system, located in a non-safety building designed to seismic Category II, and is analyzed to demonstrate structural integrity under safe-shutdown earthquake (SSE) loading conditions. In Revision 6 of the DCD, the applicant revised the Section 10.3.1.1 to reflect its response and identified the portions of this piping that will be analyzed. The applicant further stated that the integrity of the MSIV leakage path to the condenser (main steam piping, bypass piping, required drain piping, and main condenser) is not compromised by nonseismically designed SSCs. The staff finds this acceptable because it meets the guidance delineated in SRP Section 10.3.

However, in its response to RAI 10.3-1, the applicant also stated that the ASME authorized nuclear inspector (ANI) and ASME Code stamping are not required for these portions of the

system. Later, in response to RAI 3.2-1, the applicant agreed to include ANI and ASME Code stamping for all ASME Class 1, 2, and 3 piping. The staff requested that the applicant revise the response to RAI 10.3-1 and the DCD to acknowledge the commitment made in response to RAI 3.2-1. (Section 3.2 of this report discusses in detail the main steamline seismic design classification and adherence to RG 1.29). In DCD Tier 2, Revision 6, the applicant confirmed this commitment and reflected it in Table 3.2-1, “Classification Summary.” The staff’s evaluation of RAI 3.2-1 is described in 3.2 of this report. The staff considers RAI 10.3-1 resolved.

Regarding Task Action Plan Item A-1, “Water Hammer,” the “SRP Acceptance Criteria,” Item II of SRP Section 10.3, states that the system design should adequately consider water (steam) hammer and relief valve discharge loads to ensure that system safety functions can be performed and should ensure that operating and maintenance procedures include adequate precautions to prevent these effects. In DCD Tier 2, Section 10.3.2.1, the applicant addressed this issue and stated that the system design accommodates steam hammer and relief valve discharge loads. The applicant further stated that the ESBWR TMSS complies with NUREG–0927, “Evaluation of Water Hammer Occurrences in Nuclear Power Plants,” issued March 1984. Additionally, in DCD Tier 2, Section 10.3.3, the applicant stated that operating and maintenance procedures include adequate precautions to minimize the potential for water (steam) hammer. The COL applicant will develop operating and maintenance procedures that include adequate precautions to avoid steam hammer and discharge loads as described in DCD Tier 2, Revision 9, Section 10.3.3. The requirement to develop operating and maintenance procedures is included as a COL Information Item 13.5-2-A in DCD Tier 2, Section 13.5.3.

The staff reviewed the capability to detect and control system leakage and to isolate portions of the system in case of excessive leakage or component malfunction. Most of the currently licensed BWRs rely on the MSIVLCS to mitigate the radiological consequences of MSIV leakage following a design-basis loss-of-coolant-accident (LOCA) and to stay within the limits of 10 CFR Part 100, “Reactor Site Criteria,” if the MSIV leakage rate exceeds the technical specification limit. The ESBWR will not have an MSIVLCS and will rely instead on the TMSS coupled with the main condenser and the TBS to contain MSIV leakage, thus relying on plateout and holdup of fission products to limit the radiological consequences to within the 10 CFR Part 100 requirements. In response to RAI 10.3-11, the applicant stated that a procedure is needed to provide the operator actions required to ensure that the MSIV fission product leakage path is isolated from the TMSS auxiliaries. The requirement to develop procedures for operation, abnormal events, and emergencies is included as COL Information Items 13.5-2-A and 13.5-3-A in DCD Tier 2, Section 13.5.2.

The applicant stated that a drainline is connected to the low points of each main steamline, both inside and outside the containment. Both sets of drains lead to a common header and are connected with isolation valves to allow flow to the main condenser. Section 15.4 of this report discusses the maximum allowable MSIV leakage. To take credit for the TMSS and main condenser for containment and holdup of MSIV leakage, the TMSS, the main condenser, and the connections from the main steamlines to the condenser must be capable of maintaining their integrity during and following an SSE. As discussed above, the TMSS (including the drain paths) is analyzed to demonstrate structural integrity under SSE loading conditions. Section 3.2 of this report contains a detailed evaluation of the seismic analysis requirements for the TMSS and the main condenser. The staff considers RAI 10.3-11 to be resolved.

To process the MSIV leakage, a path must be ensured through the MS drainlines to the main condenser. A reliable power sources must be available so that a control operator can establish

the flow path assuming a single active failure. In RAI 10.3-10(a), the staff asked the applicant to demonstrate how the ESBWR design provides reliable methods for ensuring that flow paths can be established to process MSIV leakage through the drainlines. In response to RAI 10.3-10(a), the applicant stated that the drain valve(s) that are required to change position to establish the MSIV leakage path to the condenser will be equipped with reliable power sources or designed to fail to the required position on loss of power or air and will receive periodic inspection and testing to ensure continued reliability. The staff considered this response an insufficient basis for concluding that the design provisions to ensure availability of the power sources are acceptable. In RAI 10.3-10 S01, the staff therefore requested that the applicant provide a description similar to that provided for the ABWR, which identifies the classification of the power source (specifically, if it is a Class 1E) and configuration for all MSIV alternate leakage treatment path valves, including the turbine bypass valves. In response, the applicant submitted a supplemental response to clarify this issue where it stated that the ESBWR is designed with fail-safe air-operated valves, supplied by non-1E power sources, to fulfill the MSIV leakage path function, and therefore a 1E power source is not required. The applicant further stated that this fail-safe design is the basis for functional reliability of the subject valve(s) as required by NEDC-31858P, “BWROG Report for Increasing MSIV Leakage Rate Limits and Elimination of Leakage Control Systems,” Revision 2 and associated SER, which the staff finds acceptable.

In addition, in RAI 10.3-10(b), the staff requested that the applicant clarify if valves that are required to open the alternative leakage treatment path will be included in the COL applicant’s inservice testing (IST) program. In response to RAI 10.3-10(b), the applicant stated that a periodic test program will verify continued reliability of the valves, but the IST program will not include the valves. The staff found this unacceptable because it is not consistent with the staff’s position given in “Safety Evaluation on GE Topical Report, NEDC-31858P, Revision 2, ‘BWROG Report for Increasing MSIV Leakage and Elimination of Leakage Control Systems,’” in which the staff determined that valves required to establish and maintain the MSIV alternate leakage treatment path should be included in the plant IST program. The alternative leakage treatment path system is classified as ASME Class 2 and Quality Group B and therefore may be subject to the provisions of 10 CFR 50.55a and hence to the requirements of the ASME Code for Operation and Maintenance of Nuclear Power Plants (O&M Code). If these valves, which are relied on to mitigate the consequences of an accident, are powered from emergency power sources, then, pursuant to 10 CFR 50.55a, they are required to be included in the IST program. The staff issued RAI 10.3-10 S01. In response, the applicant clarified this issue, where it stated that according to the guidance provided in the SER for NEDC-31858P, “BWROG Report for Increasing MSIV Leakage Rate Limits and Elimination of Leakage Control Systems,” Revision 2, and 10 CFR 50.55a.f.6(ii), the valves required to open the alternative leakage treatment path are to be included in the plant’s augmented IST Program as non-code valves, which the staff finds acceptable in addressing the IST program governing the alternative leakage treatment path system valves. The staff considers RAIs 10.3-10 and 10.3-10 S01 resolved.

The applicant stated that inspection and testing will be in accordance with the requirements of DCD Tier 2, Revision 9, Section 6.6. The main steamline will be hydrostatically tested to confirm leak tightness. The staff finds this acceptable. Section 6.6 of this report discusses in detail the system pressure tests and inspections.

The requirements of GDC 2, as related to safety-related portions of the system, do not apply since there are no safety-related portions of the TMSS. However, the staff found that the TMSS meets the requirements of GDC 2 because the system is dynamically analyzed to demonstrate structural integrity under safe shutdown earthquake (SSE) loading conditions and is subject to

pertinent QA requirements of Appendix B to 10 CFR Part 50. Section 3.2 of this report contains a detailed discussion of the seismic qualification requirements.

The requirements of GDC 5 are not applicable to the ESBWR because it is designed as a single-unit facility.

The requirements of GDC 34 are not applicable to direct cycle plants (i.e., BWRs), therefore they are not applicable to the ESBWR design.

As discussed above, the TMSS includes all components and piping from the outermost containment isolation valve up to but not including the turbine stop valves. The system has no safety-related portions. Quality Group B portion of the TMSS are analyzed, fabricated, and examined to ASME Code Class 2 requirements, classified as seismic Category II. ISI will be performed in accordance with ASME Section XI requirements for Code Class 2 piping. The scope of the staff's review included layout drawings, piping and instrumentation diagrams (P&IDs), and descriptive information for the system.

10.3.4 Conclusions

Based on the above discussion, the staff concludes that the TMSS for the ESBWR satisfies the requirements of GDC 4 and meets the acceptance criteria in SRP Section 10.3, and is, therefore, acceptable.

10.3.5 Not Used

10.3.6 Steam and Feedwater System Materials

10.3.6.1 *Regulatory Criteria*

The staff reviewed DCD Tier 2, Section 10.3.6, in accordance with SRP Section 10.3.6, March 2007.

The materials selection, fabrication, and fracture toughness criteria of ASME Code Class 2 and 3 pressure boundary components of the steam and FW systems are acceptable if they meet the relevant requirements in 10 CFR 50.55a; GDC 1, "Quality standards and records," and GDC 35, "Emergency core cooling," in Appendix A to 10 CFR Part 50; and Appendix B to 10 CFR Part 50. These requirements are discussed below.

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

- Compliance with GDC 35 requires that for ferritic pressure-retaining components of a critical nature, the containment capability is assured, in part, by requiring minimum fracture toughness performance of the materials from which they are fabricated.

- Appendix B to 10 CFR Part 50 provides QA requirements for the design, construction, and operation of SSCs that are important to safety.

Descriptive information on the MS and FW systems materials, with the exception of those portions included in the reactor coolant pressure boundary (RCPB), appears in DCD Tier 2, Revision 9, Section 10.3.6.

10.3.6.2 *Summary of Technical Information*

The steam and FW component materials that are within the RCPB are addressed in DCD Tier 2, Revision 9, Section 5.2.3 and are evaluated in Section 5.2.3 of this report. The materials specified for use in ASME Code Class 2 components meet ASME Code, Sections II and III, or RG 1.84, "Design, Fabrication, and Materials Code Case Acceptability, ASME Section III." The materials used in the ASME Code Class 2 portion of the TMSS meet the fracture toughness requirements of paragraph NC-2300 of the ASME Code. The steam and FW systems in the ESBWR design contain no ASME Code Class 3 piping.

The recommendations in RG 1.71, "Welder Qualification for Areas of Limited Accessibility," and RG 1.37, "Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants," apply to the ESBWR design. The ESBWR design may employ an alternative to RG 1.71, which is discussed in DCD Tier 2, Section 5.2.3.4.2. ASME Code, Section III, paragraphs NC-2550 through 2570, will be used as the acceptance criteria for nondestructive examination of tubular products.

10.3.6.3 *Staff Evaluation*

The staff reviewed and evaluated the information in DCD Tier 2, Revision 9, Section 10.3.6, to ensure that the materials and fabrication of ASME Code Class 2 components meet the requirements detailed in SRP Section 10.3.6. The steam and FW systems in the ESBWR design have no ASME Code Class 3 piping.

10.3.6.3.1 Material Selection and Fabrication of Class 2 Components

To meet the requirements of GDC 1 and 10 CFR 50.55a, the materials used in the ASME Code Class 2 portion of the MS and FW systems must meet the requirements of Sections II and III of the ASME Code or ASME Code Cases listed in the recommendations of RG 1.84 and follow the recommendations in RG 1.71 and RG 1.50, "Control of Preheat Temperature for Welding of Low-Alloy Steel," issued May 1973.

DCD Tier 2, Section 10.3.6, did not identify material specifications for MS or FW piping. However, it referenced MS and FW components that are covered as part of the RCPB in Chapter 5. In RAI 10.3-4, the staff asked the applicant to provide a complete list of all material specifications and grades that are used in the MS and FW systems by component types, including weld filler metal, and to specify the ASME Code Class. In response to RAI 10.3-4, the applicant indicated that ASME Code Class 2 piping material used in MS and FW systems is the same as the material identified for use in the RCPB, as specified in DCD Tier 2, Table 5.2-4. The applicant indicated that it would revise DCD Tier 2, Section 10.3.6, to list MS and FW piping material specifications and grades used in Class 2 MS and FW systems. The applicant also stated that the ESBWR steam, FW, and condensate system piping has no ASME Code, Section III, Class 3/Quality Group C piping. The applicant did not provide weld filler material specifications and grades for use in ASME Code Class 2 MS and FW systems. In RAI 10.3-4(a) S02, the staff asked the applicant to provide the staff with a list of the weld filler material specifications and classifications used in Class 2 MS and FW systems. The staff identified this issue as part of Open Item 10.3-4 in the SER with open items.

The staff subsequently reviewed modifications made to the ESBWR DCD Tier 2, Revision 5, Section 10.3.6. DCD Tier 2, Revision 5, Table 10.3-2 lists MS and FW piping material specifications and grades as well as weld filler metal specifications and classifications. The staff reviewed Table 10.3-2 and verified that the materials listed for the ASME Code Class 2 MS and FW piping and weld filler materials meet the requirements of ASME Code. In addition, the weld filler materials selected are compatible with the base materials to be welded. The staff therefore finds this acceptable. RAI 10.3-4 and the associated open item related to material specifications are resolved.

In DCD Tier 2, Section 10.3.1.1, the applicant stated that the main steam system (MSS) is analyzed, fabricated, and examined to meet ASME Code Class 2 requirements, classified as nonseismic, and subject to pertinent QA requirements of Appendix B to 10 CFR Part 50. ISI will be performed in accordance with ASME Code, Section XI, requirements for Code Class 2 piping. The applicant also stated that ASME ANI and ASME Code stamping is not required.

In RAI 10.3-7, the staff asked the applicant to provide a basis for the exclusion of ASME ANI and ASME Code stamping requirements for ASME Class 2 piping and components. In response to RAI 10.3-7, the applicant stated that the N11 TMSS piping is analyzed, fabricated, and examined to ASME Code Class 2 requirements, classified as non-safety, seismic Category II, and subject to the pertinent QA requirements of Appendix B to 10 CFR Part 50. ISI will be performed in accordance with ASME Code, Section XI, requirements for Code Class 2 piping. ASME ANI and ASME Code stamping is not required. In supplemental RAI 3.2-1 S02, the staff requested additional information regarding the applicant's intended exclusion of ASME Code stamping and ASME ANI. The staff identified this issue as Open Item 10.3-7 and as part of Open Item 3.2-1 S02.

In DCD Tier 2, Revision 5, Section 10.3.6, the applicant deleted the statement that ASME authorized inspector and ASME Code stamping is not required. In addition, DCD Tier 2, Section 3.2.3.4 now states that non-safety-related SSCs that are classified seismic Category I or II and Quality Group B or C are subject to ASME Section III requirements (including N stamping) and ASME Section XI inspection requirements. The staff finds this acceptable because the ASME Code, Class 2, MS and FW systems will be fabricated and inspected in accordance with ASME Section III and be included in an applicant's ASME Code Section XI ISI program. RAI 10.3-7 and the associated Open Items are resolved. RAI 3.2-1 is resolved and is discussed in Section 3.2.2.3.2 of this report.

The guidelines listed in RG 1.71 ensure the integrity of welds in locations of restricted direct physical and visual accessibility. RG 1.50 provides staff-approved methods to control preheat temperatures before postweld heat treatment when welding low-alloy steel. ASME Code, Section III, Article D-1000, provides recommended minimum preheat temperatures used to weld carbon steel and low-alloy steel components that are acceptable to the staff. RG 1.37 provides acceptable procedures for cleaning and handling Class 2 components of the steam and FW systems.

ASME Code Class 2 components are acceptable if welds located in areas of restricted direct and visual accessibility are welded by personnel qualified according to the guidance of RG 1.71. This guide describes methods acceptable to the staff for providing better control of welder techniques in production welding. DCD Tier 2, Section 10.3.6.2, indicates that an alternative to RG 1.71 may be used as shown in DCD Tier 2, Section 5.2.3.4.2. The staff reviewed the applicant's alternative to RG 1.71 as stated in DCD Tier 2, Section 5.2.3.4.2. The staff determined that the applicant's alternative is consistent with the intent of RG 1.71. The

applicant’s alternative will provide reasonable assurance that welders working in restricted access positions will be appropriately qualified. Section 5.2.3.3.3 of this report further discusses the applicant’s level of compliance with the guidance in RG 1.71.

The ESBWR design is consistent with the guidance in RG 1.37 except as noted in DCD Tier 2, Revision 9, Table 1.9-21B. The alternative that the applicant may use is documented in Table 2-1 of NEDO-11209-04a, “GE Nuclear Energy Quality Assurance Program Description,” Class I (nonproprietary), Revision 8, dated March 31, 1989, which was approved by the NRC on March 31, 1989, and is therefore acceptable. Section 4.5.1.2.5 of this report further discusses the applicant’s level of compliance with RG 1.37. The acceptance criteria for nondestructive examination of tubular products will meet the requirements of ASME Code, Section III, paragraphs NC-2550 through NC-2570, which are consistent with the acceptance criteria in SRP Section 10.3.6.

RG 1.50 recommends that all low-alloy steel welds be maintained at the minimum preheat temperature until the performance of postweld heat treatment. In response to RAIs 5.2-44 and 6.1-4, the applicant discussed its alternative to the guidance provided in RG 1.50 for welding components such as the reactor pressure vessel (RPV) and the standby liquid control system accumulator tank, to ensure that delayed cracking of the weld metal or weld heat affected zone will not occur. The applicant’s alternative entails the use of postweld baking with times and temperatures based on the welding process used and prior qualification testing.

The staff considers the applicant’s proposal to perform postweld baking to be an acceptable alternative to the guidance in RG 1.50, which recommends the maintenance of preheat until postweld heat treatment is performed. The staff notes that this method has been successfully used in several other applications, such as fossil fuel electric generation facilities and petrochemical facilities, with materials that are much more sensitive to hydrogen cracking than those materials used in ASME Code Class 1 and 2 systems in the ESBWR design. Postweld baking is an effective measure to prevent delayed hydrogen cracking in welds that do not go directly from preheat temperature to postweld heat treatment. The staff therefore considers the applicant’s alternative to RG 1.50 acceptable, given that it provides reasonable assurance that delayed hydrogen cracking will not occur in the time that a weld is completed through completion of postweld heat treatment.

Although the staff considers the applicant’s alternative to RG 1.50 acceptable, the staff requested in RAI 10.3-4 S02 that the applicant modify the DCD to include its alternative to RG 1.50 as it applies to all ASME Code Class 1, 2, and 3 piping and components. In addition, the staff requested that the applicant modify the DCD to include its response to RAI 6.1-4, in which it states that ASME Code, Section III, Appendix D, Article D-1000, minimum preheat recommendations will be applied to all Class 1, 2, and 3 components in the ESBWR design. The staff identified these issues as part of Open Item 10.3-4.

In response, the applicant stated that it would modify Section 10.3.6.2 to provide a pointer to Section 5.2.3.3.2 regarding its intent to follow the recommendations listed in RG 1.50 and ASME Code, Section III, Appendix D, Article D-1000. The staff reviewed the modifications made to the ESBWR DCD in Revision 5 and verified that the appropriate changes were made. Therefore, RAI 10.3-4 and the associated Open Item related to RG 1.50 and minimum preheat temperatures are resolved.

10.3.6.3.2 Fracture Toughness of Class 2 Components

DCD Tier 2, Section 10.3.6.1, stated that the ASME Code, Section III, Class 2, portion of the TMSS meets the fracture toughness requirements of NC-2300. Although this is acceptable to the staff, the applicant did not indicate the fracture toughness requirements for the Class 2 FW system. In RAI 10.3-4 S02(b), the staff requested that the applicant modify DCD Tier 2, Section 10.3.6, to include the fracture toughness requirements for Class 2 FW components. The staff identified this issue as part of Open Item 10.3-4.

By letter dated August 31, 2007, the applicant stated that it would modify DCD Tier 2, Section 10.3.6 .1 to include material toughness requirements for Class 2 FW piping. The staff reviewed the modifications made to the ESBWR DCD in Revision 5 regarding fracture toughness of Class 2 piping. Section 10.3.6.1 now states that the TMSS and feedwater systems (FWS) meet the fracture toughness requirements of NC-2300. The staff finds this acceptable because the applicant included toughness requirements for the ASME Code Class 2 portion of the FWS. RAI 10.3-4 and the associated Open Item related to material fracture toughness are resolved

10.3.6.3.3 Flow-Accelerated Corrosion

ASME Code, Section III, paragraph NC-3121, requires that material subject to thinning by corrosion, erosion, mechanical abrasion, or other environmental effects shall have provision made for these effects during the design or specified life of the component by a suitable increase in or addition to the thickness of the base metal over that determined by the design formulas. The staff evaluated information supplied by the applicant in the DCD regarding material selection and design of ASME Code Class 2 MS and FW systems and non-ASME Code, Section III, FW and condensate systems to mitigate the effects of erosion/corrosion. The staff notes that historically, documents such as Generic Letter (GL) 89-08, “Erosion/Corrosion-Induced Pipe Wall Thinning,” have referred to flow-accelerated corrosion (FAC) as erosion/corrosion. Therefore, FAC and erosion/corrosion are used interchangeably throughout Section 10.4.3.3 of this report.

The applicant indicated that ASME Code Class 2 MS piping will be constructed of SA-333, Grade 6, and FW piping will be constructed of SA-335, Grade P22. In RAI 10.3-6, the staff asked the applicant to describe the mitigation steps taken in the ESBWR design related to (1) utilization of materials resistant to erosion/corrosion, (2) specification of an adequate corrosion allowance, and (3) consideration of minimizing the effects of erosion/corrosion in the design of all ESBWR FW, steam, and condensate system piping from effects such as fluid velocity, bend locations, and flashpoints. The applicant responded and stated the following:

> The TMSS piping is designed to consider the effects of erosion/corrosion for a 60 year life expectancy. Piping containing dry, single phase steam is constructed of carbon steel. Piping exposed to wet, two-phase steam is constructed of erosion/corrosion resistant low alloy steel. Velocities in the TMSS piping to the high pressure turbine are limited to reduce the potential for pipe erosion. Low point drains are provided for collecting and draining moisture and to help reduce the potential for water carryover to the high and low pressure turbines. In addition to material selection, pipe size and layout may also be used to minimize the potential for erosion/corrosion in systems containing water or two-phase flow.

The applicant’s response to RAI 10.3-6 referenced only the TMSS and did not address MS, FW, and condensate piping, as requested in the RAI. In supplemental RAI 10.3-6 S01, the staff

asked that the applicant provide a response that addresses RAI 10.3-6 for all MS, FW, and condensate system piping (ASME Code Class and non-Code piping) in the ESBWR design. In response, the applicant stated the following:

> The ESBWR standard plant has a 60-year design life. As part of the design of the condensate, FW and MS piping, an erosion-corrosion evaluation is performed. The evaluation is used to determine the expected erosion-corrosion rate, i.e., yearly reduction in wall thickness, based on the system geometry, system configuration, and chemical properties of the process fluid and piping. With the erosion rate known, the results are compared against the 60-year design life. Areas that do not meet the design life are addressed by piping configuration changes, material substitutions, or a combination of both…. The remainder of the non-ASME Code Class 1, 2, or 3 Condensate and Feedwater System piping is designed and fabricated with consideration given to the deleterious effects of erosion.

For the TMSS, the selected materials, coupled with the applicant’s evaluation to determine the expected erosion/corrosion rate based on the system geometry, system configuration, and chemical properties of the process fluid and piping, are acceptable to the staff and fulfill the design requirements of ASME Code, Section III, paragraph NC-3121. For the ASME Code Class 2 FW systems, which would tend to be more susceptible to FAC than the TMSS, the staff notes that the applicant has selected SA-335, Grade P22 (2.25-percent chromium, 1-percent molybdenum) that provides an increased level of protection against erosion/corrosion. The selection of P22, coupled with the applicant’s evaluation to determine the expected erosion/corrosion rate, is acceptable to the staff and fulfills the design requirements of ASME Code, Section III, Paragraph NC-3121.

During a teleconference between the NRC staff and the applicant on June 7, 2007, the applicant indicated that the design of non-ASME Code, Section III, systems is not yet complete. In RAI 10.3-6 S02, the staff asked that the applicant modify the DCD to include a COL information item to include materials specifications and grades for non-ASME Code, Section III, MS, FW, and condensate piping and components that could potentially be susceptible to erosion/corrosion and discuss a basis for the selection of these materials.

During subsequent teleconferences with GEH, the staff agreed with the applicant that modifying the DCD to include a discussion regarding the process used to select materials that will mitigate the effects of FAC in the non-ASME Code CFS would be acceptable in lieu of a COL information item. In Revision 5 of the ESBWR DCD, the applicant modified Section 10.4.7 to state that the CFS is potentially subject to the effects of FAC. Applicable operating experience and recommendations provided in NRC GL 89-08 and NUREG–1344 "Erosion/Corrosion-Induced Pipe Wall Thinning in U.S. Nuclear Power Plants," April 1989, are applied to the design and operation. The CFS is designed with pipe wall thicknesses that incorporate a conservative corrosion allowance commensurate with a 60-year design life. When required by analysis to meet the design life, FAC-resistant materials are utilized. The staff finds this acceptable because the ESBWR design mitigates the effects of FAC in the non-ASME Code CFS by selecting materials that will meet a 60-year design life and therefore RAI 10.3-6 and the associated Open Item are resolved.

In addition to design considerations to minimize erosion/corrosion, as described in GL 89-08, an appropriate long-term monitoring program must be implemented to detect the potential wall-thinning of high-energy piping, ASME Code, Section III, Code Class 1, 2, 3, and non-safety-

related piping, caused by erosion/corrosion. The applicant's description of the required augmented inspection program to monitor erosion/corrosion is acceptable to the staff and is located in Section 6.6.3.8 of this report.

10.3.7 Conclusions

On the basis of the information submitted, the staff concludes that the ESBWR steam and FWS materials satisfy the relevant requirements of 10 CFR 50.55a; Appendix A to 10 CFR Part 50, GDC 1 and 35; and Appendix B to 10 CFR Part 50. This conclusion is based on the fact that the ESBWR steam and FW materials satisfy ASME Code, Section III; RGs 1.37, 1.50, 1.71, and 1.84; and SRP Section 10.3.6.

10.4 Other Features of Steam and Power Conversion System

10.4.1 Main Condenser

10.4.1.1 *Regulatory Criteria*

The staff reviewed the design of the main condenser in accordance with SRP Section 10.4.1, Revision 3, issued March 2007. The design of the main condenser is acceptable if its integrated design meets the requirements of GDC 60, "Control of releases of radioactive materials to the environment," in Appendix A to 10 CFR Part 50, as they relate to the design of the system to ensure that failures do not result in excessive releases of radioactivity to the environment, do not cause unacceptable condensate quality, and do not flood areas housing safety-related equipment.

The guidance in SECY-93-087 is applicable for new BWR plants that do not incorporate a MSIVLCS and for which main condenser holdup and plateout of fission products are credited in the analysis of design-basis accident radiological consequences. The applicable guidance from SECY-93-087 states that a seismic analysis should be performed to ensure that the condenser anchorages and the piping inlet nozzle to the condenser are capable of maintaining their structural integrity during and after a SSE.

10.4.1.2 *Summary of Technical Information*

The main condenser is designed to function as the steam cycle heat sink. During normal operation, it receives, condenses, deaerates, and holds up for N-16 decay the main turbine exhaust steam and turbine bypass steam whenever the TBS is operated. The main condenser is also a collection point for other steam cycle miscellaneous drains and vents. The main condenser is used as a heat sink in the initial phase of reactor cooldown during a normal plant shutdown. The main condenser does not perform, support, or ensure any safety-related function and thus has no safety design basis. The applicant stated that it is designed with the necessary shielding and controlled access to protect plant personnel from radiation. Sections 11.1 and 11.3 of this report describe the anticipated inventory of radioactive contaminants during operation and shutdown.

DCD Tier 2, Revision 9, Section 10.4.1 describes the main condenser system of the ESBWR design. DCD Tier 2, Revision 9, Table 10.4-1, "Main Condenser Data," lists the design parameters of the condenser (such as heat transfer capability, surface area, design operating pressure, shell-side pressure, circulating water flow, and tube-side temperature rise). DCD Tier 2, Revision 9, Section 10.4.1, references this table.

The applicant stated that, during anticipated operational occurrence conditions, the condenser is designed to receive turbine bypass steam and high-level dump from the FW heaters and moisture separators/reheater (MSR) drain tanks. The condenser is also designed to receive relief valve discharges and any necessary venting from MSR vessels, FW heater shells, the gland seal steam header, steam seal regulator, and various other steam supply lines. The condenser will be designed with spray pipes and inlet baffles to preclude component or tube failures. Rupture diaphragms are also installed on the low-pressure turbine exhaust hoods to protect the condenser and turbine from overpressure damage.

10.4.1.3 *Staff Evaluation*

The staff reviewed whether the system description delineates the main condenser system capabilities including the minimum system heat transfer and system flow requirements for normal plant and turbine bypass operation. The staff also reviewed measures provided to prevent loss of vacuum, corrosion, and/or erosion of main condenser tubes and components and hydrogen buildup in the main condenser.

The staff concludes that the ESBWR design is consistent with the guidance of SECY-93-087 because the condenser structural members, supports and anchors are designed to maintain condenser integrity following an SSE. Section 3.2 of this report discusses the seismic design qualification and analysis.

In RAI 10.4-2, the staff asked the applicant to provide a detailed description of design measures to prevent the loss of the condenser. In response to RAI 10.4-2, the applicant stated that design measures to prevent the loss of condenser include treatment of circulating water to prevent algae or other growth from fouling the condenser tubes. The tube metal selected will be stainless steel or titanium, both of which are resistant to erosion, corrosion, and galvanic action. The tube sheet will be selected to complement the tube material and resist corrosion and galvanic action. Coating of the water box material will protect the circulating water system from corrosion and galvanic action resulting from the dissimilarity of the metal of the tubes and tube sheet to the water box plate material. The staff finds this acceptable and considers this RAI resolved.

Leakage will be into the condenser since it will normally be operated at a vacuum. The online instrumentation and process sampling system described in DCD Tier 2, Revision 9, Section 9.3.2, monitor leakage of circulating water into the condenser shell. Conductivity and selected impurities are continuously monitored at the discharge of the condensate pumps. High condensate conductivity, which indicates a condenser tube leak, is alarmed in the main control room. The condenser air removal system is discussed in detail in Section 10.4.2 of this report.

The staff reviewed whether the failure of the main condenser system could cause unacceptable condensate quality or flooding of areas housing safety-related components. In DCD Tier 2, Revision 9, Section 3.4.1.4.3, the applicant states that no components in the turbine building can affect the safe shutdown of the reactor. (Section 3.4.1 of this report discusses protection from flooding for safety-related equipment.) The staff finds this acceptable.

In RAI 10.4-3, the staff requested that the applicant provide a detailed description of controlling and correcting methods including alarm setpoints, operator intervention, and plant response as described in SRP Section 10.4.1. In response, the applicant committed to revise the DCD to include threshold values and recommended operator actions for chemistry excursions in the condensate system. In DCD Tier 2, Revision 2, the applicant identified this as COL Information

Item 10.4.10.5. In Revision 3, the applicant removed the information item. The staff asked the applicant to provide a justification for its decision, since the COL applicant must provide this information. In its response, the applicant stated that it would restore this COL information item in DCD Tier 2, Revision 4. In DCD Tier 2, Revision 6 this was identified as COL Information Item 10.4-1-A in DCD Tier 2, Section 10.4.6.3; therefore the staff finds this acceptable and RAI 10.4-3 is resolved.

10.4.1.4 *Conclusions*

As discussed above, the staff reviewed the design of the main condenser in accordance with SRP Section 10.4.1. On the basis of this review, the staff concludes that the main condenser system is acceptable and meets the requirements of GDC 60 with respect to controlling excessive releases of radioactivity to the environment that result from failures in the system design. The applicant meets this requirement by providing suitable radioactivity monitoring, as described in DCD Tier 2, Revision 9, Section 11.5, and measures to prevent a loss of condenser vacuum.

10.4.2 Main Condenser Evacuation System

10.4.2.1 *Regulatory Criteria*

The staff reviewed the main condenser evacuation system in accordance with the acceptance criteria in SRP, Section 10.4.2, Revision 3, issued March 2007. The design of the condenser air removal system is acceptable if its integrated design meets the requirements of GDC 60 of Appendix A to 10 CFR Part 50, as it relates to the condenser air removal system design for the control of releases of radioactive materials to the environment.

The SRP includes RG 1.33, “Quality Assurance Program Requirements (Operation),” in the acceptance criteria. RG 1.123, “Quality Assurance Requirements for Control of Procurement of Items and Services for Nuclear Power Plants,” has been withdrawn and is therefore no longer applicable. The applicant may meet the requirements of GDC 60 and 64 by using the guidance contained in the following RGs and industrial standard:

- RG 1.26, “Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants,” as it relates to the condenser air removal system quality group classification that may contain radioactive materials but is not part of the RCPB and is not important to safety

- RG 1.33, as it relates to the QA programs for the condenser air removal system components that may contain radioactive materials

- The Heat Exchanger Institute’s “Standards for Steam Surface Condensers,” 6th Edition, as it relates to the condenser air removal system components that may contain radioactive materials

10.4.2.2 *Summary of Technical Information*

The condenser air removal system, as depicted in DCD Tier 2, Revision 9, Figure 10.4-2 is designed to remove noncondensable gases from the power cycle. The condenser air removal system removes the hydrogen and oxygen produced by radiolysis of water in the reactor and other power cycle noncondensable gases and exhausts them to the offgas system during plant

power operation and to the turbine building compartment exhaust (TBCE) system during plant startup, cooldown, and low power operation. Condenser vacuum is established and maintained during power operation by either of the two 100-percent capacity, double-stage SJAEs, or by two 50-percent capacity mechanical vacuum pumps during early startup. One SJAE unit is normally in operation and the other is on standby or they can be operated simultaneously in partial load.

The SJAEs are placed in service to remove the gases from the main condenser after vacuum is established in the main condenser by the mechanical vacuum pumps and when sufficient nuclear steam pressure is available. During normal power operations, the SJAEs are normally driven by MS. Auxiliary steam from the auxiliary boiler system (ABS) may be available for normal use of the SJAEs during early startup, as an alternative to the MS or if the mechanical vacuum pumps are unavailable. Section 9.3.12 of this report discusses the ABS.

10.4.2.3 *Staff Evaluation*

The staff reviewed the condenser air removal system to determine the flow paths of gases through the system, including all bypasses, and the points of release of gaseous wastes to the environment or other systems.

In RAI 10.4-10, the staff asked the applicant to revise DCD Tier 2, Revision 2, Figure 10.4-2, to include the location of the auxiliary steam and MS supply connections, which the figure does not show. In response to RAI 10.4-10, GEH committed to revise the drawing to delete the specific reference to the auxiliary steam system because the SJAEs can be supplied from several steam sources. In DCD Tier 2, Revision 3, the applicant revised the drawing to reflect this commitment. The staff finds this acceptable and considers this RAI resolved.

RGs 1.33 and 1.28, “Quality Assurance Program Requirements (Design and Construction),” are applied as they relate to the QA programs for the condenser air removal system components that may contain radioactive materials. The applicant stated in DCD Tier 2, Revision 9, Section 10.4.2.2, that the applicability of Regulatory Guide 1.33 during construction and operation is addressed in Section 17.2 of the DCD. DCD Tier 2, Revision 9, Section 17.2, states that QA responsibilities during construction and operations are in the scope of the COL applicant. This was identified in DCD Tier 2, Revision 9, as COL Information Item 17.2.1-A, which requires that the COL applicant describe the QA Program for the construction and operations phases. The staff finds this acceptable.

The components of the condenser air removal system are designed to Quality Group D as defined in RG 1.26 and are not designed to SSE seismic standards. The applicant stated that the quality standards meet the requirements of 10 CFR 50.55a for water- and steam-containing components that may contain radioactive materials but are not part of the RCPB. Section 3.2 of this report discusses the seismic and quality group classification of components in detail. Based on the staff’s evaluation in Section 3.2 of this report, the staff finds this acceptable.

The staff reviewed whether the condenser air removal system design meets the intent of GDC 64, “Monitoring radioactive releases,” as it relates to the design for monitoring of releases of radioactive materials to the environment. The offgas from the main condenser is one source of radioactive gas in the station. The applicant stated that the discharge of the vacuum pump is routed to the TBCE system because at that point there is very low effluent radioactivity present. Section 11.3 of this report discusses an inventory of radioactive contaminants in the effluent from the SJAEs. Radiation detectors in the TBCE system and plant vent stack will alarm in the

main control room if they detect abnormal radioactivity in the steam being supplied to the condenser. The staff finds this acceptable.

DCD Tier 2, Section 10.4.2.3 states that steam supply to the second-stage ejector is maintained at a minimum specified flow to ensure adequate dilution of hydrogen and prevent the offgas from reaching the flammability limit of hydrogen. In addition, maximum power limits are placed on operation of the mechanical vacuum pumps to ensure that the flammability limit of hydrogen is not reached. In RAI 10.4-5, the staff asked the applicant to provide minimum steam flow, maximum power limit on the operation of the vacuum pump, and design steam content volume percentage, in accordance with SRP Section 10.4.2, to ensure that hydrogen flammability levels are not reached. In response to RAI 10.4-5, the applicant stated that the staff concerns are applicable only if the ESBWR design includes a hydrogen water chemistry (HWC) system. The applicant stated that the HWC system is an option that the owner may choose as a later plant modification and is not offered in the ESBWR standard plant design. The staff finds this acceptable and considers this RAI resolved.

10.4.2.4 *Conclusions*

On the basis of the above discussion, the staff concludes that the condenser air removal system design meets the requirements of GDC 60 and 64 with respect to the control of releases of radioactive materials to the environment and is, therefore, acceptable.

10.4.3 Turbine Gland Seal System (TGSS)

10.4.3.1 *Regulatory Criteria*

The staff reviewed the design of the TGSS in accordance with SRP Section 10.4.3, Revision 3, issued March 2007. The design of the TGSS is acceptable if it meets the requirements of GDC 60, as it relates to the control of releases of radioactive materials to the environment

10.4.3.2 *Summary of Technical Information*

The TGSS, depicted in DCD Tier 2, Revision 9, Figure 10.4-3, minimizes the escape of radioactive steam from the turbine shaft/casing penetrations and valve stems and limits air in-leakage through subatmospheric turbine glands. The TGSS does not perform, ensure, or support any safety-related function and thus has no safety design basis. The high-pressure turbine shaft seals must accommodate a range of turbine shell pressures from full vacuum to full load operating pressure in the shell at the glands. The low-pressure turbine shaft seals operate against a vacuum during normal operation. The gland seal outer portion steam/air mixture is exhausted to the gland steam condenser via the seal vent annulus, which is maintained at a slight vacuum. In addition, the auxiliary boiler steam system is designed to provide a 100-percent backup to the normal gland seal process steam supply. Section 9.3 of this report discusses the auxiliary boiler steam system.

The annular space through which the turbine shaft penetrates the casing is sealed by steam supplied to the shaft seals. Where the gland seals operate against positive pressure, the sealing steam flows either inwards for collection at an intermediate leak-off point or outwards and into the vent annulus. Where the gland seals operate against vacuum, the sealing steam either is drawn into the casing or leaks outward to a vent annulus. At all gland seals, the vent annulus is maintained at a slight vacuum and receives air in-leakage from the outside. From each vent annulus, the air-steam mixture is drawn to the gland steam condenser.

A pressure controller automatically regulates the seal steam header pressure. MS is supplied during normal low load operations. At all loads, including startup and low-load operation, the auxiliary boiler can supply the seal steam. The outer portion of all glands of the turbine and MS valves is connected to the gland steam condenser, which is maintained at a slight vacuum by the exhauster blower. During plant operation, the gland steam condenser and one of the two installed 100-percent capacity motor-driven blowers are in operation. The exhauster blower to the TBCE system effluent stream is continuously monitored before being discharged. The gland steam condenser is cooled by main condensate flow. The TGSS returns the condensed steam to the condenser and exhausts the noncondensable gases, through the TBCE system, to the plant vent.

The applicant stated that the TGSS has enough capacity to handle steam and airflows resulting from greater than normal packing clearances. The TGSS provides for the collection and condensation of sealing steam and the venting and treatment of noncondensable gases. The applicant stated that components are designed to Quality Group D standards, as defined in RG 1.26, and, consistent with the guidance in RG 1.26, the components are not designed to SSE seismic standards.

10.4.3.3 *Staff Evaluation*

The staff reviewed the TGSS to determine the source of sealing steam and the disposition of steam and noncondensables vented from the gland seal to determine if the design meets GDC 60. The TGSS includes the equipment and instruments to provide a source of sealing steam to the annulus space where the turbine and large steam valve shafts penetrate their casings. The scope of the review included the source of sealing steam and the provisions incorporated to monitor and control releases of radioactive material in effluents.

RGs 1.33 and 1.28 are applied as they relate to the QA programs for the TGSS components that may contain radioactive materials. The applicant stated in DCD Tier 2, Revision 9, Section 10.4.3.2.1, that the applicability of Regulatory Guide 1.33 during construction and operation is addressed in Section 17.2 of the DCD. DCD Tier 2, Revision 9, Section 17.2, states that QA responsibilities during construction and operations are in the scope of the COL applicant. This was identified in DCD Tier 2, Revision 9, as COL Information Item 17.2.1-A, which requires that the COL applicant describe the QA Program for the construction and operations phases. The staff finds this acceptable.

The staff reviewed the TGSS with respect to monitoring releases of radioactive materials to the environment. The applicant stated that the TGSS effluents are first monitored by a system-dedicated, continuous, radiation monitor installed on the gland steam condenser exhauster blower discharge. High monitor readings are alarmed in the main control room. The system effluents are then discharged to the TBCE system and the plant vent stack, where further effluent radiation monitoring occurs. Section 12.3.3.3 of this report describes the staff's evaluation of the radiological analysis of the effluents from the turbine building for offsite doses.

Section 11.5 of this report describes the staff's evaluation of the associated radiation monitoring equipment and the COL applicant measures applied to control and monitor effluent releases. Based on the staff's evaluations in Section 11.5 and 12.2 of this report, the staff finds this acceptable.

In RAI 10.4-6, the staff asked the applicant to provide ITAAC in DCD Tier 1 for the TGSS. In response to RAI 10.4-6, the applicant stated that the TGSS does not perform or support safety-

related functions nor does it qualify as important to safety because its failure would not result in an accident. The staff did not agree with this position and held subsequent discussions with the applicant. Although the TGSS is not safety related or important to safety, it does a have a role in controlling and monitoring releases of radioactive materials to the environment, as required by GDC 60 and GDC 64. Subsequently, in DCD Tier 1, Revision 5, GEH included an ITAAC table for the TGSS that will verify the as-built system functional arrangement. The staff finds this acceptable. The staff considers RAI 10.4-6 to be resolved.

10.4.3.4 *Conclusions*

Based on the preceding discussion, the staff concludes that the TGSS is acceptable because it meets the requirements of GDC 60 and 64 for controlling and monitoring releases of radioactive material to the environment. The system also meets the acceptance criteria of SRP Section 10.4.3.

10.4.4 Turbine Bypass System (TBS)

10.4.4.1 *Regulatory Criteria*

The staff reviewed the design of the TBS in accordance with SRP, Section 10.4.4, Revision 3, issued March 2007. The acceptability of the system design is based on meeting the following GDC as described in the SRP:

- GDC 4, as it relates to the system's being designed in such a way that a failure of the system (because of a pipe break or system malfunction) does not adversely affect safety-related systems or components

- GDC 34, "Residual heat removal," as it relates to the ability to use the TBS to shut down the plant during normal operations by removing residual heat without using the turbine generator

10.4.4.2 *Summary of Technical Information*

The TBS provides the capability to discharge MS from the reactor directly to the condenser to minimize step load reduction transient effects on the nuclear boiler system. The TBS is also used to discharge MS during startup reactor hot standby and cooldown operations. Operation of the TBS eliminates the need to rely solely on safety-related systems for shutting down the plant during normal operations.

The TBS, in combination with the reactor systems, provides the capability to accept a full load rejection without reactor trip. The turbine bypass valves are opened by redundant signals received from the turbine SB&PC whenever the actual steam pressure exceeds the preset steam pressure by a small margin. This occurs when the turbine cannot use the entire amount of steam generated by the reactor. This bypass demand signal causes fluid pressure to be applied to the operating cylinder, which opens the first of the individual valves. As the bypass demand increases, additional bypass valves are opened, dumping the steam to the condenser. The bypass valves are equipped with fast-acting solenoid valves to allow rapid opening of bypass valves upon turbine trip or generator load rejection.

10.4.4.3 *Staff Evaluation*

The TBS will not perform or support any safety-related function. There is no safety-related equipment in the vicinity of the TBS, except four position sensors at each bypass valve that provide valve status to the reactor protection system (RPS) logic. In response to RAI 10.4-11, the applicant stated that these sensors are not relied on to shut down the reactor and mitigate the consequences of a postulated piping failure outside containment, and thus are not considered essential components. In addition, the four position sensors, which are mounted on each turbine valve, are fail safe, such that if the bypass valve fails to open or the switch fails to change state during the approximately 200 milliseconds (ms) after the detection of a fast turbine control valve closure or turbine stop valve closure, the RPS scram is not bypassed, and thus the position sensors cannot prevent actuation of the reactor protection function. The staff finds this acceptable and considers this RAI resolved. Section 7.2 of this report discusses RPS operational bypasses. Sections 15.2 and 15.3 of this report discuss failures of the TBS during anticipated operational occurrences and during infrequent events, respectively.

Although the TBS will not be required to serve or support any reactor safety function, it will have a post-LOCA function for the ESBWR. In the absence of an MSIVLCS, the MS lines and condenser will be used to collect MSIV leakage following a LOCA. Therefore, the TBS must be capable of maintaining its integrity following an SSE. The turbine bypass line from the bypass valve to the condenser will be seismically analyzed to demonstrate that it is capable of sustaining the SSE loading conditions without failure. (Section 3.2 of this report contains additional discussion and evaluation of the capability of the turbine bypass piping to meet this requirement.)

The TBS includes turbine bypass valves (TBV) connected to the TMSS main steam lines via TMSS piping. The outlets of the TBVs are connected to the main condenser visa pressure reducers. The scope of review of the TBS for the ESBWR design included layout drawings, P&IDs, and descriptive information for the TBS and the auxiliary supporting systems that are essential to its operation.

The applicant stated that all turbine bypass valves will be tested for operability. The steamlines will be hydrostatically tested to confirm leak-tightness. Pipe weld joints will be inspected accordance with ASME III, Class 2, requirements upstream of the bypass valves and in accordance with ASME B31.1 downstream. The bypass valves will be tested while the unit is in operation. Periodic inspections will be performed on a rotating basis within a preventive maintenance program in accordance with the manufacturer's recommendations. The staff finds this acceptable.

10.4.4.4 *Conclusions*

The basis for accepting the design, design criteria, and design bases of the TBS is their conformance to GDC 4 and 34 of Appendix A to 10 CFR Part 50, as explained below:

- The ESBWR TBS design meets the requirements of GDC 4 such that its failure will not prevent the plant's safe shutdown.
- The ESBWR design meets the requirements of GDC 34 with respect to the ability to use the TBS to shut down the plant during normal operations. The TBS is designed such that sufficient steam can be bypassed to the main condenser so that the plant can be shut down during normal operations without using the turbine generator.

Based on the preceding, the staff concludes that the design of the TBS conforms to SRP Section 10.4.4, meets the requirements of GDC 4 and 34 and is, therefore, acceptable.

10.4.5 Circulating Water System (CIRC)

10.4.5.1 *Regulatory Criteria*

The staff reviewed the CIRC in accordance with SRP Section 10.4.5, Revision 3, issued March 2007. Acceptability of the system is based on meeting the requirements of GDC 4, as they relate to provisions in the ESBWR design to accommodate the effects of discharging water that may result from a failure of a component or piping in the CIRC. Compliance with GDC 4 is based on meeting the relevant acceptance criteria specified in the SRP, such as the following:

- Means to prevent, detect, and control flooding of safety-related areas resulting from leakage from the CIRC

- Means to prevent adverse effects of malfunction or failure of CIRC piping on functional capabilities of the safety-related systems or components

- Control of water chemistry, corrosion, and organic fouling in the CIRC

10.4.5.2 *Summary of Technical Information*

The CIRC consists of condenser water boxes and piping and valves, as well as water box drain subsystem. The cooling water is circulated by four motor-driven pumps. The pumps are arranged in parallel, and discharge lines combine into two parallel circulating water supply lines to the main condenser. Each circulating water supply line connects to a low pressure condenser shell inlet water box. An interconnecting line fitted with a butterfly valve is provided to connect both circulating water supply lines. The discharge of each pump is fitted with a remotely operated valve. This arrangement permits isolation and maintenance of any one pump while the others remain in operation and minimizes the backward flow through a tripped pump.

The CIRC and condenser are designed to permit isolation of each set of the three series-connected tube bundles to permit repair of leaks and cleaning of water boxes while operating at reduced power. The CIRC includes water box vents to help fill the condenser water boxes during startup and removes accumulated air and other gases from the water boxes during normal operation.

A chemical additive subsystem is also provided to prevent the accumulation of biological growth and chemical deposits within the wetted surfaces of the system.

10.4.5.3 *Staff Evaluation*

The staff reviewed the CIRC to verify that it meets GDC 4, as it relates to accommodating the effects of discharging water that may result from a failure of a component or piping in the CIRC by providing a means to prevent or detect and control flooding of safety-related areas. Level switches in the turbine building condenser area trip the pumps and close the valves of the CIRC in case of a system component failure. The flooding signal initiates from the detection of a high water level.

The staff reviewed the system to verify that a malfunction or failure of a component or piping will not have unacceptable adverse effects on the functional performance capabilities of safety-related systems or components. The CIRC provides cooling water for removal of the power cycle waste heat from the main condensers and transfers this heat to the normal power heat sink. The applicant stated that CIRC does not interface with any safety-related SSC, and a CIRC failure could not adversely affect any safety-related SSC.

The applicant performed a flooding analysis of the turbine building, postulating a complete rupture of a single expansion joint. If a circulating water system pipe, water box, or expansion joint failure is not detected and isolated, the water discharged would cause internal turbine building flooding up to slightly above grade level, with excess water potentially spilling over on site. If a failure occurred within the condensate system (condenser shell side), the resulting flood level would be below grade level, because of the relatively small hotwell inventory compared to the turbine building capacity. The staff finds this acceptable. Section 3.4 of this report contains a detailed description of general flooding provisions.

The applicant stated that certain portions of the system are conceptual design information and are outside the scope of the ESBWR standard plant. These include the (1) screen house and intake screens, (2) pumps and pump discharge valves, and (3) related support facilities such as the makeup water system, water treatment, inventory blowdown, tube cleaning system, and maintenance equipment. In addition, the DCD states that some site-dependent system design features and additional information are also outside the scope of the ESBWR design certification. These include the (1) compatible design as described in DCD Section 10.4.5.2, (2) evaluation per DCD Section 10.4.5.3, (3) tests and inspections per DCD Section 10.4.5.4, (4) instrument applications per DCD Section 10.4.5.5, and (5) flood protection per DCD Section 10.4.5.6. Before Revision 3, the applicant had identified this information in the DCD as COL Action Item 10.4.10.4. However, in Revision 3, the applicant removed this action item and identified this information as conceptual design information in the DCD. The staff finds this to be acceptable.

10.4.5.4 *Conclusions*

On the basis of its review the staff concludes that the design of the CIRC meets the requirements of GDC 4, with respect to the effects of discharging water that may result from a failure of a component or piping in the CIRC. Acceptance is based on the following design provisions:

- The CIRC is designed to prevent flooding of safety-related areas so that leakage from the CIRC will not preclude the intended safety function of a system or component.
- The CIRC is designed to detect and control flooding of safety-related areas so that leakage from the CIRC will not preclude the intended safety function of a system or component.
- Malfunction of a component or piping of the CIRC, including an expansion joint, will not have unacceptable adverse effects on the functional performance capabilities of safety-related systems or components.

The staff concludes that the design of the CIRC meets the acceptance criteria of SRP Section 10.4.5 and thereby, the requirements of GDC 4. The staff, therefore, finds the design acceptable.

10.4.6 Condensate Purification System

10.4.6.1 *Regulatory Criteria*

The staff reviewed the CPS description in accordance with SRP Section 10.4.6. Staff acceptance of the design is based on compliance with the requirements of GDC 14, "Reactor coolant pressure boundary" as it relates to the water chemistry control being capable of preventing adverse chemistry conditions that could degrade the primary coolant boundary integrity.

RG 1.56, "Maintenance of Water Purity in Boiling Water Reactors," Revision 1, July 1978, describes a method acceptable to the NRC staff for implementing the criteria with regard to minimizing the probability of corrosion-induced failure of the RCPB in BWRs by maintaining acceptable purity levels in the reactor coolant. It further describes instrumentation acceptable to the NRC staff for determining the condition of the reactor coolant and coolant purification system.

10.4.6.2 *Summary of Technical Information*

The condensate purification system (CPS) purifies and treats the condensate to maintain reactor FW purity. The CPS uses filtration to remove suspended solids, including corrosion products, and demineralizers to remove dissolved solids from condenser leakage and other impurities. The CPS consists of the following major components:

- Filters
- Demineralizers
- Resin storage tank
- Resin receiver tank
- Filter backwash tank

The CPS does not perform any safety-related functions.

10.4.6.3 *Staff Evaluation*

The staff reviewed the CPS description in accordance with SRP Section 10.4.6. Staff acceptance of the design is based on compliance with the requirements of GDC 14 as related to assuring the integrity of the RCPB.

The CPS removes dissolved and suspended solids from the condensate in addition to some radioactive material, activated corrosion products, and fission products that are carried over from the reactor, to maintain a high quality of FW to the reactor under all normal plant operating conditions. The CPS will also remove corrosion products from the condensate to limit any accumulation of corrosion products in the cycle.

The CPS consists of six back-washable filters and eight mixed-bed demineralizers arranged in parallel. One demineralizer is normally on standby. Demineralizers are equipped with a resin trap downstream of each vessel to prevent resin from entering the effluent and to catch resin fine leakage as much as possible. Demineralizers have a bypass valve which can be controlled manually or automatically from the main control room. The CPS operates continuously to maintain FW purity levels at all times. Waste generated in the CPS is sent to the radwaste system for treatment and/or disposal.

The CPS contains instrumentation that monitors different parameters throughout the system. The parameters monitored in the CPS are conductivity, differential pressure, and flow. Conductivity of the condensate flow is measured just before entrance to the system and at the outlet flow of the demineralizers. Measuring conductivity just before entrance to the system helps detect condenser leakage, whereas conductivity measured at the outlet flow of the demineralizers provides indication of resin exhaustion. Differential pressure is measured across each filter vessel, demineralizer vessel, and across each vessel discharge resin strainer to help detect flow blockage. Condensate flow is measured through each demineralizer and used as input to ensure that the flow is distributed evenly through all operating demineralizing vessels.

All of these parameters are indicated at the CPS local control panel. Any parameter that is not within its required value will be alarmed in the control panel, which is connected to the main control room where all these alarms are directed.

The applicant stated that the CPS complies with RG 1.56 “Maintenance of Water Purity in Boiling Water Reactors”, Rev 1, 1978. However, the application was unclear as to whether the CPS complies with EPRI Report NP-4947-SR, “BWR Hydrogen Water Chemistry Guidelines,” 1987 Revision. Therefore, in RAI 10.4-1, the staff asked the applicant to clarify whether the CPS complies with the guidelines in EPRI Report NP-49-47-SR. In response, the applicant stated that the HWC system is not offered in the ESBWR standard plant design, although provisions have been made to install the system as a COL applicant option. The applicant stated that if the COL applicant considers the option to include the HWC system, the CPS will be modified, as required, to comply with the subject EPRI chemistry guidelines. The staff finds the applicant’s response acceptable and considers RAI 10.4-1 resolved.

The CPS components and related support facilities are located in the turbine building and other non-safety-related buildings. Any component failure of the CPS will not compromise any safety-related system or component nor will it preclude the ability to achieve and maintain a safe shutdown.

10.4.6.4 *Conclusions*

The CPS includes all components and equipment necessary for the removal of dissolved and suspended impurities that may be present in the condensate.

Based on its review of the applicant’s proposed design criteria and design bases for the CPS and the requirements for operation of the system, the staff concludes that the design of the CPS and supporting systems is acceptable and meets the primary boundary integrity requirements of GDC 14. The staff reached this conclusion because the applicant’s design meets the requirements of GDC 14 as it relates to maintaining acceptable chemistry control for reactor coolant during normal operation and anticipated operational occurrences by reducing corrosion of reactor system components. The design of the CPS meets the regulatory positions of RG 1.56, Revision 1.

Based on this information, the staff concludes that the CPS design for the ESBWR is acceptable.

10.4.7 Condensate and Feedwater System

10.4.7.1 *Regulatory Criteria*

The staff reviewed the CFS in accordance with SRP Section 10.4.7, Revision 4, issued 2007. Conformance with the acceptance criteria of the SRP forms the basis for concluding that the CFS satisfies the following criteria:

- GDC 2, with respect to withstanding the effects of natural phenomena (such as earthquakes, tornadoes, and floods)
- GDC 4, with respect to withstanding the effects of possible fluid flow instabilities (such as water hammers)
- GDC 5, "Sharing of structures, systems, and components," with respect to the ability of the shared systems and components important to safety to perform required safety functions
- GDC 44, "Cooling water," with respect to the capability to transfer heat loads from the reactor system to a heat sink under both normal operating and accident conditions
- GDC 45, "Inspection of cooling water system," with respect to permitting periodic ISI of systems, components, and equipment
- GDC 46, "Testing of cooling water system," with respect to design provisions to permit functional testing of the system and components for structural integrity and leak-tightness

10.4.7.2 *Summary of Technical Information*

The CFS consists of the piping, valves, pumps, heat exchangers, controls, and instrumentation and the associated equipment and subsystems that supply the reactor with heated FW in a closed steam cycle utilizing regenerative FW heating. The system is divided into two subsystems: (1) piping and components extending from the RPV inside the containment, to the seismic interface restraint located upstream of the outermost FW isolation valve, outside of the containment, and (2) piping, pumps, valves, heat exchangers, controls, and instrumentation extending from the main condenser outlet to, but not including, the seismic interface restraint. DCD Tier 2, Revision 9, Section 5.4.9, describes subsystem (1), as discussed above.

The FW lines are routed from the turbine building to the MS and FW pipe tunnel, through containment penetrations, at which point they branch into six lines that connect to the RPV in the upper drywell. There is a connection at each of the two lines for detection and monitoring of differential pressure between the two FW lines. The six branch lines inside containment provide FW flow distribution to the RPV. The control rod drive system injection line connects to the reactor water cleanup/shutdown cooling (RWCU/SDC) system loop "A" return line, which is connected to a thermal sleeve in the "B" FW line in the tunnel. The fuel and auxiliary pool cooling system low-pressure coolant injection line connects to the RWCU/SDC system loop "B" return line, which connects to the "A" FW line in the tunnel.

The CFS consists of four 33.3- to 37-percent capacity condensate pumps (three normally operating and one on automatic standby), four 33.3-percent nominal capacity FW booster pumps (three normally in operation and one in automatic standby), four 33.3- to 45-percent capacity reactor FW pumps (three normally in operation and one on automatic standby), three

stages of low-pressure closed FW heaters, a direct contact FW heater (FW tank), and three stages of high-pressure FW heaters, piping, valves, and instrumentation. The condensate pumps take suction from the condenser hotwell and discharge the deaerated condensate into one common header, which feeds the CPS. Downstream of the CPS, the condensate is taken by a single header, through the auxiliary condenser/coolers, one gland steam exhauster condenser, and two sets of SJAE condensers and offgas recombiner condensers (coolers). The condensate then branches into parallel strings of low-pressure FW heaters. Each string contains three stages of low-pressure FW heaters that join together at a common header, which is routed to the open FW tank. The FW booster pumps take suction from the open FW tank and provide adequate suction head for the reactor FW booster pumps.

The reactor FW pumps discharge into two parallel high-pressure FW heater strings, each with three stages of high-pressure FW heaters. Downstream of the high-pressure FW heaters, the two strings are then joined into a common header, which divides into two FW lines that connect to the reactor. A bypass is provided around the FW tank and reactor FW pumps to permit the supply of FW to the reactor during early startup without operating the FW pumps, using only the condensate pumps.

One more bypass, equipped with a flow control valve, is provided around the high-pressure heaters to provide a flow path around a single string for heater maintenance/failure or for reducing final FW temperature to extend the end of the fuel cycle. During power operation, the condensate is deaerated in the condenser, and continuous oxygen injection is used to maintain the level of oxygen content in the final FW. To minimize corrosion product input to the reactor during startup, recirculation lines to the condenser are provided from the high-pressure FW heater outlet header.

The DCD states that before plant startup, FW cleanup is accomplished by allowing the system to recirculate through the condensate polishers for treatment before feeding any water to the reactor during startup. Section 10.5.6 of this report discusses the condensate cleanup system.

During operation, radioactive steam and condensate are present in the FW heating portion of the system, which includes the extraction steam piping, FW heater shells, heater drain piping, and heater vent piping. Chapter 12 of this report discusses shielding and access control provisions.

10.4.7.3 *Staff Evaluation*

The staff reviewed the system to determine that it meets GDC 2 as it relates to the ability to withstand the effects of earthquakes. The FW lines are designed as Quality Group A and ASME Section III, Class 1, from the RPV through the outboard isolation check valves, and Quality Group B and ASME Section III, Class 2, through the isolation shutoff valves to the seismic interface restraint. The FW lines are seismic Category I from the RPV to the seismic interface restraint upstream of the isolation shutoff valve, seismic Category II to the last FW heater, and nonseismic thereafter. (Section 3.2 of this report discusses the details of seismic classification.) The staff finds this acceptable.

The staff reviewed the system to determine if it meets GDC 4 with regard to protection against the effects of high-energy pipe ruptures and with respect to withstanding the effects of possible fluid flow instabilities (such as water hammer). The piping design pressure and temperature of the Class 1 portions are, respectively, 8.62 MPa gauge (1250 psig) and 302 degrees C (576 degrees F).

The applicant stated that the FW control system is designed to ensure that there could not be large sudden changes in FW flow that could induce water hammer. During normal operation, FW flow is varied as needed by using the adjustable speed of the motor-driven feed pumps, which eliminates the need for flow control valves and thus minimizes the likelihood of a water hammer event. During low-flow conditions (less than 25 percent of rated reactor power), the FW control system uses single-element control based on vessel water level.

Single-element control reduces the likelihood of water hammer events by minimizing valve cycling at low loads when compared to three-element controllers. In this mode, the conditioned level error from the master level controller is used to determine the demand to either the low-flow control valve or to an individual feed pump adjustable speed drive. Section 7.7 of this report provides a detailed discussion of the FW control system.

The staff finds that the FW control system includes adequate design considerations to avoid water hammer events and is consistent with the guidelines of NUREG–0927, "Evaluation of Water Hammer Occurrences in Nuclear Power Plants." DCD Tier 2, Revision 9, Section 10.4.7, states that the operating and maintenance procedures include adequate precautions to avoid steam hammer. Section 3.4.1 of this report discusses protection of safety-related equipment from flooding. Based on the preceding discussion, the staff finds that the ESBWR FW system includes adequate considerations to avoid and withstand the effects of high-energy pipe ruptures and of fluid flow instabilities as required by GDC 4.

The requirements of a GDC 5 are not applicable to the ESBWR design because it is designed as a single unit.

The staff concludes that the system meets GDC 44 requirements, as it relates to the capability to transfer heat loads from the reactor system to a heat sink under both normal operations and regarding provisions for redundancy and isolation of components, subsystems, or piping. The staff concludes that failure of the CFS will not compromise any safety-related system or function or prevent safe shutdown as demonstrated by the results of the CFS component failure analysis provided in DCD Tier 2, Revision 9, Table 10.4-6. The CFS trip logic and control schemes respectively use coincident logic and redundant controllers, and input signals to assure that plant availability goals are achieved and spurious trips are avoided. This specifically includes all FW heater level controllers, all CFS flow and minimum flow controllers, pump suction pressure trips, FW heater string isolation/high-level trips, and CFS bypass system(s) operation.

The staff concludes that the system meets GDC 45, as related to permitting periodic ISI of system components and equipment, and GDC 46, as related to design provisions to permit appropriate functional testing of the system and components to assure structural integrity and leak-tightness, operability and performance of active components, and capability of the integrated system to function as intended during normal, and shutdown conditions. The performance status, leak-tightness, and structural leak-tight integrity of all system components are demonstrated by continuous operation. The applicant stated that each FW heater and condensate pump receives a shop hydrostatic test, which is performed in accordance with applicable codes. All tube joints of FW heaters are shop leak tested. Before initial operation, the complete CFS will receive a field hydrostatic and performance test and inspection. Periodic tests and inspections of the system are performed in conjunction with scheduled maintenance outages. The Class 1 portions of the system are inspected and tested in accordance with the requirements of the ASME Boiler and Pressure Vessel Code, as discussed in Section 5.2.4 of this report.

Section 10.3.6 of this report discusses considerations for piping systems, including material standards and inspection programs, to avoid erosion and corrosion effects and compliance with GL 89-08 and the guidelines, in EPRI NCAC-202L-R2, “Recommendations for an Effective Flow Accelerated Corrosion Program.” Section 6.6 of this report discusses preservice inspection and inservice inspection provisions. Based on the staff evaluation described in Section 6.6, the staff finds this acceptable.

DCD Revision 3, Section 3.9.3.2, states that the FW nozzle design incorporates the requirements in NUREG–0619, “BWR Feedwater Nozzle and Control Rod Drive Return Line Nozzle Cracking: Resolution of Generic Technical Activity A-10,” Revision 1, issued November 1980; GL 80-95, “Generic Activity A-10,” dated November 13, 1980; and GL 81-11 “BWR Feedwater Nozzle and Control Rod Drive Return Line Nozzle Cracking.” However, in addition to the design considerations, the staff requested the applicant to confirm that the ESBWR complies with all NUREG–0619 provisions, including an FW nozzle PSI and ISI program. Specifically, in RAI 10.4-12, the staff requested GEH to confirm that the ESBWR FW nozzles are designed to provide access for the examinations described in NUREG–0619, in accordance with ASME Section XI requirements. The staff also asked the applicant to include a COL action item to ensure that the COL applicant will include the provisions of NUREG–0619 in its PSI and ISI inspection programs. The applicant provided a supplemental response to clarify this issue which the staff found acceptable. The applicant has included COL Information Item 5.2-1-A in DCD Tier 2, Revision 9, Section 5.2.6 to address development of the PSI and ISI programs by the COL applicant. The staff considers RAI 10.4-12 resolved.

10.4.7.4 *Conclusions*

On the basis of its review, the staff concludes that the design of the CFS meets the NRC regulations in GDC 2, 4, 44, 45, and 46 and is, therefore, acceptable. The following provides the basis for this conclusion:

- The ESBWR meets the requirements of GDC 2 with respect to the system’s ability to withstand the effects of earthquakes by conforming to RG 1.29.

- The ESBWR meets the requirements of GDC 4 with respect to the dynamic effects associated with high-energy piping failures and possible fluid flow instabilities.

- The ESBWR meets the requirements of GDC 44 because the applicant demonstrated that failure of this system cannot compromise any safety-related system or function, or prevent safe shutdown.

- The ESBWR meets the requirements of GDC 45 and GDC 46 because the system will be tested and inspected in accordance with the applicable codes and regulatory requirements. Periodic tests and inspections of the system are performed in conjunction with scheduled maintenance outages.

The requirements of a GDC 5 are not applicable to the ESBWR design because it is designed as a single unit.

10.4.8 Steam Generator Blowdown System (PWR)

Not applicable to the ESBWR design.

10.4.9 Auxiliary Feedwater System (PWR)

Not applicable to the ESBWR design.

11.0 RADIOACTIVE WASTE MANAGEMENT

11.1 Source Terms

11.1.1 Regulatory Criteria

No specific regulatory criteria are directly applicable to this section. The U.S. Nuclear Regulatory Commission (the NRC or staff) evaluated the information in Chapter 11 of the economic simplified boiling-water reactor (ESBWR) design control document (DCD), Tier 2, Revision 9, against the criteria in Chapter 11 of NUREG–0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants (LWR Edition)," March 2007 (hereafter referred to as the SRP). The staff used the source terms provided by the applicant in Section 11.1, of the DCD Tier 2, Revision 9, to evaluate the liquid waste management system (LWMS) and gaseous waste management system (GWMS) described in DCD Tier 2, Revision 9, Sections 11.2 and Section 11.3. Sections 11.2 and 11.3 of this report detail the staff's evaluation of DCD Tier 2, Revision 9, Sections 11.2 and 11.3, respectively.

The following acceptance criteria apply to Sections 11.2 and 11.3 of this report:

- Title 10, of the *Code of Federal Regulations* (CFR), Part 20, "Standards for Protection Against Radiation," as it relates to dose limits for members of the public and effluent concentration limits in unrestricted areas

- 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," Appendix I, "Numerical Guides for Design Objectives and Limiting Conditions for Operation to Meet the Criterion 'As Low as Is Reasonably Achievable' for Radioactive Material in Light-Water-Cooled Nuclear Power Reactor Effluents," as it relates to the numerical guidelines for design objectives and limiting conditions for operation to meet the criterion to keep exposures "as low as is reasonably achievable" (ALARA) in Appendix I

- 10 CFR Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants," General Design Criterion (GDC) 60, "Control of releases of radioactive materials to the environment," as it relates to radioactive waste management system (RWMS) designs to control releases of gaseous and liquid radioactive effluents and to handle solid radioactive wastes produced during normal operation

The staff also used the source terms provided by the applicant in DCD Tier 2, Revision 9, Section 11.1 in evaluating plant radioactive sources described in DCD Tier 2, Revision 9, Sections 12.2 and 12.3. Sections 12.3 and 12.4 of this report describe the staff's evaluation of DCD Tier 2, Revision 9, Sections 12.2 and 12.3, respectively.

The regulatory positions and guidance in the following NRC regulatory guides (RGs) and industry standards apply to this section:

- RG 1.112, Revision 1, "Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Light-Water-Cooled Power Reactors," as it relates to the method of calculating releases of radioactive materials in effluents from nuclear power plants;

- American National Standards Institute/American Nuclear Society (ANSI/ANS) 18.1-1999, "Radioactive Source Term for Normal Operation of Light-Water Reactors;"

- NUREG–0016, Revision 1, “Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Boiling Water Reactors” (BWR-GALE code).

The staff performed its primary review of this section of the DCD using SRP Section 11.1, 1981. The staff then performed a comparison of the SRP used during the review of the DCD (i.e., SRP Section 11.1, 1981 version) with the 2007 version of the SRP. The 2007 version includes additional guidance on the use of methods to develop primary coolant and steam radioactive source terms other than that described in the earlier version of the SRP. The staff considered this additional guidance in its review of the DCD. Discussions and dispositions of these additional items are provided in this and other supporting sections of this report. Therefore, the staff concludes that the version of the SRP used, in combination with the additional review performed by the staff, is adequate for this review.

11.1.2 Summary of Technical Information

The ESBWR RWMS controls the handling and treatment of liquid, gaseous, and solid radioactive wastes. The system comprises the LWMS, the GWMS, and the solid waste management system (SWMS). DCD Tier 2, Revision 9, Section 11.1 describes the sources of radioactivity (source terms) processed by the RWMS.

DCD Tier 2, Revision 9, Section 11.1 defines the radioactive source terms in reactor coolant and steam as the design bases and normal operation source terms for the gaseous, liquid, and solid RWMS. Radioactive fission products are generated within the fuel assemblies and can leak to the reactor coolant system during normal plant operation, including anticipated operational occurrences (AOOs). As operational plant events, AOOs include unplanned releases of radioactive materials associated with equipment failures, operator errors, and administrative errors, with radiological consequences that are not considered accident conditions. The applicant described two types of source terms for the reactor primary coolant and steam. The first addresses the design basis, and the second describes the anticipated average concentrations in reactor coolant and steam over the life of a boiling-water reactor (BWR). These source terms serve as a basis for the RWMS design and shielding analysis and for the purpose of assessing doses to members of the public. DCD Tier 2, Revision 9, Tables 11.1-1 through 11.1-7b give the source terms and supporting assumptions.

11.1.2.1 *Design-Basis Source Term*

The design-basis source term is used for the plant equipment design and radiation shielding requirements.

The first category of design-basis source terms is the noble gas source term, which assumes a fuel defect level that produces 3700 megabecquerels (MBq) per second (s) (0.1 curie [Ci] per second) of noble gases after 30-minute decay. The applicant chose the noble gas source term rate after 30-minute decay as a measure of the fuel defect leakage rate based on BWR operating experience with offgas systems (OGSs) (General Electric licensing topical reports NEDO-10871 and NEDO-21159, as referenced in DCD Tier 2, Revision 9, Section 11.1).

The second category of design-basis source terms is the radioiodine source term, which is associated with leakage from failed fuel. The presence of radioiodines is based on a leak rate of 26 MBq per second (700 microcuries [µCi] per second) from the fuel. The applicant assumed the ratio of the concentration of radioiodines in coolant to that of reactor steam to be 0.02, using ANSI/ANS 18.1-1999 as the basis.

The third design-basis source term category is the fission products source term, which includes all radionuclides other than noble gases and radioiodines. The fission products included in the source term are based on ANSI/ANS 18.1-1999 and include transuranics. The applicant assumed the ratio of the concentration of fission products in reactor coolant to that of reactor steam to be 0.001, using ANSI/ANS 18.1-1999 as the basis.

The last category of design-basis source terms includes coolant activation products, non-coolant activation products, and argon. Coolant activation products include nitrogen–16 (N–16) and tritium. The concentrations of N–16 and tritium are based on ANSI/ANS 18.1-1999 tabulations. The presence of N–16 results from the neutron activation of naturally occurring oxygen–16. The presence of tritium in coolant primarily results from the activation of naturally occurring deuterium in water and, to a lesser extent, appears as a fission product in fuel. The applicant assumed the reactor coolant, process water, and steam to have a common tritium concentration, as tritium is not reduced by coolant cleanup systems or liquid waste treatment systems. The source term for argon–41 (Ar–41), an activation product of naturally occurring Ar-40, is based on NUREG–0016, but adjusted to a power level of 4500 megawatts thermal. The level of Ar–41 in coolant primarily depends on air in-leakage into the primary coolant system. Neutron activation of circulating impurities and corrosion of irradiated system materials are the cause of non-coolant activation products in the coolant. The concentration of non-coolant activation products is based on ANSI/ANS 18.1-1999 tabulations. The applicant assumed the ratio of the concentration of non-coolant activation products in reactor coolant to that of reactor steam to be 0.001, using ANSI/ANS 18.1-1999 as the basis.

11.1.2.2 *Normal Operation Source Term*

The normal operation source term is used to calculate the quantity of radioactive materials released annually in liquid and gaseous effluents during normal plant operation, including AOOs, to demonstrate compliance with the effluent concentration limits in Table 2 of Appendix B, "Annual Limits on Intake (ALIs) and Derived Air Concentrations (DACs) of Radionuclides for Occupational Exposure; Effluent Concentrations; Concentrations for Release to Sewerage," to 10 CFR Part 20, the dose limits set forth in 10 CFR 20.1302, "Compliance with dose limits for individual members of the public;" and the ALARA design objectives of Appendix I to 10 CFR Part 50.

The normal operation source term is the expected average concentration of the principal radionuclides in the reactor coolant and steam over the life of a BWR. The applicant assumed a realistic design-basis fuel defect level of 740 MBq per second (0.02 Ci per second) of noble gases released after 30-minute decay. For radioiodines, the estimated release rate is 3.7 MBq per second (100 µCi per second). The applicant determined these values using the model in ANSI/ANS 18.1-1999 and NUREG–0016.

11.1.3 Staff Evaluation

DCD Tier 2, Revision 9 did not use the methods and parameters described in NUREG–0016. Rather, the radioactive source terms defined in the DCD derive from ANSI/ANS 18.1-1999. Calculating the source term using ANSI/ANS 18.1-1999 is an alternative method listed in RG 1.112. RG 1.112 defines expected long-term radionuclide concentrations in the coolant and steam of BWRs. RG 1.112 provides a uniform approach for determining concentrations of principal radionuclides for a reference BWR plant and provides a method for adjusting radionuclide concentrations to a specific plant design. The data defining the reference plant reflect industry experience at operating BWR plants. The adjustment of radionuclide

concentrations from the reference plant to a specific plant design requires information for various plant system parameters. The major parameters include plant thermal power, mass of water in the reactor vessel, cleanup demineralizer flow rate, steam flow rate, and ratio of condensate demineralizer to steam flow rate. Other parameters address factors characterizing the types of systems used to purify reactor coolant and cleanup efficiencies of such systems by class of radionuclides.

The source terms provide the bases for estimating typical concentrations of the principal radionuclides for operating BWR plants. The applicant used the source terms, in part, in DCD Tier 2, Sections 11.2, 11.3, and 12.2.2 to calculate the quantity of radioactive materials released annually in liquid and gaseous effluents during normal plant operation, including AOOs, and to demonstrate compliance with the effluent concentration limits in Table 2 of Appendix B to 10 CFR Part 20, dose limits to members of the public in 10 CFR 20.1302, and the ALARA design objectives of Appendix I to 10 CFR Part 50.

While reviewing prior versions of the DCD Tier 2, the staff asked the applicant to provide additional information, as requests for additional information (RAIs). The staff issued a number of RAIs, not listed here for the sake of brevity, during the review of the application. These RAIs involved requests for the applicant to provide clarifications for technical completeness, provide details supporting design bases and design descriptions in demonstrating compliance with regulatory requirements, revise technical and regulatory references, and provide information for the staff to conduct independent evaluations of results presented in the application. These RAIs were satisfactorily resolved by the applicant and closed by the staff in DCD Tier 2, Revision 6. The following paragraphs discuss the staff's evaluations of the applicant's responses to these RAIs on important technical and regulatory topics.

In RAI 11.1-1a, the staff asked the applicant to provide information on the parameters used to calculate concentrations of radioactive materials in primary and secondary coolant to ensure that they were consistent with NUREG–0016. The applicant responded that this information is addressed in DCD Tier 2, Section 12.2.2 and in the responses provided to RAIs in DCD Tier 2, Chapter 12. The staff found in DCD Tier 2, Section 12.2.2 and in the responses provided to RAIs in DCD Tier 2, Chapter 12 that the applicant used ANSI/ANS 18.1-1999 to calculate concentrations of radioactive materials in the primary coolant and steam as an acceptable alternative to that of RG 1.112 and NUREG–0016. Based on the applicant's response, this RAI is resolved. The staff confirmed that this change was included in DCD Tier 2, Revision 3.

In RAI 11.1-1b, the staff asked the applicant to provide information on all normal and potential sources of radioactive effluents delineated in Section 11.1, Subsection I, of NUREG–0800. The applicant provided a list of the sources for both BWR liquid and gaseous wastes, which are addressed in DCD Tier 2, Section 12.2. The applicant's normal and potential sources of radioactive effluents are consistent with those listed in SRP Section 11.1 as sources of liquid and gaseous waste. Based on the applicant's response, this RAI is resolved. The staff confirmed that this change was included in DCD Tier 2, Revision 3.

In RAIs 11.1-2 and 11.1-3, the staff asked the applicant to provide average operational source terms for fission, activation, and corrosion products in reactor water and steam and to provide all calculation parameters used to determine the average source terms. The applicant provided in DCD Tier 2, Revision 5, Tables 11.1-1 through 11.1-7b the average operational source terms. The staff performed an independent confirmatory calculation of the average operational source terms using the methodology provided in ANSI/ANS 18.1-1999 and finds that the applicant's

values are acceptable. Based on the applicant's response, this RAI is resolved. The staff confirmed that this change was included in DCD Tier 2, Revision 3.

In RAI 11.1-4, the staff asked the applicant to clarify the values provided for noble gases and zinc–65 (Zn–65) in the coolant source term. The applicant provided the adjustment factor used in its calculation for Zn–65 and described how the design-basis noble gas leakage rate was used to determine the noble gas concentrations. The staff performed an independent confirmatory calculation of the Zn–65 and noble gas concentration using the methodology provided in ANSI/ANS 18.1-1999 and in the applicant's information, and it finds that the applicant's values are acceptable. Based on the applicant's response, this RAI is resolved. The staff confirmed that this change was included in DCD Tier 2, Revision 3.

11.1.4 Conclusions

The staff reviewed the reactor coolant and steam source terms for the ESBWR design. Based on the information discussed above, the staff concludes that the applicant calculated the ESBWR coolant and steam source terms in accordance with the guidance of RG 1.112. Therefore, the staff further concludes that the source terms provided in DCD Tier 2, Revision 9, Section 11.1 are acceptable for evaluating the LWMS and GWMS described in DCD Tier 2, Revision 9, Sections 11.2 and 11.3 respectively.

11.2 Liquid Waste Management System

11.2.1 Regulatory Criteria

The staff reviewed DCD Tier 2, Revision 9, Section 11.2 in accordance with the guidance and acceptance criteria described in SRP Section 11.2. The following acceptance criteria are applicable:

- 10 CFR 20.1302, as it relates to limits on doses to persons and liquid effluent concentrations in unrestricted areas (these criteria apply to releases resulting from the LWMS during normal plant operations and AOOs)
- 10 CFR 20.1406, "Minimization of contamination," as it relates to facility design and operational procedures for minimizing facility contamination and the generation of radioactive waste
- 10 CFR 50.34a, as it relates to the inclusion of sufficient design information to demonstrate the design objectives for equipment necessary to control releases of radioactive effluents to the environment
- Sections II.A and II.D of Appendix I to 10 CFR Part 50, as they relate to the numerical guidelines for dose design objectives to meet the ALARA criterion and cost-benefit analysis
- GDC 60, as it relates to the design of LWMS to control releases of liquid radioactive effluents
- GDC 61, "Fuel storage and handling and radioactivity control," as it relates to the design of LWMS to ensure adequate safety under normal operations and postulated accident conditions

The following RGs contain the regulatory positions and guidance for meeting the relevant requirements of the regulations identified above:

- RG 1.109, “Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I,” as it relates to demonstrating compliance with the numerical guidelines for dose design objectives and the ALARA criterion of Appendix I to 10 CFR Part 50

- RG 1.110, “Cost-Benefit Analysis for Radwaste Systems for Light-Water-Cooled Nuclear Power Reactors (for comment),” as it relates to performing a cost-benefit analysis for reducing the cumulative dose to the population by using available technology

- RG 1.143, Revision 2, “Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants,” as it relates to the seismic design and quality group classification of components used in the LWMS and the structures housing this system, as well as the provisions used to control leakages

- RG 4.21, “Minimization of Contamination and Radioactive Waste Generation: Life-Cycle Planning”

- SRP Section 11.2, Branch Technical Position (BTP) 11-6, Revision 3, “Postulated Radioactive Releases Due to Liquid-Containing Tank Failures,”

The staff performed a comparison of the SRP (Section 11.2, 1981 version) used during the review of the DCD with the 2007 version of the SRP. The 2007 version includes additional acceptance criteria and guidance addressing the requirements of 10 CFR 20.1406, when compared to the prior version of the SRP. However, the requirements of 20.1406 were considered in the staff‘s review of the DCD, given the 2007 version of the SRP. Discussions and dispositions of these items are provided in this and other supporting sections of this report. Therefore, the staff concludes that the version of the SRP used, in combination with the additional review performed by the staff, is adequate for this review.

11.2.2 Summary of Technical Information

DCD Tier 2, Revision 9, Section 11.2 describes the design of the LWMS and its functions in controlling, collecting, processing, storing, and disposing of liquid radioactive waste generated as a result of normal operation, including AOOs. The LWMS is a nonsafety-related system and serves no safety functions except for the isolation of radioactive releases during planned discharges. Failure of the LWMS does not compromise safety-related systems or components and does not prevent the safe shutdown of the plant. DCD Tier 2, Revision 9, Section 3.2 describes the seismic and quality group classification and corresponding codes and standards that apply to the design of the LWMS components and piping and the structures housing the system. The LWMS is designed to the seismic criteria of RG 1.143, Class RW-IIa. All waste collection and processing tanks have level-indication gauges and provisions for high-level alarms in the main control room. Local indications and controls are available on displays located in the radwaste building control room. DCD Tier 2, Revision 9, Figure 11.2-1 provides an overview of the LWMS process diagram depicting all subsystems, while Figures 11.2-1a, 11.2-1b, 11.2-3, and 11.2-4 present specific design details for each subsystem. DCD Tier 2, Revision 9, Figure 11.2-2 provides an LWMS process stream information directory and simplified flow diagram. DCD Tier 2, Revision 9, Sections 9.3, 9.2 and 10.4 describe the

equipment and floor drain drainage systems and origins and discharges of nonradioactive effluents. DCD Tier 2, Revision 9, Figures 1.2-21 to 1.2-25 present the general arrangements of the radwaste building in which the LWMS is located. The LWMS does not normally process nonradioactive secondary system effluent and has no interconnections with the potable and sanitary water systems, as described in DCD Tier 2, Revision 9, Section 9.2.4.

The LWMS and its components are housed in the radwaste building and located in radiologically controlled access areas. DCD Tier 2, Revision 9, Figure 11.2-1 provides a process diagram showing the LWMS tanks, processing equipment, pumps, valves, ion exchangers, filters, and other components of the subsystems. All LWMS tank overflows are routed to building sumps and drains, which are pumped to their respective drain tanks. Subsystem tanks and components are vented to the radwaste building ventilation system, as described in DCD Tier 2, Revision 9, Section 9.4. Cubicles, where tanks are located, are lined with steel liners to avoid releases of radioactive materials in the environment. Concrete walls are coated with sealants for additional protection and minimization of radioactive waste, e.g., in the form of contaminated concrete. The LWMS treatment system components are arranged in shielded enclosures to minimize exposure of plant personnel during operation, inspection, and maintenance. There are provisions for periodic inspection of major components to ensure the capability and integrity of the subsystems.

The LWMS is comprised of four subsystems, as permanently installed equipment and connected to other installed equipment. The LWMS subsystems are:

- Equipment (low conductivity) drain subsystem
- Floor (high conductivity) drain subsystem
- Chemical drain subsystem
- Detergent drain subsystem

The LWMS is divided into several subsystems so that liquid wastes from various sources can be segregated and processed separately. The subsystems maintain the segregation to support the most appropriate treatment of the waste by the LWMS. Cross-connections between subsystems provide additional flexibility in processing wastes by alternate methods and provide redundancy if one subsystem becomes inoperable. The LWMS normally operates on a batch basis. The system provides for sampling at process points, as discussed in Section 11.5.2 of this report. Administrative controls and detection and alarm signaling of abnormal conditions protect against accidental discharges into the environment.

The treatment subsystems rely on mixed bed demineralizers, charcoal beds and cartridge filters, reverse osmosis, and organic and neutralization treatments. The subsystems use plant service utilities for their operations, such as compressed air, water, electricity, ventilation, and radiation monitoring. DCD Tier 2, Revision 9, Tables 11.2-1 through 11.2-4 list the descriptions and design parameters for these systems, which are depicted in DCD Tier 2, Revision 9, Figures 11.2-1 to 11.2-4. DCD Tier 2, Revision 9, Table 11.2-3 lists the decontamination factors (DFs) by types of process streams and types of processing methods. DCD, Revision 9, Section 9.3.3, provides additional design details. Descriptions of the associated process and effluent radiological monitoring and sampling systems appear in DCD, Revision 9, Sections 11.5.3 and 11.5.5. DCD Tier 2, Revision 9, Section 12.2.2.3 presents estimates of liquid effluent radionuclide concentrations and average annual releases, and DCD, Revision 9, Section 12.2.2.4, describes associated doses to the maximally exposed individual located in unrestricted areas. The LWMS processes four major categories of radioactive wastes, including the following:

(1) Equipment drains from various plant sources;

(2) Floor drains from various sumps in the reactor, turbine, and radwaste buildings;

(3) Chemical drains from the laboratory and other relatively small-volume sources; and

(4) Detergent drains from laundry and personnel decontamination and decontamination waste water from the reactor and turbine buildings.

The equipment drain subsystem processes liquid wastes (high purity) from the reactor water cleanup (RWCU) and shutdown cooling (SDC) systems, fuel auxiliary pools cooling system (FAPCS), condensate demineralizer, and equipment drains in the reactor, fuel, turbine, and radwaste buildings. This subsystem can also receive liquid waste from the floor drains subsystem. The permanently installed equipment includes three collection tanks, each with a capacity of about 140,000 liters (37,000 gallons), and two sample tanks with the same capacity. The associated treatment subsystem consists of pre- and main filters, reverse osmosis units, pretreatment and polishing resin ion exchangers, and a resin trap. The subsystem's processing capacity is rated at about 330 liters per minute (88 gallons per minute). The subsystem's design includes process sampling points to assess its performance and compliance with regulatory requirements for the disposition or recycling of treated liquid waste.

The floor drain subsystem processes liquid wastes (low purity) from the reactor drywell and from floor drains in the reactor, fuel, turbine, and radwaste buildings. This subsystem can also receive liquid waste from the equipment drain, chemical drain, or detergent drain subsystems. The equipment includes two collection tanks, each with a capacity of about 130,000 liters (34,000 gallons), and two sample tanks with the same capacity. The associated treatment subsystem consists of pre- and main filters, a reverse osmosis unit, pretreatment and polishing resin ion exchangers, and a resin trap. The subsystem's processing capacity is rated at about 250 liters per minute (66 gallons per minute). The subsystem's design includes process sampling points to assess its performance and compliance with regulatory requirements for the disposition or recycling of treated liquid waste.

The chemical drain subsystem processes liquid wastes from the turbine and radwaste buildings, and from the detergent drain collection tank, if needed. The permanently installed equipment includes one collection tank with a capacity of about 4,000 liters (1,060 gallons) and no sample tanks. Chemicals are added to the tank for pH control or other chemical adjustments, as needed. The subsystem's design includes process sampling points to assess its performance and compliance with regulatory requirements for the disposition or recycling of treated liquid wastes.

The detergent drain subsystem processes liquid wastes from the hot-laundry and hot-shower facilities, and decontamination drains from the reactor, turbine, and radwaste buildings. This subsystem can also receive liquid waste from the chemical drains subsystem. The permanently installed equipment includes two collection tanks, each with a capacity of about 15,000 liters (4,000 gallons) and two sample tanks with the same capacity. The associated treatment subsystem consists of an organic pretreatment unit, which includes the pre- and main cartridge and charcoal filters, with a rated processing capacity of about 33 liters per minute (9 gallons per minute). The subsystem's design includes process sampling points to assess its performance and compliance with regulatory requirements for the disposition or recycling of treated liquid wastes.

When liquid wastes are processed, treated wastes returned to the LWMS for eventual discharge to the environment or are returned to the condensate storage tank (CST) for recycling, as described in DCD Tier 2, Revision 9, Section 9.2.6. Liquid wastes that cannot be discharged are returned to their specific collection tanks for reprocessing or reuse in plant systems. Any liquid wastes that cannot be treated onsite are placed in tanks or containers and shipped offsite for processing and disposal. Process discharge paths are normally aligned to one of the subsystem sample tanks. Before discharge, liquid wastes are sampled for radiological analysis and compliance with state and local requirements for non-radioactive contaminants, based on procedures for combined license (COL) conditions. DCD Tier 2, Revision 9, Sections 9.3.2 and 11.5.5 describe the features of the LWMS process sampling system. DCD Tier 2, Revision 9, Table 9.3-1 identifies process and effluent streams that are to be evaluated for the presence of radioactivity.

All LWMS discharges are made through a single liquid waste discharge line to the discharge canal using procedures developed by COL Licensees. The release of processed liquid wastes from any sample tank to the environment is permitted only when the analysis of the tank's contents indicates that such a release meets the requirements of Appendix B to 10 CFR Part 20 for liquid effluent concentrations and Appendix I to 10 CFR Part 50 for doses to members of the public. During discharge, liquid wastes are mixed with and diluted by other water in the discharge canal, at a flow rate of about 20,000 liters per minute (5,300 gallons per minute), as described in DCD, Revision 9, Table 12.2-20a. The discharge flow rate from the LWMS is controlled to ensure that radionuclide concentration levels in unrestricted areas comply with effluent concentration limits Table 2, Column 2, of Appendix B to 10 CFR Part 20.

Based on DCD Tier 2, Revision 9, Table 11.2-4, the combined normal generation rate of liquid wastes serviced by the four subsystems is estimated to be about 98,000 liters per day (25,600 gallons per day). The estimated maximum flow rate is about 240,000 liters per day (63,400 gallons per day), based on listed sources of liquid wastes. The estimated time needed to process the maximum anticipated flow rate varies from nearly 7 hours for the floor drain subsystem to about 0.2 hours for the chemical drain subsystem. The combined storage capacity and processing rates are expected to provide an adequate margin for handling surges in the generation of liquid wastes serviced by these subsystems.

The liquid radwaste discharge radiation monitor tracks all discharges from the LWMS before in-plant dilution and subsequent release to the discharge canal. The monitor is located on the common discharge line downstream of the LWMS sample tanks, as shown in DCD Tier 2, Revision 9, Figure 11.5-1. The radiation monitor provides a signal to terminate liquid waste release before discharge concentrations exceed predetermined set points, based on effluent limits in Table 2, Column 2, of Appendix B to 10 CFR Part 20. DCD Tier 2, Revision 9, Tables 11.5-1, 11.5-3, 11.5-4, 11.5-5, 11.5-7, and 11.5-9 describe the sampling requirements and operational characteristics of the liquid radwaste discharge radiation monitor. The radiation monitoring system used to control and monitor releases of radioactive materials in liquid effluents to unrestricted areas conforms to the requirements of GDC 60.

In DCD Tier 2, Revision 9, Section 11.2.6 identifies two COL information items for COL applicants. These COL information items address the implementation of Inspection and Enforcement (IE) Bulletin (BL) 80-10, "Contamination of Non-Radioactive System and Resulting Potential for Unmonitored, Uncontrolled Release of Radioactivity to Environment," dated May 6, 1980, on the protection of nonradioactive systems, and operational procedures for the minimization of contamination under the requirements of 10 CFR 20.1406. The COL information items are:

- COL Information Item 11.2-1-A: Implementation of IE Bulletin 80-10—The COL applicant is responsible for identifying LWMS subsystems interface and connections that are considered nonradioactive but that could later become radioactive through improper interfaces with radioactive systems. In identifying such connections, the applicant applies the guidance and information in IE BL 80-10.

- COL Information Item 11.2-2-A: Implementation of Part 20.1406—The COL applicant is responsible for demonstrating compliance with 10 CFR 20.1406 as this section relates to the design and operational procedures of treatment subsystems to minimize contamination, facilitate eventual decommissioning, and minimize the generation of radioactive waste.

11.2.3 Staff Evaluation

The staff reviewed the LWMS in accordance with the guidance in SRP Section 11.2. Staff acceptance of the LWMS is based on the design meeting the requirements of 10 CFR 50.34a and GDC 60 and 61. Under the requirements of 10 CFR 50.34a, the applicant must provide sufficient design information to demonstrate that it has met the design objectives for equipment necessary to control releases of radioactive effluents to the environment. GDC 60 requires that the LWMS is designed to control releases of liquid radioactive effluents, and GDC 61 stipulates that the LWMS is designed to ensure adequate safety under normal operations and postulated accident conditions.

In response to staff inquiries, the applicant stated in DCD Tier 2, Revision 7, Section 11.2 that the LWMS complies with the guidance in RG 1.143, Revision 2. Specifically, the guidance addresses the design and construction methods, materials specifications, welding, and inspection and testing standards for the LWMS pumps and piping. The COL Licensee is responsible for testing all liquid waste processing subsystems installed in the plant, as described in DCD Tier 2, Chapter 14. Chapter 14 of this report addresses the adequacy of the preoperational testing programs.

The LWMS tanks (floor and equipment drain tanks, sample tanks, detergent drain tanks, and chemical drain tanks) are located in the radwaste building. The LWMS is designed to the seismic criteria of RG 1.143, as Class RW-IIa. The associated subsystems and components, such as ion exchangers, filters, pumps, applicable valves, and waste processing equipment, are also located in the radwaste building. All LWMS tank overflows are routed to watertight rooms or cubicles within the radwaste building and drained to local sumps, which are pumped to their appropriate waste collection tanks. All tanks are vented through filtration systems and monitored for radioactivity before being discharged to the environment through the radwaste building stack. The staff finds the above design aspects of the LWMS acceptable with respect to meeting the design guidance specified in RG 1.143.

Regarding the presence of outdoor tanks, the applicant confirmed that, other than the CST, there are no LWMS tanks located in yard areas outside of buildings (see DCD Tier 2, Revision 9, Section 9.2.6). The CST is the only outdoor tank that is expected to contain low levels of radioactivity. The CST is located in a catch basin that is designed to hold the entire volume of the tank. Tank overflow also discharges in the same basin. The design of the catch basin includes a sump, with provisions to pump water to the LWMS for treatment or to the site surface water drainage system if radioactivity levels are in compliance with the requirements of Appendix B to 10 CFR Part 20 for liquid effluent concentrations and Appendix I to 10 CFR Part 50 for doses to members of the public. The staff finds the above design aspects of the CST acceptable with respect to meeting the design guidance in RGs 1.143 and 4.21.

No other tank interfaces are necessary with the LWMS. The LWMS has no interconnections with the potable and sanitary water systems, and it does not normally process non-radioactive secondary system effluent. The applicant confirmed that all releases of radioactive effluents from the LWMS will be tracked by a continuous liquid effluent radiation monitor on the LWMS discharge line. The relevant requirements of GDC 60 and 61 are met by using the regulatory positions contained in RG 1.143, as it relates to the seismic design and quality group classification of components used in the LWMS, structures housing the systems, provisions used to control leakage, and definition of the discharge path beginning with interfaces with plant primary systems and terminating at the point of controlled discharge to the environment.

The LWMS is designed to handle most process and effluent streams and other anticipated events. However, for events occurring at low frequencies, or producing effluents not compatible with currently used processing equipment, additional or specialized temporary processing equipment may be brought into the radwaste building treatment system bay. Connections to various portions of LWMS subsystems facilitate the use of additional skid-mounted processing equipment. These connections allow for the use of skid-mounted equipment applied in series with or parallel to installed equipment as an alternative to returning treated liquid wastes to the LWMS, or the use of skid-mounted equipment as a pumping point into tanks for shipment, treatment, and disposal by third-party waste processors. The design includes the use of mobile shield walls to reduce ambient radiation levels and minimize exposure to workers during operation and maintenance. The COL Licensee is responsible for confirming that the use of any additional processing equipment complies with the DCD design bases and meets the NRC regulations on the discharge of liquid effluents, dose limits for members of the public, and radiation protection for workers during the operation and maintenance of such equipment.

The use of skid-mounted systems is expected to result in more efficient liquid waste processing by matching optimum treatment methods to waste streams, based on their chemical and radiological properties. The selection of specific treatment methods and ion exchange and adsorbent media depends on current and future developments of ion exchange and filtration technologies and known characteristics of liquid radwaste streams to be treated by waste processing subsystems. DCD Tier 2, Revision 9, Table 11.2-3 lists the DFs by types of generic process streams and types of processing methods. The DFs were found to be consistent with the NRC guidance, as described in NUREG–0016, Revision 1. A COL applicant referencing the ESBWR certified design should confirm that the performance characteristics or types of adsorbent media it plans to use for all liquid waste processing subsystems will rely on the use of ion exchange or filtration media that will meet or exceed the DFs listed in DCD Tier 2, Revision 9, Table 11.2-3 for the purpose of complying with liquid radioactive effluent concentration limits and doses to members of the public, as evaluated in DCD Tier 2, Revision 9, Section 12.2.2. In applying the guidelines of RG 1.143, the staff will review the proposed use of any additional processing equipment for treating liquid radwaste on a plant-specific basis for particular COL applications. A COL applicant should discuss how such processing equipment intended for the treatment of liquid radwaste would be integrated with the design of permanently installed equipment and confirm that it meets the guidelines of RG 1.143 and the design objectives of Sections II.A and II.D of Appendix I to 10 CFR Part 50. The staff's evaluation of whether the design of the LWMS is acceptable and meets the requirements of 10 CFR 20.1301, "Dose limits for individual members of the public," and 10 CFR 20.1302 and the design objectives of Appendix I to 10 CFR Part 50 is considered part of its review of DCD Tier 2, Revision 9, Section 12.2.2 and is addressed in Section 12.3 of this report.

The staff reviewed the system construction standards; system process flow outlines and descriptions; sources of liquid input volumes; collection points of liquid waste; flow paths of

liquids through subsystems, including potential bypasses; provisions for monitoring radioactivity levels in effluent releases; and point of release of liquid effluents to the environment. The LWMS design includes provisions for sampling at specific process points and protecting against accidental discharges by the detection and alarm signaling of abnormal conditions, as managed under administrative controls by the COL Licensee. The system incorporates design and operational flexibility by providing redundancy in processing wastes through cross-connections to route effluents among subsystems and sufficient storage capacity using multiple collection drain and sample tanks. There are provisions for periodic inspection of major components to ensure the capability and integrity of LWMS subsystems. The COL Licensee is responsible for testing any additional skid-mounted liquid waste processing systems installed in the plant. The staff finds the design acceptable with respect to meeting the criteria of 10 CFR 50.34a, GDC 60 and 61, and the guidance in RG. 1.143, given the quality assurance (QA) program described in DCD Tier 2, Revision 9, Chapter 17. Specifically, the QA requirements address the design, fabrication, procurement, and installation of permanently installed liquid waste processing systems or of such permanently installed systems combined with the use of skid-mounted processing equipment.

The staff's evaluation of the assessment of a potential release of radioactive liquids following the postulated failure of a tank and its components, located outside of containment, is part of the review of DCD Tier 2, Revision 9, Section 15.3.16. The assessment considers the potential impacts of the release of radioactive materials on the nearest potable water supply located in an unrestricted area, unless the design includes specific engineering provisions to contain the expected amount of liquid radioactive waste and avoid a release of radioactivity into the environment. Chapter 15 of this report addresses this issue and presents the results of the staff's analysis.

DCD Tier 2, Revision 9, Section 12.3.1.5 addresses compliance with 10 CFR 20.1406 and RG 4.21, as they relate to facility design and operational procedures for permanently installed subsystems in minimizing the contamination of the facility and generation of radioactive waste.

In addition, the DCD commits the COL applicant to follow the guidance of IE BL 80-10 to avoid the cross-contamination of nonradioactive systems, and avoid unmonitored, uncontrolled radioactive releases to the environment. These aspects are addressed under COL Information Items 11.2-1-A and 11.2-2-A. The COL applicant is responsible for identifying LWMS subsystems interface and connections that are considered nonradioactive but that could later become radioactive through improper interfaces with radioactive systems. In identifying such connections, the applicant applies the guidance and information in IE BL 80-10. The applicant does not list specific design features, while the detailed operational features of each subsystem are left to the COL applicant to define in specifications developed for the procurement of each subsystem through qualified vendors. DCD Tier 2, Revision 9, Section 12.3.1.5 outlines design concepts and features that are expected to reduce contamination levels using the guidance in RG 4.21. The staff's evaluation is presented in Section 12.4 of this report.

A COL applicant referencing the ESBWR certified design should either include the operational set points of the liquid radwaste discharge radiation monitor in its plant-specific offsite dose calculation manual (ODCM), or include a description of the methodology for establishing the set points in the description of the Operational Program for the ODCM. In addition, the COL applicant should describe standard radiological effluent controls (SRECs) in monitoring and controlling releases of radioactive materials to the environment; thereby, eliminating the potential for unmonitored and uncontrolled release. The staff will review this information for

each COL application. Section 11.5 of this report addresses this as COL Information Item 11.5-2-A.

Under the requirements of Sections II.A and II.D of Appendix I to 10 CFR Part 50, a COL applicant referencing the ESBWR certified design is responsible for addressing the requirements of the design objectives in Appendix I to 10 CFR Part 50 in controlling doses to a hypothetical maximally exposed member of the public and populations living near the proposed nuclear power plant. The requirements define dose criteria for liquid effluents and stipulate the conduct of a cost-benefit analysis in justifying installed processing and treatment systems as permanently installed equipment and in combination with any additional skid-mounted subsystems, using the guidance in RG 1.110. Section 12.3 of this report addresses these aspects as COL Information Item 12.2-3-A for liquid effluents.

In reviewing prior revisions of DCD Tier 2, the staff could not confirm that some aspects of the ESBWR LWMS design were consistent with NRC regulatory requirements and guidance. The staff issued a number of RAIs, not listed here for the sake of brevity, during the review of the application. These RAIs involved requests for the applicant to provide clarifications for technical completeness, provide details supporting design bases and design descriptions in demonstrating compliance with regulatory requirements, revise and update system drawings for consistency with system descriptions, revise technical and regulatory references, and provide information for the staff to conduct independent evaluations of results presented in the application. These RAIs were satisfactorily resolved by the applicant and closed by the staff in DCD Tier 2, Revision 6. The following paragraphs discuss the staff's evaluations of the applicant's responses to the staff's RAIs on important technical and regulatory topics.

In RAI 11.2-4, the staff asked the applicant to revise DCD Tier 2, Revision 2, Table 11.2-1 to reflect the guidance in RG 1.143, Revision 2, for atmospheric tanks. In its response, the applicant provided a revised table which was included in DCD Tier 2, Revision 3. The staff found that the applicant retained a footnote that added the use of fiberglass reinforced tanks constructed in accordance with the requirements of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (BPVC) Section X. The use of fiberglass tanks is not consistent with the guidance in RG 1.143. BPVC Section X does not have any specific guidance on the use of fiberglass tanks in radiation zones or on the retention of radioactive liquids. According to 10 CFR 50.34(h)(3), the applicant should justify deviations from the established review criteria, as stated in the applicable SRP section. Therefore, the staff asked the applicant either to provide documentation to demonstrate that the use of fiberglass reinforced tanks for retention of liquids containing radioactive waste is acceptable and will not pose a risk to the health and safety of the public or plant workers, or to remove the provision to use fiberglass reinforced plastic tanks. In a letter dated July 19, 2007, the applicant agreed to delete Footnote 3 in DCD Tier 2, Revision 3, Table 11.2-1 about using fiberglass tanks. Based on the applicant's response, this RAI is resolved. The staff confirmed that this change was included in DCD Tier 2, Revision 7.

In RAI 11.2-5, the staff asked the applicant to revise Table 11.2-1 to reflect the guidance in RG 1.143, Revision 2, for tanks rated in the pressure range of 0-0.1 megapascal (MPa) (0–15 pounds [lbs] per square inch). In its response, the applicant agreed to revise the table to comply with RG 1.143. The staff reviewed the revised table attached to the applicant's response letter and included in DCD Tier 2, Revision 3. The staff found that the applicant retained a footnote that adds the use of fiberglass reinforced tanks constructed in accordance with the requirements of ASME BPVC Section X. Based on the same reasons discussed in the above evaluation for the RAI 11.2-4 response, the staff found the response to RAI 11.2-5

unacceptable. In response to a supplemental RAI, the applicant agreed to delete the provision (Footnote 3 to Table 11.2-1) allowing the use of fiberglass tanks to contain liquid radioactive waste. Based on the applicant's response, this RAI is resolved. The staff confirmed that this change was included in DCD Tier 2, Revision 7.

In RAI 11.2-11, the staff asked for additional details on DCD Tier 2, Figure 11.2-1. For example, this figure did not show sufficient detail to allow identification of all sources of liquid input volumes; the points of collection of liquid waste; the flow paths of liquids through the system, including potential bypasses; and the specific point of release of liquid effluents to the environment. The level of detail should be sufficient to allow the staff to conduct its review in accordance with the Review Criterion III.1 in SRP Section 11.2, (Revision 2, July 1981). In its response, the applicant stated that DCD Tier 2 would be revised to include a new Figure 11.2-2, "Liquid Waste Management System Process Stream Information Directory." Additionally, the applicant added a description of Figure 11.2-1b in the text of DCD Tier 2, Revision 3, Section 11.2. The staff reviewed the revised figures in DCD Tier 2, Revision 3 and still could not find the specific point(s) of release of liquid effluents to the environment (e.g., interfacing with the circulating water system). In a letter dated July 19, 2007, the applicant agreed to revise DCD Tier 2, Revision 3, Figures 11.2-1b and 11.2-3 to identify the release point(s). Based on the applicant's response, this RAI is resolved. The staff confirmed that this change was included in DCD Tier 2, Revision 7.

In RAI 11.2-13, the staff asked the applicant to describe how the classifications and design criteria apply to the LWMS (including piping, tanks, and structures used to contain leakage) and how the criteria satisfy the requirements of GDC 61 with respect to designing radioactive waste systems to ensure adequate safety under accident conditions. In its response, the applicant stated that the LWMS was designed to Quality Group D and modified by RG 1.143, Revision 2, Section 7 and Table 1. Referring to the response to RAI 11.2-9 and 11.2-10, the applicant addressed the compliance of the LWMS with RG 1.143 guidance. The staff reviewed the response to RAI 11.2-13. It previously had reviewed the responses to RAI 11.2-6 through 11.2-10, related to the compliance of the LWMS with RG 1.143, Revision 2. Based on SRP Section 11.2, the compliance with RG 1.143 forms the bases for satisfying GDC 61. A COL applicant referencing the ESBWR certified design should describe the QA program for design, fabrication, procurement, construction of structures, and installation of permanent LWMS systems and components in the plant in accordance with its overall QA program. However, DCD Tier 2, Revision 3, Section 11.2.6 did not commit the COL applicant to conform with the QA guidance specified in RG 1.21, Revision1, "Measuring, Evaluating, and Reporting Radioactivity in Solid Wastes and Releases of Radioactive Materials in Liquid and Gaseous Effluents from Light-Water-Cooled Nuclear Power Plants," RG 1.33, Revision 2 "Quality Assurance Program Requirements (Operation)," and RG 4.15, Revision 2 "Quality Assurance for Radiological Monitoring Programs (Inception through Normal Operations to License Termination)-Effluent Streams and the Environment." In RAI 11.2-13 S01, the staff asked the applicant to update the DCD to address this aspect. In the response to RAI 11.2-13 S01 and RAI 11.5-44, the applicant proposed changes to all sections of Chapter 11 related to this topic and stated that the applicable QA requirements are described in DCD Tier 2, Revision 4, Table 17.0-1. In a letter dated July 23, 2007, the applicant committed to placing a reference to QA requirements of DCD Tier 2, Revision 4, Chapter 17 for the design, fabrication, procurement, and installation of the liquid radioactive waste system in accordance with the COL Licensee's overall QA program. Based on the applicant's response, this RAI is resolved. The staff confirmed that this change was included in DCD Tier 2, Revision 7.

In RAIs 11.2.3-1, 11.2.3-2, and 11.2.2-4, the staff asked the applicant to clarify the basis of the decontamination factors (DFs) listed in DCD Tier 2, Revision 1, Table 11.2-3 and their applications in deriving the estimated radioactive liquid effluent radioactive source term identified in DCD Tier 2, Revision 3, Section 12.2.2.3. DCD Tier 2, Revision 3, Table 11.2-3 presented updated DFs assigned by types of liquid wastes and groupings of radionuclides. The revised DFs are consistent with those presented in NUREG–0016 for general purpose ion-exchange and adsorbent media and filtration systems. Accordingly, the staff finds the response to RAI 11.2.3-1 acceptable. However, the staff's review of DCD Tier 2, Revision 3 noted that Section 11.2.6 did not commit the COL applicant to the performance of installed mobile processing equipment with that described in DCD Tier 2, Revision 3, Tables 11.2-2c and 11.2-3. For example, a COL applicant referencing the ESBWR certified design should either describe the performance requirements of ion-exchange and adsorbent media and filtration, or identify the types of ion-exchange and adsorbent media and filtration systems it plans to use depending on the expected characteristics of liquid process and effluent streams. In RAI 11.2.3-1 S01, the staff asked the applicant to update the DCD to address this aspect. In a response to this supplemental RAI, the applicant noted that DCD Tier 2, Section 11.2.2 and Table 11.2-3 would be revised to state that the processing equipment and adsorbent media used to treat liquid radioactive wastes will meet or exceed the DFs given in DCD Table 11.2-3 for the purpose of complying with liquid effluent concentration limits and doses to members of the public. In a letter dated July 19, 2007, the applicant committed to placing this information in DCD Tier 2. Based on the applicant's response, this RAI is resolved. The staff confirmed that this change was included in DCD Tier 2, Revision 7.

In RAI 11.2-16, the staff asked the applicant to revise DCD Tier 2, Revision 3, Section 11.2.2 and DCD Tier 1, Revision 3, Section 2.10.1 to indicate that the then proposed mobile liquid radioactive waste processing system was a conceptual design and that DCD Tier 2 included a COL information item committing the COL applicant to provide complete descriptions and specifications of the mobile LWMS and its subsystems so as to meet the performance specifications described in DCD Tier 2, Revision 3, Table 11.2-3, and radiological liquid effluent source terms and doses to members of the public presented in DCD Tier 2, Revision 3, Section 12.2.2. The staff's evaluation of the LWMS and the use of mobile waste processing systems concluded that the design of the LWMS is conceptual and, therefore, not in the scope of design certification, given the requirements of 10 CFR 52.47(a)(24) and 52.47(a)(25). Alternatively, the applicant may provide final descriptions and specifications of the mobile LWMS and its subsystems in the DCD rather than conceptual design information, with the inspections, tests, analyses, and acceptance criteria (ITAAC) included as appropriate. In the context of DCD Tier 1 requirements, design descriptions and interface requirements are intended to serve as binding requirements for the purpose of confirming that the plant will be built given the design descriptions and ITAACs described in Tier 1. In the context of COL information items, a COL applicant referencing the ESBWR certified design is responsible for ensuring that the plant is built in accordance with the design features described in DCD Tier 2, Section 11.2 and other relevant and supporting DCD sections, and tested using the initial test program described in DCD Tier 2, Section 14.2.

In a letter dated March 17, 2008, the applicant committed to replacing the initially proposed conceptual design of the mobile liquid waste processing subsystems with full design descriptions, including flow diagrams for each of the four subsystems. The design of each subsystem is described in DCD Tier 2, Revision 5, Section 11.2.2 and shown in Figures 11.2-1 to 11.2-4. Similarly, DCD Tier 2, Section 11.2.3 contains the safety evaluation; Section 11.2.4 describes the testing and inspection requirements; Section 11.2.5 describes the types of instrumentation used; and Section 11.2.6 identifies COL information items to be addressed by

COL applicants. Based on the applicant's response, this RAI is resolved. The staff confirmed that these changes were included in DCD Tier 2, Revision 5.

In addressing 10 CFR 52.47(b)(1), which requires that a design certification application contain the proposed ITAAC, the applicant has included specific ITAAC for the LWMS. These ITAAC are described in DCD Tier 1, Revision 9, Section 2.10.1 and Tables 2.10.1-1 and 2.10.1-2. Two ITAAC address the descriptions and functional arrangements of the LWMS and assess the integrity of the LWMS when subjected to hydrostatic testing pressures expected during operation. An ITAAC is assigned to confirm that the initial loading of subsystem demineralizers and vessels includes the appropriate types of filtration and adsorption media in meeting or exceeding the DFs listed in Table 11.2-3 of DCD Tier 2, Revision 9. An ITAAC addresses the loading of filtration and adsorption media in process subsystems. A further commitment assigns an ITAAC for the installation of steel liners in cubicles housing LWMS tanks and vessels to ensure that, in the event of a tank rupture, the effluent concentration limits of Table 2 (Column 2) of Appendix B to 10 CFR Part 20 will not be exceeded in unrestricted areas. Another ITAAC focuses on a test to confirm that the liquid radioactive waste discharge radiation monitor provides automatic closure of the discharge isolation valve on receipt of a high radiation signal exceeding a set-point value. If the inspections, tests, and analyses are performed and the acceptance criteria met, the proposed ITAAC provide reasonable assurance that a plant that incorporates the ESBWR design certification and operates in accordance with the design certification will meet the provisions of the *Atomic Energy Act*, as amended (AEA) and NRC regulations.

In RAI 11.0-1, the staff's review indicated that some listed references were not cited in the text, e.g., Ref. 11.2-8 and the reference list included improper regulatory citations, e.g., Ref. 11.2-2. In RAI 11.1-1, the staff asked the applicant to make the appropriate corrections. In a response dated November 13, 2008, the applicant agreed to make the appropriate corrections and provided proposed changes to be included in Revision 6 of the DCD. The staff finds the proposed changes acceptable. Based on the applicant's response, this RAI is resolved. The staff confirmed that these changes were included in DCD Tier 2, Revision 6.

11.2.4 Conclusions

Based on the information discussed above, the staff concluded that the LWMS (as a permanently installed subsystem and in combination with the use of skid-mounted processing equipment) includes the equipment necessary to manage and treat process streams and control releases of radioactive materials in liquid effluents in accordance with 10 CFR 20.1301, 10 CFR 20.1302, and 10 CFR 20.1406; Appendix I to 10 CFR Part 50; the requirements of GDC 60 and 61; and the requirements of 10 CFR 50.34a. This conclusion is based on the following requirements that:

- The ESBWR design meets the dose requirements of 10 CFR 20.1301 and 10 CFR 20.1302 by ensuring that the annual average concentration of radioactive materials in liquid effluents released into unrestricted areas will not exceed the limits specified in Appendix B to 10 CFR Part 20, Table 2, Column 2, as demonstrated in DCD Tier 2, Revision 9, Section 12.2.2.

- The ESBWR design complies with the requirements set forth in Section II.A of Appendix I to 10 CFR Part 50 in ensuring that offsite individual doses resulting from liquid effluent releases will not exceed dose criteria, as demonstrated in DCD Tier 2, Revision 9, Section 12.2.2.

- The ESBWR design demonstrates compliance with 10 CFR 50.34a, as it relates to the provision of sufficient design information and an ITAAC confirming the initial loading of the appropriate types of filtration and adsorption media in subsystem demineralizers, as set forth in the above discussion.

- By preparing a plant-specific cost-benefit analysis in accordance with RG 1.110, a COL applicant referencing the ESBWR certified design is required to demonstrate compliance with the requirements for offsite individual doses and population doses resulting from liquid effluents treated by installed waste processing subsystems, as stipulated in Sections II.A and II.D of Appendix I to 10 CFR Part 50. These requirements are the subject of COL Information Item 12.2-3-A in DCD Tier 2, Revision 9, Section 12.2.4.

- The ESBWR design has met the requirements of GDC 60 and 61 with respect to controlling releases of liquid effluents by radiation monitoring of releases from the LWMS. A radiation monitor tracks all releases and will generate a signal to terminate liquid radwaste releases before the discharge concentration exceeds a predetermined set point. The COL Licensee is required to identify appropriate operational set points for its LWMS radiation monitor in its plant-specific ODCM, as described in DCD Tier 2, Revision 9, Section 11.5.4.

- The applicant demonstrates compliance with the requirements of GDC 61 by meeting the guidelines of RG 1.143, as supported with additional commitments described in DCD Tier 2, Revision 9, Section 12.2.2 and an ITAAC confirming the installation of steel liners in cubicles where liquid radioactive waste tanks are located. These commitments also fulfill the requirements of 10 CFR 20.1406 and guidance in RG 4.21 by minimizing the contamination of the facility and generation of radioactive waste and of IE BL 80-10 by avoiding the cross-contamination of nonradioactive systems and unmonitored and uncontrolled radioactive releases to the environment. These requirements are the subject of COL Information Items 11.2-1-A and 11.2-2-A. DCD Tier 1, Revision 9, Section 2.10.1, and Table 2.10.1-2 describe the ITAAC for the installation of steel liners.

- The applicant demonstrates compliance with the requirements of 10 CFR 52.47(b)(1) with the inclusion of ITAAC for the LWMS. These ITAAC address the descriptions and functional arrangements of the LWMS, the integrity of the LWMS under expected operating pressures, the initial loading of the appropriate types of filtration and adsorption media LWMS subsystems, the installation of steel liners in cubicles housing LWMS tanks and vessels to ensure that in the event of a tank rupture, the effluent limits of Table 2 (Column 2) in Appendix B to 10 CFR Part 20 will not be exceeded, and the proper operation of the liquid radioactive waste discharge radiation monitor will provide automatic closure of the discharge isolation valve on receipt of a high radiation signal. If the inspections, tests, and analyses are performed and the acceptance criteria met, the proposed ITAAC provide reasonable assurance that a plant that incorporates the ESBWR design certification and operates in accordance with the design certification will meet the provisions of the AEA and NRC regulations.

11.3 Gaseous Waste Management System

11.3.1 Regulatory Criteria

The staff reviewed DCD Tier 2, Revision 9, Section 11.3 in accordance with the guidance and acceptance criteria described in SRP Section 11.3. The following acceptance criteria are applicable:

- 10 CFR 20.1302, as it relates to limits on doses to persons and gaseous effluent concentrations in unrestricted areas (these criteria apply to releases resulting from the GWMS during normal plant operations and AOOs)

- 10 CFR 20.1406, as it relates to facility design and operational procedures for minimizing the contamination of the facility and generation of radioactive waste

- 10 CFR 50.34a, as it relates to providing sufficient design information to demonstrate the effectiveness of design objectives for equipment necessary to control releases of radioactive gaseous effluents to the environment

- GDC 3, "Fire protection," as it relates to protecting gaseous waste handling and treatment systems from the effects of explosive mixtures of hydrogen and oxygen

- GDC 60, as it relates to the design of the GWMS to control releases of gaseous radioactive effluents

- GDC 61, as it relates to the control of radioactivity in the GWMS and building ventilation systems associated with fuel storage and handling areas

- Sections II.B, II.C, and II.D of Appendix I to 10 CFR Part 50, as they relate to the numerical guidelines for dose design objectives to meet the ALARA criterion and cost-benefit analysis

The following RGs contain the regulatory positions and guidance for meeting the relevant requirements of the regulations identified above:

- RG 1.109, as it relates to demonstrating compliance with the numerical guidelines for dose design objectives and the ALARA criterion of Appendix I to 10 CFR Part 50

- RG 1.110, as it relates to performing a cost-benefit analysis for reducing the cumulative dose to the population by using available technology

- RG 1.140, Revision 2, "Design, Inspection, and Testing Criteria for Air Filtration and Adsorption Units of Normal Atmosphere Cleanup Systems in Light-Water-Cooled Nuclear Power Plants," as it relates to the design, testing, and maintenance of normal ventilation exhaust systems at nuclear power plants

- RG 1.143, as it relates to the seismic design and quality group classification of components used in the GWMS and structures housing this system, as well as the provisions for controlling leakage

- RG 4.21, as it relates to minimizing the contamination of equipment, plant facilities, and environment, and minimizing the generation of radioactive waste during plant operation

- SRP Section 11.3, BTP 11-5, Revision 3, "Postulated Radioactive Releases Due to a Waste Gas System Leak or Failure," as it relates to assessing offsite doses associated with such a failure.

The staff performed a comparison of the SRP (Section 11.3, 1981 version) used during the review of the DCD with the 2007 version of the SRP. The 2007 version includes additional acceptance criteria and guidance addressing the requirements of 10 CFR 20.1406, when

compared to the prior version of the SRP. However, the requirements of 10 CFR 20.1406 were considered in the staff's review of the DCD, given the 2007 version of the SRP. Discussions and dispositions of these items are provided in this and other supporting sections of this report. Therefore, the staff concludes that the version of the SRP used, in combination with the additional review performed by the staff, is adequate for this review.

11.3.2 Summary of Technical Information

There are two main sources of plant gaseous radioactive effluents. One source is from building ventilation systems servicing radiologically controlled areas, and the other is from the power cycle OGS. DCD Tier 2, Revision 9, Section 11.3 describes the GWMS and its OGS used to control, collect, process, hold for decay, and discharge gaseous radioactive wastes generated during normal operation, including AOOs. The major components include preheaters, recombiners, cooler/condensers, dryers, activated charcoal beds (guard and delay), and associated valves, pumps, and instrumentation. The OGS is located in the turbine building. Section 3.2 of DCD Tier 2, Revision 9 discusses the seismic and quality group classification and corresponding codes and standards that apply to the design of the GWMS/OGS components and piping, and the structures housing the GWMS/OGS. The OGS equipment and piping are classified as non-seismic, but are designed to meet the requirements of RG 1.143.

The OGS provides a means of treating non-condensible gases removed from the condenser by the evacuation system. The sources are gases that leak into the system through components such as pump seals and valve packing; gases that become entrained in solution while in auxiliary systems, such as the CST; and any gases created through the radiolytic decomposition of water in reactor coolant. The gases removed from the condenser may be radioactive and, therefore, must be treated before being released into the environment to ensure effluent radioactivity is reduced to acceptable levels.

The OGS consists of processing equipment, with its associated monitoring instrumentation, and control components. The OGS treats the removed gases in two ways. The first method reduces the volume of the gases by recombining the hydrogen and oxygen into water. The recombination also reduces the explosion potential within the OGS. The water is removed to protect the carbon beds and returned for plant process use. Because a buildup of explosive mixtures of hydrogen and oxygen is possible, the OGS must be designed either to withstand the effects of a hydrogen explosion or have design features that preclude the formation or buildup of explosive gas mixtures in accordance with SRP Section 11.3 guidelines. The ESBWR OGS is designed to be detonation resistant and to meet the requirements of RG 1.143. The second method for treating removed gases is to provide a holdup of gases through temporary retention. The holdup allows time for the decay of radioactive materials in the remaining gases removed from the main condenser. The delay is sufficient to achieve adequate decay of the radioactivity before the plant discharges process offgases and to ensure that radioactivity levels released into the environment meet regulatory requirements. The OGS is housed in a reinforced concrete structure to provide adequate shielding and minimize radiation exposures to personnel during operation and maintenance.

The design uses redundant, cross-connected flow paths to ensure availability of the system during maintenance or malfunction of a component. Plant operators can isolate functional groups or single units to respond to operational needs, maintenance, or equipment malfunctions, while ensuring the proper treatment of the processed gas before it is released into the environment. The system's operational configuration can be scaled to match plant power levels from startup to 100 percent power. The normal operational configuration of the OGS is in

the “treat mode.” The design allows the OGS to be bypassed during periods of startup or if the process gas activity is acceptable; however, the bypass can only be activated by the use of dual keyed permissive commands of the OGS.

The major inputs to the GWMS are offgases from the main condenser evacuation system, which is described in DCD Tier 2, Revision 9, Section 10.4.2. The flow through the OGS consists of hydrogen and a carrier gas (air from in-leakage), fission and activation gases, and water vapors. For each train, gaseous influents flow through the following major process stages of the OGS:

(1) A preheater, which preheats gases for improving recombiner efficiency

(2) A hydrogen/oxygen recombiner, which recombines radiolytic hydrogen and oxygen into water

(3) Cooler/condenser units, which remove moisture from cooled gases to protect the charcoal beds

(4) A dryer, which removes residual moisture from gases out of the cooler/condenser

(5) A charcoal guard bed, which protects the delay beds from abnormal moisture carryover, or chemical contaminants, by removing them from the gas stream

(6) Two charcoal trains, each consisting of four 100 percent capacity beds, which adsorb and retain radioactive isotopes of krypton, xenon, nitrogen, oxygen, and iodines

(7) An offgas post-treatment radiation monitor, which measures levels of radioactivity in the treated gaseous process stream before the gases are vented and monitored through the turbine building stack radiation monitor

DCD Tier 2, Revision 9, Tables 11.3-1 and 11.3-2 list the components of the OGS and the system’s design parameters, which are also shown in Figure 11.3-1. The temperature in the charcoal vault, located in the turbine building, is monitored and controlled. The recombiner-dryer portion of the system consists of two trains (trains A and B), which are connected to charcoal beds consisting of tanks. The charcoal vault houses two charcoal guard tanks, and two trains of four charcoal tanks each. Each guard tank contains about 7,500 kilograms (kg) (16,500 lbs) of charcoal, and each adsorber tank contains about 27,750 kg (61,180 lbs) of charcoal, for a total amount of 222 metric tons (490,000 lbs) of charcoal. The design includes provisions to bypass the charcoal beds in the event of a fire, when excessive moisture is present, and during plant preoperational testing and startup. A nitrogen purge line and an air supply line connection are provided to the first charcoal bed. A nitrogen purge would be used if a fire were detected in charcoal beds. The air supply line would be used to dry the charcoal bed if it became saturated with moisture. A nitrogen line is also provided in servicing the main charcoal beds. The OGS includes various types of instrumentation, including oxygen and hydrogen analyzers; flow, temperature, and pressure measurements; radiation monitoring; and provisions for gas sampling. Control and monitoring occur locally and remotely in the plant’s control room. Liquid waste generated by the coolers, condensers, and dryers is processed by the LWMS or routed to the condenser hot well. Radiation monitoring equipment is provided to measure radioactivity levels in the pre- and post-treatment streams leading in to and out of the charcoal vault.

The GWMS minimizes and controls releases of radioactive materials using activated charcoal adsorber beds, to retain radioactive isotopes of krypton, xenon, nitrogen, oxygen, and iodines through dynamic adsorption, resulting in significant delays during their transit through the beds. The estimated holdup time for xenon radioactive gases in charcoal beds is about 60 days. Radioiodines are adsorbed and retained on the charcoal beds. Radioactive particles are removed either through condensation by the system's cooler and condenser components or retained in charcoal beds. DCD Tier 2, Revision 9, Section 11.3.2 and Tables 11.3-1 through 11.3-3 describe process functions, equipment, and operational parameters for the GWMS and OGS. The radiation monitoring system includes the offgas pre-treatment and offgas post-treatment monitors. The offgas post-treatment monitor provides automatic closure of the OGS discharge and charcoal bypass valves on receipt of a high radiation signal exceeding a set-point value. The description of the design includes an analysis identifying potential malfunctions by specific types of components, including those that could result in increased releases of radioactivity, and precautionary design features for dealing with such malfunctions.

Monitoring of the discharge side of the OGS charcoal beds tracks the presence of radioiodines, noble gases, and particulates. The system includes provisions for the collection of grab samples for radiological analysis. Discharges from the OGS are routed to the turbine building stack, through the turbine building compartment exhaust, where gaseous effluents are monitored by the process radiation monitoring system (PRMS), as described in DCD Tier 2, Revision 9, Section 11.5.3. DCD Tier 2, Revision 9 Tables 11.5-1, 11.5-2, 11.5-3, 11.5-6, 11.5-8, and 11.5-9 describe the sampling requirements and operational characteristics of the OGS post-treatment and turbine building stack radiation monitor, see DCD Tier 2, Revision 9, Section 11.5, Figure 11.5-1.

The turbine gland steam sealing (TGSS) system exhaust and the condenser air removal system (CARS) exhaust are routed to a common header that discharges to the environs through the turbine building compartment exhaust subsystem and turbine building stack. During startup and low-load operation, the TGSS system uses clean steam from the auxiliary boiler system, with main steam used as a backup supply, as described in DCD Tier 2, Revision 9, Section 10.4.3. At plant high-power levels, the TGSS system may be supplied with steam from high-pressure turbine exhaust or from the auxiliary boiler system, as described in DCD Tier 2, Revision 9, Sections 10.3.2 and 10.4.3. At startup, the CARS exhaust is routed to the turbine building compartment exhaust subsystem. During plant operation, the CARS exhaust is discharged to the GWMS/OGS, where it is processed as discussed earlier.

DCD Tier 2, Revision 9, Section 11.3.7 presents an analysis of the radiological impact of a postulated failure or leak of the waste gas system, as well as the justifications for the assumptions used in that analysis. DCD Tier 2, Revision 9, Tables 11.3-3 through 11.3-7 present the assumptions and system parameters used in the analysis and also provide the results in assessing the consequences of the postulated accident, as specified in BTP 11-5 of SRP Section 11.3. The applicant states that the results of the analysis show that the associated doses are in compliance with the SRP acceptance criteria of 25 millisieverts (mSv) (2500 millirems [mrem]) for systems designed to withstand the effects of hydrogen explosions and earthquakes.

Airborne radioactive materials present in buildings are associated with process leakage and steam discharges and are handled through each building's exhaust ventilation system. These releases are in addition to those from the GWMS and OGS. With the exclusion of ventilation systems servicing clean areas of the plant, radioactive materials are released from the following buildings and systems:

- The reactor building heating, ventilation, and air conditioning (HVAC) system, consisting of the reactor building contaminated area HVAC, the refueling and pool area HVAC, and the reactor building HVAC purge exhaust

- The turbine building HVAC system, consisting of the turbine building exhaust, turbine building compartment exhaust, and turbine building decontamination room exhaust

- The fuel building HVAC system, consisting of the fuel building general area HVAC, and the fuel building fuel pool area

- The radwaste building HVAC system, consisting of the radwaste building general area HVAC

Although exhaust flows from plant building exhaust systems are not normally filtered before their release, the ventilation systems servicing the reactor building and refueling building incorporate design features that provide automatic isolation and filtration of exhaust flows before their release under certain circumstances. Specifically, a high-radiation signal from specific monitors located in or next to exhaust ducts will result in isolation of the normal supply and (unfiltered) exhaust ducts to the affected area and route the respective ventilation exhausts to the reactor building HVAC purge exhaust, where it is filtered before being discharged through the reactor building stack. The reactor building HVAC purge exhaust is also used to treat the exhaust from the fuel building. The exhaust of the radwaste building is filtered. Releases from these buildings, as well as from the turbine and radwaste building, are conducted through their individual stacks. DCD Tier 2, Revision 9, Sections 9.4.2, 9.4.3, 9.4.4, and 9.4.6 describe the design bases, operation, and monitoring of such ventilation systems. The PRMS provides for the monitoring and control of gaseous and particulate releases, as discussed in DCD Tier 2, Revision 9, Section 11.5.3. DCD Tier 2, Revision 9, Tables 11.5-1, 11.5-2, 11.5-6, 11.5-8, and 11.5-9 describe the sampling requirements and operational characteristics of the related radiation monitors, including those servicing the discharge stacks of the three buildings.

11.3.3 Staff Evaluation

The staff reviewed the GWMS in accordance with the guidance of SRP Section 11.3. Staff acceptance of the GWMS is based on the design’s meeting the requirements of 10 CFR 50.34a and GDC 3, 60, and 61. Under 10 CFR 50.34a requirements, the applicant must provide sufficient design information to demonstrate that the design objectives of the equipment necessary to control releases of radioactive effluents into the environment have been met. GDC 3 requires that the design protect gaseous waste handling and treatment systems from the effects of an explosive mixture of hydrogen and oxygen gases. The relevant requirements of GDC 60 and 61 are met by using the regulatory positions contained in RG 1.143, Revision 2, as it relates to the seismic design, quality group classification of components used in the GWMS and structures housing the systems, the provisions used to control leakage, and definitions of discharge paths beginning with interfaces with plant primary systems and terminating at the point of controlled discharges to the atmosphere through their respective building stacks.

In reviewing the GWMS, the staff evaluated the system construction standards, seismic design, and quality group classification of components. The structures housing these systems should conform to the guidelines of RG 1.143, Revision 2. The design should include precautions to stop continuous leakage paths. The staff reviewed the system process flow outlines and system descriptions and materials. The OGS review included an examination of the adequacy of the design to withstand the effects of a hydrogen explosion. The applicant did not exercise

the option of using gas analyzers with automatic control functions to preclude the formation or buildup of explosive mixtures; instead, the ESBWR OGS is designed to be detonation resistant.

The OGS minimizes and controls releases of radioactive materials by delaying the flow of gases using activated charcoal adsorber beds. The charcoal adsorber beds retain radioactive isotopes of krypton, xenon, nitrogen, oxygen, and iodines through dynamic adsorption, resulting in significant delays during their transit through the beds. The estimated holdup time for xenon radioactive gases in charcoal beds is about 60 days, about 80 hours for krypton, and about 30 hours for argon based on the stated dynamic absorption coefficients. Radioiodines are adsorbed and retained on the charcoal beds. Because the charcoal bed system design contains about 222 metric tons (490,000 lbs) of charcoal, the iodine removal efficiency is assumed to be about 99.99 percent using the guidance of RG 1.140, Revision 2, given the large amount of charcoal used. Radioactive particles are removed either through condensation by the system's cooler and condenser components or retained in charcoal beds. There are provisions for periodic inspection of major components to ensure the capability and integrity of the subsystems. The COL Licensee will subject the GWMS and OGS to preoperational tests in accordance with DCD Tier 2, Section 14.2. Chapter 14.2 of this report addresses the adequacy of the preoperational testing program for the GWMS. As a result, the OGS satisfies GDC 60, as it provides sufficient holdup capacity for the retention of radioactive gaseous effluents.

The GWMS and OGS generate a liquid radioactive waste phase from the associated coolers/condensers, where the liquid phase can potentially cross-contaminate nonradioactive systems and result in unmonitored and uncontrolled radioactive releases. In DCD Tier 2, Revision 9, Sections 11.3.1 and 11.3.2, the applicant states that the design of the OGS follows the guidance of IE BL 80-10 and 10 CFR 20.1406. The design includes drains and vents to route radioactive process or waste streams and avoids interconnections between plant systems that could become radioactive through improper interfaces with radioactive systems. The liquid phases from coolers and condensers are routed to the turbine hotwell or the LWMS. The air supply and nitrogen purge systems are protected from backflow by dual check valves and tell-tale leak-off connections to prevent the contamination of clean air and nitrogen supply sources. The staff finds such design features acceptable and in compliance with the requirements of 10 CFR 20.1406 and the guidelines of IE BL 80-10 and RG 4.21. DCD Tier 2, Revision 9, Section 12.3 outlines design concepts and features to address such concerns using the guidance in RG 4.21.

In DCD Tier 2, Revision 9, Section 11.3.7 the applicant provided the analysis of a waste gas system leak or failure, as well as the justification for the assumptions used in that analysis. The applicant performed the analysis to demonstrate that the OGS design meets the applicable guidelines of BTP 11-5. This BTP stipulates that the total body dose at the exclusion area boundary (EAB) as a result of the release of radioactivity for 2 hours from a postulated failure of the OGS, calculated in accordance with BTP assumptions, should not exceed 25 mSv (2.5 rem). The applicant analyzed the accident using a short-term (0–2 hours) X/Q of 2×10^{-3} seconds per cubic meter at the EAB, a release duration of 1 hour, instead of 2 hours, as suggested by the BTP, and a noble gas release rate of 16,700 MBq per second (450,000 μCi per second). The applicant justified a release duration of 1 hour as consistent with the isolation time of the system. The applicant calculated a total body dose of 6.2 mSv (0.62 rem) over the assumed duration of the event. The dose result is in compliance with the guideline of BTP 11-5 for systems designed to withstand the effects of hydrogen explosions and earthquakes, given the acceptance criterion of 25 mSv (2.5 rem). Based on the above, the staff finds the results of this analysis acceptable.

In DCD Tier 2, Revision 9, Sections 9.4.2, 9.4.3, 9.4.4, and 9.4.6 state that exhaust air filtration units are equipped with air filtration systems that comply with the guidelines of RG 1.140. The containment purge system has high-efficiency particulate air (HEPA) filters and charcoal adsorbers for mitigating and controlling releases of radioactive materials from the reactor building and fuel building. The air filtration units are designed and tested in accordance with ASME N-509-2002 and ANSI/ASME AG-1-2003 standards. These standards discuss requirements for the installation, inspection, and verification of system airflow rates, air temperatures, and filter pressure drops. On the basis of the above discussion and the evaluation presented in Section 9.4 of this report, the staff finds that the GWMS complies with GDC 61 and meets the guidelines of RG 1.140, as they relate to normal ventilation exhaust systems and design features to control releases of radioactivity through the turbine building stack.

The PRMS provides monitoring and control of gaseous and particulate releases, as discussed in DCD Tier 2, Revision 9, Section 11.5.3. In DCD Tier 2, Revision 9, Tables 11.5-1, 11.5-2, 11.5-6, 11.5-8, and 11.5-9 describe the sampling requirements and operational characteristics of the related detectors of the radiation monitoring systems (RMS). The staff finds these design features acceptable. Based on the above, the staff finds that the GWMS/OGS complies with GDC 60 and 61, as they relate to monitoring and controlling radioactivity releases from ventilation systems associated with fuel storage and handling areas. The applicant's conclusion that the designs of the GWMS and OGS meet the requirements of 10 CFR 20.1301 and 10 CFR 20.1302 and Sections II.B and II.C of Appendix I to 10 CFR Part 50 was part of the review of DCD Tier 2, Revision 9, Section 12.2.2. Section 12.3 of this report addresses the staff's evaluation of radiological impacts associated with releases of radioactive materials from the GWMS/OGS via the turbine building stack and all building ventilation systems (reactor and fuel building stack, and radwaste building stack). The staff finds that the results of the applicant's analysis comply with 10 CFR 20.1301, 10 CFR 20.1302, and Sections II.B and II.C of Appendix I to 10 CFR Part 50.

A COL applicant referencing the ESBWR certified design should either identify the operational set points for its GWMS and turbine building stack radiation monitors in its plant-specific ODCM, or include a description of the methodology for establishing these set points in the description of the operational program for the ODCM. In addition, the COL applicant should describe the SRECs for monitoring and controlling releases of radioactive materials into the environment, which thus eliminate the potential for unmonitored and uncontrolled releases. The staff will review this information on a plant-specific basis for each COL application, including the following:

- The building stacks RMS (reactor and fuel building, turbine building, and radwaste building)
- The reactor building HVAC exhaust RMS and its subsystems
- The containment purge exhaust RMS
- The turbine building combined ventilation exhaust RMS and its subsystems
- The radwaste building ventilation exhaust RMS
- The fuel building combined ventilation exhaust and its RMS subsystems

Section 11.5 of this report addresses these aspects as a COL information item.

Under the requirements of Sections II.B, II.C, and II.D of Appendix I to 10 CFR Part 50, a COL applicant is responsible for addressing the requirements of the design objectives in Appendix I to 10 CFR Part 50 in controlling doses to a hypothetical maximally exposed member of the public and populations living near the proposed nuclear power plant. The requirements define

dose criteria for gaseous effluents and mandate a cost-benefit analysis in justifying installed processing and treatment systems as permanently installed equipment. Section 12.3 of this report addresses these aspects as a COL Information Item 12.2-2-A.

In reviewing the prior versions of DCD Tier 2, the staff found that it did not have sufficient information to determine the acceptability of the GWMS. The staff issued a number of RAIs, not listed here for the sake of brevity, during the review of the application. These RAIs involved requests for the applicant to provide clarifications for technical completeness, provide details supporting design bases and design descriptions in demonstrating compliance with regulatory requirements, revise and update system drawings for consistency with system descriptions, revise technical and regulatory references, and provide information for the staff to conduct independent evaluations of results presented in the application. These RAIs were satisfactorily resolved by the applicant and closed by the staff in DCD Tier 2, Revision 6. The following paragraphs discuss the staff's evaluation of the applicant's responses to the staff's RAIs on important technical and regulatory topics.

In RAI 11.3-2, the staff asked the applicant to describe how the classifications and design criteria applied to the OGS satisfy the requirements of GDC 61 with respect to designing radioactive waste systems to ensure adequate safety under accident conditions. In its response, the applicant stated that the OGS was designed to Quality Group D and modified in accordance with the guidance set forth in RG 1.143, Revision 2, Section 7 and Table 1. DCD Tier 2, Revision 6, Section 11.3.7.1 states that the OGS meets all criteria in RG 1.143. The staff reviewed the response to RAI 11.3-2 and DCD Tier 2, Revision 6, relating to whether the OGS was consistent with RG 1.143, Revision 2. The compliance with RG 1.143 forms the bases for satisfying GDC 61, and the staff finds the response acceptable. Based on the applicant's response, this RAI is resolved. The staff confirmed that this change was included in DCD Tier 2, Revision 6.

In RAI 11.3-3, the staff asked the applicant to describe how the OGS design pressure of the components was selected to enable them to withstand an internal hydrogen explosion. In addition, the staff asked the applicant to provide numerical performance criteria for the hydrostatic test demonstrating this capability. In its response, the applicant stated that the ESBWR OGS design used the methodology outlined in the GEH report NEDE-11146, "Pressure Integrity Design Basis for New Gas Systems," to establish hydrogen explosion pressure integrity in off-gas piping. The NRC has previously approved NEDE-11146, which was submitted for the staff to evaluate and establish design pressure integrity for the Grand Gulf Nuclear Station's OGS for internal hydrogen explosions. The staff evaluated the specifications and performance of the hydrogen and oxygen recombiner system and related gas analyzer instrumentation used to monitor and control the presence of explosive gas mixtures. DCD Tier 2, Revision 6, Sections 11.3.2.2, 11.3.5, and 11.3.6 describe the system. The staff's evaluation of the OGS system, as described in DCD Tier 2, Table 11.3-2 reveals that it is designed to sustain an internal explosion without loss of integrity. The staff finds this methodology to be adequate, and Section 11.3.2.6 of DCD Tier 2, Revision 6 references the NEDE report. In addition, the applicant identified a COL information item in Section 11.3.8 of DCD Tier 2, Revision 2. The staff asked the applicant to define the OGS design parameters— major equipment items as well as other system data— as shown in DCD Tier 2, Table 11.3-2. This COL information item addressed a portion of the RAI and was identified as COL Information Item 11.3.8-1. However, in DCD Tier 2, Revision 3, the applicant removed COL Information Item 11.3.8-1. In a letter dated July 23, 2007, the applicant explained the reasons for the removal of this COL information item using the following rationale:

> The COL item was removed because the OGS is a GEH permanent plant designed system without mobile systems that are used in the liquid and solid radioactive waste system designs. Table 11.3-2 in DCD Tier 2, Revision 5 is the final OGS major equipment design parameters. If a COL Applicant chooses to make changes to the GEH permanent plant OGS design, a departure with justification and design details will be required in the COL application.

The staff finds the above reasons acceptable; therefore, RAI 11.3-3 is resolved. Based on the applicant's response, this RAI is resolved. The staff confirmed that this change was included in DCD Tier 2, Revision 6.

A COL applicant referencing the ESBWR certified design should describe the QA program for design, fabrication, procurement, construction of structures, and installation of permanent GWMS and OGS systems and components in the plant in accordance with its overall QA program. However, DCD Tier 2, Revision 5, Section 11.3.8 did not commit the COL applicant to conform with the QA guidance specified in RGs 1.21, 1.33, and 4.15. In a global response to RAI 11.5-44, the applicant proposed changes to all related sections of Chapter 11 on this topic and stated that the applicable QA requirements are described in DCD Tier 2, Table 17.0-1. As a result, the applicant has revised DCD Tier 2, Section 11.3.5, to reference the QA requirements of Chapter 17 for the design, fabrication, procurement, and installation of the gaseous radioactive waste system in accordance with the COL Licensee's overall QA program. In a letter dated July 23, 2007, the applicant committed to placing this information in DCD Tier 2. Based on the applicant's response, this RAI is resolved. The staff confirmed that this change was included in DCD Tier 2, Revision 6.

As part of RAI 12.2-9, the staff asked the applicant to provide information describing the amounts of charcoals present in each charcoal guard and main tank and to include this information either in DCD Tier 2, Revision 3, Table 11.3-2 or Table 12.2-15. The staff finds the inclusion of this information necessary in evaluating the performance of the charcoal delay beds and assessing releases of noble gases into the environment. In DCD Tier 2, Revision 5, the applicant provided information on the amounts of charcoal contained in each type of tank in Tables 12.2-15 and 11.3-1. The staff finds the inclusion of this information adequate. Based on the applicant's response, this RAI is resolved. The staff confirmed that this change was included in DCD Tier 2, Revision 6.

DCD Tier 2, Revision 9, Section 9.4 describes the exhaust ventilation systems servicing buildings containing radioactive systems that are expected to generate airborne radioactivity. The reactor building and refueling building incorporate design features that automatically isolate and filter exhaust flows before their release in some circumstances. The exhaust of the radwaste building is filtered. Releases from these buildings, as well as from the turbine building and radwaste building, are conducted through their respective stacks. As part of RAI 12.2-9, the staff asked the applicant to confirm the use of charcoal and HEPA filters in controlling radioactive releases into the environment for consistency with the HVAC system descriptions in DCD Tier 2, Revision 3, Section 9.4. The staff found the inclusion of this information important for evaluating the design of the HVAC systems and assessing releases of radioactivity into the environment. In DCD Tier 2, Revision 6 the applicant updated the listing of systems using charcoal and HEPA filters in Section 9.4 and Tables 9.4-4, 9.4-7, and 9.4-15. The staff finds the inclusion of this information satisfactory. Based on the applicant's response, this RAI is resolved. The staff confirmed that this change was included in DCD Tier 2, Revision 6.

In addressing the requirements of 10 CFR 52.47(b)(1), which states that a design certification application must contain the proposed ITAAC, the applicant has included specific ITAAC for the GWMS. These ITAAC are described in DCD Tier 1, Revision 9, Section 2.10.3 and Table 2.10.3-1. Three ITAAC address the descriptions and functional arrangements of the GWMS, confirm the integrity of the GWMS to withstand internal hydrogen explosions, and assess leakage when subjected to testing pressures expected during operation. An ITAAC is assigned to confirm that the initial loading of the appropriate amounts of charcoal adsorbers in the guard beds and decay beds will meet or exceed the delay times listed in DCD Tier 2, Revision 9, Table 11.3-1. Another ITAAC focuses on a test to confirm that the offgas post-treatment radiation monitor provides automatic closure of the OGS discharge and charcoal bypass valves on receipt of a high radiation signal exceeding a set-point value. If the inspections, tests, and analyses are performed and the acceptance criteria met, the proposed ITAAC provide reasonable assurance that a plant that incorporates the certified ESBWR design and operates in accordance with the design certification will meet the provisions of the AEA and NRC regulations.

A review of DCD Tier 2, Revision 5, Section 11.3.2, indicated a number of inconsistencies in the description of offgas equipment design criteria and code requirements, and in equipment descriptions. In RAI 11.3-13, the staff asked the applicant to address the following:

(a) The preheater tube side design temperature is 302 degrees Celsius (^{o}C) (575 degrees Fahrenheit [^{o}F]), but the shell side design is 232^{o}C (450^{o}F). The applicant was requested to clarify what safety considerations were taken in the event of tube failure when 302^{o}C (575^{o}F) gas leaks into the shell side. Also, the applicant was requested to clarify pressure design considerations in the event of tube failure, where tube-side design pressure is 8.6 MPa gauge (1250 pounds per square inch (PSI) gauge) and shell-side design pressure is 2.41 MPa gauge (350 PSI gauge) based on applicable ASME code specifications.

(b) There is a TEMA C code requirement for the cooler-condenser, but not for the preheater and catalyst (both are shell and tube (S&T) heat exchangers). The applicant was requested to identify the proper design codes for the preheater and catalyst.

(c) A review of DCD Tier 2, Revision 5, Figure 11.3-1, OGS flow diagram shows a preheater- recombiner- cooler, as one assembly. This was found confusing since in DCD Tier 2, Section 11.3.2.2, the recombiner assembly includes a preheater, catalyst, and condenser sections. The applicant was requested to clarify this inconsistency in the text and flow diagram.

(d) The OGS flow diagram, Figure 11.3-1, also shows eight charcoal beds and two guard beds, with DCD Tier 2, Table 11.3-2 calling for 10 vessels to be “filled with activated charcoal.” The applicant was requested to clarify this inconsistency in the flow diagram.

(e) A review of DCD Tier 2, Revision 5, Section 11.3.2.5.4 (Drying) did not make it clear as to what type of dryer design will be used (e.g., a refrigerant dryer or a desiccant dryer?). The applicant was requested to provide specific details.

(f) DCD Tier 2, Revision 5, Figure 11.3-1, OGS, has a note “Material per requirements of RG 1.143.” However, Table 1 of this RG is very specific as to the types of materials and grade being required for pressure retaining parts, while DCD Tier 2, Revision 5, Table 11.3-2 did not specify the actual type of materials and grade. The applicant was requested to provide more details on materials and grade used in the design of this system.

In the response to this RAI, the applicant provided technical clarifications for the above noted items and discussed the bases of the turbine auxiliary steam system (TASS) design temperature and operating temperature. The staff finds the responses acceptable as to the inclusion of additional information and technical clarifications, with one exception. A review of DCD Tier 2, Revision 5, Table 11.3-1 identified that the steam supply temperature was not given in the design parameters listed in the table. Therefore, in supplemental RAI 11.3-13 S01, the staff asked the applicant to include the TASS temperature design value in Table 11.3-1. In its response, the applicant provided a proposed revision to DCD Tier 2, Table 11.3-1 and Section 11.3.2.5.1 for inclusion in DCD Tier 2, Revision 6. The proposed revision updated the information on design temperature specifications. The staff finds the proposed changes acceptable. Based on the applicant’s response, these RAIs are resolved. The staff confirmed that these changes were included in DCD Tier 2, Revision 6.

The staff’s review revealed that the description of OGS components in DCD Tier 2, Revision 5, Section 11.3.2.3, under process facilities, was incomplete as it did not include all equipment described in DCD Tier 2, Section 11.3.2.2. Specifically, the text of DCD Tier 2, Section 11.3.2.3, beyond the first paragraph, repeated some of the information already presented in DCD Tier 2, Section 11.3.2.1 and did not include a discussion of process equipment and the locations in process facilities. For example, DCD Tier 2, Section 11.3.2.3 did not address the OGS pre-heaters, recombiners, dryers, and monitoring instrumentation and controls. Accordingly, in RAI 11.3-14, the staff asked the applicant to revise the discussions in DCD Tier 2, Section 11.3.2.3 to include all equipment described in DCD Tier 2, Section 11.3.2.2. In its response, the applicant provided a proposed revision to DCD Tier 2, Sections 11.3.2.2 and 11.3.2.3 for inclusion in DCD Tier 2, Revision 6. The proposed revision updates the information on system descriptions and locations of major components in the turbine building. The staff finds the proposed changes acceptable. Based on the applicant’s response, this RAI is resolved. The staff confirmed that these changes were included in DCD Tier 2, Revision 6.

The staff’s review also found an improper reference to a DCD Tier 2, Revision 5, Section 12.2, Table 12.2-18b which should be Table 12.2-17, some listed references that were not cited in the text, e.g., Ref. 11.3-10 and the reference list included improper regulatory citations, (e.g., Ref. 11.3-1). In RAI 11.0-1, the staff asked the applicant to make the appropriate corrections. In its response, the applicant agreed to make the appropriate corrections and provided proposed changes to be included in DCD Tier 2, Revision 6. The staff finds the proposed changes acceptable. Based on the applicant’s response, this RAI is resolved. The staff confirmed that these changes were included in DCD Tier 2, Revision 6.

In RAI 11.3-15, the staff noted that Section 11.3.2.6.2 states that the OGS’s radioactive gaseous pressure relief discharge is piped to the main condenser, but it is not clear if the design considers the effects of back pressure on relief setting and capacity. The applicant was requested to explain if back pressure was taken into consideration in the design, since excessive back pressure in the condenser can affect the relief setting and relieving capacity. The DCD should confirm that back pressure spikes will not compromise pressure relief setting

and relieving capacity. In the response, the applicant described the operational features of the condenser and pressure trip points at which alarms would be activated in the control room, and a turbine trip and reactor scram would occur, followed by the closure of the main steam isolation valves, if internal pressure levels were to increase further. In its discussion, the applicant refers to supporting information presented in DCD Tier 2, Revision 5, Section 10.4.1, Table 10.4-1. The staff finds the description of these operational features satisfactory. Based on the applicant's response, this RAI is resolved. The staff confirmed that this change was included in DCD Tier 2, Revision 6.

In RAI 11.3-16, the staff noted that Section 11.3.2.6.8 states that channeling in the OGS charcoal beds is prevented by a high charcoal bed height-to-particle diameter ratio. The word "particle" is deemed to be incorrect in the proposed context. The applicant was requested to consider whether "particle" should be changed to read "diameter" instead, since flow channeling is affected first by bed-height to bed-diameter ratio of the vessel. In the response, the applicant described the relationship between charcoal bed-to-particle diameter ratio and flow profiles across the cross-sectional area of the charcoal bed as a function of charcoal particle size. In its discussion, the applicant refers to supporting information on charcoal particle sizes used in the design of the OGS, as presented in DCD Tier 2, Revision 5, Section 11.3.1, Table 11.3-1. The staff finds the supplemental information satisfactory. Based on the applicant's response, this RAI is resolved. The staff confirmed that this change was included in DCD Tier 2, Revision 6.

11.3.4 Conclusions

Based on the information discussed above, the staff concludes that the GWMS/OGS (as permanently installed systems) and building HVAC systems include the equipment necessary to manage and treat process streams and control releases of radioactive materials in gaseous effluents in accordance with 10 CFR 20.1302 and 10 CFR 20.1406; Appendix I to 10 CFR Part 50; GDC 3, 60, and 61; and 10 CFR 50.34a. This conclusion is based on the following requirements that:

- The ESBWR design meets the dose requirements of 10 CFR 20.1302 by ensuring that the annual average concentration of radioactive materials in gaseous effluents released into unrestricted areas will not exceed the limits specified in Appendix B to 10 CFR Part 20, Table 2, Column 1, as demonstrated in DCD Tier 2, Revision 9, Section 12.2.2.

- The ESBWR design complies with the requirements of Sections II.B and II.C of Appendix I to 10 CFR Part 50, in ensuring that offsite individual doses resulting from gaseous effluent releases will not exceed dose criteria, as demonstrated in DCD Tier 2, Revision 7, Section 12.2.2. These requirements are the subject of COL Information Item 12.2-2-A in DCD Tier 2, Revision 9, Section 12.2.4.

- The ESBWR design demonstrates compliance with 10 CFR 50.34a requirements for sufficient design information, as set forth in the above discussion.

- When preparing a plant-specific cost-benefit analysis in accordance with RG 1.110, a COL applicant referencing the ESBWR certified design is required to demonstrate compliance with the requirements of Sections II.B, II.C, and II.D of Appendix I to 10 CFR Part 50 for offsite individual doses and population doses resulting from gaseous effluents treated by the GWMS and OGS systems.

- The ESBWR design meets the requirements of GDC 3 in protecting the OGS from the effects of explosive gas mixtures of hydrogen and oxygen.

- The portions of the GWMS design features requiring normal ventilation and venting of specific components, as described in DCD Tier 2, Revision 9, Sections 9.4 and 12.2, satisfies the guidance in RG 1.140.

- The design features of the OGS satisfy the guidance in RG 1.143, as it relates to the certification of pressure-retaining components and material specifications in withstanding an explosion without the loss of integrity.

- The design features of the OGS charcoal delay bed ensure conformance with the BTP 11-5 dose guidelines for the analysis of a postulated failure of a component for a receptor located at the EAB.

- The ESBWR design meets the requirements of GDC 60 and 61 with respect to controlling releases of gaseous effluents by radiation monitoring of releases from the GWMS. Radiation monitors track all releases and will generate an alarm, a signal, or both to divert gaseous effluent releases before discharge concentrations exceed a predetermined set point. A COL Licensee will identify the operational set points for its GWMS/OGS radiation monitors in its plant-specific ODCM, or discuss the process in description of the operational program for the ODCM, as discussed in DCD Tier 2, Revision 9, Section 11.5.4.

- Compliance with the requirements of GDC 61 has been demonstrated by meeting the guidelines in RGs 1.140 and 1.143. This commitment also fulfills the requirements of 10 CFR 20.1406 and guidance in RG 4.21 by minimizing the contamination of the facility and the generation of radioactive waste and in IE BL 80-10 in avoiding the cross-contamination of nonradioactive systems and unmonitored and uncontrolled radioactive releases to the environment.

- The applicant demonstrates compliance with the requirements of 10 CFR 52.47(b)(1) with the inclusion of three ITAAC for the GWMS. These ITAAC address the descriptions and functional arrangements of the GWMS, the integrity of the GWMS under expected operating pressures and internal hydrogen explosions, the initial loading of the appropriate amounts of charcoal media, and the proper operation of the offgas post treatment radiation monitor in providing automatic closure of OGS discharge isolation valves on receipt of a high radiation signal. If the inspections, tests, and analyses are performed and the acceptance criteria met, the proposed ITAAC provide reasonable assurance that a plant that incorporates the ESBWR design certification and operates in accordance with the design certification will meet the provisions of the AEA and NRC regulations.

11.4 Solid Waste Management System

11.4.1 Regulatory Criteria

The staff reviewed DCD Tier 2, Revision 9, Section 11.4 in accordance with the guidance and acceptance criteria described in SRP Section 11.4. The following acceptance criteria are applicable:

- 10 CFR 20.1302, as it relates to radioactive materials released in gaseous and liquid effluents and doses to persons in unrestricted areas (criteria that apply to releases resulting from the SWMS during normal plant operations and AOOs)

- 10 CFR 20.1406, as it relates to facility design and operational procedures for minimizing the contamination of the facility and the generation of radioactive waste

- 10 CFR 20.2006 and Appendix G, "Requirements for Transfers of Low-Level Radioactive Waste Intended for Disposal at Licensed Land Disposal Facilities and Manifests," to 10 CFR Part 20, as they relate to the transfer and manifesting of radioactive waste for disposal at licensed land disposal facilities

- 10 CFR 50.34a, as it relates to providing adequate system design information to demonstrate that design objectives have been met for equipment necessary to control releases into the environment of radioactive effluents resulting from SWMS operation

- GDC 60, as it relates to the design of the SWMS incorporating the means to handle solid wastes produced during normal plant operation, including AOOs

- GDC 63, "Monitoring fuel and waste storage," as it relates to the design of the radioactive management systems to control releases of radioactivity

- 10 CFR Part 61, "Licensing Requirements for Land Disposal of Radioactive Waste," as it relates to the classification, processing, and disposal of solid radioactive wastes

- 10 CFR Part 71, "Packaging and Transportation of Radioactive Material," as it relates to the packaging of radioactive materials

- 49 CFR Parts 171–180, as they relate to the packaging of waste, labeling of waste containers, placarding of waste shipments, and transportation of radioactive materials

Specific acceptance criteria for the relevant requirements identified above are as follows:

- SRP Section 11.4, BTP 11-3, Revision 3, "Design Guidance for Solid Radioactive Waste Management Systems Installed in Light-Water-Cooled Nuclear Power Reactor Plants"

- Appendix 11.4-A, "Design Guidance for Temporary Storage of Low-Level Radioactive Waste," to SRP Section 11.4. Appendix 11.4-A addresses the guidance of Generic Letter (GL) 80-009 on low level radioactive waste disposal; GL 81-038 on the storage of low level radioactive waste at reactor sites; and GL 81-039 on the NRC low-level radioactive waste volume reduction policy

- RG 1.143, with respect to specific guidelines for solid radwaste systems; seismic qualification; general guidelines for design, construction, and testing criteria for radwaste systems; and general QA guidelines for radwaste management systems

- RG 4.21, as it relates to minimizing the contamination of equipment, plant facilities, and environment, and minimizing the generation of radioactive waste during plant operation

- RG 8.8, Revision 3, "Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations Will Be as Low as Is Reasonably Achievable"

- RG 8.10, Revision 1-R, "Operating Philosophy for Maintaining Occupational Radiation Exposures as Low as Is Reasonably Achievable,"

- The provisions of GL 89-001, "Implementation of Programmatic and Procedural Controls for Radiological Effluent Technical Specifications" (Supplement No. 1, dated November 14, 1990), as it relates to the restructuring of the radiological effluent technical specification (RETS) and process control program (PCP)

- Guidance of NUREG–1302, "Offsite Dose Calculation Manual Guidance: Standard Radiological Effluent Controls for BWRs," as it relates to the development of a plant-specific PCP. Alternatively, a COL applicant may use Nuclear Energy Institute (NEI) PCP Template 07-10A (Rev. 0, March 2009) for the purpose of meeting this regulatory milestone until a plant-specific PCP is prepared, before fuel load, under the requirements of a license condition described in FSAR Section 13.4 of the COL application. The results of the staff's evaluation are presented in ML082910077 and the NEI PCP Template 07-10A is presented in ML091460236.

- NRC Regulatory Issue Summary (RIS) 2008-32, "Interim Low Level Radioactive Waste Storage at Reactor Sites," December 30, 2008

The staff performed a comparison of the SRP (Section 11.4, 1981 version) used during the review of the DCD with the 2007 version of the SRP. The 2007 version includes additional acceptance criteria and guidance addressing the requirements of 10 CFR 20.1406, when compared to the prior version of the SRP. The requirements of 20.1406 were considered in the staff's review of the DCD, given the 2007 version of the SRP. Discussions and dispositions of these items are provided in this and other supporting sections of this report. Therefore, the staff concludes that the version of the SRP used, in combination with the additional review performed by the staff, is adequate for this review.

11.4.2 Summary of Technical Information

DCD Tier 2, Revision 9, Section 11.4 describes the SWMS used to control, collect, handle, process, package, and temporarily store wet and dry solid radioactive wastes before shipment. Radioactive wastes will be generated during normal operation and AOOs. The SWMS is located in the radwaste building. The SWMS has no safety-related function. Failure of the subsystem does not compromise any safety-related system or component, nor does it prevent the safe shutdown of the plant. No interface with the Class IE electrical system exists. The SWMS is designed to meet the requirements in RG 1.143, with regard to its seismic qualifications. DCD Tier 2, Revision 9, Sections 3.2 and 3.8 discuss the seismic and quality group classification and corresponding codes and standards that apply to the design of the SWMS components and piping and the structures housing the SWMS.

The SWMS processes wastes from the LWMS, RWCU/SDC system, FAPCS, and condensate purification system. DCD Tier 2, Revision 9, Figures 11.4-1 through 11.4-4 show the functional arrangements of SWMS components, which are described in DCD Tier 2, Revision 9, Table 11.4-1. The SWMS can be operated from local panels and from the radwaste building control room. The instrumentation monitors such features as tank levels, process flow rates, and radiation levels. There are no provisions to release liquid wastes from the SWMS. Releases of liquid wastes are conducted through the LWMS. The SWMS is comprised of the following four subsystems:

(1) The SWMS collection subsystem;
(2) The SWMS processing subsystem;
(3) The dry solid waste accumulation and conditioning subsystem; and
(4) The container storage subsystem.

The SWMS collection subsystem consists of high and low activity resin holdup tanks, phase separators, a condensate resin holdup tank, decant pumps, sampling points, control panels, instrumentation, vents and drains, and high and low activity transfer pumps. The operation of the system is supported by plant service utilities, such as compressed air, water, electricity, ventilation, and radiation monitoring. Radioactive wastes processed by the SWMS collection subsystem include spent resins from the RWCU system and the FAPCS, resins from the equipment and floor drain ion-exchangers, dewatering fill head, concentrated wastes, condensate filter backwash drains, equipment and floor drain filter backwash drain, reject waste from the equipment and floor drain osmosis units, chemical drain collection tanks, and condensate demineralizers. Tank overflows are routed to radwaste equipment drains and tanks are vented through filtration systems and monitored for radioactivity before being discharged to the environment via the radwaste building stack.

The SWMS design includes six tanks: high-activity and low-activity resin holdup tanks to receive processed wastes, each with a nominal capacity of about 70,000 liters (18,500 gallons); two low-activity low-phase separator tanks with a nominal capacity of about 55,000 liters (14,500 gallons) each; one condensate resin holdup tank with a nominal capacity of about 70,000 liters (18,500 gallons); and one concentrated waste tank with a nominal capacity of about 60,000 liters (15,800 gallons).

This SWMS design includes four decant pumps, each with a nominal flow rate of about 330 liters per minute (88 gallons per minute); four high and low activity resin transfer pumps, each with a nominal flow rate of about 380 liters per minute (100 gallons per minute); two circulation concentrated waste pumps, each with nominal flow rates of about 1333 liters per minute (352 gallons per minute); and two resin transfer pumps with a nominal flow rate of about 379 liters per minute (100 gallons per minute) each.

The SWMS processing subsystem consists of a vented dewatering and fill head atop a liner, and an associated dewatering pump. Radioactive wastes processed by the SWMS processing subsystems include concentrated wastes and resins from the SWMS collection subsystem, and resins and sludge from the spent resin and phase separator tanks. The dewatering skid returns the liquid waste to the low and high activity phase separators for reuse or further processing. Condensate water may be used for flushing purposes through the fill head. The two dewatering pumps have a rated capacity of 75 liters per minute (20 gallons per minute) each. The dewatering skid drain is routed to radwaste equipment drains, and the fill head is vented through filtration systems and monitored for radioactivity before being discharged to the environment through the radwaste building stack.

The container storage subsystem and the dry solid waste accumulation and conditioning subsystem are designed to process solid wastes. Solid wastes include spent filter cartridges, HEPA filters, paper, rags, plastics, protective clothing, tools, and contaminated equipment generated during plant operations and refueling and maintenance outages. DCD Tier 2, Revision 9, Figures 11.4-1 and 11.4-4 provide a conceptual description of the process flow used in handling dry solid and wet wastes. The COL Licensee will address the actual process under operational programs and procedures developed by taking into consideration the regulatory requirements for the processing, storage, packaging, shipment, radiological monitoring, and

disposal of radioactive wastes of the NRC, the U.S. Department of Transportation (DOT), and State and local agencies.

Spent activated charcoals from the GWMS/OGS are not expected to be routinely disposed of as radioactive waste. Rather, spent activated charcoals will be regenerated in place within the OGS. The COL Licensee will address the replacement of the charcoals in affected beds under operational programs and procedures, in the event that activated charcoals contained in the guard or main beds become contaminated with chemicals or saturated with water.

DCD Tier 2, Revision 9, Table 11.4-2 lists the expected amounts of radioactive waste generated yearly. The estimates include about 363 m^3 (12,830 cubic feet) for dry active solid waste, and 111 m^3 (3,922 cubic feet) for wet solid wastes. Dry solid wastes include combustible and compressible materials and other unspecified waste forms. Wet solid wastes are comprised of spent resins, filter sludge, and waste concentrates from the LWMS. The estimated generated amounts are about 55 m^3 (1,943 cubic feet) for spent resins, about 6 m^3 (212 cubic feet) for filter sludge, and about 50 m^3 (1770 cubic feet) for waste concentrates. The estimated amounts of mixed waste are about 0.42 m^3 (14.7 cubic feet).

Onsite storage capacity is designed for 6 months of waste generation and stored as packaged waste. Waste packaging includes 55-gallon (about 210-liter) drums, high-integrity containers (HICs), and shielded filter containers. The specific design features of the solid waste processing subsystem are not described in the DCD Tier 2, Revision 9, Section 11.4, the COL Licensee will define specifications and procurement through qualified vendors. The services may include skid-mounted waste treatment systems and the use of offsite waste processing services, such as for waste compaction, treatment, and decontamination. The COL applicant is expected to assess whether expanded low-level waste (LLW) storage capacity, beyond 6 months, is required in light of operating practices, as actual waste or projected generation rates, and whether the COL applicant has access to LLW disposal facilities. Appendix 11.4-A, "Design Guidance for Temporary Storage of Low-Level Radioactive Waste," to SRP Section 11.4 and RIS 2008-32 provide guidance on waste storage at reactor sites.

The SWMS is serviced by the exhaust system of the radwaste building, which includes a HEPA filtration system. Airborne effluent releases from this building are conducted and monitored through the radwaste building stack. DCD, Revision 9, Section 9.4.3, describes the design bases, operation, and monitoring of the radwaste building ventilation system. The PRMS provides for the monitoring and control of gaseous and particulate releases from the radwaste building stack, as described in DCD Tier 2, Revision 9, Section 11.5.3. DCD Tier 2, Revision 9, Tables 11.5-1, 11.5-2, 11.5-6, 11.5-8, and 11.5-9, and Figure 11.5-1 describe the sampling requirements and operational characteristics of the related radiation monitors. All liquid radioactive effluents are processed and discharged through the LWMS. DCD Tier 2, Revision 9, Sections 11.2 and 11.3 describe plant systems used to process and treat liquid and gaseous effluents, respectively. DCD Tier 2, Revision 9, Section 12.2.2 describes the methods used to assess doses to members of the public associated with liquid and gaseous effluent releases from the SWMS, as combined with all other radioactive releases from the radwaste building.

In DCD Tier 2, Revision 9, Section 11.4.6 the applicant identified five COL information items. These items address requirements associated with the guidance in RG 1.143 and RG 8.8, Revision 3, "Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations Will Be as Low as Is Reasonably Achievable," for the testing and operation of all SWMS subsystems, identifying system connections to non-radioactive systems that could become contaminated through improper interfaces, description of a plant-specific PCP, the

consideration of LLW storage in an overall site waste management plan, and compliance with 10 CFR 20.1406 in minimizing contamination of the facility. The COL information items are:

- COL Information Item 11.4-1-A: SWMS Processing Subsystem Regulatory Guide— The COL applicant is responsible for ensuring that SWMS subsystems comply with the guidance in RG 1.143, Revision 2 and RG 8.8 Revision 3, for the testing and operation of all SWMS subsystems.

- COL Information Item 11.4-2-A: Compliance with IE Bulletin 80-10— The COL applicant is responsible for evaluating SWMS subsystems, using the guidance and information in IE BL 80-10, for the purpose of identifying and rectifying connections to systems that are considered nonradioactive but that could become radioactive through improper interfaces with radioactive systems (i.e., a nonradioactive system that could become contaminated through leakage, valving errors, or other operating conditions in radioactive systems).

- COL Information Item 11.4-3-A: Process Control Program— The COL applicant is responsible for the description of a plant-specific PCP addressing operating procedures and technical specifications, as they relate to the classifying, treatment, and disposal of radioactive wastes processed by the SWMS in accordance with regulatory requirements of the NRC, DOT and State and local agencies.

- COL Information Item 11.4-4-A: Temporary Storage Facility— The COL applicant is responsible for the development of an overall site management plan for the storage of radioactive waste using the guidance in Section 11.4 of the SRP. The NRC guidance also includes RIS 2008-32.

- COL Information Item 11.4-5-A: Compliance with Part 20.1406— The COL applicant is responsible for including site specific information describing the implementation of operating programs and procedures in accordance with the requirements of 10 CFR 20.1406 and guidance of RG 4.21. The objectives are to minimize the contamination of plant facilities and environment, facilitate decommissioning, and minimize the generation of radioactive wastes.

11.4.3 Staff Evaluation

The staff reviewed the SWMS in accordance with the guidance of SRP Section 11.4. Staff acceptance of the SWMS is based on the design meeting the requirements of 10 CFR 50.34a, GDC 60, 61, and 63, and RGs 1.143 and 8.8. Under 10 CFR 50.34a, an applicant is required to provide sufficient design information to demonstrate that the design objectives of equipment necessary to control releases of radioactive effluents into the environment have been met. An applicant meets the relevant GDC requirements by using the regulatory positions in RG 1.143, as they relate to the seismic design and quality group classification of components used in the SWMS and structures housing the systems, and those addressing leakage control. RG 8.8 addresses design and operational features to ensure occupational exposures from ambient radiation levels remain ALARA.

The staff reviewed the system design according to the guidelines of RG 1.143 and BTP 11-3. The seismic design and quality group classification of components used in the SWMS and structures housing these systems should conform to the guidelines of RG 1.143. The staff reviewed the system construction standards and proposed construction methods. The staff reviewed the system process flow outlines and evaluated the anticipated operational

requirements. The staff reviewed material specifications and potential leakage paths for those areas that conduct fluid separations.

The ESBWR design to process liquid, wet, and solid wastes relies on the use of two processing subsystems integrated with the operation of other permanently installed equipment. The other two elements of the SWMS, the container storage subsystem and the dry solid waste accumulation and conditioning subsystem, are conceptual descriptions of methods for COL Licensees to handle and process solid wastes and packaged solid wastes. As such, the process is described without the inclusion of equipment. DCD Tier 2, Revision 9, Figures 11.4-1 and 11.4-4 provide conceptual overviews of the processes used in handling dry solid and wet wastes. The COL Licensee will define and implement the actual process under operational programs and procedures developed by taking into consideration the regulatory requirements for the processing, storage, packaging, shipment, radiological monitoring, and disposal of radioactive wastes of the NRC, DOT, and State and local agencies.

The subsystems are designed to process waste efficiently, provide operational versatility, and minimize the generation of extraneous radioactive wastes. The types of waste processing methods and waste processing capacities are selected to be commensurate with the expected types of wastes to be generated and waste generation rates. The following paragraphs summarize the operation of the proposed waste processing subsystems:

- For liquid and wet wastes, processing subsystems will be used to treat spent resins, filter and tank sludge, and concentrated wastes. When sufficient amounts of waste have been collected in the high- or low-activity holdup tank, they will be mixed and routed to the appropriate mobile waste processing system. Pumps are used to decant, circulate, and transfer wet wastes to various tanks and waste processing units. The subsystem, in conjunction with other permanently installed equipment, is used to further process wet wastes and to convey liquid and wet wastes to containers for storage or shipment, with excess water routed back to high or low-activity phase separators or to equipment and floor drain collection tanks, based on water quality. Depending on radioactivity and radiation levels, the COL Licensee may erect additional temporary radiation shielding around waste processing units to minimize radiation exposures and doses to plant workers.

- For dry solid wastes, the processing subsystem will be used to process waste collected in containers at specific workstations and brought to the radwaste building. Such stations include control points located throughout the plant or set up to support specific plant evolutions, such as refueling and other types of outages. Given that most of the solid waste is characterized by low levels of radioactivity, the applicant expects that dry waste containers will be handled manually and by forklifts and stored in the radwaste building. Before shipment, wastes will be sorted and packaged into suitable containers that meet DOT shipping and disposal facility requirements or specifications of an offsite waste processor. The waste will be separated into specific categories, such as non-contaminated wastes, contaminated compressible wastes, and contaminated non-compressible wastes. Contaminated compressible wastes include such items as discarded anti-contamination clothing, plastic, glass, paper, and HEPA filters. Contaminated non-compressible wastes include such items as discarded tools, wood, components, and debris. Depending on radioactivity and radiation levels, the COL Licensee may erect temporary radiation shielding around specific containers to minimize radiation exposures and doses to plant workers.

DCD Tier 2, Revision 9, Section 11.4 provides design features, operating characteristics, and piping and instrumentation diagrams for the SWMS collection subsystem and the SWMS

processing subsystem. The staff has reviewed the system construction standards; system process flow outlines and descriptions; sources of liquid input volumes; collection points of liquid waste; flow paths of liquids through the system, including potential bypasses; provisions for monitoring radioactivity levels in effluent releases; and points of release of liquid effluents to the environment. The SWMS design includes provisions for sampling at specific process points and protects against accidental discharges by the detection of abnormal conditions, as managed under administrative controls by the COL Licensee. The system incorporates design and operational flexibility by providing redundancy in processing wastes to route process streams among subsystems and sufficient storage capacity using multiple collection tanks. The applicant describes provisions for periodic inspection of major components to ensure the capability and integrity of SWMS subsystems. The staff finds the design acceptable with respect to meeting the criteria of GDC 60, 61, 63, and 64; 10 CFR 20.1406; 10 CFR 50.34a; Appendix I to 10 CFR Part 50; and the design guidelines of SRP Section 11.4; and RGs 1.143, 8.8, and 8.10, Revision 1-R. The applicant has indicated that the SWMS is covered by the overall QA program described in Chapter 17. Specifically, the QA requirements address the design, fabrication, procurement, and installation of radioactive waste processing systems.

Once all waste processing subsystems are installed, the COL applicant will subject each to the preoperational tests described in DCD Tier 2, Revision 9, Section 14.2 and associated QA tests. The COL Licensee will conduct periodic inspections of subsystem components to confirm the performance and integrity of all operational functions. The COL applicant and Licensee will be responsible for ensuring that the initial installations and future modifications of processing subsystems comply with the requirements of 10 CFR 20.1406 and the guidance in IE BL 80-10 and RG 4.21 to avoid the cross-contamination of non-radioactive systems and unmonitored and uncontrolled radioactive releases into the environment, and to minimize the contamination of the facility and environment. The staff finds this approach acceptable.

DCD Tier 2, Revision 9, Sections 11.4.2 and 11.4.6 state that waste disposal containers will be selected from options that meet (1) the disposal requirements of 10 CFR Part 61, (2) the specific criteria of the chosen disposal facility or waste processor, and (3) the radioactive waste transportation requirements of 10 CFR Part 71 and relevant DOT regulations under 49 CFR Parts 171–180. The verification of waste characteristics, waste packaging, and waste disposal are within the purview of the COL Licensee. The staff expects that the COL applicant, referencing the ESBWR certified design, will develop a plant-specific PCP, in compliance with 10 CFR Part 61, that identifies the operating procedures (i.e., boundary conditions for a set of process parameters, such as settling time, drain time, drying time) for processing wet solid wastes and parallel sets of conditions to process and prepare dry solid wastes. Therefore, for each COL application, the staff will review the PCP, including dewatering, stabilization, solidification (if performed), and compaction, and determine whether the COL application demonstrates that the SWMS complies with the requirements of 10 CFR 61.55, 10 CFR 61.56, 10 CFR Part 71; and relevant DOT regulations. The scope of the SWMS PCP should include a discussion of conformance to RGs 1.143, 8.8, and 4.21, and it should address the issues raised in GL 80-009, “Low Level Radioactive Waste Disposal,” dated January 29, 1980; GL 81-039, “NRC Volume Reduction Policy,” dated November 30, 1981; and GL 89-001, and the guidelines of SRP Section 11.4, including BTP 11-3, Appendix 11.4-A, and NRC RIS 2008-32 for short and extended storage capabilities. It should also include a discussion of equipment containing wet and liquid wastes located in the non-seismic-rated radwaste building. In DCD Tier 2, Revision 9, Section 11.4.6, the applicant identifies COL information items to meet the above requirements and guidance concerning the processing of wet and dry solid wastes. The staff finds the proposed approach and the integration of SWMS operational requirements into the PCP acceptable. The staff also finds the COL information items acceptable.

The design of components and subsystems of mobile waste processing systems that are used by contractors to process wet and solid wastes and chemical wastes on behalf of a COL Licensee are not within the scope of the ESBWR certified design. The portion of the SWMS that is within the scope of the ESBWR certified design complies with the provisions of RG 1.143, with respect to specific guidelines for solid radwaste systems; general guidelines for design, construction, and testing criteria for radwaste systems; and general guidelines for providing QA for radwaste management systems. DCD Tier 2, Revision 9, Sections 3.2 and 3.8 discuss how the design of the SWMS and the radwaste building meet the applicable guidelines of RG 1.143 and the codes and standards listed in Table 1 of RG 1.143. The COL Licensee is also responsible for testing all waste processing subsystems installed in the plant. Chapter 14 of this report addresses the adequacy of the preoperational testing program for the SWMS.

The design of the radwaste building includes an onsite storage capacity of up to 6 months. Based on the applicant's projected waste generation rates, the staff finds that the ESBWR design has sufficient onsite storage capacity only in the short term. The need for storage space capacity beyond 6 months is left to the determination of the COL applicant or Licensee. The design conforms to the guidelines of BTP 11-3 and Appendix 11.4-A to SRP Section 11.4. In GL 81-038, "Storage of LLW at Power Reactor Sites," dated November 10, 1981, the NRC provides guidance to licensees on the addition of onsite storage facilities for LLW generated onsite. Appendix 11.4-A, to SRP Section 11.4 and RIS 2008-32 provide guidance on waste storage at reactor sites. Appendix 11.4-A addresses the guidance of GL 80-009 on LLW disposal; GL 81-038 on the storage of LLW at reactor sites; and GL 81-039 on the NRC LLW volume reduction policy. The guidance addresses technical issues in considering the duration of the intended storage, types and forms of wastes, selection and expected long-term integrity of storage containers, and amounts of radioactive materials contained in wastes to ensure public health and safety, minimize doses to operating personnel, and protection of the environment. In considering the design and construction of an onsite LLW storage facility or modifications to existing storage capacity, the COL Licensee is expected to follow the requirements of the change process that will be outlined in the ESBWR design certification rule (similar to the process included in 10 CFR 50.59), as it relates to facility modifications, changes in structures, systems, and components that could affect performance and compliance with the requirements in 10 CFR Part 20 and Part 50, and changes in methods described in the FSAR and operating procedures.

The staff recognizes that the need for additional onsite storage capacity for LLW is a plant-specific consideration, which depends, in part, on whether the State or a regional LLW compact has provided a facility for long-term storage and disposal. The availability of offsite LLW storage space is beyond the control of the COL applicant or Licensee. Consequently, when offsite storage or disposal capacity becomes available, the COL applicant or Licensee should submit to the NRC the details of arrangements about long-term onsite storage or disposal of LLW. The COL applicant or Licensee should evaluate the need for any additional waste storage capability and the design features of such a facility using the requirements of the change process that will be outlined in the ESBWR design certification rule and the technical guidance in SRP Section 11.4, NRC RIS 2008-32, and RGs 1.143, 4.21, 8.8, and 8.10. The staff will review and evaluate such a proposed additional plant-specific facility against the guidelines in GL 81-038, which is similar to the guidance in Appendix 11.4-A to SRP Section 11.4. In light of the above considerations, the applicant revised DCD Tier 2, Revision 4 Section 11.4.6 to identify the need for LLW storage as part of an overall site management plan as COL Information Item 11.4-4-A. The staff finds the proposed approach and revision to DCD Tier 2, Revision 4, Section 11.4.6 acceptable.

According to the dose objectives in Appendix I to 10 CFR Part 50, the COL applicant is responsible for addressing the requirements for controlling doses to a hypothetical maximally exposed member of the public and populations living near the proposed nuclear power plant. Sections II.A, II.B, II.C, and II.D contain the requirements. The requirements define dose objectives for liquid and gaseous effluents and require a cost-benefit analysis in justifying installed processing and treatment systems for liquid and gaseous radioactive wastes. The LWMS and GWMS will control liquid and gaseous effluents, respectively, generated by the SWMS. Accordingly, compliance with the requirements of Appendix I for the SWMS is subsumed in the respective COL information items noted in Section 11.2 of this report for the LWMS and Section 11.3 of this report for the GWMS.

The SWMS subsystems generate liquid and wet radioactive wastes from the associated operation of the SWMS collection subsystem and the SWMS processing subsystem. Such liquid and wet wastes could potentially cross-contaminate non-radioactive systems, and result in the contamination of nearby facilities and equipment, as well as unmonitored and uncontrolled radioactive releases to the environment. In DCD Tier 2, Revision 9, Sections 11.4.1 and 11.4.2, the applicant states that the design of SWMS subsystems follows the guidance in IE BL 80-10 and the requirements of 10 CFR 20.1406. The design includes drains and vents to route radioactive process or waste streams and avoids interconnections between plant systems that could become radioactive through improper interfaces with radioactive systems. In addition, the DCD commits the COL applicant to ensure that system interfaces and connections and component design features comply with the associated requirements and guidance. The staff finds such design features and COL commitments acceptable and in compliance with the requirements of 10 CFR 20.1406 and the guidelines of IE BL 80-10 and RG 4.21. DCD Tier 2, Revision 9, Section 12.3 outlines design concepts and features that are expected to address such concerns using the guidance of RG 4.21.

In reviewing the prior versions of DCD Tier 2, the staff found that some information was not sufficient for it to determine the acceptability of the SWMS. The staff issued a number of RAIs, not listed here for the sake of brevity, during the review of the application. These RAIs involved requests for the applicant to provide clarifications for technical completeness, provide details supporting design bases and design descriptions in demonstrating compliance with regulatory requirements, revise and update system drawings for consistency with system descriptions, revise technical and regulatory references, and provide information for the staff to conduct independent evaluations of results presented in the application. These RAIs were satisfactorily resolved by the applicant and closed by the staff in DCD Tier 2, Revision 6. The applicant responded to the staff's RAI, and the following paragraphs discuss the staff's evaluations of responses on important technical and regulatory topics.

In RAI 11.4-13, the staff requested additional information on how large system components (e.g., pumps, vessels, etc.) or voluminous amounts of waste (e.g., spent charcoal) will be handled and disposed of as radioactive wastes. In response, the applicant stated that such wastes will be handled on a specialized basis using offsite waste processors, as needed. For OGS spent charcoal adsorbers, the approach describes a method by which spent activated charcoals will be regenerated within the OGS. If activated charcoals in the guard or main beds become contaminated with chemicals or saturated with water, the COL Licensee will address the replacement of the charcoals in affected beds under operational programs and procedures. In general, large components and other voluminous amounts of waste can be temporarily held in the radwaste building or in other staging areas, or they can be decontaminated and shipped to offsite facilities for processing, storage, and disposal, given access to appropriate disposal facilities. Alternatively, a COL applicant or Licensee may propose the design and construction

of a separate onsite radioactive storage building to supplement the storage capacity of the radwaste building. The decision to build a dedicated onsite radioactive waste storage building may depend, in part, on the availability of waste storage and disposal space provided by the State or regional LLW compacts. In either case, the staff finds such considerations plausible and acceptable.

In addition, in the response to RAI 11.4-13, the applicant indicated a similar approach for managing mixed wastes (i.e., those with radiological and chemical hazardous properties). The facility will collect mixed wastes and store them in appropriate containers, such as 55-gallon (208-liter) drums, and ship them offsite to authorized processing facilities. In some instances, the plant may use other types of containers, such as HICs, based on the radiological and chemical properties of specific mixed wastes. Regulations of the NRC and the U.S. Environmental Protection Agency (EPA) control the storage of mixed wastes, which must be shipped in accordance with applicable EPA and DOT requirements. Some States require a COL applicant or the Licensee to comply with additional regulations addressing the characterization, treatment, transportation, and disposal of mixed wastes. The staff finds this approach acceptable in dealing with requirements governing the presence of any other toxic or hazardous properties of materials that may be disposed of under the NRC regulations. Based on the applicant's response, RAI 11.4-13 is resolved. The staff confirmed that this change was included in DCD Tier 2, Revision 6.

In RAI 11.4-15, the staff requested the addition of ITAAC to verify that the plant configuration is consistent with the described operations and process diagram. In its response, the applicant stated that the SWMS is not safety-related and does not qualify as a regulatory treatment of non-safety systems (RTNSS), and thus, it is not safety significant. Therefore, under the guidance in SRP Sections 14.3 through 14.3.11 and RG 1.206, "Combined License Applications for Nuclear Power Plants," issued June 2007, only the system name is required to be included in DCD Tier 1. DCD Tier 1 currently contains some design descriptions without an ITAAC table and, therefore, already contains more information than is required. Consequently, DCD Tier 1 requires no additional information for the SWMS. The staff reviewed the above response to RAI 11.4-15 and found the response not acceptable. The staff determined that the safety significance of the SWMS is at the same level as that of the LWMS and GWMS. The level of detail for the SWMS ITAAC should be similar to that of LWMS and GWMS, which include an ITAAC table to describe "design commitment," "inspection, tests, and analyses," and "acceptance criteria." In response to a supplemental RAI, the applicant proposed to include specific ITAAC for the SWMS addressing the requirements of 10 CFR 52.47(b)(1). These ITAAC are described in DCD Tier 1, Revision 6, Section 2.10.2 and Tables 2.10.2-1 and 2.10.2-2. Two ITAAC address the descriptions and functional arrangements of the SWMS, confirm the integrity of the SWMS against leakage when subjected to testing pressures expected during operation, and verify the nominal capacities of the major processing tanks, including the high and low activity resin holdup tanks, the condensate resin holdup tank, the phase separator tanks, and the concentrated waste tank. If the inspections, tests, and analyses are performed and the acceptance criteria met, the proposed ITAAC provide reasonable assurance that a plant that incorporates the ESBWR design certification and operates in accordance with the design certification will meet the provisions of the AEA and NRC regulations. Based on the applicant's response, this RAI is resolved. The staff confirmed that this change was included in DCD Tier 2, Revision 6.

A COL applicant referencing the ESBWR certified design should describe the QA program for design, fabrication, procurement, construction of structures, and installation of permanent or skid-mounted SWMS and its components in the plant in accordance with its overall QA

program. However, DCD Tier 2, Revision 3, Section 11.4.6 did not commit the COL applicant to conform to the QA guidance specified in RGs 1.21, 1.33, and 4.15. In a global response to RAI 11.5-44, the applicant proposed changes to all related sections of Chapter 11 on this topic and stated that the applicable QA requirements are described in DCD Tier 2, Table 17.0-1. As a result, the applicant has revised the text of DCD Tier 2, Section 11.4.4 to reference the QA requirements of Chapter 17 for the design, fabrication, procurement, and installation of solid and wet radioactive waste systems in accordance with the COL Licensee's overall QA program. In a letter dated July 23, 2007, the applicant committed to placing this information in DCD Tier 2, Revision 4. Based on the applicant's response, this RAI is resolved. The staff confirmed that this change was included in DCD Tier 2, Revision 4.

In RAI 11.4-18, the staff asked the applicant to revise DCD Tier 2, Revision 3, Section 11.4.2 and DCD Tier 1, Revision 3, Section 2.10.2 to indicate that the solid and wet radioactive waste mobile processing system is a conceptual design and should include a COL information item committing the COL applicant to provide complete descriptions and specifications of the mobile SWMS and its subsystems so as to meet the specifications described in DCD Tier 2, Revision 3, Table 11.4-1 and Figure 11.4-1. The staff evaluated the SWMS and use of mobile waste processing systems and concluded that the design of the SWMS is conceptual and, therefore, not in the scope of design certification, given the requirements of 10 CFR 52.47(a). Alternatively, the applicant may provide final descriptions and specifications of the mobile SWMS and its subsystems in the DCD rather than conceptual design information, with ITAACs included as appropriate. In the context of DCD Tier 1 requirements, design descriptions and interface requirements are intended to serve as binding requirements for the purpose of confirming that the plant will be built according to the design features and specifications described in DCD Tier 1.

In responses dated November 16, 2007 and March 17, 2008, the applicant agreed to remove the conceptual designs of the SWMS processing systems from the DCD and instead provide full descriptions of SWMS subsystems in DCD Tier 2, Revision 6, Section 11.4 and DCD Tier 1, Section 2.10.2. Based on the applicant's response, this RAI is resolved. The staff confirmed that this change was included in DCD Tier 2, Revision 6.

The staff's review found that Figure 11.4-4 was included but not cited in the text, some of the listed references were not cited in the text, e.g., Ref. 11.4-5, and the reference list included improper regulatory citations, e.g., Ref. 11.4-8. In RAI 11.0-1, the staff asked the applicant to make the appropriate corrections. In its response dated November 13, 2008, the applicant agreed to make the appropriate corrections and provided proposed changes to be included in DCD Tier 2, Revision 6. The staff finds the proposed changes acceptable. Based on the applicant's response, this RAI is resolved. The staff confirmed that these changes were included in DCD Tier 2, Revision 6.

In DCD Tier 2, Revision 9, Section 11.4.6 the applicant identified COL Information Items 11.4-1-A through 11.4-5-A. The five COL information items identify responsibilities in following the guidance of RG 1.143, Revision 2, RG 8.8 and IE BL 80-10. The COL information items also address compliance with 10 CFR 20.1406 and the guidance of RG 4.21 in minimizing the contamination of plant facilities and environment. Finally, the COL information items assign responsibilities for the management and storage of LLW via the implementation of plant-specific PCP. The staff finds the inclusion of these five COL information items acceptable.

In addressing, Task Action Plan, Item C-17, "Interim Acceptance Criteria for Solidification Agents for Radioactive Solid Waste," DCD Tier 2, Section 11.4 describes the design features of

the SWMS to collect, process, and package wet and dry solid wastes before shipment to disposal sites or offsite waste processors. As a result, the COL Licensee is responsible for the implementation of a plant-specific PCP presenting operating procedures and technical specifications for the classification, treatment, and disposal of radioactive wastes in accordance with regulatory requirements of the NRC, DOT and State and local agencies. The parameters and criteria used to process, treat, store, and ship wastes are to be included in a plant-specific PCP and implementing procedures. Guidance on the development of a plant-specific PCP is contained in GL 89-001 and NUREG–1302. The commitment to develop a PCP is identified under COL Information Item 11.4-3-A in DCD Tier 2, Section 11.4. In fulfilling this requirement, the COL applicant has two options, (a) prepare a plant-specific PCP using NRC criteria and guidance, or (b) adopt by reference NEI PCP Template 07-10A, Revision 0 in meeting this regulatory milestone until a plant-specific PCP is prepared before fuel load under the requirement of a license condition described in FSAR Section 13.4 of COL applications. The results of the staff's evaluation of the NEI PCP template are presented in the final safety evaluation for NEI 07-09. Accordingly, the option of preparing and submitting a plant-specific PCP under FSAR COL Information Item 11.4-3-A, or adopting by reference NEI PCP Template 07-10A and preparing a plant-specific PCP before fuel load is deemed acceptable by the staff. Either option is acceptable in complying with Item C-17 of the Task Action Plan.

11.4.4 Conclusions

Based on the information as discussed above, the staff concluded that the SWMS (as a permanently installed system and in combination with other plant systems) includes the equipment necessary to manage and treat process and waste streams and control releases of radioactive materials in liquid and gaseous effluents in accordance with 10 CFR 20.1302 and 10 CFR 20.1406, Appendix I to 10 CFR Part 50, GDC 60, 61, 63 and 64, and 10 CFR 50.34a. This conclusion is based on the following requirements:

- In conjunction with the LWMS and GWMS, the SWMS design meets the dose requirements of 10 CFR 20.1302 by ensuring that the annual average concentration of radioactive materials in liquid and gaseous effluents released into unrestricted areas will not exceed the limits specified in Appendix B to 10 CFR Part 20, Table 2, Columns 1 and 2, as demonstrated in DCD Tier 2, Revision 9, Section 12.2.2.

- In conjunction with the LWMS and GWMS, the SWMS design complies with the requirements set forth in Sections II.A, II.B, and II.C of Appendix I to 10 CFR Part 50, by ensuring that offsite individual doses resulting from liquid and gaseous effluent releases will not exceed dose criteria, as demonstrated in DCD Tier 2, Revision 9, Section 12.2.2.

- The SWMS design provides sufficient information to demonstrate that it is in compliance with 10 CFR 50.34a, as set forth in the above discussion.

- A COL applicant referencing the ESBWR certified design will demonstrate compliance by preparing a plant-specific cost-benefit analysis in accordance with the guidance in RG 1.110 and the requirements of Sections II.A, II.B, II.C, and II.D of Appendix I to 10 CFR Part 50 for offsite individual and population doses resulting from the operation of waste processing subsystems to treat solid and wet wastes. These requirements are the subject of two COL information items in DCD Tier 2, Revision 9, Section 12.2.4 (i.e., COL Information Items 12.2-2-A, and 12.2-3-A).

- The SWMS design meets the requirements of GDC 60, 61, 63, and 64 with respect to controlling releases of liquid and gaseous effluents by radiation monitoring of such releases in conjunction with the operation of the LWMS and GWMS. Radiation monitors track all releases and will generate a signal to alert or terminate effluent releases before the discharge concentration exceeds a predetermined set point. A COL Licensee will identify the operational set points for its LWMS and GWMS radiation monitors in its plant-specific ODCM, as described in DCD Tier 2, Revision 9, Section 11.5.4.

- The applicant demonstrates compliance with the requirements of GDC 61 by meeting the guidelines in RG 1.143. This commitment also fulfills the requirements of 10 CFR 20.1406 to minimize the contamination of the facility and the generation of radioactive waste and the guidance in IE BL 80-10 and RG 4.21 concerning the avoidance of cross-contamination of nonradioactive systems and unmonitored and uncontrolled radioactive releases into the environment.

- The design of the radwaste building can provide up to 6 months of onsite storage for processed solid and wet wastes. The design conforms to the guidelines of BTP 11-3 and Appendix 11.4-A to SRP Section 11.4. The need for storage capacity beyond 6 months is left to the determination of the COL applicant or the Licensee.

- A COL applicant referencing the ESBWR certified design will be responsible for the description of a PCP. The applicant's proposed PCP should address operating procedures and technical specifications, as they relate to the classifying, treatment, and disposal of radioactive wastes processed by the SWMS in accordance with the requirements of 10 CFR 20.2006; 10 CFR 20.2007, 10 CFR 20.2108, 10 CFR Part 61 and 10 CFR Part 71; and applicable DOT regulations under 49 CFR Parts 171–180.

- In addressing 10 CFR 52.47(b)(1), which requires that a design certification application contain the proposed ITAAC, the applicant has included specific ITAAC for the SWMS. DCD Tier 1, Revision 9 Section 2.10.2 and Tables 2.10.2-1 and 2.10.2-2 describe the ITAAC. These ITAAC address the descriptions and functional arrangements of the SWMS, confirm the integrity of the SWMS against leakage during operation, and verify the nominal capacities of major processing tanks. If the inspections, tests and analyses are performed and the acceptance criteria met, the proposed ITAAC provide reasonable assurance that a plant that incorporates the ESBWR design certification and operates in accordance with the design certification will meet the provisions of the AEA and NRC regulations.

11.5 Process Radiation Monitoring System

11.5.1 Regulatory Criteria

The staff reviewed DCD Tier 2, Revision 9, Section 11.5 in accordance with the guidance and acceptance criteria provided in SRP Section 11.5. The following acceptance criteria are applicable:

- 10 CFR 20.1301 and 10 CFR 20.1302, as they relate to limits on doses to persons and on liquid and gaseous effluent concentrations in unrestricted areas. These criteria apply to all effluent releases resulting from normal plant operations and AOOs

- 10 CFR 20.1406, as it relates to facility design and operational procedures for minimizing the contamination of the facility and the generation of radioactive waste

- GDC 19, “Control room,” as it relates to provisions used in controlling radiation exposures and doses to control room operators during normal operations and postulated accident conditions

- GDC 60, as it relates to controlling releases of radioactive materials into the environment

- GDC 63, as it relates to monitoring fuel and waste storage

- GDC 64, “Monitoring radioactivity releases,” as it relates to monitoring radioactive releases from the containment and effluent discharge pathways in plant environs

- 10 CFR 50.34a, as it relates to the design of equipment and procedures to control releases of radioactive materials into the environment within the numerical guidance provided in Appendix I to 10 CFR Part 50

- Appendix I to 10 CFR Part 50, as it relates to numerical guides for design objectives to meet the requirements of 10 CFR 50.34a and 50.36a, “Technical specifications on effluents from nuclear power reactors,” which specify that radioactive effluents released to unrestricted areas will be kept ALARA

- 10 CFR 50.34(f)(2)(xvii) and 50.34(f)(2)(xxvii), as they relate to monitoring radiation and radioactivity levels for routine operating and accident conditions, consistent with the requirements of GDC 63 and 64 (TMI-related requirements II.F.1, and III.D.3.3)

- 10 CFR 50.34(f)(2)(viii), as it relates to providing the ability to obtain and analyze samples from the reactor coolant system and containment without exceeding occupational radiation exposure dose limits (TMI-related requirement II.B.3)

- 10 CFR 50.34(f)(2)(xxviii), as it relates to monitoring radiation and radioactivity levels and control room habitability, consistent with the requirements of GDC 19 (TMI-related requirement III.D.3.4)

The relevant requirements of the regulations identified above are met by using the regulatory positions and guidance contained in the following RG and industry standards:

- The design of systems should meet the provisions of the applicable regulatory positions given in RGs 1.21 Revision 1; 1.33 Revision 2; 1.97 Revision 4, “Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants,” and 4.15; and guidance from Appendix 11.5-A, “Design Guidance for Radiological Effluent Monitors Providing Signals for Initiating Termination of Flow or Other Modification of Effluent Stream Properties,” to SRP Section 11.5, as well as in RG 1.45 Revision 1, “Guidance on Monitoring and Responding to Reactor Coolant System Leakage,” and RG 4.21.

- Monitoring and sampling of the gaseous and liquid process streams, or effluent release points, should occur according to Tables 1 and 2 of SRP Section 11.5

- The design of aerosol sampling systems should follow the guidance in ANSI/Health Physics Society (HPS) ANSI/HPS N13.1-1999

- The design of continuous RMS should follow the guidance of ANSI N42.18-2004

- The design of the instrumentation and sampling systems used in the event of a postulated accident should meet the provisions of SRP Sections 9.3.2, 11.2, and 11.3

- The description of the operational program should address the development of the plant's SRECs, ODCM, and radiological environmental monitoring program (REMP), which should meet the provisions of GL 89-001 (Supplement No. 1), Radiological Assessment Branch Technical Position (Revision 1, November 1979) included as Appendix A in NUREG–1302, as ODCM guidance for BWR plants, and the guidance in NUREG–0133, "Preparation of Radiological Effluent Technical Specifications for Nuclear Power Plants," October 1978. Alternatively, a COL applicant may use NEI ODCM Template 07-09A, Revision 0 to meet this regulatory milestone until a plant and site-specific ODCM is prepared, before fuel load, under the requirements of a license condition described in FSAR Section 13.4 of the COL application. The staff has reviewed NEI ODCM Template 07-09A and found it acceptable. (See the staff's analysis in "Final Safety Evaluation for NEI 07-09, Revision 4, 'Generic Final Safety Analysis Report Template Guidance for Offsite Dose Calculation Manual (ODCM) Program Description,' " dated January 27, 2009.)

The staff performed a comparison of the SRP (Section 11.5, 1981 version) used during the review of the DCD with the 2007 version of the SRP. The 2007 version includes additional acceptance criteria and guidance addressing the requirements of 10 CFR 20.1406, when compared to the prior version of the SRP. However, the requirements of 20.1406 were considered in the staff's review of the DCD, given the 2007 version of the SRP. Discussions and dispositions of these items are provided in this and other supporting sections of this report. Therefore, the staff concludes that the version of the SRP used, in combination with the additional review performed by the staff, is adequate for this review.

11.5.2 Summary of Technical Information

The primary purpose of the PRMS is to provide information characterizing the types and amounts of radioactivity contained in process streams and liquid and gaseous effluents. Other objectives are to alert control room operators of abnormal levels of radioactivity in process streams and liquid and gaseous effluents, and to provide signals that initiate automatic safety functions, isolate process streams, and terminate effluent discharges if predetermined radioactivity levels or release rates exceed alarm set points. Another function of the PRMS is to provide the means to collect samples from process and effluent streams for radiological analysis. The design objectives and criteria of the PRMS are based on requirements that address the following:

- Radiation monitoring instrumentation required for plant safety and protection
- Radiation instrumentation required for monitoring plant operation

The PRMS consists of skid-mounted and permanently installed sampling and monitoring equipment designed to indicate operational radiation levels and releases of radioactive materials, equipment or component failures, and system malfunctions or improper operation. The PRMS includes beta and gamma radiation sensitive detectors working in redundant channels, as required for each subsystem. The radiation detectors are capable of detecting the types and energies of radiation emitted from fuel, radioactive wastes, and process and effluent streams. Local readout and alarm modules are located at specific areas to provide information on the radiological status of plant systems and to alert personnel of abnormal or accident conditions. The PRMS generates signals to initiate the operation of certain safety-related equipment to control radioactive releases under normal and abnormal operations and accident

conditions. The COL Licensee will subject the PRMS to preoperational tests. The COL Licensee also is responsible for testing all skid-mounted RMS installed in the plant. There are provisions for periodic inspection of major components to ensure the capability and integrity of all PRMS subsystems.

DCD Tier 2, Revision 9, Sections 11.5.1 and 11.5.2, Table 11.5-3, and Figure 11.5-1 list the design bases and criteria and describe the locations of the PRMS components in plant buildings. DCD Tier 2, Revision 9, Tables 11.5-1, 11.5-2, 11.5-4, and 11.5-9 describe the key operational features of the PRMS, including configurations, dynamic detection ranges, principal radionuclides on which initial instrumentation responses are based and types of trip and alarm functions. DCD Tier 2, Revision 9, Section 12.2, presents information on expected radiation or radioactivity levels in various plant systems. DCD Tier 2, Revision 9, Tables 11.5-5 through 11.5-8 describe provisions for sampling and analyzing process and effluent streams. Figure 11.5-2 presents the PRMS interface with the plant's instrumentation and control system, as described in DCD Tier 2, Revision 9, Sections 7.1 and 7.5. Section 11.5.4 presents a regulatory evaluation of the PRMS basis for the selection of the locations of subsystem components, expected radiation or radioactivity levels, instrumentation and sample collection, and requirements for establishing alarm or trip instrumentation set points.

DCD Tier 2, Revision 9, Section 11.5.2.1 indicates that the PRMS subsystems required for plant safety and protection incorporate the following major design requirements:

- Be capable of withstanding the effects of natural phenomena without the loss of operational function;

- Perform intended safety related functions during normal and abnormal conditions;

- Meet the reliability, testability, independence, and failure mode requirements of engineered safety systems;

- Use redundant channels satisfying the separation and single-failure criteria for the initiation of safety functions;

- Provide compatibility with expected radiation levels and ranges under normal operation, abnormal operation, and accident conditions;

- Provide the means for checking the availability and operational status of each RMS channel and calibration and functional checks;

- Provide continuous RMS output and alarm levels in the plant's control room;

- Initiate protective action when operational limits are exceeded; and

- Register full-scale if radiation detection levels exceed full-scale.

The following PRMS subsystems provide signals and initiate automatic safety functions for the building HVAC exhausts:

- Reactor building HVAC exhaust RMS
- Refuel handling area HVAC exhaust RMS
- Control building air intake HVAC RMS

- Isolation condenser vent exhaust RMS
- Fuel building general area HVAC RMS
- Fuel building fuel pool HVAC RMS
- Containment purge exhaust RMS

The safety-related portions of the PRMS are classified as safety Class 2, Seismic Category I, and conform to the QA requirements of Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," to 10 CFR Part 50.

DCD Tier 2, Revision 9, Section 11.5.2.2 states that the PRMS subsystems required for plant operation incorporate the following major functional requirements:

- Provide the operational range response of each subsystem under normal operation, AOOs, and accident conditions;

- Provide self-diagnosis for instrumentation malfunctions, with annunciation provided in the plant's control room and isolation of effluent discharges;

- Ensure compatibility with expected radiation levels and ranges under normal operation, abnormal occurrences, and accident conditions;

- Monitor a representative sample of bulk stream or volume of process and effluent streams;

- Incorporate provisions for instrumentation calibration and functional checks;

- Register full-scale if radiation detection levels exceed full scale;

- Monitor selected non-radioactive systems for the intrusion of radioactivity.

The other subsystems of the PRMS monitor plant operations and provide information on levels of radioactivity present in process streams and liquid and gaseous effluents.

As compared to the prior revisions of the DCD Tier 2, Revision 5 incorporated a major design change associated with the original plant stack design. The prior design (DCD Tiers 1 and 2, up to Revision 4) included a single plant stack for all buildings. The single stack was designed as the single point of releases for all gaseous effluents. In DCD Tiers 1 and 2, Revision 5 the design included three discharge stacks, the reactor and fuel building as one, and separate stacks for the turbine building and radwaste building. Given this change, the PRMS design includes three PRMS subsystems, one for each of the three building stacks. The following PRMS subsystems meet design criteria and provide the means to collect process and effluent samples for radiological analysis:

- PRMS subsystems for gaseous effluents
  - Reactor and fuel building stack RMS
  - Turbine building stack RMS
  - Radwaste building stack RMS
  - Turbine building normal ventilation air HVAC RMS
  - Turbine building compartment area air HVAC RMS
  - Turbine building combined ventilation exhaust RMS
  - Radwaste building ventilation exhaust RMS

- Main turbine gland seal steam condenser exhaust RMS
- Fuel building combined ventilation exhaust RMS

- PRMS subsystems for liquid effluents
  - Liquid radwaste discharge RMS

- PRMS subsystems for gaseous process streams
  - Main steamline RMS
  - Off-gas pretreatment RMS
  - Off-gas post-treatment RMS
  - Charcoal vault ventilation RMS
  - Drywell fission product RMS

- PRMS subsystems for liquid process streams
  - Reactor component cooling water intersystem leakage RMS
  - Drywell sumps low and high conductivity waste discharge RMS

- PRMS subsystems for gaseous intake streams

- Technical support center HVAC air intake RMS

DCD Tier 2, Revision 9, Sections 11.5.5, 7.5.3, and 9.3.2 describe the features of the process monitoring and sampling that would be used for normal operations and under accident conditions. DCD, Revision 9, Tables 11.5-1, 11.5-2, 11.5-5 through 11.5-8, and Table 9.3.1 describe the design for monitoring and sampling these process and effluent streams. The system consists of permanently installed sampling lines, sampling panels with analyzers and associated sampling equipment, provisions for local sampling, and permanently installed radiation shielding. The descriptions include a list of process and effluent systems with operational features, the selection of locations for the placement of RMS monitors, the number of RMS channels, provisions for grab sampling, expected radiation levels, and types of alarms and trips.

Sampling stations or points are provided for the following systems:

- Reactor building
  - RWCU/SDC system
  - FAPCS

- Fuel building
  - Spent fuel pool treatment system

- Turbine building
  - Condensate and feedwater system
  - Moisture separator and reheater system
  - Heater drain and vent system
  - Generator cooling system
  - Turbine main steam system

- Condensate polishing
  - Condensate and feedwater system
  - Condensate purification system

- Condenser
  - Main condenser OGS and auxiliaries

- Radwaste building
  - Equipment and floor drain input
  - Chemical waste drain
  - Detergent waste drain
  - Sample tanks

- Local grab sampling stations and points
  - Reactor component cooling water system
  - Turbine component cooling water system
  - Plant service water system
  - Chilled water system
  - Circulating water system
  - Standby liquid control system
  - Condensate storage and transfer system
  - CST basin sump
  - Equipment and floor drain system
  - Storm and underdrain water system (COL applicant item)
  - Non-contaminated waste water system (COL applicant item)

For gaseous effluents, the system provides for continuous and representative sampling of radioactive airborne particulates, radioiodines, and noble gases from the three building stacks. The PRMS subsystems also provide the means for the grab sampling of noble gases, radioiodines, particulates, and tritium for the listed gaseous radwaste discharges. For liquid process and effluent streams, the system provides grab sampling and analysis capability for gross radioactivity determination, identification of principal radionuclides and alpha emitters, and measurement of their concentrations. DCD Tier 2, Revision 9, Sections 7.5.1 through 7.5.3, and 9.3.2 describe the features of the postaccident sampling system and process sampling system. DCD Tier 2, Section 9.3.2.6 commits the COL applicant (per the COL Information Item 9.3.2-1-A) to develop a postaccident sampling program to monitor plant systems listed in DCD Tier 2, Table 9.3-1.

DCD Tier 2, Revision 9, Sections 11.5.3.1.4 and 11.5.3.2.12 describe the designs of the PRMS subsystems used to monitor the air intakes of the control building and technical support center, respectively, as being compliant with GDC 19. Each RMS subsystem includes provisions to initiate the isolation of the outside air intake and exhaust dampers and startup of the emergency air filtration system when doses to control room operators and occupants of the technical support center are expected to exceed 0.05 Sv (5 rem) during a postulated accident.

DCD Tier 2, Revision 9, Section 11.5.6 describes the requirements for the calibration, inspection, testing, and maintenance of the PRMS. The PRMS includes provisions for self-diagnosis and online calibrations of process monitors that operate continuously. Each monitor

channel has provisions to conduct periodic calibrations using standard radiation sources or electronic test signals. The PRMS includes design features to facilitate such maintenance using modules that can be removed for repairs or replacement. The derivation of each subsystem's lower dynamic range and sensitivity (as the lower limit of detection) is left to the COL applicant, based on site-specific conditions, types of RMS installed, and operating characteristics of each installed subsystem.

In DCD Tier 2, Revision 9, Section 11.5.7, the applicant listed the following five COL information items:

- COL Information Item 11.5-1-A: Sensitivity or Subsystem Lower Limit Detection— The COL applicant is required to derive the lower limit of detection for each effluent PRMS subsystem and response sensitivity for each process PRMS subsystem installed, taking into consideration plant and site-specific conditions.

- COL Information Item 11.5-2-A: Offsite Dose Calculation Manual— The COL applicant is required to develop a plant- and site-specific ODCM for calculating offsite doses resulting from liquid and gaseous effluents and planned discharge flow rates.

- COL Information Item 11.5-3-A: Process and Effluent Monitoring System— The COL applicant is responsible for implementing the requirements in RGs 1.21 and 4.15, and ANSI/HPS N13.1-1999 in developing a process to monitor and extract samples from all identified process and effluent streams.

- COL Information Item 11.5-4-A: Site Specific Offsite Dose Calculation— The COL applicant is responsible for addressing the requirements of the dose objectives in Appendix I to 10 CFR Part 50 for controlling doses to a hypothetical maximally exposed member of the public and populations living near the proposed nuclear power plant. Sections II, III, and IV of Appendix I to 10 CFR Part 50 contain the requirements. A separate set of COL Information Items (DCD Tier 2, Revision 9, Section 12.2.4) addresses the requirements in complying with dose objectives for liquid and gaseous effluents and the conduct of a cost-benefit analysis in justifying installed systems for processing and treating liquid and gaseous radioactive wastes.

- COL Information Item 11.5-5-A: Instrument Sensitivity— The COL applicant is responsible for defining instrumentation response sensitivities, and sampling and analytical frequencies for all listed liquid and gaseous samples extracted from process and effluent streams.

11.5.3 Staff Evaluation

The staff reviewed the PRMS in accordance with the guidance of SRP Section 11.5. The staff's acceptance of the PRMS is based on the design meeting the requirements of 10 CFR 20.1301 and 10 CFR 20.1302; 10 CFR 20.1406; 10 CFR 50.34a; 10 CFR 50.36a; Appendix I to 10 CFR Part 50; GDC 60, 63, and 64; and 10 CFR 50.34(f)(2)(viii), 50.34(f)(2)(xvii), (f)(2)(xxvii), and 50.34(f)(2)(xxviii).

Under 10 CFR 50.34a and 50.36a, the applicant is required to demonstrate that sufficient design information is provided to comply with the ALARA design objectives of Appendix I to 10 CFR Part 50 for equipment necessary to control releases of radioactive effluents into the environment. The relevant requirements of GDC 60, 63, and 64 are met by using the regulatory

positions in RG 1.143, as they relate to the seismic design and quality group classification of components used in plant systems and structures housing the PRMS.

DCD Tier 2, Revision 9, Sections 11.5.1 and 11.5.2, Table 11.5-3, and Figure 11.5-1 list the design bases and criteria and place the locations of the PRMS components in plant buildings. Section 11.5.2.1 identifies radiation monitors required for plant safety and protection, and Section 11.5.2.2 describes radiation monitors required for plant operation. Tables 11.5-1, 11.5-2, 11.5-4, and 11.5-9 describe the key operational features of the PRMS, including configurations, dynamic detection ranges, principal radionuclides on which instrumentation responses are based, and types of trip and alarm functions. DCD Tier 2, Revision 9, Section 12.2, presents information on expected radiation or radioactivity levels in various plant systems. DCD Tier 2, Revision 9, Tables 11.5-5 through 11.5-8 describe provisions for the sampling and analyzing of process and effluent streams. Figure 11.5-2 presents the PRMS interface with the plant's instrumentation and control system, as described in DCD Tier 2, Revision 9, Sections 7.1 and 7.5. Section 11.5.4 presents a regulatory evaluation of the PRMS that addresses the basis for the selection of the locations of subsystem components, instrumentation and sample collection, and requirements for establishing alarm or trip instrumentation set points. The staff evaluated the safety-related portions of the PRMS, classified as safety Class 2, Seismic Category I by the applicant, and considered whether those portions of the PRMS conform to the QA requirements of Appendix B to 10 CFR Part 50.

In addressing the sampling and analysis of process and effluent streams, the applicant follows the NRC and industry guidance. RG 1.21 addresses requirements associated with the ability to perform specific types of radiological analysis, and RG 4.15 covers requirements to calibrate, maintain and inspect instrumentation used to monitor the presence of radioactivity in process and effluent streams, as well as methods to measure effluent discharge flow and radioactivity release rates. Finally, ANSI/HPS N13.1-1999 provides guidance on sampling and monitoring from building stacks, vents, and ducts containing radioactivity; and the ANSI Standard ANSI N42.18-2004 addresses instrumentation designed for continuous monitoring. In DCD Tier 2, Revision 9, Section 11.5.4.6, the applicant stated that the requirements of these two RGs (1.21 and 4.15) and industry guidance are endorsed by reference and are the responsibility of the COL applicant under COL Information Item 11.5-3-A. Two other COL information items (11.5-1-A and 11.5-5-A) require COL applicants to define appropriate PRMS instrumentation detection limits and response sensitivities for process and effluent monitors as well as the frequencies and basis for liquid and gaseous sample collection and analysis. DCD Tier 2, Revision 9, Tables 11.5-7 and 11.5-8 present summaries of the radiological sampling and analyses programs for liquid and gaseous effluents, based on the guidelines in RGs 1.21 and 4.15. The staff finds this approach acceptable.

Under the requirements of 10 CFR 50.34(f)(2)(xvii) and 10 CFR 50.34(f)(2)(xxvii), the applicant must provide the means to monitor radiation and radioactivity levels for routine operating and accident conditions, consistent with the requirements of GDC 63 and 64. The staff finds the range provided in DCD Tier 2, Revision 9, Tables 11.5-1 and 11.5-9 for radiation measurement and sampling of noble gases, particulates, and radioiodines from potential release points to be acceptable because it meets the range criterion for such monitors specified in NUREG–0737, "Clarification of TMI Action Plan Requirements," Three Mile Island (TMI) Item II.F.1, Attachment 3, "Containment High-Range Radiation Monitor," dated November 1980. DCD Tier 2, Revision 9, Section 12.3 and Section 12.4 of this report evaluate the high-range containment radiation monitors.

The staff finds the ranges specified in DCD Tier 2, Revision 9, Table 11.5-1 for the control building radiation monitor to be acceptable as they are consistent with applicable NRC guidance. RGs 1.45, and RG 1.97 present guidance for sampling and monitoring process and effluent streams and analyzing samples, including the proposed analytical programs, during postulated accidents, in accordance with the requirements of 10 CFR 50.34(f)(2)(viii) (TMI-related requirement II.B.3), 10 CFR 50.34(f)(2)(xvii) (TMI-related requirement II.F.1), and 10 CFR 50.34(f)(2)(xxvii) (TMI-related requirement III.D.3.3). DCD Tier 2, Revision 9, Sections 13.3, 13.5, 7.5.1, and 9.3.2 and BTP 7-10, "Guidance on Application of Regulatory Guide 1.97," of SRP Section 7.5 provide more specific information on the application of Revision 4 of RG 1.97, to the ESBWR design. The applicant has adopted the option of BTP 7-10 to define instrumentation response ranges using the provisions of Revision 3 of RG 1.97 (DCD Tier 2, Revision 9, Tables 7.1-1 and 1.9-21). In addition, the staff has determined that these aspects will be integrated as part of the human factors engineering process described in DCD Tier 1, Section 3.7. Section 7.5 of this report presents the staff's evaluation of the provisions associated with the guidelines of Revision 4 of RG 1.97. On the basis of the above discussions, the staff finds that these special-purpose monitors comply with GDC 60 and 64 in terms of their ability to control and monitor the release of radioactive materials into the environment.

Under 10 CFR 50.34(f)(2)(xxviii) (TMI-related requirement III.D.3.4), the applicant must provide the means to monitor radiation and radioactivity levels and control room habitability, consistent with the requirements of GDC 19, during normal operations and postulated accident conditions. DCD Tier 2, Revision 9, Sections 11.5.3.1.4 and 11.5.3.2.12 describe the designs of the PRMS subsystems used to monitor the air intakes of the control building and technical support center, respectively, as complying with GDC 19. Each PRMS subsystem includes provisions to initiate the isolation of the outside air intake and exhaust dampers and the startup of the emergency air filtration system when doses to control room operators and occupants of the technical support center are expected to exceed 0.05 Sv (5 rem) during a postulated accident. The staff finds the design and provisions for automatic closure of the air intake and initiation of the emergency air intake system to be acceptable. DCD Tier 2, Revision 9, Section 6.4 and Section 6.4 of this report discuss the habitability of the control building.

The COL Licensee will subject the PRMS, in conjunction with sampling equipment and portions of process or effluent system components that are activated by the PRMS, to preoperational tests and calibration, as well as maintenance. DCD Tier 2, Revision 9, Section 11.5.6 presents the requirements for the operational programs involving calibration, maintenance, inspections, and tests. The staff finds the scope of the program to be acceptable. Chapter 14 of this report addresses the adequacy of the preoperational testing program for the PRMS.

In DCD Tier 2, Revision 9, Sections 11.5.2, 11.5.3, and 11.5.4.6, the applicant states that the PRMS is designed in accordance with ANSI/HPS N13.1-1999 and applicable RGs 1.21 and 4.15. DCD Tier 2, Section 11.5.7 states that the COL applicant referencing the ESBWR certified design, is responsible for ensuring that the process and effluent monitoring and sampling program conforms to the guidelines of ANSI/HPS N13.1-1999 and RGs 1.21 and 4.15. This requirement is identified as COL Information Item 11.5-3-A. The staff finds this approach acceptable.

DCD Tier 2, Revision 9, Section 11.5.5.9 discusses provisions to collect radioactive samples from radioactive process streams. The applicant states that the sample points are described in DCD Tier 2, Sections 7.5.2 and 9.3.2 and listed in DCD Tier 2, Revision 7, Table 9.3-1. The sampling system is designed according to the requirements and guidelines of

10 CFR 20.1101(b); 10 CFR 50.34(f)(2)(viii) (TMI-related requirement II.B.3), 10 CFR 50.34(f)(2)(xvii) (TMI-related requirement II.F.1), and 10 CFR 50.34(f)(2)(xxvii) (TMI-related requirement III.D.3.3); GDC 19, 60, 63, and 64; RGs 1.21, 1.33, 1.97, and 8.8; NUREG–0737 (TMI Action Plan); and ANSI/HPS N13.1-1999. The systems identified in DCD Tier 2, Revision 9, Table 9.3-1 include the RWCU/SDC, FAPCS, main steam line, condensate purification system, and liquid radwaste system effluent sample tank. Additional sampling stations are provided for other systems, including the condensate and feedwater system, turbine main steam, reactor component cooling water system, standby liquid control system, LWMS, and GWMS. The types of measurements are identified as broad categories, such as gross activity, activity caused by corrosion and activation products, iodine-131, gaseous fission products (xenon and krypton), and principal radionuclides and alpha emitters. DCD Tier 2, Revision 9, Section 9.3.2.6 requires a COL Information Item (9.3.2-1-A) to develop a postaccident sampling and monitoring program based on the information presented in DCD Tier 2, Revision 9, Table 9.3-1 and the NRC guidance of SRP Section 9.3.2. DCD Tier 2, Chapter 12 describes plant design features (shielding and ventilation) and operational programs that will maintain occupational radiation exposures within NRC limits and ALARA during accident conditions in complying with TMI action plan items. The process sampling system consists of permanently installed lines, sampling panels equipped with instrumentation and the associated equipment, provisions for local grab sampling, provisions for obtaining representative samples, heat tracing and cooling for sample conditioning, provisions to purge and flush sampling lines, and permanent shielding. The design also includes provisions to minimize leakage and spillage, return flushing fluids to their appropriate process streams or send them to the radwaste system, and reduce radiation exposures to plant personnel while working at sampling stations. Based on the above, the staff finds that the design is acceptable.

In reviewing DCD Tier 2, Revision 1 the staff found some of the information was insufficient to determine the acceptability of the PRMS and requested additional information. The staff issued a number of RAIs, not listed here for the sake of brevity, during the review of the application. These RAIs involved requests for the applicant to provide clarifications for technical completeness, provide details supporting design bases and design descriptions in demonstrating compliance with regulatory requirements, revise and update system drawings for consistency with system descriptions, revise technical and regulatory references, and provide information for the staff to conduct independent evaluations of results presented in the application. These RAIs were satisfactorily resolved by the applicant and closed by the staff in DCD Tier 2, Revision 6. The following paragraphs discuss the staff's evaluations of the applicant's responses to RAIs on important technical and regulatory topics.

In RAI 16.2-9, as it relates to the submission of an ODCM, as described in DCD Tier 2, Revision 3, Section 11.5.7.2 and the administrative requirements of DCD Tier 2, Revision 1, Section 16.5.5.1.c, and the COL items listed in DCD Tier 2, Revision 3, Section 11.5.7, the staff finds the listed items adequate given that the development process of a plant- and site-specific ODCM, and its associated documents is required to follow the NRC requirements and guidance. A COL applicant should base its ODCM, SREC, and REMP, or the description of their associated operational programs, on the guidance of NUREG–1302 for BWR plants; NUREG–0133; RGs 1.21, 1.33, and 4.15; ANSI/HPS N13.1-1999 and ANSI N42.18-2004; Appendix 11.5-A (Section 11) to the SRP; GL 89-001 (Supplement No. 1); and Radiological Assessment Branch Technical Position (Revision 1). Alternatively, a COL applicant may use NEI ODCM Template 07-09A, Revision 0 to meet this regulatory milestone until a plant and site-specific ODCM is prepared, before fuel load, under the requirements of a license condition described in FSAR Section 13.4 of the COL application. The staff has reviewed NEI ODCM Template 07-09A and finds it acceptable.

In this context, the ODCM, or the description of the operational program for the ODCM, should present the plant's SREC and the REMP. The ODCM, or the description of the operational program for the ODCM, should describe programs and identify procedures used in implementing effluent discharges, define effluent discharge flow rates, provide the basis for liquid effluent dilution factors and atmospheric dispersion and deposition parameters for gaseous effluents, and identify exposure pathways and dose receptors using data from the current local land-use census. The ODCM, or the description of the operational program for the ODCM, should contain the methodology and parameters used for calculating offsite doses to members of the public from gaseous and liquid effluents to demonstrate compliance with the numerical objectives of Appendix I to 10 CFR Part 50; the dose limits of 10 CFR 20.1301 for members of the public; the effluent concentration limits of Appendix B (Table 2) to 10 CFR Part 20; and the compliance requirements of 10 CFR 20.1302. The ODCM, or the description of the operational program for the ODCM, should present methods and parameters used to determine operational set points for effluent radiation monitors in limiting releases of radioactive materials into the environment within the liquid and gaseous effluent concentration limits of Table 2 of Appendix B to 10 CFR Part 20. The ODCM, or the description of the operational program for the ODCM, should also provide instructions for identifying and eliminating the potential for unmonitored and uncontrolled releases. In DCD Tier 2, Revision 9, Section 11.5.7 the applicant states that the development of the ODCM, or the description of the operational program for the ODCM, is the responsibility of the COL applicant under COL Information Items 11.5-2-A and 11.5-4-A. DCD Tier 2, Revision 9, Section 13.4 identifies these milestones as being due before fuel loading, given the requirements of SECY-05-0197, "Review of Operational Programs in a Combined License Application and Generic Emergency Planning Inspections, Tests, Analyses, and Acceptance Criteria," dated October 28, 2005; RG 1.206, and SRP Section 13.4. The staff finds this approach acceptable. Based on the applicant's response, this RAI is resolved. The staff confirmed that this change was included in DCD Tier 2, Revision 6.

In RAI 11.5-5, the staff asked the applicant to provide further elaboration on complying with 10 CFR 20.1406 and using the guidance of IE BL 80-10. In DCD Tier 2, Revision 3, Sections 11.5.2, 11.5.4, and 11.5.5, the applicant did not indicate whether the design of the process and effluent sampling systems follows the guidance of IE BL 80-10 and whether the design avoids interconnections with non-radioactive systems that could become radioactive through improper interfaces with radioactive systems. Similarly, the applicant did not indicate whether the design of the process and effluent sampling systems complies with the requirements of 10 CFR 20.1406, as it relates to the design and operational procedures to minimize contamination and the generation of radioactive wastes. While DCD Tier 2, Revision 3, Section 12.6, addresses some requirements associated with 10 CFR 20.1406, the discussions of DCD Tier 2, Section 12.6 are broadly generic and do not focus on specific design issues for the PRMS. In response to a supplemental RAI, the applicant proposed, in a letter dated August 31, 2007, to revise DCD Tier 2, Sections 11.5.6.4 and 11.5.6.5 by providing more technical details to demonstrate compliance with the guidance of IE BL 80-10 and implementation of 10 CFR 20.1406. The response acknowledges that a potential exists for interconnections with non-radioactive systems and describes design features to prevent the contamination of non-radioactive systems and to minimize radioactive contamination during operation. The applicant describes provisions to protect the clean supply of purge air and water used to flush contaminated subsystems and makeup water used in filling loop seals, and measures to prevent spills and leaks. With the supplemental information presented in DCD Tier 2, Revision 5, Sections 9.3.2, 11.5.6.4, 11.5.6.5, and 12.6. The staff finds the response acceptable. Based on the applicant's response, this RAI is resolved. The staff confirmed that this change was included in DCD Tier 2, Revision 5.

In RAI 11.5-6, as it relates to DCD Tier 2, Revision 1, Sections 11.5.3 and 11.5.4 the staff asked the applicant to describe how the reactor building HVAC exhaust system captures discharges from the isolation condenser vent exhaust. In DCD Tier 2, Revision 3, Section 11.5.3.1.5, the discussion regarding the air exhaust from the atmospheric area above each condenser pool is incomplete. Although the exhaust is monitored by the isolation condenser vent exhaust RMS, it was not clear from this discussion and the information presented in DCD Tier 2, Revision 3, Sections 5.4.6.5 and 5.1.2 and Figure 5.1-3 what design features are provided to prevent the exhaust from the atmospheric area above each condenser pool from becoming an uncontrolled and unmonitored release into the environment. In response to a supplemental RAI, the applicant proposed, to revise DCD Tier 2, Section 11.5.3.1.5 (now 11.5.3.1.6 in Revision 9), to amplify the operational description of the isolation condenser vent exhaust and the related response of radiation monitors that initiate closure of the containment isolation valves for the affected condensers in the event of a condenser tube leak. The supplemental information indicates that the condenser pool is filled with non-radioactive water supplied by the makeup water system. During normal operation, the pool does not become radioactive and the steam generated during boiloff is removed as moisture by a dryer and drained back to the pool. The radiological consequences following a leak from the isolation condenser and the closure of the affected isolation condenser by the radiation monitor is treated generically in DCD Tier 2, Section 15.4.8. The staff finds the response acceptable. Based on the applicant's response, this RAI is resolved. The staff confirmed that this change was included in DCD Tier 2, Revision 6.

In RAI 11.5-8, as it relates to DCD Tier 2, Revision 1, Sections 11.5.3 and 11.5.4, the staff asked the applicant to resolve inconsistencies in addressing competing objectives of RGs 1.21 and 1.97 in describing dynamic response ranges and expected activity levels. The specific information was presented in DCD Tier 2, Revision 1, Tables 11.5-1, 11.5-2, 11.5-4, and 11.5-9. In DCD Tier 2, Revision 3, Section 11.5.2.1 and Table 11.5-9, the applicant stated that the PRMS dynamic instrumentation response ranges are consistent with system designs and qualifications under the provisions of RG 1.97. A review of DCD, Revision 3, Section 7.5, indicated that the instrumentation design requirements are based on Revision 4 of RG 1.97. A review of Revision 4 of the RG indicates that it did not provide criteria for instrumentation variables as do Revisions 2 and 3 of the same guide. In Revision 4, the RG states that the basis and numerical values for instrumentation are to be established in the licensing basis documentation, which is nonexistent at this time, given the endorsement of IEEE Std 497-2002 in the regulatory position of RG 1.97, Revision 4. In addressing conformity with RG 1.97, DCD, Revision 3, Section 7.5.1.3, stated that conformance to these requirements will be addressed during the design process using inputs from various design analyses and human factor engineering. In response to RAI 11.5-8, supplemental RAI 11.5-46, and other RAIs issued against DCD Tier 2, Section 7.5.1, the applicant discussed compliance with RG 1.97 for postaccident radiation monitoring instrumentation described in DCD Tier 2, Section 7.5.1.3, and noted that these aspects will be integrated as part of the human factors engineering process described in DCD Tier 1, Section 3.7. As part of this process, the applicant agreed, in DCD Tier 2, Revision 5, to address design and performance criteria in defining instrumentation response ranges using the provisions of RG 1.97, Revision 4, against the postaccident monitoring variables applicable to radiation monitors installed in building stacks. Section 7.5.1 of this report presents the staff's evaluation of the provisions associated with compliance of Revision 4 of RG 1.97 and associated BTP of Section 7.5 of the SRP. The staff finds the response acceptable and the two RAIs are closed in the context of DCD Tier 2, Revision 9, Section 11.5.

The staff's review of DCD Tier 2, Revision 1, Sections 10.4.2, 11.3, 11.5, 12.3.1, and 12.3.2 found no discussion addressing plant design features to mitigate radiation exposures and doses

to members of the public associated with the production of N-16 and skyshine outside of the turbine building, in the context of 10 CFR 20.1302, 10 CFR 20.1301(e), and 40 CFR Part 190, "Environmental Radiation Protection Standards for Nuclear Power Operations." The staff sent the applicant RAI 11.5-23, which asked for such a discussion. In response, the applicant stated that the related topics were also addressed in the responses to RAI 12.3-5 and 12.3-5 S01. DCD Tier 2, Revision 3, Section 12.2.1.3 discusses the analysis and dose results, with references given for other sections of DCD Tier 2 containing information on the N-16 radiological source term and shielding provided by structures, systems, and components. The applicant proposed to add a requirement in DCD Tier 2, Section 11.5.7, that a COL Licensee consider in its ODCM site-specific conditions and requirements to assess radiation exposures and doses to members of the public located in unrestricted areas, in accordance with the requirements of 10 CFR 20.1301(e) and 10 CFR 20.1302. The staff finds these responses acceptable in the context of DCD Tier 2, Section 11.5. In a letter dated July 19, 2007, the applicant committed to placing this information in DCD Tier 2, Revision 4. Based on the applicant's response, this RAI is resolved. The staff confirmed that this change was included in DCD Tier 2, Revision 4.

The staff's review of DCD Tier 2, Revision 1, 11.5 and SRP Sections 9.3.2 and 11.5, found no discussions on whether the design considered the acceptance criteria and guidance of SRP Section 9.3.2.II on the process sampling system and post-accident sampling system. In RAI 11.5-24, the staff asked the applicant to:

- Address how the applicable requirements of SRP Section 9.3.2.II were met in DCD Tier 2, Sections 11.5.5 and 9.3.2 for gaseous/liquid process and effluent streams;

- Update the text in DCD Tier 2, Sections 11.5.5 and 9.3.2 and DCD Tier 2, Tables 9.3-1 and 11.5-1 to reflect the applicable criteria of SRP Section 9.3.2.II;

- Update the text in DCD Tier 2, Section 11.5.5 by adding internal cross-references to DCD Tier 2, Section 9.3.2; and

- Describe the operational considerations that would be addressed by the COL applicant in DCD Tier 2, Section 11.5.7 and SRP Sections 9.3.2 and 11.5.7.

In the response to RAI 11.5-24, the applicant stated that a new section (11.5.5.9) would be created to address these issues and that it would include a cross-reference to DCD Tier 2, Revision 5, Section 9.3.2 where information can be found on the consideration of station layout and design criteria in selecting locations for sampling from process and effluent streams against specific GDC, regulatory requirements, and regulatory guidance. Also, DCD Tier 2, Section 11.5.5.9 documents the requirements to maintain radiation exposure to workers ALARA and reducing leakages and spills. In a letter dated July 19, 2007, the applicant committed to placing this information in DCD Tier 2, Revision 4. The staff finds the applicant's technical response acceptable. Based on the applicant's response, this RAI is resolved. The staff confirmed that this change was included in DCD Tier 2, Revision 4.

The staff's review indicated that DCD Tier 2, Revision 3, Section 11.5.7 did not commit the COL applicant to conforming to the QA guidance specified in RGs 1.21, 1.33, and 4.15. A COL applicant referencing the ESBWR certified design should describe the QA program for design, fabrication, procurement, construction of structures, and installation of permanent or skid-mounted PRMS subsystems and components in the plant in accordance with its overall QA program. In a global response to RAI 11.5-44, the applicant proposed changes to all related

sections of Chapter 11 on this topic and stated that the applicable QA requirements are described in DCD Tier 2, Chapter 17, Table 17.0-1. As a result, the applicant has revised the text of DCD Tier 2, Section 11.5.6.1 to reference the QA requirements of Chapter 17 for the design, fabrication, procurement, and installation of process and effluent radiation monitoring subsystems in accordance with the COL Licensee's overall QA program. In a letter dated July 23, 2007, the applicant committed to placing this information in DCD Tier 2, Revision 4. The staff finds the applicant's technical response acceptable. Based on the applicant's response, this RAI is resolved. The staff confirmed that this change was included in DCD Tier 2, Revision 4.

In RAI 11.5-46, the staff noted that DCD Tier 2, Revision 3, Section 11.5.7 should commit the COL applicant to establishing operational procedures for the associated post-accident RMS. DCD Tier 2, Revision 3, Sections 11.5.1, 11.5.4, and 11.5.5 describe operational requirements of the post-accident sampling system and the operational range of each PRMS to ensure that they are consistent with the requirements of 10 CFR 50.34(f)(2)(viii), 10 CFR 50.34(f)(2)(xvii), 10 CFR 50.34(f)(2)(xxvii), and 10 CFR 50.34(f)(2)(xxviii) and the guidance of RG 1.97 and NUREG–0737 (TMI-related Item II.F.1). However, DCD, Revision 3, Section 11.5.7, did not commit the COL applicant to establish operational procedures for the associated RMS. The staff asked the applicant to update this section of DCD Tier 2 to add this COL information item. In a letter dated July 23, 2007, the applicant committed to clarifying compliance with the guidance of RG 1.97 and TMI-related action items in DCD Tier 2, Revision 4, Sections 11.5.2, 11.5.3, and 13.5.3.4 without a COL information item. The staff finds the applicant's technical response acceptable given more specific information provided in DCD Tier 2, Revision 4, Sections 2.5, 9.3.2, 12.3 and 12.5. Based on the applicant's response, this RAI is resolved. The staff confirmed that this change was included in DCD Tier 2, Revision 4.

In RAI 11.5-47, the staff noted that in DCD Tier 2, Revision 3, Chapter 11 the applicant identified COL holder items encompassing operational programs including ODCM, PCP, REMP, radiological effluent technical specifications, and SRECs. In accordance with SECY-05-0197, COL applicants should fully describe these operational programs in their COL applications and should propose implementation milestones (license conditions) for staff's review. The staff asked the applicant to revise the DCD to include COL applicant items rather than COL holder items for these operational programs. In a letter dated July 23, 2007, the applicant committed to changing "COL holder items" to "COL applicant items" in DCD Tier 2, Revision 4, Sections 11.5.4 and 11.5.7. DCD Tier 2, Revision 7, Section 13.4 assigns the development of operational programs and implementation milestones as two COL information items (COL 13.4-1-A and 13.4-2-A) being due before fuel loading. The staff finds the applicant's technical response acceptable. Based on the applicant's response, this RAI is resolved. The staff confirmed that this change was included in DCD Tier 2, Revision 4.

In DCD Tier 2, Revision 3, Section 11.5.7 the applicant identified COL Information Items 11.5-1-A through 11.5-5-A. The staff finds the inclusion of these five COL information items acceptable, based on the above discussions.

In addressing 10 CFR 52.47(b)(1), which requires that a design certification application contain the proposed ITAAC, the applicant has included specific ITAAC for the PRMS. These ITAAC are described in DCD Tier 1, Revision 9, Section 2.3.1 and Tables 2.3.1-1 and 2.3.1-2 and Figure 2.3.1-1. The ITAAC address the descriptions and functional arrangements of the PRMS for safety and nonsafety-related subsystems. For safety-related PRMS subsystems, ITAAC are assigned to confirm the source of electrical power, seismic qualifications, instrumentation indications of radiation or radioactivity levels, alarms on exceeding set-point values, alarms on

inoperative conditions, and initiation of protective actions and isolation or termination of plant processes or effluent releases. For nonsafety-related PRMS subsystems, ITAAC are assigned to confirm instrumentation indications of radiation or radioactivity levels, alarms on exceeding set-point values, and alarms on inoperative conditions. The ITAAC refer to DCD Tier 1, Revision 9, Tables 2.10.1-2 and 2.10.3-1 for PRMS subsystems designed for the initiation of isolation or termination of plant effluent releases in demonstrating compliance with 10 CFR Part 20.1301 doses to members of the public or effluent concentration limits in Table 2 (Columns 1 and 2) of Appendix B to 10 CFR Part 20. If the inspections, tests and analyses are performed and the acceptance criteria met, the proposed ITAAC provide reasonable assurance that a plant that incorporates the ESBWR design certification and operates in accordance with the design certification will meet the provisions of the AEA and NRC regulations.

The review of DCD Tier 2, Revision 5, revealed a number of inconsistencies in internal citations and references, incorrect conversions from the International System of Units (système internationale d'unités) to conventional radiological units, and an incomplete list of radiological sampling points. Specifically, in RAI 11.01-1.a to d, the staff asked the applicant to review and resolve these items in Tables 11.5-2 and 11.5-9; to resolve inconsistent references to stack design changes in Sections 11.5.3.2.13 and 7.5.3; and to correct Section 9.2.6.2 and Table 11.5-5, which lacked a line item entry identifying sampling provisions for condensate water that might be present in the CST basin before discharge to the storm drain. In its response dated November 13, 2008, the applicant agreed to make the appropriate corrections and provided proposed changes to be included in DCD Tier 2, Revision 6. The staff finds the proposed changes acceptable, except for that proposed in response to RAI 11.01-1.d.

The staff reviewed GEH's response to RAI 11.0-1, item (d). The staff finds that the intended purpose of Footnote 9 in DCD Tier 2, Table 11.5-5 was acceptable, but that its defined objective was improperly stated. The purpose of the RAI was to ensure that sampling of contaminated condensate water in the retention basin would be accomplished following a tank rupture or overflow. DCD Tier 2, Section 9.2.6 acknowledges the possibility of such an event and states that sampling would be performed to assess whether condensate water held in the retention basin could be released to the storm drain or pumped back to the LWMS depending on radioactivity levels. However, the proposed Footnote 9 refers to sampling following a "rain event." Given the design features of the condensate storage and transfer system and the purpose of the retention basin, reference to a "rain event" is inconsistent with the system's design bases and underlying radiological concerns. In RAI 11.0-1 S01, the staff asked the applicant to remove "rain event" from Footnote 9 as the rationale for sampling and instead indicate that manual sampling will be performed following the observation of water in the retention basin. Also, the applicant should make Footnote 9 consistent in its terminology with other sections of the DCD, as it refers to "CST Containment Dike" while DCD Tier 2, Section 9.2.6 refers to a "retention area" and "retention basin." Accordingly, the staff asked the applicant to revise proposed Footnote 9 to Table 11.5-5 to make it consistent with system descriptions and design bases of DCD Tier 2, Section 9.2.6. In a response dated February 24, 2009, the applicant agreed to remove "rain event" from Footnote 9 and use a consistent nomenclature for the retention area around the condensate storage tank (CST). The staff reviewed the proposed changes to be included in DCD Tier 2, Revision 6 and finds the corrections acceptable. Based on the applicant's response, these RAIs are resolved. The staff confirmed that these changes were included in DCD Tier 2, Revision 6.

In addressing Task Action Plan, Subtask 1 of Item B-67 "Effluent and Process Monitoring Instrumentation," for normal plant operation and AOO effluents, DCD Tier 2, Section 11.5 is consistent with the acceptance criteria and guidance of SRP Section 11.5. The associated TMI-

related items in monitoring radioactive effluents under accident conditions are covered in DCD Tier 2, Sections 7.5.1, 7.5.3, and 9.3.2. The staff's evaluations of these DCD sections are addressed in their respective sections of this report.

In addressing Task Action Plan, Subtask 2 of Item B-67, the radiological impacts at the EAB associated with an OGS leak or component failure is addressed in DCD Tier 2, Section 11.3.7. The assumptions and dose results of the radiological analysis are found to be in compliance with the SRP acceptance criteria and guideline of SRP Section 11.3, BTP 11-5, for systems designed to withstand the effects of hydrogen explosions and earthquakes. The staff finds the results of this analysis acceptable. Section 11.3 of this report addresses this issue and presents the results of the staff's analysis.

In addressing Task Action Plan, Subtask 3 of Item B-67, the radiological impact associated with the failure of a liquid radwaste tank is addressed in DCD Tier 2, Section 15.3.16. The assessment considers the potential impacts of the release of radioactive materials on the nearest potable water supply located in an unrestricted area. The design features include the use of steel liners in cubicles where radwaste storage tanks are located in containing liquid radioactive waste and avoiding releases of radioactive materials in the environment. The staff finds such design features and results of the analysis acceptable. Chapter 15 of this report addresses this issue and presents the results of the staff's analysis.

DCD Tier 2, Section 11.4 addresses Task Action Plan, Subtask 4 of Item B-67 and describes the installation and use of permanently installed radwaste processing subsystems. This approach is consistent with the acceptance criteria and guidance of SRP Section 11.4. The staff finds this acceptable. Section 11.4 of this report addresses this issue and presents the results of the staff's analysis.

11.5.4 Conclusions

Based on the information provided and the COL information items discussed above, the staff concludes that the PRMS (as permanently installed system components in combination with skid-mounted RMS) includes equipment necessary to measure and control releases of radioactive materials in plant process streams and liquid and gaseous effluents; alert the control room of abnormal levels of radioactivity in process streams and liquid and gaseous effluents; provide signals that initiate automatic safety functions, isolate process streams, and terminate effluent discharges if predetermined radioactivity levels or release rates exceed alarm set points; and provide the means to collect samples from process and effluent streams for radiological analysis. Based on this evaluation, the staff finds the PRMS to be in compliance with the requirements of GDC 19, 60, 63, and 64; 10 CFR 50.34a and 50.36a; Appendix I to 10 CFR Part 50; 10 CFR 20.1301 and 20.1302; 10 CFR 50.34(f)(2)(viii), 10 CFR 50.34(f)(2)(xvii), (f)(2)(xxvii), and 50.34(f)(2)(xxviii); and associated guidance in RGs 1.21, 1.45, 1.97, 4.15, and 4.21. This conclusion is based on the following:

- The staff reviewed the provisions proposed in DCD Tier 2 for automatic termination of effluent releases and for control over discharges, in accordance with GDC 60 and 63. Sections 11.2 and 11.3 of this report discuss systems used in controlling releases of radioactive materials from the GWMS exhaust and LWMS discharge line. The PRMS monitors discharges or releases from the reactor and fuel building stack, turbine building stack, and radwaste building stack. The PRMS also monitors exhaust and process streams for the reactor building HVAC exhaust and its subsystems, containment purge exhaust,

turbine building combined ventilation exhaust and its subsystems, radwaste building ventilation exhaust, and fuel building combined ventilation exhaust and its subsystems.

- The staff reviewed the provisions proposed in DCD Tier 2 that are required for plant safety. These PRMS subsystems provide signals and initiate automatic safety functions for the following systems: reactor building HVAC exhaust, refuel handling area HVAC exhaust, control building air intake HVAC, isolation condenser vent exhaust, fuel building general area HVAC, fuel building fuel pool HVAC, and containment purge exhaust. The safety-related portions of the PRMS are classified as safety Class 2, Seismic Category I, based on the QA requirements of Appendix B to 10 CFR Part 50.

- The staff reviewed the provisions proposed in DCD Tier 2, that are required for plant operation. These PRMS subsystems provide signals and initiate automatic functions for the following plant systems: main steam line, off-gas pre-treatment, off-gas post-treatment, charcoal vault ventilation, drywell fission product, reactor component cooling water intersystem leakage, drywell sumps LCW/HCW discharge, liquid radwaste discharge line, and technical support center HVAC air intake.

- The staff reviewed the provisions in DCD Tier 2 for systems to sample and monitor plant effluents in accordance with GDC 64. These systems include instrumentation to monitor and sample radioactivity in contaminated liquid and gaseous process and effluent streams. The staff evaluated the design features provided for process and effluent streams identified in DCD Tier 2, Revision 9, Section 11.5.3, Table 11.5-1 and Tables 11.5-5 through 11.5-8.

- The staff reviewed the provisions for conducting sampling and analytical programs in accordance with RGs 1.21, 1.45, and 4.15, as well as the provisions for sampling and monitoring process and effluent streams during postulated accidents in accordance with RG 1.97. Section 9.3.2 of this report presents the staff's evaluation of the compliance of the related design provisions of the process sampling system.

- The staff reviewed the requirements specified in 10 CFR 50.34(f)(2)(xvii) and 50.34(f)(2)(xxvii) in DCD Tier 2, Sections 11.5, 7.5.2, and 9.3.2, for monitoring gaseous effluents from potential accident release points. DCD Tier 2, Revision 5, Section 9.3.2 commits the COL applicant to develop a postaccident sampling program to monitor the parameters identified in DCD Tier 2, Table 9.3-1. Section 7.5.1 of this report addresses the design features of the postaccident monitoring instrumentation and compliance with RG 1.97 in defining the response range of instrumentation. Chapters 6 and 12 of this report address design features and operational programs in maintaining occupational radiation exposures under NRC limits and ALARA as they relate to 10 CFR 50.34(f)(2)(viii) and 10 CFR 50.34(f)(2)(xxviii).

- The staff reviewed the provisions proposed in DCD Tier 2 that are required for the development and implementation of operational programs. The DCD identifies the implementation of technical specifications/SREC, ODCM, and REMP, as COL Information Item 11.5-4-A. The operational programs include administrative programs. Operational procedures associated with their implementation by the COL Licensee should be consistent with the guidance of GL 89-001 and NUREG–1302 for BWR plants; NUREG–0133; RG 1.21, Revision 1, 1.33 Revision 2, 4.1 Revision 1, "Programs for Monitoring Radioactivity in the Environs of Nuclear Power Plants," and 4.15, guidance from RC Generic Letter 79-65, "Radiological Environmental Monitoring Program Requirements –Branch Technical Position

[Concerning Direct Radiation Measurements] Revision 1 (Generic Letter 79-65)," November 27, 1979," and RIS 2008-03.

- The design of the PRMS, operating in conjunction with the LWMS, GWMS, and SWMS to control and monitor radioactive effluent releases into the environment, meets the dose requirements of 10 CFR 20.1301 and 10 CFR 20.1302 by ensuring that annual average concentrations of radioactive materials in liquid and gaseous effluents released into unrestricted areas will not exceed the limits specified in Appendix B to 10 CFR Part 20 (Table 2, Columns 1 and 2). DCD Tier 2, Revision 9, Section 12.2.2 presents this information and Section 12.3.3.2 of this report presents the staff's evaluation.

- In conjunction with the operation of the LWMS, GWMS, and SWMS, the design of the PRMS complies with the requirements of Sections II.A, II.B, and II.C of Appendix I to 10 CFR Part 50 in ensuring that offsite individual doses resulting from liquid and gaseous effluent releases will be ALARA, will not exceed dose criteria, and will comply with the requirements of 10 CFR 50.34a and 50.36a. DCD Tier 2, Revision 9, Section 12.2.2 addresses the requirements associated with Section II.D of Appendix I on the conduct of cost-benefit analyses and ALARA in assessing the augmentation of effluent treatment systems. DCD Tier 2, Section 12.2.4 addresses these requirements under two COL information items (12.2-2-A and 12.2-3-A) and Section 12.3.3.2 of this report presents the staff's evaluation.

- The staff reviewed the applicant's QA provisions for the PRMS, the quality group classifications used for PRMS components, and the seismic design applied to structures housing these systems. The design of the systems and the structures housing these systems meets the guidance of RG 1.143, as described in DCD Tier 2, Revision 9 Sections 3.2 and 3.8. Sections 3.2 and 3.8 of this report present the staff's evaluation.

- The applicant demonstrates compliance with the requirements of GDC 61 by meeting the guidelines of RGs 1.143 and 4.21. This commitment also fulfills the requirements of 10 CFR 20.1406 to minimize the contamination of the facility and the generation of radioactive waste and the guidance of IE BL 80-10 and RG 4.21 to avoid cross-contamination of non-radioactive systems and un-monitored and uncontrolled radioactive releases into the environment, and to minimize the contamination of the facility.

- In addressing the requirements of 10 CFR 52.47(b)(1), which requires that a design certification application contain the proposed ITAAC, the applicant has included specific ITAAC for the PRMS. The ITAAC address the descriptions and functional arrangements of the PRMS for safety and nonsafety-related subsystems. ITAAC are assigned to confirm the source of electrical power, seismic qualifications, instrumentation indications of radiation or radioactivity levels, alarms on exceeding set-point values, alarms on inoperative conditions, and initiation of protective actions and isolation or termination of plant processes or effluent releases in demonstrating compliance with 10 CFR 20.1301 doses to members of the public or effluent concentration limits in Table 2 (Columns 1 and 2) of Appendix B to 10 CFR Part 20. If the inspections, tests and analyses are performed and the COL acceptance criteria met, the proposed ITAAC provide reasonable assurance that a plant that incorporates the ESBWR design certification and operates in accordance with the design certification will meet the provisions of the AEA and NRC regulations.

12.0 RADIATION PROTECTION

12.1 Introduction

The economic simplified boiling-water reactor (ESBWR) design control document (DCD), Tier 2, Revision 9, Chapter 12, “Radiation Protection,” describes the kinds and quantities of radioactive materials expected to be produced in the operation of the ESBWR reactor and the means for controlling and limiting radiation exposures to within the requirements in Title 10 of the *Code of Federal Regulations* (10 CFR) Part 20, “Standards for Protection against Radiation.” The ESBWR reactor design incorporates radiation protection measures intended to ensure that internal and external radiation exposures to station personnel, contractors, and the general population resulting from plant conditions, including anticipated operational occurrences (AOOs), will be within regulatory criteria and will be as low as reasonably achievable (ALARA).

The U.S. Nuclear Regulatory Commission (the NRC or staff) evaluated the information in Chapter 12 of DCD Tier 2, Revision 9, against the criteria in Chapter 12 of NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants (LWR Edition),” March 2007 (hereafter referred to as the SRP). Compliance with these criteria provides assurance that doses to workers will be maintained within the occupational dose limits of 10 CFR Part 20. These occupational dose limits, applicable to workers at NRC-licensed facilities, restrict the sum of the external whole-body dose (deep-dose equivalent) and the committed effective equivalent doses resulting from radioactive material deposited inside the body (deposited through injection, absorption, ingestion, or inhalation) to 50 millisievert (mSv) (5 roentgen equivalent in man [rem]) per year with a provision (i.e., by planned special exposure) to extend this dose to 100 mSv (10 rem) per year with a lifetime dose limit of 250 mSv (25 rem) resulting from planned special exposures.

Compliance with the SRP acceptance criteria also provides assurance that radiation doses resulting from exposure to radioactive sources both outside and inside the body can be maintained well within the limits of 10 CFR Part 20 and ALARA. The balancing of internal and external exposure necessary to ensure that the sum of the doses is ALARA is an operational concern. An applicant seeking a combined license (COL) must address these operational concerns, as well as programmatic radiation protection concerns.

The staff has received sufficient information from GE-Hitachi Nuclear Energy (GEH or the applicant) to conclude that the radiation protection measures incorporated in the ESBWR reactor design offer reasonable assurance that occupational doses during all plant operations will be maintained ALARA and will be within the limits of 10 CFR Part 20. The following sections present the bases for the staff’s conclusions.

12.2 Ensuring That Occupational Radiation Doses Are As Low As Reasonably Achievable

12.2.1 Regulatory Criteria

The applicable criteria and guidance include the following:

- 10 CFR 20.1101 and 10 CFR 20.1704

- 10 CFR 50.34(b)(3), as it relates to the kinds and quantities of radioactive materials produced and the means for controlling and limiting radiation exposures within the limits of 10 CFR Part 20

- Regulatory Guide (RG) 1.8, Revision 3, “Qualification and Training of Personnel for Nuclear Power Plants,” May 2000.

- RG 1.206, “Combined License Applications for Nuclear Power Plants (LWR Edition),” June 2007.

- RG 8.8, Revision 3, “Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations Will Be as Low as Is Reasonably Achievable,” June 1978.

- RG 8.10, Revision 1-R, “Operating Philosophy for Maintaining Occupational Radiation Exposures as Low as Is Reasonably Achievable,” May 1977.

The staff compared the SRP (Section 12.1, 1981 version) used during its initial review of the DCD with the 2007 version of the SRP and incorporated any additional guidance from the 2007 SRP during the staff’s subsequent review of Section 12.1 of the DCD. Therefore, the staff concludes that the version of the SRP used, in combination with the staff’s additional review, is appropriate for this review.

12.2.2 Summary of Technical Information

In addition to providing radiation exposure limits for workers and members of the public, 10 CFR 20.1101(b) requires that, to the extent practical, procedures and engineering controls based on sound radiation protection principles be employed to achieve occupational doses and doses to the public that are ALARA. In addition, 10 CFR 20.1704(a) requires that the intake of airborne radioactive materials be consistent with maintaining total effective dose equivalent ALARA. RG 8.8 provides specific guidance and criteria on designing, constructing, and operating a nuclear power plant to meet this regulatory requirement. The scope of this design certification does not include programmatic and policy considerations associated with plant operations that are needed to ensure that radiation doses will be ALARA (as discussed in RGs 1.8, 8.8, and 8.10).

The applicant has identified the following COL information items (see Section 12.2.3.1 below) to ensure that license applicants referencing the ESBWR design will address these issues:

- (COL 12.1-1-A) Regulatory Guide 8.10—The COL applicant will demonstrate compliance with RG 8.10.

- (COL 12.1-2-A) Regulatory Guide 1.8—The COL applicant will demonstrate compliance with RG 1.8.

- (COL 12.1-3-A) Operational Considerations—The COL applicant will provide the criteria and conditions under which it will implement various operating procedures and techniques to ensure that occupational radiation exposures remain ALARA using the guidance of NUREG–1736, “Consolidated Guidance: 10 CFR Part 20—Standards for Protection Against Radiation,” to the level of detail provided in RG 1.206.

- (COL 12.1-4-A) Regulatory Guide 8.8—The COL applicant will demonstrate that its policy considerations regarding plant operations (i.e., establishment of a program to maintain occupational radiation exposures ALARA, establishment of a radiation control program to plan and supervise jobs performed in radiation areas, and maintenance of adequate radiation protection facilities, instrumentation, and equipment) are in accordance with the applicable guidance contained in RG 8.8.

12.2.3 Staff Evaluation

The staff reviewed the information in DCD Tier 2, Revision 9, Section 12.1, to assess adherence to the guidelines in RG 1.206, as well as to the criteria in Section 12.1 of the SRP regarding the radiation protection aspects of the ESBWR reactor design. Specifically, the staff reviewed Section 12.1 of DCD Tier 2, Revision 9, to ensure that the applicant had either committed to adhere to the criteria of the RGs and staff positions referenced in Section 12.1 of the SRP or had provided acceptable alternatives.

12.2.3.1 *Policy Considerations*

In DCD Tier 2, Revision 9, Section 12.1, the applicant described the design, construction, and operational policies that ensure that ALARA considerations are factored into each stage of the ESBWR design process. The applicant has committed to ensure that the ESBWR plant will be designed and constructed in a manner consistent with the guidelines of RG 8.8. In particular, DCD Tier 2, Revision 9, Section 12.1.1.1, states that the ALARA philosophy was applied during the initial design of the ESBWR. GEH performed a detailed review of the plant design for ALARA considerations and modified the design as necessary during the design phase. Experience related to ALARA performance gained from operating plants was continuously integrated during the design phase of the ESBWR standard plant. This ALARA policy is consistent with the guidelines of RG 8.8 and is therefore acceptable.

The requirements of 10 CFR Part 20 specify that all licensees must develop, document, and implement a radiation protection program. Specifically, this program must encompass the ALARA concept and provide for maintaining radiation doses and intakes of radioactive materials ALARA. The operational ALARA policy forms the basis for the operating station's ALARA manual. However, the scope of this design certification review does not include the detailed policy considerations regarding overall plant operations and implementation of such a radiation protection program.

To maintain doses to plant personnel ALARA, the applicant stated, in DCD Tier 2, Revision 9, Section 12.1.4, that the COL applicant will present, consistent with the criteria in RG 1.206, the operating procedures and techniques it will implement to ensure that occupational radiation doses are maintained ALARA (COL Information Item 12.1-3-A). In addition, a COL applicant referencing the ESBWR certified design will demonstrate how its operational ALARA policy conforms to the requirements of 10 CFR Part 20 and the recommendations of Revision 3 of RG 1.8 (COL Information Item 12.1-2-A), RG 8.8 (COL Information Item 12.1-4-A), and Revision 1-R of RG 8.10 (COL Information Item 12.1-1-A).

12.2.3.2 *Design Considerations*

The plant radiation protection design should ensure that individual doses and collective total effective dose equivalent (person-rem) to plant workers and to members of the public are maintained ALARA and that individual doses are maintained within the limits of 10 CFR Part 20.

DCD Tier 2, Revision 9, Section 12.1.2, describes the objectives for the general design and shielding. Specifically, Section 12.1.2 states that the basic management philosophy guiding the ESBWR design is to ensure that exposures are ALARA by designing structures, systems, and components (SSCs) to achieve the following objectives:

- Minimize the necessity for and the amount of time spent in radiation areas.

- Minimize radiation levels in routinely occupied plant areas in the vicinity of plant equipment expected to require personnel attention.

The staff finds that these design objectives are consistent with the guidelines in RG 8.8.

Section 12.1.2 of DCD Tier 2, Revision 9, describes several design features that satisfy the objectives of the plant's radiation protection program. Examples of these features include the following:

- To the extent practicable, materials in contact with the reactor coolant system (RCS) have low concentrations of cobalt and nickel. This reduces the amounts of cobalt-60 and cobalt-58 introduced in the RCS. (Cobalt-60 and cobalt-58 are the major sources of radiation exposure during shutdown, maintenance, and inspection activities at light-water reactors.)

- Central control panels to permit remote operation of all safety-related instrumentation and controls (e.g., the control rod drive [CRD] maintenance control panel and the remote shutdown control panels) are located in separate, shielded rooms in the lowest radiation zone possible.

- Adequate spacing and laydown areas facilitate access for maintenance and inspection. Separate low background rooms are provided for CRD and hydraulic control unit maintenance.

- The time spent in radiation areas will be minimized by providing ease of access to equipment, instruments, and sampling stations that require routine maintenance, calibration, operation, or inspection. In addition, where practicable, components are designed for ease of disassembly for replacement or removal to a lower radiation area for repair or servicing.

- Radioactive systems are separated from nonradioactive systems, and high-radiation sources are located in separate shielded cubicles.

- Equipment requiring periodic service or maintenance (e.g., pumps, valves, and control panels) is separated from more radioactive sources (i.e., tanks and piping).

- Valves located in high-radiation areas are equipped with reach rods or motor operators to minimize operator exposure.

- Equipment and piping are designed to minimize the accumulation of radioactive materials.

- Drains are located at low points of systems and components.

- Piping is seamless, and the number of fittings is minimized, thereby reducing the radiation accumulation at seams and welds.

- Use of flushing connections minimizes the buildup of crud in system components.

- Adequate space and means are provided for the use of movable radiation shielding to provide personnel protection from radioactive sources, when required.

These design considerations incorporate the basic management philosophy guiding the ESBWR design effort and are consistent with the guidelines in RG 8.8. Therefore, the staff finds them acceptable.

In addition to the features described above, the ESBWR reactor design incorporates the following features that represent improvements over many currently operating plants:

- The ESBWR design uses natural circulation, resulting from thermal convective forces in the reactor vessel, to circulate coolant through the core. This design eliminates the need for reactor water recirculation system piping and associated active pumps and valves, which historically have been significant sources of personnel exposure in current boiling-water reactor (BWR) designs.

- Material selection for the ESBWR design includes minimizing the use of cobalt-bearing components in the reactor water systems. The ESBWR main condenser has titanium or stainless steel tubes and tube sheets to minimize service water in-leakage and the resultant activation of reactor water contaminants.

The second bullet of DCD Tier 2, Revision 1, Section 12.1.2.3.2, stated that the design of the reactor pressure vessel (RPV) shield wall in the upper drywell permitted continued operation in the upper drywell during refueling and provided shielding in the case of a refueling accident. In request for additional information (RAI) 12.2-19 and follow-up supplements to this RAI, the staff asked the applicant to verify that the shielding around the reactor vessel is sufficient to allow personnel access to the upper drywell during fuel-handling operations and in the event of a refueling accident in which an extended burnup fuel assembly is dropped onto the vessel flange/refueling pool seal diaphragm.

In response RAI 12.2-19 S02, GEH stated that its initial estimates of the dose rates in the upper drywell area from a postulated refueling accident (provided in the GEH initial response to RAI 12.2-19) were in error and that the revised estimated dose rates in the upper drywell from a dropped fuel assembly were a factor of 50 percent higher than initially estimated. GEH stated that this error resulted from the use of the incorrect energy groups when converting fluence outputs from the ORIGEN computer code to be compatible with the MCNPX Monte Carlo computer code input requirements. In RAI 12.2-19 S03, the staff requested verification that, because of the applicant’s revised estimated dose rates in the upper drywell, personnel working in the upper drywell would be able to evacuate this area in the event of a refueling accident. The staff tracked RAI 12.2-19 as an open item in the SER with open items.

In response to RAI 12.2-19 S03, GEH modified the DCD (in DCD Tier 2, Revision 6) to delete the aforementioned bullet (found in DCD Tier 2, Revision 1, Section 12.1.2.3.2), which stated that continued operation in the upper drywell would be permitted during refueling operations. In addition, in DCD Tier 2, Revision 6, GEH modified the radiation zone designation for Room 1570 in the upper drywell from Zone E (less than 1 mSv per hour [mSv/h] [100 millirem per hour [mrem/h]]) to Zone G (less than 100 mSv/h [10 roentgens per hour [R/h]]) during spent fuel transfer activities. Because the application no longer proposes to permit continued operation in the upper drywell during refueling operations, but, instead, proposes to implement

controlled access to this area due the possibility of transient dose conditions, the staff's concern regarding the evaluation of potential high doses to personnel in the upper drywell resulting from a postulated fuel assembly drop accident is resolved. In addition, the staff finds the revised radiation zone designations to be acceptable because these zone designations reflect the resulting increase in estimated dose rates in this area due to the use of the revised source term. Based on the applicant's response, RAI 12.2-19 is resolved.

The design features described in Section 12.1.2 of DCD Tier 2, Revision 9, are intended to minimize personnel exposures and comply with the guidelines of RG 8.8. As such, these design features should help maintain individual doses and total person-rem doses to plant workers and to members of the public ALARA, while maintaining individual doses within the limits of 10 CFR Part 20. The staff therefore finds these design features to be acceptable for that purpose.

12.2.3.3 *Operational Considerations*

The scope of this design certification review does not include operational considerations regarding the implementation of a radiation protection program. Section 12 of the SRP lists the following regulatory guides that pertain to DCD Tier 2, Chapter 12:

- RG 8.2, "Guide for Administrative Practices in Radiation Monitoring," February 1973.
- RG 8.7, Revision 2, "Instructions for Record Keeping and Recording Occupational Radiation Exposure Data," November 2005.
- RG 8.9, Revision 1, "Acceptable Concepts, Models, Equations, and Assumptions for a Bioassay Program," July 1993.
- RG 8.13, Revision 3, "Instruction Concerning Prenatal Radiation Exposure," June 1999.
- RG 8.15, Revision 1, "Acceptable Programs for Respiratory Protection," October 1999.
- RG 8.20, Revision 1, "Applications of Bioassay for I-125 and I-131," September 1979.
- RG 8.25, Revision 1, "Air Sampling in the Work Place," June 1992.
- RG 8.26, "Applications of Bioassay for Fission and Activation Products," September 1980.
- RG 8.27, "Radiation Protection Training for Personnel at Light-Water Cooled Nuclear Power Plants," March 1981.
- RG 8.28, "Audible-Alarm Dosimeters," August 1981.
- RG 8.29, Revision 1, "Instructions Concerning Risks from Occupational Radiation Exposure," February 1996.
- RG 8.34, "Monitoring Criteria and Methods to Calculate Occupational Radiation Doses," July 1992.
- RG 8.35, "Planned Special Exposures," June 1992.

- RG 8.36, “Radiation Dose to the Embryo/Fetus,” July 1992.

- RG 8.38, Revision 1, “Control of Access to High and Very High Radiation Areas in Nuclear Power Plants,” May 2006.

Addressing the above RGs is outside the scope of this design certification review. In DCD Tier 2, Revision 9, Section 12.1.3, the applicant stated that the COL applicant will address operational considerations of the SRP, consistent with the level of detail provided in RG 1.206, and describe how it will comply with the recommendations of (or provide acceptable alternatives to) the preceding regulatory guides (COL Information Item 12.1-3-A).

12.2.4 Conclusions

The design features described by the applicant are intended to maintain individual doses and total person-rem doses to plant workers and to members of the public ALARA, while maintaining individual doses within the limits of 10 CFR Part 20. Therefore, the staff concludes that the ESBWR design features meet the criteria of Section 12.1 of the SRP.

The COL applicant will address the policy and operational considerations for the ESBWR. The staff has determined that the COL information items described in this section are complete and adequately describe the actions necessary for the COL applicant. The staff finds it acceptable to defer the discussion of the material addressed by COL Information Items 12.1-1-A, 12.1-2-A, 12.1-3-A, and 12.1-4-A until the COL review. The staff will determine compliance with the requirements of 10 CFR Part 20 in these areas at that time. The staff, therefore, finds the material contained in Section 12.1 of DCD Tier 2, Revision 9, acceptable.

12.3 Radiation Sources

12.3.1 Regulatory Criteria

The applicable regulatory criteria and guidance include the following:

- 10 CFR 20.1301 and 10 CFR 20.1302

- 10 CFR 50.34a

- 10 CFR 50.34(f)(2)(vii), as it relates to the conduct of radiation and shielding design reviews of spaces around systems that may contain accident source term radioactive materials

- 10 CFR Part 20, Appendix B, “Annual Limits on Intake (ALIs) and Derived Air Concentrations (DACs) of Radionuclides for Occupational Exposure; Effluent Concentrations; Concentrations for Release to Sewerage.”

- 10 CFR Part 50, “Domestic Licensing of Production and Utilization Facilities,” Appendix I, “Numerical Guides for Design Objectives and Limiting Conditions for Operation to Meet the Criterion ‘As Low as Is Reasonably Achievable’ for Radioactive Material in Light-Water-Cooled Nuclear Power Reactor Effluents.”

- 10 CFR Part 50, Appendix A, “General Design Criteria for Nuclear Power Plants,” General Design Criterion (GDC) 19, “Control room.”

- 10 CFR Part 50, Appendix A, GDC 60, “Control of releases of radioactive materials to the environment.”

- 10 CFR Part 50, Appendix A, GDC 61, “Fuel storage and handling and radioactivity control.”

- RG 1.70, Revision 3 “Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants (LWR Edition),” November 1978.

- NUREG–0016, Revision 1 “Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Boiling Water Reactors (BWR),“ (BWR-GALE Code), January 1979.

- NUREG–1465, “Accident Source Terms for Light-Water Nuclear Power Plants,” February 1995.

The staff compared the SRP (Section 12.2, 1981 version) used during its initial review of the DCD with the 2007 version of the SRP and incorporated any additional guidance from the 2007 SRP during the staff’s subsequent review of Section 12.2 of the DCD. Therefore, the staff concludes that the version of the SRP used, in combination with the staff’s additional review, is appropriate for this review.

12.3.2 Summary of Technical Information

The applicant will use the contained source terms described in DCD Tier 2, Revision 9, as the basis for the radiation design calculations (shielding and equipment qualification) and personnel dose assessment. The applicant will use the airborne radioactive source terms in DCD Tier 2, Revision 9, for the design of ventilation systems and for assessing personnel dose. The staff reviewed the source terms in DCD Tier 2, Revision 9, to ensure that the applicant had either committed to follow the guidelines of the RGs and staff positions in Section 12.2 of the SRP or provided acceptable alternatives. Based on the staff’s review, the staff has concluded that the design meets the relevant requirements of 10 CFR Part 20 and GDC 61.

The applicant has identified the following COL information items to ensure that license applicants referencing the ESBWR design will address these issues:

- (COL 12.2-2-A) Airborne Effluents and Doses—The COL applicant is responsible for ensuring that offsite dose (using site-specific parameters) resulting from radioactive airborne effluents complies with the regulatory dose limits in Sections II.B and II.C of Appendix I to 10 CFR Part 50. In addition, the COL applicant is responsible for complying with Section II.D of Appendix I to 10 CFR Part 50; the airborne effluent concentration limits of Appendix B to 10 CFR Part 20 (Table 2, Column 1); and the dose limits of 10 CFR 20.1301 and 10 CFR 20.1302 to members of the public.

- (COL 12.2-3-A) Liquid Effluents and Doses—The COL applicant is responsible for ensuring that offsite dose (using site-specific parameters) resulting from radioactive liquid effluents complies with the regulatory dose limits in Section II.A of Appendix I to 10 CFR Part 50. In addition, the COL applicant is responsible for complying with Section II.D of Appendix I to 10 CFR Part 50; the liquid effluent concentration limits of Appendix B to 10 CFR Part 20 (Table 2, Column 2); and the dose limits of 10 CFR 20.1301 and 10 CFR 20.1302 to members of the public.

- (COL 12.2-4-A) Other Contained Sources—The COL applicant will address any additional contained radiation sources (including sources for instrumentation and radiography) not identified in DCD Tier 2, Revision 7, Section 12.2.1.5.

12.3.3 Staff Evaluation

The staff reviewed the descriptions of the radiation sources given in DCD Tier 2, Revision 9, Chapter 11 and DCD Tier 2, Revision 9, Section 12.2, to assess their completeness as compared to the guidelines in RG 1.70 and the criteria in Section 12.2 of the SRP.

12.3.3.1 *Contained Sources*

In DCD Tier 2, Revision 9, Section 12.2.1, the applicant described the shielding design source terms, including location, and all pertinent quantitative source parameters during normal full-power operation, shutdown, and design-basis accident events. The noble gas source terms are consistent with a BWR operating offgas rate of 100,000 microcuries per second (μCi/s) after a 30-minute delay. The source terms associated with systems and components carrying radioactively contaminated fluids were calculated consistent with the guidance in American National Standards Institute/American Nuclear Society (ANSI/ANS)-18.1-1999, "Radioactive Source Term for Normal Operation of Light Water Reactors." Filters and ion exchange beds in such systems were assumed to contain their maximum radioactivity before filter backwash or resin exchange.

The activation product, nitrogen–16, with its short half-life and energetic gamma emissions, is the predominant radionuclide during plant operations. Since the ESBWR design does not have reactor coolant recirculation loops, nitrogen–16 is somewhat less of a consideration for primary containment shielding design. However, during power operation of the ESBWR, nitrogen–16 activity is a factor in the radiation sources for the components of the steam and condensate systems located outside of primary containment. The fraction of nitrogen–16 produced in the reactor core that is released into steam depends on reactor water chemistry. Hydrogen Injection into the reactor coolant to minimize the potential for stress corrosion of piping and components in contact with the reactor coolant results in significant increases in the concentration of nitrogen–16 in BWR steam. Reducing the amount of hydrogen injection necessary, by pretreating the reactor system with noble metals, mitigates the nitrogen–16 increase. The nitrogen–16 source term used in the ESBWR design considers both hydrogen injection and noble metal treatment of the reactor system. The applicant used this elevated nitrogen–16 source term, which is six times the concentration of steam leaving the reactor vessel specified in ANSI/ANS-18.1-1999, to calculate the annual skyshine contribution from nitrogen-16 at two typical site boundary distances. In both cases, the resulting annual dose from the nitrogen–16 skyshine resulting from operation of the ESBWR is a small fraction of the 10 CFR 20.1301(a) and 10 CFR 20.1301(e) dose limits.

The applicant used the design-basis source term values for the various radionuclides in determining the shielding design necessary to obtain the desired plant area radiation levels for the ESBWR. In arriving at the design-basis corrosion product activity levels for the ESBWR, the applicant used a set of values that are reasonably conservative relative to current operating plant experience.

In accordance with the criteria in Section 12.2 of the SRP, Section 12.2.2 of DCD Tier 2, Revision 9, describes the large contained sources of radiation used as the basis for designing the radiation protection program and completing shield design calculations. These sources

include the reactor core; the reactor water cleanup/shutdown cooling system; spent fuel and the fuel and auxiliary pools cooling system; the main steam and feedwater lines; the liquid, gaseous, and solid radwaste systems; and other miscellaneous sources. For each of these contained sources, the applicant provided either the source strength by energy group or the associated maximum activity levels listed by isotope. The DCD provides system layouts within rooms or cubicles, as well as information about the type and size of components in these systems.

In RAI 12.3-8, the staff asked GEH to clarify the meaning of the "before and after" dose rates listed in DCD Tier 2, Revision 3, Table 12.2-5. In DCD Tier 2, Revision 4, the applicant addressed the staff's concerns by modifying this table to indicate that the dose rates shown reflected the dose rates from the upper CRD components both before and after cleaning of these components. Furthermore, GEH stated that these components would be cleaned when removed for maintenance or repair. This response clarified the meaning of the information contained in this table and the staff, therefore, finds the response acceptable. RAI 12.3-8 was being tracked as an open item in the SER with open items. Based on the applicant's response, RAI 12.3-8 is resolved.

Section 12.2 of the SRP also states that this section of the DCD should describe any radiation sources containing byproduct, source, and special nuclear materials. However, DCD Tier 2, Revision 4, Section 12.2.1, did not describe radiation sources (such as calibration sources) needed to construct and operate an ESBWR plant. The absence of this information was the basis for RAI 12.3-9. After reviewing the applicant's response to RAI 12.3-9, the staff issued RAI 12.3-9 S01. This supplemental RAI stated that, if the COL applicant were to provide any sources containing byproduct, source, and special nuclear materials, then GEH must identify this as a COL information item and indicate that the COL applicant is responsible for identifying these additional sources and describing any features implemented to minimize the dose from these sources. In response, GEH modified DCD Tier 2, Revision 5, Section 12.2.1.5, to add a new COL information item to address the staff's concerns. This new COL information item, COL Information Item 12.2-4-A, states that "the COL applicant will address any additional contained radiation sources (including sources for instrumentation and radiography) not identified in Section 12.2.1." Because the addition of this COL information item ensures that any radiation sources containing byproduct, source, or special nuclear materials will either be described in the DCD or by the COL applicant, as specified in Section 12.2 of the SRP, the staff finds the response to RAI 12.3-9 acceptable. RAI 12.3-9 was being tracked as an open item in the SER with open items. Based on the applicant's response, RAI 12.3-9 is resolved.

The ESBWR core activity release model for a core melt accident is based on the source term model from NUREG–1465. NUREG–1465 uses updated information on fission product releases to provide a revised source term which is more realistic than the 1962 Technical Information Document (TID) -14844 source term.

In SECY-94-302, the staff stated that the revised source term as given in NUREG–1465 is appropriate for use in the licensing review of evolutionary and passive LWR designs. 10 CFR 50.34(f)(2)(vii) states that the applicant shall perform radiation and shielding design reviews of spaces around systems that may, as a result of an accident, contain accident source term radioactive materials, to ensure that these areas can be adequately accessed following an accident. In reviewing DCD Tier 2, Section 12.3, the staff was unable to ascertain what source term assumptions the applicant had used to develop the postaccident radiation zone maps in DCD Tier 2, Revision 1, Section 12.3. In order to obtain this information, the staff issued RAI 12.3-10.

In RAI 12.3-10, the staff asked the applicant to verify that it used the source term assumptions in NUREG–1465 to determine the in-plant postaccident source terms and to provide the source term assumptions it used to determine the dose rates indicated on the postaccident radiation zone maps in DCD Tier 2, Revision 1, Section 12.3. In response, GEH described the source term assumptions used and calculated the postaccident dose rates, as well as worker doses incurred during vital area access and activities following an accident, using the resulting source strengths. In response to RAI 12.4-31, the applicant, in Revision 5 of the DCD, revised Figures 12.3-43 through 12.3-51, and verified that the postaccident dose rates shown in these figures incorporate the source term assumptions in NUREG–1465. The staff finds GEH's response to be acceptable because GEH verified that it had used the source term assumptions described in NUREG–1465 in determining the in-plant postaccident source terms, and the staff agrees that GEH provided acceptable source term assumptions used to determine the dose rates indicated on the postaccident radiation zone maps in the DCD. RAI 12.3-10 was being tracked as an open item in the SER with open items. Based on the applicant's response, RAI 12.3-10 is resolved.

Based on its review of the material contained in Section 12.2.1 of DCD Tier 2, Revision 9, the staff finds that the applicant's description of contained sources complies with the applicable requirements of 10 CFR Parts 20 and 50. The staff, therefore, finds the information contained in this section to be acceptable.

12.3.3.2 *Airborne and Liquid Effluent Source Terms and Doses*

The staff reviewed DCD Tier 2, Revision 9, Section 12.2.2, in accordance with the guidance and acceptance criteria provided in SRP Sections 11.2 and 11.3. The staff's evaluation addressed compliance with the requirements of 10 CFR 20.1301 and 10 CFR 20.1302 and the design objectives of Appendix I to 10 CFR Part 50 under Sections II.A, II.B, and II.C. Compliance with Section II.D of Appendix I to 10 CFR Part 50 is left to the COL applicant in evaluating the cost-effectiveness of liquid and gaseous effluent treatment systems.

In reviewing DCD Tier 2, Revision 3, the staff could not confirm that the gaseous and liquid effluent radiological source terms, methodology, and assumptions used in estimating doses to members of the public, as well as gaseous and liquid effluent concentrations in unrestricted areas, were consistent with the guidance provided in SRP Sections 11.2 and 11.3 and associated regulatory guidance.

The staff asked the applicant to provide additional information addressing the basis of the radiological source terms and associated doses to members of the public. The applicant responded to these RAIs, and the staff's evaluations of these responses are discussed below. Sections 11.2 and 11.3 of this safety evaluation report (SER), respectively, present the staff's evaluation of whether the designs of the liquid waste management system (LWMS) and gaseous waste management system (GWMS) are acceptable and meet the requirements of 10 CFR 20.1301 and 10 CFR 20.1302 and the design objectives in Appendix I to 10 CFR Part 50.

12.3.3.2.1 Airborne Effluent Releases

In reviewing DCD Tier 2, Revision 1, the staff could not confirm that the gaseous effluent radiological source term, methodology, and assumptions used in estimating doses to members of the public, as well as gaseous effluent concentrations in unrestricted areas, were consistent with the guidance in SRP Section 11.3 and associated regulatory guidance. The staff's

evaluation addressed compliance with the requirements of 10 CFR 20.1301 and 10 CFR 20.1302 and the design objectives of Appendix I to 10 CFR Part 50 under Sections II.B and II.C. Section 11.3 of this report presents the staff's review of the GWMS, as it relates to the design requirements of 10 CFR 50.34a and GDC 60 and 61.

In reviewing DCD Tier 2, Revision 3, Chapter 12, the staff found that some information remained insufficient to determine the acceptability of the applicant's analysis and results. The staff, therefore, issued RAI 12.2-9 and follow-up supplements to this RAI requesting additional information. The applicant responded to this RAI and its supplements, and the staff's evaluations of these responses are discussed below.

In RAI 12.2-9 (with its two supplements), the staff was unable to duplicate the estimates of annual airborne activity releases presented in DCD Tier 2, Revision 1, Chapter 12, Table 12.2-16, using the information presented in DCD Tables 9.4-1, 10.4-2, 11.1-1, 11.3-1, and 12.2-15, including information provided by the applicant in response to RAI 11.1-3. The staff asked the applicant to address these issues and provide information describing all input parameters used in the BWR-GALE code (NUREG–0016). In DCD Tier 2, Revision 3, Section 12.2.2.1 and Table 12.2-15, the applicant submitted an updated source term for all gaseous effluent releases.

In response, the applicant provided new information used in deriving the estimates of total airborne radioactivity releases. DCD Tier 2, Revision 3, Chapter 12, Table 12.2-16, lists these estimates. The new information presents models, equations, and values for specific parameters, either given in the new information or extracted from NUREG-0016. The staff independently confirmed the approach and most results, except in a few instances in which specific results could not be duplicated or additional clarifications were needed because of specific assumptions or values used in the calculations. In RAI 12.2-9 S02, the staff sought further clarification to resolve outstanding issues regarding adjustment factors for gaseous effluents (apparent data mismatches in power ratings, system liquid mass); clarification of data, such as equation symbols and steam mass flow rates, for the various calculational equations for source term and release rates; noble gas delay times; and removal efficiencies. The staff tracked this RAI as an open item in the SER with open items. In response to this RAI, the applicant provided additional information, clarification, or correction for the various topics and appended a portion of its response as appendices, which the staff confirmed to be Appendices 12A and 12B to DCD Tier 2, Revision 5, Chapter 12. The staff finds that the clarifications in Appendices 12A and 12B addressed the treatment of airborne radioactive materials and airborne radioactivity release calculations, consistent with the guidance in NUREG–0016. The staff's independent assessments confirmed that the assumptions and the stated resulted were reasonable, and the staff therefore finds the applicant's responses to be acceptable. Based on the applicant's response, RAI 12.2-9 is resolved.

In DCD Tier 2, Revision 3, Section 12.2.4.2, the COL information item did not provide sufficient information to demonstrate compliance with NRC regulations for airborne effluents. In addition to demonstrating compliance with the dose objectives of Sections II.B and II.C of Appendix I to 10 CFR Part 50, the staff determined that a COL applicant will also need to demonstrate compliance with Section II.D of Appendix I to 10 CFR Part 50; the airborne effluent concentration limits of Appendix B (Table 2, Column 1) to 10 CFR Part 20; and the dose limits of 10 CFR 20.1301 and 10 CFR 20.1302 to members of the public. In RAI 12.2-21, the staff asked the applicant to update this COL information item to fully reflect all applicable NRC regulations. The applicant identified this as COL Information Item 12.2.4.2 for airborne effluents, pending confirmation in DCD Tier 2, Revision 4. The staff tracked RAI 12.2-21 as a confirmatory item in the SER with open items. The staff confirmed that, in DCD Tier 2, Revision 6, Chapter 12, the

applicant redesignated Sections 12.2.2.2 and 12.2.4 as COL 12.2-2-A, which states the following:

> The COL Applicant is responsible for ensuring that offsite dose (using site-specific parameters) due to radioactive airborne effluents complies with the regulatory dose limits in Sections II.B and II.C of 10 CFR Part 50, Appendix I. In addition, the COL applicant is responsible for compliance with Section II.D of 10 CFR Part 50, Appendix I; airborne effluent concentration limits of 10 CFR Part 20 Appendix B (Table 2, Column 1); and dose limits of 10 CFR 20.1301 and 20.1302 to members of the public.

The staff finds that the applicant's revision of the COL information item and corresponding cross-references to the applicable NRC regulations and guidance regarding dose limits and airborne radioactive effluents identified the applicable regulatory requirements, and the staff therefore determined that the RAI response is acceptable. Based on the applicant's response, RAI 12.2-21 is resolved.

For the reasons discussed above, DCD Tier 2, Revision 9 satisfies the regulatory criteria and guidance by providing sufficient detail to demonstrate that the equipment of the GWMS will support the design objectives of Appendix I to 10 CFR Part 50 under Sections II.B and II.C and the gaseous effluent concentration limits of Appendix B (Table 2, Column 1) to 10 CFR Part 20.

In addition, a COL applicant referring to the ESBWR certified design is responsible for ensuring that offsite doses to members of the public, based on site-specific parameters, comply with the design objectives of Appendix I to 10 CFR Part 50 for gaseous effluents under Sections II.B and II.C, the effluent concentration limits of Appendix B (Table 2, Column 1) to 10 CFR Part 20, and Section II.D of Appendix I to 10 CFR Part 50 in conducting a cost-benefit analysis of installed gaseous effluent treatment systems.

In DCD Tier 2, Revision 6, Chapter 12, based on its reevaluation of condensate demineralizer flow in DCD Tier 2, Revision 6, Chapter 11, the applicant further revised the source term and the estimates of annual airborne radioactivity releases for some of the radionuclides listed in Table 12.2-19b. As a result of the staff's review of these changes in DCD Tier 2, Revision 6, the staff issued RAI 12.2-28 to resolve additional questions regarding source terms, with a focus on the radioactivity levels in liquid systems and effluents (see Section 12.3.3.2.2 of this report). The RAI requested that the applicant perform additional analyses to determine the effect of the changes in the source term on the effluent releases, resultant offsite doses, and other analyses. In its responses to RAI 12.2-28, the applicant provided additional analysis of in-plant dose evaluations and further revised liquid and airborne concentration and release information. This included numerous changes to the DCD to resolve inconsistencies and clarify the information on doses to workers and releases to the environment, and involved a complete recalculation of the design values in Tables 12.2-16, 12.2-17, and 12.2-18b.

One of the revisions to the DCD that was included in response to RAI 12.2-28 was a significant change to the long-term atmospheric dispersion values stated in DCD Tier 2, Table 12.2-15, which the applicant had arithmetically calculated to achieve an estimate of offsite doses that would not exceed the guidelines of 10 CFR Part 50, Appendix I, given the re-calculated airborne effluent release concentrations.

The staff reviewed the proposed revisions, and confirmed that the proposed changes were incorporated in DCD Tier 2, Revision 7. The staff conducted an independent evaluation of the

dispersion values, which is provided in Chapter 2 of this report, and additional information was needed to evaluate the appropriateness of the applicant's approach to the calculations of dispersion of airborne effluent releases. As part of an audit of the applicant's supporting information to address the suitability of the revised dispersion values, the applicant proposed further revisions to DCD Tier 2, Chapters 2 and 12. These changes included changing the nature of the site parameters to be included in COL Information Item 12.2-2-A, "Airborne Effluents and Doses," such as deleting the site boundary from Table 12.2-15 as site-specific information.

The staff also conducted an independent evaluation of the estimated offsite doses, consistent with GDC 60, and the guidelines in Appendix I to 10 CFR Part 50. The applicant had performed its revised offsite dose analyses based on site-specific information from multiple potential sites. The staff performed independent offsite dose analyses using revised inputs (independently-derived site specific dispersion values for these same sites) to the LADTAP II and GASPAR II codes. The changes discussed in Revision 6 to the routine source term did not affect accident release concentrations and projected dose, and RAI 12.2-28 did not address accident analyses in the context of the revised routine operations source term. The staff found that the results were generally consistent with the applicant's calculations and stated results and that the results did not exhibit significant variation. The staff found that the maximum doses calculated did not exceed the guidelines of 10 CFR Part 50, Appendix I, and were unlikely to substantially underestimate the actual exposure of an individual through appropriate pathways. After the staff evaluated the proposed changes in response to RAI 12.2-28 that were incorporated in DCD Tier 2, Revision 7, and Revision 1 to the response to RAI 12.2-28 S01, the staff was able to finalize the conclusion regarding the acceptability of the GWMS as a permanently installed system, which includes the equipment necessary to control releases of radioactive materials in gaseous effluents in accordance with 10 CFR 20.1301 and 10 CFR 20.1302; Appendix I to 10 CFR Part 50; the requirements of GDC 60 and 61; and the requirements of 10 CFR 50.34a. Section 11.3 of this report presents the staff's evaluation of the functions of the GWMS. Therefore, the staff agrees that the maximum doses are within NRC dose guidelines and are thus acceptable. Based on the applicant's responses, RAI 12.2-28 is resolved.

12.3.3.2.2 Liquid Effluent Releases

In reviewing DCD Tier 2, Revision 1, Chapters 11 and 12, the staff could not confirm that the liquid effluent radiological source term, methodology, and assumptions used in estimating doses to members of the public, as well as liquid effluent concentrations in unrestricted areas, were consistent with the guidance in Section 11.2 of the SRP and the associated regulatory guidance. The staff's evaluation addressed compliance with the requirements of 10 CFR 20.1301 and 10 CFR 20.1302 and the design objectives of Appendix I to 10 CFR Part 50 under Section II.A. Section 11.2 of this report presents the staff's review of the LWMS, as it relates to the design requirements of 10 CFR 50.34a and GDC 60 and 61.

In reviewing DCD Tier 2, Revision 3, Chapter 12, the staff found that the applicant did not provide sufficient details regarding effluent behavior. The staff issued RAI 12.2-15 asking the applicant to provide this information. After reviewing the applicant's response to RAI 12.2-15, which included additional changes to the dose assessment parameters, the staff issued RAI 12.2-15 S01. The basis of RAI 12.2-15 S01 was to resolve additional questions regarding offsite dose receptors, modeling parameters using the methodology of the BWR-GALE code (NUREG-0016) and LADTAP II code and the technical reference and user guide (NUREG/CR–4013), changes in the annual average source term, and effluent concentrations released in liquid effluents.

Upon evaluating the applicant's response to RAI 12.2-15 S01, the staff then issued Supplements 2 and 3 to this RAI requesting further clarification of the applicant's responses. The applicant subsequently provided additional updates, incorporated into DCD Tier 2, Revision 5, Chapters 11 and 12, amending the calculations of releases and resulting dose assessments. The staff finds that the amended calculations and updates to the tables of results are sufficient to address the information requested in RAI 12.2-15 and enable the staff to perform an independent confirmation of the results, including sensitivity analyses; the results are consistent with the staff's confirmatory review and, accordingly, the responses are acceptable. RAI 12.2-15 was being tracked as an open item in the SER with open items. Based on the applicant's responses, RAI 12.2-15 is resolved.

Subsequent to satisfactory closure of RAI 12.2-15, the applicant made a significant change in the description of the condensate demineralizer system flow. Throughput and capacity changes resulted in a complete revision of the estimated long-term radioactivity levels in the coolant and liquids, which resulted in minor changes to the long-term estimate of liquid effluents, but significantly changed the long-term airborne effluents. However, the modeling parameters associated with the issues raised in RAI 12.2-15, and estimates of accidental releases were not affected. The effect on airborne effluents and the resulting changes to the estimates of offsite dose from airborne effluents is addressed in Section 12.3.3.2.1, of this report.

In DCD Tier 2, Revision 3, Section 12.2.4.3, the COL information item did not provide sufficient information to demonstrate compliance with NRC regulations for liquid effluents. In addition to demonstrating compliance with the dose objectives of Section II.A of Appendix I to 10 CFR Part 50, the staff determined that a COL applicant will also need to demonstrate compliance with Section II.D of Appendix I to 10 CFR Part 50; the liquid effluent concentration limits of Appendix B (Table 2, Column 2) to 10 CFR Part 20; and the dose limits of 10 CFR 20.1301 and 10 CFR 20.1302 to members of the public. In RAI 12.2-22, the staff asked the applicant to update this COL information item to fully reflect all applicable NRC regulations. The staff tracked this RAI as a confirmatory item in the SER with open items. The applicant identified COL Information Item 12.2.4.3 for liquid effluents, which the staff confirmed in DCD Tier 2, Revision 4, Chapter 12. As of DCD Tier 2, Revision 6, the applicant redesignated this COL information item as COL 12.2-3-A. COL Information Item 12.2-3-A states the following:

> The COL Applicant is responsible for ensuring that offsite dose (using site-specific parameters) due to radioactive liquid effluents complies with the regulatory dose limits in Section II.A of 10 CFR Part 50, Appendix I. In addition, the COL applicant is responsible for compliance with Section II.D of 10 CFR Part 50, Appendix I; liquid effluent concentration limits of 10 CFR Part 20 Appendix B (Table 2, Column 2); and dose limits of 10 CFR 20.1301 and 20.1302 to members of the public. (COL 12.2-3-A)

The staff finds that the applicant's revision of the COL information item and corresponding cross-references to the applicable NRC regulations and guidance regarding dose limits and liquid radioactive effluents identify the applicable regulatory requirements, and the staff therefore finds that the RAI response is acceptable. Based on the applicant's response, RAI 12.2-22 is resolved.

For the reasons discussed above, the applicant satisfied the regulatory criteria and guidance that require sufficient details to demonstrate that the equipment of the LWMS will support the design objectives of Appendix I to 10 CFR Part 50 under Section II.A and the liquid effluent concentration limits of Appendix B (Table 2, Column 2) to 10 CFR Part 20.

In addition, a COL applicant referring to the ESBWR certified design is responsible for ensuring that offsite doses to members of the public, based on site-specific parameters, comply with the design objectives of Appendix I to 10 CFR Part 50 for liquid effluents under Section II.A; the effluent concentration limits of Appendix B (Table 2, Column 2) to 10 CFR Part 20; and Section II.D of Appendix I to 10 CFR Part 50 in conducting a cost-benefit analysis of installed liquid effluent treatment systems.

The staff was able to finalize its conclusion regarding the acceptability that the LWMS (as a permanently installed system) includes the equipment necessary to control releases of radioactive materials in liquid effluents in accordance with the requirements of 10 CFR 20.1301 and 10 CFR 20.1302, Appendix I to 10 CFR Part 50, GDC 60 and 61, and 10 CFR 50.34a. The staff finds that the amended calculations and updates to the tables of results are sufficient to address the information requested in RAI 12.2-15 in order to enable the staff to perform an independent confirmation of the results, including sensitivity analyses; the results are consistent with the staff's confirmatory review and, accordingly, the responses are acceptable.

12.3.3.3 *Airborne Radioactive Material Sources*

In DCD Tier 2, Revision 7, Section 12.2.3, the applicant described the sources of airborne radioactivity for the ESBWR reactor design and described actions taken to minimize radioactive airborne concentrations in various parts of the plant.

The main source of airborne activity in the reactor building (RB) during operation is leakage of primary coolant. The containment drywell is not accessible during normal operation; during maintenance, the drywell air is purged before access is permitted. In RB areas outside the drywell, the ventilation system routes air from areas of lower potential airborne contamination (i.e., corridors) to areas of higher potential airborne contamination (i.e., equipment rooms).

During refueling, some of the sources of airborne radioactivity are water evaporation from reactor internals and the fuel pools. Evaporation from reactor internals will be minimized by keeping surfaces of reactor internals (i.e., the steam dryer and separator) wetted or covered when removed from the reactor vessel. Fuel pool evaporation will be minimized by lowering the temperatures in the fuel pools and using the fuel pool ventilation system to sweep air from the fuel pool surface to prevent pool evaporation releases from mixing with the area atmosphere. The applicant estimated that the resulting airborne concentrations in the RB will be below the limits established in Table 1, Column 3, of Appendix B to 10 CFR Part 20.

The source of airborne radioactivity in the fuel building (FB) is the spent fuel storage pool and equipment areas. Similar to procedures in the RB, fuel pool evaporation will be minimized by lowering the temperature in the fuel pool and using the fuel pool ventilation system to sweep air from the fuel pool surface to prevent pool evaporation releases from mixing with the area atmosphere. The applicant estimated that the resulting airborne radioactive material concentrations in the FB will be below the limits established in Table 1, Column 3, of Appendix B to 10 CFR Part 20.

The main potential source of airborne radioactivity in the turbine building is leakage from valves on large lines carrying high-pressure steam. The design provides for collection of this leakage and its transport back to the condenser. By circulating air from areas of lower potential airborne contamination to areas of higher potential airborne contamination, the applicant plans to minimize sources of airborne radioactivity from equipment leakage in occupied areas. The applicant estimates that the resulting airborne radioactive material concentrations in the turbine

building will be below the limits established in Table 1, Column 3, of Appendix B to 10 CFR Part 20.

The corridors and routine access operating areas within the radwaste building are not expected to have significant airborne radioactivity levels. The vents from tanks in the radwaste building are vented directly to the building ventilation system. Pumps and valves for radioactive systems are located in separate compartments that are not normally occupied. The radwaste building ventilation system routes air from areas of lower potential airborne contamination to areas of higher potential airborne contamination. The applicant estimates that the resulting airborne radioactive material concentrations in the radwaste building will be below the limits established in Table 1, Column 3, of Appendix B to 10 CFR Part 20.

The applicant uses airborne radioactive source terms in the design of ventilation systems and for personnel dose assessment. RG 1.70 states that Section 12.2 of DCD Tier 2 should include a tabulation of the calculated concentrations of airborne radioactive material, by nuclide, for areas normally occupied by operating personnel. DCD Tier 2, Revision 6, Section 12.2.3, describes the assumptions and parameters used to determine the maximum expected airborne radioactivity concentration levels during normal operations in the RB, FB, turbine building, and radwaste building. The staff finds that this approach constitutes an acceptable basis for satisfying the requirements of Appendix B to 10 CFR Part 20.

12.3.4 Conclusions

Based on its review of the information on radiation sources supplied by the applicant for the ESBWR, as described above, the staff concludes that the applicant has committed to follow the guidelines of the RGs and staff positions outlined in Section 12.2 of the SRP. The staff finds that Section 12.2 of DCD Tier 2, Revision 9, is consistent with the guidance contained in these RGs and staff positions. The staff therefore concludes that the design meets the relevant requirements of 10 CFR Part 20 and the applicable sections of 10 CFR Part 50, including Appendix A, GDC 60 and 61. Thus, the staff finds the material contained in DCD Tier 2, Revision 9, Section 12.2 acceptable.

The staff finds it acceptable for the applicant to defer discussion of the material addressed by COL Information Items 12.2-2-A, 12.2-3-A, and 12.2-4-A. The staff will determine compliance with these COL information items during the COL review.

12.4 Radiation Protection Design

12.4.1 Regulatory Criteria

The applicable regulatory criteria and guidance include the following:

- 10 CFR Part 20
- 10 CFR 20.1406
- 10 CFR 50.34(f)(2), as it relates to requirements related to Three Mile Island
- 10 CFR 50.68
- 10 CFR 70.24

- 10 CFR Part 50, Appendix A, GDC 19, “Control room.”
- 10 CFR Part 50, Appendix A, GDC 64, “Monitoring radioactivity releases.”
- RG 1.69, “Concrete Radiation Shields for Nuclear Power Plants,” December 1973.
- RG 1.70
- RG 1.183, “Alternative Radiological Source Terms for Evaluation Design Basis Accidents at Nuclear Power Reactors,” July 2000.
- RG 1.97, Revision 4, “Criteria for Accident Monitoring Instrumentation For Nuclear Power Plants,” June 2006.
- RG 4.21, “Minimization of Contamination and Radioactive Waste Generation: Life-Cycle Planning.”
- RG 8.2
- RG 8.8
- NUREG/CR–3587, “Identification and Evaluation of Facilitation Techniques for Decommissioning Light Water Power Reactors,” June 1986.
- NUREG–0737, “Clarification of TMI Action Plan Requirements,” November 1980.

The staff compared the SRP (Sections 12.3 and 12.4, 1981 version) used during the review of the DCD with the 2007 version of the SRP. The 2007 version includes additional guidance, including additional acceptance criteria and guidance addressing the requirements of 10 CFR 20.1406. The staff incorporated this additional guidance from the 2007 version of the SRP during the staff’s subsequent review of Sections 12.3 and 12.4 of the DCD. Therefore, the staff concludes that the version of the SRP used, in combination with its additional review, is appropriate for this review.

12.4.2 Summary of Technical Information

The purpose of this review was to ensure that the applicant had either followed the guidelines of the RGs and applicable staff positions or offered acceptable alternatives for facility design features, minimization of contamination, shielding, ventilation, and area and airborne radiation monitoring to maintain occupational radiation exposures ALARA. For cases in which the DCD adheres to these RGs and staff positions, the staff can conclude that the design meets the relevant requirements of 10 CFR Part 20, 10 CFR Part 50, and 10 CFR Part 70, “Domestic Licensing of Special Nuclear Material.”

Under the license termination provisions of 10 CFR 20.1406, Subpart E, applicants for a new license are required, among other things, to describe how the facility design and procedures for operation will facilitate eventual decommissioning of the facility by minimizing, to the extent practicable, contamination of the facility and the environment, and the quantities of radioactive wastes generated. The staff reviewed the applicant’s design features for minimizing contamination against the requirements of 10 CFR 20.1406. The staff ensured that the applicant had either followed the criteria of the applicable guidance or provided acceptable

alternatives. Where the DCD adheres to these staff positions, the staff can conclude that the design meets the relevant requirements of 10 CFR 20.1406.

The applicant has identified the following COL information items to ensure that license applicants referencing the ESBWR design will address these issues:

- (COL 12.3-2-A) Operational Considerations—Airborne radiation monitoring operational considerations, such as the procedures for operations and calibration of the monitors, as well as the placement of the portable monitors, are the COL applicant's responsibility.

- (COL 12.3-4-A) Compliance with 10 CFR 20.1406—The COL applicant will address the operational and postconstruction objectives of RG 4.21.

12.4.3 Staff Evaluation

The staff reviewed the facility design features, shielding, ventilation, and area and airborne radiation monitoring instrumentation contained in DCD Tier 2, Revision 9, Section 12.3, for adherence to the guidelines in RG 1.70 and the criteria in Section 12.3-12.4 of the SRP.

12.4.3.1 *Facility Design Features*

The ESBWR reactor design incorporates several features to help maintain occupational radiation exposures ALARA, in accordance with the guidance in RG 8.8. These design features are founded on the ALARA design considerations described in DCD Tier 2, Revision 9, Section 12.1 and discussed in Section 12.2.3.2 of this report.

The ESBWR natural circulation design eliminates the need for reactor coolant pumps and reactor coolant piping typically found in BWR designs. Maintenance and inspection of these components (and supporting activities, such as insulation removal and replacement) are significant sources of occupational radiation exposure in operating nuclear power plants. The simpler design of the ESBWR also facilitates personnel access and equipment maintainability in the upper and lower drywells. Work platforms are also provided for accessibility to main steam isolation valves and other equipment requiring routine maintenance. The lower reactor head area is designed with a minimum of equipment interference to facilitate CRD mechanism access for maintenance. In addition, a trolley system provides transport of the CRDs from the lower drywell to a dedicated maintenance area with lower radiation levels.

Equipment and piping layout are designed to reduce the exposure of personnel required to inspect or maintain equipment. Major sources of radiation are located in separate cubicles from their associated piping and pumps, as well as from each other, to reduce personnel radiation exposure from these components during maintenance. Pumps located in radiation areas are designed to minimize the time required for maintenance.

Quick-change cartridge-type seals on pumps and pumps with back pullout features that permit removal of the pump impeller or mechanical seals without disassembly of attached piping are used to minimize exposure time during pump maintenance. The configuration of piping surrounding pumps is designed to provide sufficient space for efficient pump maintenance. Heat exchangers are constructed of stainless steel or copper-nickel tubes to minimize the possibility of failure and reduce maintenance requirements. Fill and drain fittings are provided on radioactive systems and components that facilitate system or component flushing to reduce radiation dose rates during maintenance.

Lighting is designed to provide sufficient illumination in radiation areas to allow quick and efficient surveillance and maintenance operations. To reduce the need for immediate replacement of defective bulbs, multiple lighting fixtures are provided in shielded cubicles. Incandescent lamps, which require less time for servicing, are the only type of lamp used within the primary containment, the main steam tunnel, and the refueling level of the RB.

The ESBWR design has many features to minimize the spread of contamination within the plant. Contaminated piping systems are welded, to the extent practical, to minimize leaks through screwed or flanged fittings. For systems containing highly radioactive fluids, drains are hard-piped directly to equipment drain sumps so that contaminated fluid does not flow across the floor to a floor drain. Smooth epoxy-type coatings are employed to facilitate decontamination in the event of spills or leaks. Pump casing drains are employed on radioactive systems whenever possible to remove fluids from the pump before disassembly. On the refueling floor, a circular stand in the reactor vessel head laydown area prevents contamination from inside the reactor vessel cover from spreading to the outside of the cover when the cover is in its storage space. In addition, the applicant can plasticize the floor inside the stand and the area of the cover storage point to control potential contamination releases.

In addition to designing equipment to comply with ALARA guidelines, the ESBWR plant layout is designed to reduce personnel exposures. The design provides adequate work and laydown space at each inspection and maintenance station. In addition, it provides for rigging and lifting equipment to facilitate the removal, transport, or replacement of equipment and the use of portable shielding during maintenance activities. Adequate support services (e.g., power, compressed air, water, ventilation, and communications) will be available at work stations. Floor drains with appropriately sloped floors are provided in shielded cubicles where the potential for spills exists. Valves associated with highly radioactive components will be separated from other components and located in shielded valve galleries. Major components in radioactive systems will be located in shielded compartments where practicable. To minimize radiation streaming through wall penetrations, the ESBWR design calls for shield wall penetration rooms with offsets between the radioactive source and the normally accessible areas.

Radioactive piping will be routed through shielded pipe chases or shielded equipment cubicles, wherever possible, to minimize personnel exposures. Some short feed-through sections of piping may be embedded in concrete. By limiting the length of embedded piping to short sections, to the extent practicable, the applicant will facilitate the dismantlement of the systems and the decommissioning of the facility, as required by 10 CFR 20.1406. The equipment and layout design features described above conform to the guidelines of RG 8.8 for maintaining occupational radiation exposures ALARA. Therefore, the staff finds these features acceptable.

The ESBWR design also incorporates several features to minimize the buildup, transport, and deposition of activated corrosion products in the RCS and auxiliary systems. The DCD states that the ESBWR design will reduce or eliminate the use of materials containing cobalt that are in contact with reactor coolant, except in cases in which the use of these materials is necessary for reliability purposes. Stainless steel is used in portions of the system, such as the reactor internal components and heat exchanger tubes, where high corrosion resistance is required. The nickel content of the stainless steel is in the range of 9 to 10.5 percent and is controlled in accordance with applicable material specifications of the American Society of Mechanical Engineers. Cobalt content is controlled to less than 0.05 percent in the XM-19 alloy used in the CRDs. To the extent practicable, Colmonoy is used for hard facings of components in the core area as an alternative to Stellite and other high-cobalt alloys.

The use of butt welds instead of sleeve-welded joints will minimize the potential for creating crud traps in the weld areas of piping for those systems carrying radioactive liquids. Tanks containing radioactive liquid will have drainpipes connected at the lowest part of the tank and convex or sloped-bottom designs to minimize radioactivity deposition. Pipes are seamless and are adequately sloped for avoiding stagnation. Piping configurations are designed to minimize the number of "dead legs" and low points in piping runs to avoid accumulation of radioactive crud and fluids in the line. Straight-through valve configurations are used, where practical, to minimize crud traps and radiation exposure associated with maintenance on these valves. Valve packing and gasket material are selected for long operating life to minimize required maintenance. Valves have back seats to minimize the leakage through the packing. Equipment and piping containing radioactive materials will have provisions for draining and flushing. These design features, which are intended to minimize the buildup, transport, and deposition of activated corrosion products in the RCS and auxiliary systems, are based on the guidelines in RG 8.8 and are, therefore, acceptable.

The applicant provided the staff with detailed drawings of the ESBWR plant layout that show the 10 radiation zones used in the plant design. These radiation zones serve as a basis for classifying occupancy and access restrictions for various areas within the plant during normal operations and accident conditions. On this basis, the applicant established the maximum design dose rates for each zone and used these as input for shielding of the respective zones. Based on its review of the detailed zoning drawings, the staff concludes that the applicant's method of plant zoning, for normal operations, is consistent with the guidance in RG 1.70 and the SRP. Therefore, the staff finds this method acceptable.

Areas in which an individual could receive a dose in excess of 5 Sv (500 rem) within a period of 1 hour at 1 meter from a radiation source or 1 meter from any surface that the radiation penetrates are posted with "Very High Radiation Area" signs. The COL applicant will control access to these very high radiation areas (VHRAs) in response to COL Information Item 12.5-3-A (described in DCD Tier 2, Revision 9, Section 12.5).

DCD Tier 2, Revision 2, Section 12.3.1.3, did not contain incremental zone designations for area dose rates above 1 millisievert per hour (mSv/h) (100 mrem/h). The staff requested, in RAI 12.4-4, that the applicant amend the DCD to include additional incremental zone designations for higher dose rates and identify on the plant layout drawings all plant areas having dose rates exceeding 1 Sv/h (100 rads/h). The staff also requested that the applicant identify each area of the plant that meets the definition of a VHRA, as provided in 10 CFR Part 20. In response, the applicant modified DCD Tier 2, Revision 3, Section 12.3.1.3 to expand the radiation zone classifications to include the upper radiation ranges requested. The applicant also provided a listing of all plant areas having dose rates exceeding 1 Sv/h (100 rads/h) and all VHRAs. By modifying the DCD plant layout figures to include the upper radiation zone ranges, the staff was able to assess the estimated integrated doses to personnel accessing these higher dose rate areas and evaluate GEH's dose assessment, in accordance with the guidance provided in RG 8.19. In addition, the staff performed some confirmatory shielding calculations of selected plant areas to verify the accuracy of the applicant's radiation zone designations. In each of these cases, the staff verified that the applicant had used the correct radiation zone designations. 10 CFR 20.1602 states that, in addition to the requirements in 10 CFR 20.1601 for control of access to high radiation areas, licensees shall institute additional measures for control of access to areas identified as VHRAs. By providing a listing of all VHRAs in the ESBWR design, and describing the access controls in place to control access to these areas, GEH has complied with the requirements of 10 CFR 20.1602. For these reasons, the staff finds the response to RAI 12.4-4 to be acceptable. RAI 12.4-4 was being

tracked as an open item in the SER with open items. Based on the applicant's response, RAI 12.4-4 is resolved.

Section 12.3.2 of RG 1.70 states that an applicant should provide information on the shielding for each of the sources identified in DCD Tier 2, Section 12.2. Since DCD Tier 2, Revision 1, did not initially contain this information, the staff, in RAI 12.4-6, asked the applicant to provide the composition and thickness of each radiation shield depicted in DCD Tier 2, Revision 1, Figures 12.3-1 through 12.3-22. In response, the applicant amended DCD Tier 2, Revision 3, to add Table 12.3-8, which provided a listing of the wall, ceiling, and floor thicknesses for the rooms with the most significant plant radiation sources. As part of its response, the applicant stated that the COL applicant will define any special shielding features required. Since the applicant did not specify the COL information item that would require the COL applicant to provide this information, the staff issued RAI 12.4-6 S02 requesting the applicant to clarify its response. In response, the applicant stated in DCD Tier 2, Revision 5, that COL Information Item 12.5-3-A, which requests that the COL applicant describe the operational radiation protection program, would address this information. By adding to the DCD a listing of the shielding thicknesses of the rooms with the most significant plant radiation sources and providing the shielding information for each of the radiation sources called for in Section 12.3.2 of RG 1.70, GEH's response resolved the staff's RAI. GEH's addition of the COL information item to the DCD further resolved the staff's supplement to this RAI by committing the COL applicant to provide a description of the shielding for any additional sources not described in the DCD. RAI 12.4-6 was being tracked as an open item in the SER with open items. Based on the applicant's response, RAI 12.4-6 is resolved.

In RAI 12.4-11, the staff asked the applicant to provide additional information regarding the purpose of the "wash down bays" listed in DCD Tier 2, Revision 1, Figure 12.3-4. In its response, the applicant modified DCD Tier 2, Revision 4, Section 12.3.1.2.6, to state that the wash down bays would be used to remove contamination from the spent fuel cask and its transporter before leaving the plant. In this revised section, the applicant also described several design features associated with these wash down bays to minimize the spread of contamination. These features include walls or curbs to contain potential contamination fluid leakage, sloped floor surfaces to drains, and the use of concrete surfaces protected with non-porous coatings. These features are in accordance with the guidance provided in RG 4.21 to minimize the spread of contamination and are acceptable. The staff finds that GEH's detailed description of the purpose of the wash down bays and the description of the associated design features thus adequately responded to the staff's RAI. RAI 12.4-11 was being tracked as an open item in the SER with open items. Based on the applicant's response, RAI 12.4-11 is resolved.

For the reasons stated above, the staff finds that the information in Section 12.3.1 of DCD Tier 2, Revision 9, adequately addresses the relevant requirements and guidance of 10 CFR Part 20 and RG 8.8. Therefore, the staff finds the information contained in this section acceptable.

12.4.3.1.1 Minimization of Contamination and Radioactive Waste Generation

The requirements in 10 CFR 20.1406 state that each applicant must describe how it intends to minimize, to the extent practicable, contamination of the facility and of the environment, as well as the generation of radioactive waste. Applicants are also required to describe how they will facilitate decommissioning. The guidance in Section 12.3-4 of the SRP states that design features described by the applicant should facilitate eventual decommissioning and minimize, to the extent practicable, contamination of the facility and the environment, as well as the

generation of radioactive waste. RG 4.21 contains a basis acceptable to the staff for complying with the requirements of 10 CFR 20.1406. For those cases in which the applicant adhered to this guidance, the staff can have reasonable assurance of compliance with 10 CFR 20.1406.

Initially, DCD Tier 2, Revision 1, Section 12.6 described how the ESBWR is designed to minimize contamination of the facility and environment, facilitate decommissioning, and minimize the generation of radioactive waste, in compliance with 10 CFR 20.1406. However, DCD Tier 2, Revision 1, Section 12.6 was prepared before the issuance of RG 4.21 and did not contain sufficient information for the staff to determine whether the ESBWR design complies with the requirements of 10 CFR 20.1406. To obtain the information necessary to make this determination, the staff issued RAIs 12.7-1, 12.7-2, and 12.7-3.

NUREG/CR–3587 lists several decommissioning facilitation techniques that are applicable during the design and construction phase of a commercial nuclear power reactor. RAI 12.7-1 requested that the applicant amend DCD Tier 2, Section 12.6, to describe to what extent each of the features described in this NUREG/CR document were incorporated in the ESBWR design. RAI 12.7-2 requested that the applicant describe how the ESBWR design minimizes the generation of radioactive waste during decommissioning operations. RAI 12.7-3 requested that the applicant identify ESBWR piping or components that have the potential for leaking radioactively contaminated fluids and which are designed to be below the grade of the plant site.

In responding to RAIs 12.7-1, 12.7-2, and 12.7-3, the applicant amended DCD Tier 2, Revision 3, Section 12.6 to describe additional design features to facilitate the eventual decommissioning of the plant, to minimize the generation of radioactive waste during decommissioning, and to identify below-grade systems that have the potential for leaking radioactive fluids. As stated above, the staff had asked these three RAIs to obtain additional information to permit the staff to make a determination as to whether the applicant's design features to minimize contamination and the generation of radioactive waste satisfied the requirements of 10 CFR 20.1406. RG 4.21 was issued after the staff received the applicant's responses to these RAIs. RG 4.21 contains a listing of design and operational objectives to minimize contamination and the generation of radioactive waste and describes a basis acceptable to the staff for complying with the requirements of 10 CFR 20.1406. In reviewing the applicant's responses to RAIs 12.7-1, 12.7-2, and 12.7-3, the staff found that the applicant had listed a number of acceptable design features to facilitate decommissioning operations, and to minimize the generation of radioactive waste during decommissioning operations. The applicant has also described the ESBWR below-grade piping or components that have the potential for leaking radioactively contaminated fluids. However, the staff had additional concerns about the completeness of the information added to the DCD in response to these RAIs and found that this information did not adequately address the guidance contained in RG 4.21. The staff, therefore, issued RAI 12.7-5, which requested that the applicant provide the additional information needed to address the guidance contained in RG 4.21. Because RAI 12.7-5 also addressed the remaining staff concerns with the applicant's responses to RAIs 12.7-1, 12.7-2, and 12.7-3, which were being tracked as open items in the SER with open items, the staff closed out RAIs 12.7-1, 12.7-2, and 12.7-3 and stated that the staff would track these concerns as part of the applicant's response to RAI 12.7-5. As requested by the staff, the applicant incorporated its additional responses to RAIs 12.7-1, 12.7-2, and 12.7-3 as part of the response to RAI 12.7-5. The basis for the staff's acceptance of the applicant's response to RAI 12.7-5 is discussed below. As discussed above, with the issuance of RAI 12.7-5, RAIs 12.7-1, 12.7-2, and 12.7-3 are closed.

In RAI 12.7-5, the staff requested that the applicant relocate the information contained in Section 12.6 of the DCD into Section 12.3 of the DCD. The staff also requested that the applicant supplement this information by describing how the ESBWR design meets each of the RG 4.21 design objectives and how the COL applicant will meet each of the operational objectives contained in RG 4.21 (through the creation of new COL information items). The staff requested that the applicant include a listing in the DCD of several examples of ESBWR design features that illustrate how the ESBWR design meets these design objectives. For those instances in which other sections of the DCD describe the design features incorporated to meet the design objectives, the staff requested that the applicant provide cross-references in Section 12.3 of the DCD directing the reader to the appropriate section of the DCD that addresses each of these objectives in a more detailed manner. Finally, RAI 12.7-5 provided a listing of several plant systems that could generate radioactive waste or could result in the contamination of nonradioactive systems and requested that the applicant amend the DCD to describe the specific design feature associated with each of the systems incorporated into the ESBWR design to comply with the requirements of 10 CFR 20.1406.

In response to RAI 12.7-5, GEH indicated that it had amended the DCD, in Revision 6, to add a new section, Section 12.3.1.5. In this new section (described in detail below), the applicant provided a detailed description of both generic and specific ESBWR design features to minimize contamination and the generation of radioactive waste and to facilitate decommissioning. In response to RAI 12.7-5, the applicant also added a table to DCD Tier 2, Revision 6, to provide a crosswalk between the structures and systems that address the RG 4.21 design objectives specified in the staff's RAI and the applicable DCD chapters and sections in which they are discussed. To address how the operational objectives specified in RG 4.21 will be met, the applicant amended the DCD, in Revision 6, to add the new COL Information Item 12.3-4-A, which states that the COL applicant will be responsible for addressing the operational and post-construction objectives of RG 4.21.

In reviewing the applicant's response to RAI 12.7-5, the staff noted that the applicant did not specifically describe the ESBWR SSCs that had associated buried piping which could potentially carry radioactive or potentially radioactive fluids. Accordingly, in RAI 12.7-5 S01, the staff asked the applicant to amend the DCD to describe any such ESBWR SSCs and to discuss the associated design features that the applicant had implemented to minimize the potential for unmonitored, uncontrolled releases of radioactivity to the environment from the SSC or its associated buried piping. In response, the applicant modified the DCD, in Revision 6, to describe the requested design features associated with three ESBWR SSCs that had underground piping.

In RAI 12.7-5 S02, the staff stated that, in addition to describing the 10 CFR 20.1406 related design features associated with these three SSCs, the applicant must also describe how it plans to monitor the cooling tower blowdown line for leakage in order to minimize the potential for unmonitored, uncontrolled releases of radioactivity to the environment. In the applicant's response, GEH modified the DCD, in Revision 7, to state that the underground piping associated with the three SSCs initially described, as well as the cooling tower blowdown lines, are designed to preclude inadvertent or unidentified leakage to the environment. As described above, in response to RAI 12.7-5 and its supplemental RAIs, the applicant has shown how the ESBWR is designed to minimize contamination and the generation of radioactive waste in accordance with the requirements of 10 CFR 20.1406 and the guidance provided in RG 4.21. The applicant's response describes how each of the design objectives contained in RG 4.21 are addressed by design features incorporated in the ESBWR design. The staff has reviewed these design features and finds that these features are designed to minimize contamination and the

generation of radioactive waste in accordance with 10 CFR 20.1406. The following paragraphs list the design objectives contained in RG 4.21 and provide numerous examples of how the ESBWR design addresses each of these design objectives. In the applicant's response to this RAI, the addition of the new COL information item commits the COL applicant to be responsible for addressing the operational and post-construction objectives of RG 4.21. On this basis, the staff concludes that GEH has adequately described how the ESBWR design addresses the design objectives of RG 4.21. Therefore, RAI 12.7-5 is resolved.

DCD Tier 2, Revision 9, Section 12.3.1.5, describes a design philosophy of prevention and early detection of leaks, such that occupational doses are maintained ALARA, contamination is minimized, and decommissioning is facilitated. The general design features described by the applicant are consistent with this design philosophy and demonstrate compliance with the requirements of 10 CFR 20.1406. As more fully described below, these features include measures to minimize facility contamination and contamination of the environment, as well as features to facilitate decommissioning.

The applicant has stated that the ESBWR design incorporates the following design objectives, which address the objectives contained in RG 4.21:

- Minimize leaks and spills and provide containment in areas where such events might occur.

- Provide for adequate leak detection capability to provide prompt detection of leakage for any SSC that has the potential for leakage.

- 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 (e.g., spent fuel pools; tanks that are in contact with the ground; and buried, embedded, or subterranean piping) to avoid release of contamination.

- 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.

- Facilitate decommissioning by (1) minimizing embedded and buried piping and (2) designing the facility to facilitate the removal of any equipment and components that may require removal or replacement during facility operation or decommissioning.

- Minimize the generation and volume of radioactive waste both during operation and during decommissioning (by minimizing the volume of components and structures that become contaminated during plant operation).

The following discussion of several ESBWR design features described in DCD Tier 2, Revision 9, Chapter 12, addresses the above-listed design objectives.

To minimize leaks of radioactive gases, equipment drain sumps that contain airborne contaminants from discharges to the sump are hard-piped directly to their respective building heating, ventilation, and air conditioning (HVAC) system. The drains for systems containing highly radioactive fluids are hard-piped directly to equipment drain sumps to eliminate the potential for the flow of contaminated fluid across the floor to a floor drain. For other radioactive systems, shielded cubicles in which the potential for spills exists include appropriately sloped floor drains to limit the extent of contamination. Equipment and floor drain sumps are lined in

stainless steel to reduce crud buildup and to provide surfaces easily decontaminated. To facilitate the cleanup of leaks and spills, epoxy-type wall and floor coverings have been selected which provide smooth, nonporous surfaces for ease of decontamination. To prevent any potential water releases from high-activity areas in the radwaste building to the environment, tank cubicle concrete is provided with a sealant and a tank cubicle steel liner. In addition, the radwaste building is designed to contain any liquid release by locating all high-activity tanks in watertight rooms designed to contain the maximum liquid release for that room. Penetrations through outer walls of a building containing radiation sources are sealed to prevent miscellaneous leaks to the environment. Curbs or walls are also provided to limit contamination and simplify washdown operations. A basin surrounding the condensate storage tank is designed to prevent uncontrolled runoff in the event of a tank failure.

The process radiation monitoring system (PRMS) will monitor all radioactive release points and paths within the plant. This system provides continuous monitoring and display of the radiation measurements during normal, abnormal, and accident conditions and allows for the content of radioactive material in various gaseous and liquid process and effluent streams to be determined. The ESBWR fuel and auxiliary pools cooling system is designed to detect and monitor potential leaks from the spent fuel pools, as well as from the auxiliary pools and isolation condenser/passive containment cooling system pools, and drain this leakage to the LWMS. The process sampling system collects and analyzes representative liquid and gaseous samples for indications of system leaks. Before discharge to the environment, radioactive releases from tanks will be sampled and analyzed to ensure that the activity concentration is consistent with the discharge criteria of 10 CFR Part 20.

Operating experience has shown that effluent discharge piping can be a source of low-level environmental contamination. In particular, operating experience has shown that the following SSCs have experienced piping-related occurrences that have resulted in unmonitored, uncontrolled releases of radioactivity to the environment: condensate storage tank and associated piping, radwaste/effluent discharge piping, and cooling tower blowdown line. In response to RAI 12.7-5 (discussed above) concerning these SSCs, the applicant stated that, for the ESBWR design, segments of piping of the three above-mentioned SSCs, as well as portions of the hot machine shop drain, will be run underground. The applicant stated 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 either will be enclosed within a guard pipe and monitored for leakage or 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.

The applicant also stated that fittings will be kept to a minimum and no in-line components (e.g. valves) will be incorporated into these lines. These features will reduce the potential for unmonitored and uncontrolled releases to the environment and meet the applicable guidelines of RG 4.21.

Plant equipment containing radioactive material is designed to minimize the buildup of radioactive material by using sloped lines, minimizing the number of “dead legs” and low points, and using welded versus flanged or screwed connections to the most practical extent. The design employs straight-through valve configurations, where practical, instead of valve configurations that exhibit flow discontinuities or internal crevices to minimize crud trapping. 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

crud traps. Equipment in contact with liquid and solid radioactive wastes will be designed with adequate finish or linings to prevent the adherence of corrosion products to facilitate decontamination. The HVAC systems are designed to make airborne radiation exposures to plant personnel and releases to the environment ALARA. These systems maintain potentially contaminated areas at a negative pressure to minimize exfiltration of potentially contaminated air and maintain airflow from areas of lower potential for contamination to areas of greater potential for contamination.

To facilitate decommissioning, the reactor, fuel, turbine, and radwaste buildings are designed for the removal of large equipment by providing ample space around components and including equipment hatches on many cubicles. Lifting points, monorails, and other installed devices provided to facilitate equipment handling during maintenance can also be used to facilitate decommissioning. The radwaste process systems are skid mounted and located in the radwaste building to allow truck access and system skid loading and unloading. Wherever possible, piping carrying radioactive fluids is separated from piping carrying nonradioactive fluids. In some cases, short feed-through sections of piping may be embedded in concrete. However, the use of these will be minimized to facilitate the eventual dismantlement of the systems and the decommissioning of the facility. 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 ESBWR is designed to minimize the generation and release of radioactive materials in their gaseous, liquid, and solid forms. The liquid and solid radioactive waste management systems are divided into several subsystems to segregate wastes which allows for efficient processing and the minimization of the overall quantity of liquid and solid waste. The offgas system minimizes and controls the release of radioactive material into the atmosphere by delaying the release of the radioactive offgas process stream. The ESBWR 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.

As described above, DCD Tier 2, Revision 9, Chapter 12, describes numerous ESBWR design features that address the RG 4.21 design objectives and demonstrates their compliance with the requirements of 10 CFR 20.1406. Several of the design features described above are also described in other chapters and sections of the DCD. DCD Tier 2, Revision 9, Section 12.3, Table 12.3-18, provides a comprehensive crosswalk of applicable DCD chapters and sections which describe design features that address the above-listed RG 4.21 design objectives.

In addition to the design objectives listed above, RG 4.21 contains the following operational and post-construction objectives which address the requirements of 10 CFR 20.1406:

- 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 pathways from the release sources to the receptor points.

The applicant stated, in DCD Tier 2, Revision 6, Section 12.3.7, that the COL applicant will address the operational and postconstruction objectives of RG 4.21. The applicant identified this issue as COL Information Item 12.3-4-A.

Based on the staff's review of the ESBWR design features provided to minimize contamination, the staff concludes that the applicant followed the guidelines in RG 4.21 and thus complies with the requirements of 10 CFR 20.1406. Therefore, the staff finds the material contained in DCD Tier 2, Revision 9, Section 12.3.1.5, to be acceptable.

12.4.3.2 *Shielding*

The objective of the plant's radiation shielding is to minimize plant personnel and population exposures to radiation during normal operation (including AOOs and maintenance) and during accident conditions while maintaining a program of controlled personnel access to and occupancy of radiation areas. The ESBWR design also includes shielding, where required, to mitigate the possibility of radiation damage to materials.

DCD Tier 2, Revision 9, states that radioactive components and piping will be separated from nonradioactive components and piping to minimize personnel exposure during maintenance and inspection activities. When radioactive piping must be routed through corridors or other low-radiation zones, shielded pipe chases are provided. Where applicable, pumps and other support equipment for components that contain radioactive material are separated from the more highly radioactive components by locating them outside the component cubicle in separate shielded cubicles. Shielded compartments have labyrinth entrances to minimize radiation streaming directly through access openings.

Penetrations are located to preclude a direct line of sight from the radioactive source to adjacent occupied areas. In selected situations, provisions are made for shielding major radiation sources during inservice inspection (ISI) to reduce exposure to inspection personnel. These shielding techniques comply with the guidelines contained in RG 8.8 for protecting plant personnel and the public against exposure from various sources of ionizing radiation in the plant. Therefore, the staff finds these techniques acceptable.

The applicant applied the provisions of RG 1.69, ANSI/ANS 6.4, "Nuclear Analysis and Design of Concrete Shielding for Nuclear Power Plants," and ANSI/ANS 6.4.2, "Radiation Shielding Materials," to the design of the ESBWR radiation shielding.

During the review of the plant layout figures in DCD Tier 2, Revision 1, Section 12.3, the staff noted that the radwaste piping gallery housed both radwaste piping and electrical equipment. The staff issued RAI 12.4-16 to ascertain what design features or administrative precautions were in place to ensure that the dose to personnel who would have to enter the radwaste piping

gallery to perform maintenance on the electrical equipment located in the piping gallery would be minimized. In the response, the applicant amended DCD Tier 2, Revision 4, Section 12.3.1.2.4 to state that the nonsafety-related electrical cables in the piping gallery would be separated from the radwaste piping by a shield wall to reduce the potential dose to electrical equipment and to personnel in the piping gallery who may be performing inspection or maintenance of the electrical cabling in the gallery. In addition, the applicant stated that electrical cable replacement, though infrequent, would be performed during shutdown or when no waste transfer operations were occurring, in accordance with plant maintenance and radiation protection procedures. Because the applicant's response described the design features and administrative precautions in place to ensure that doses to personnel performing inspections or maintenance in the radwaste piping gallery would be ALARA, in accordance with the guidelines of RG 8.8, the staff finds this response acceptable to resolve this RAI. RAI 12.4-16 was being tracked as an open item in the SER with open items. Based on the applicant's response, RAI 12.4-16 is resolved.

In reviewing the plant layout drawings in DCD Tier 2, Revision 1, the staff noted that several rooms in the radwaste building (depicted in Figures 12.3-19 and 12.3-20) were missing radiation zone designations. In RAI 12.4-17, the staff requested that the applicant modify the DCD layout drawings to include the missing radiation zone designations. In DCD Tier 2, Revision 3, the applicant provided the missing information. RAI 12.4-17 was being tracked as an open item in the SER with open items. Based on the applicant's response, RAI 12.4-17 is resolved.

During the review of the ESBWR plant layout drawings provided in DCD Tier 2, Revision 1, Section 12.3, the staff noted that, since the inclined fuel transfer system (IFTS) spanned several plant elevations between the buffer pool in the RB (elevation 27,000 millimeters [mm] [88.6 feet (ft)]) and the incline fuel transfer tube pit in the FB (elevation -10,000 mm [32.8 ft]), numerous rooms and corridors appeared to be adjacent to the IFTS. Since the IFTS would be used to transfer irradiated spent fuel assemblies from the IFTS pool in the RB to the spent fuel storage pool in the FB, the staff was concerned about the potential radiation levels in these areas during spent fuel transfer through the IFTS. In RAI 12.4-19, the staff requested that the applicant provide detailed radiation shielding calculations showing peak dose rates for each area adjacent to the IFTS system in the RB and FB during irradiated fuel transfer through the IFTS system. In response to this RAI, GEH identified the various rooms and corridors adjacent to the IFTS, described the thickness of the concrete separating each of these areas from the IFTS, and provided an estimate of dose rates in most of these areas during spent fuel transfer through the IFTS. After reviewing the applicant's response to RAI 12.4-19, the staff noted that the initial response to this RAI did not clearly address the accessibility and dose rates for several areas adjacent to the IFTS during fuel transfer through the IFTS.

In RAI 12.4-19 S01, the staff requested that the applicant provide this information, as well as a description of the thickness of concrete separating two areas in the FB from the IFTS. In performing confirmatory shielding calculations to verify the applicant's calculated dose rates in various areas adjacent to the IFTS during spent fuel transfer, the staff identified an error in the applicant's shielding calculation. This error led to an underestimation of dose rates for areas adjacent to the IFTS by approximately a factor of 10. In RAI 12.4-19 S02, the staff informed the applicant of this error and requested that GEH submit revised estimated dose rates for those areas adjacent to the IFTS.

In response to Supplements 1 and 2 to RAI 12.4-19, the applicant provided a table listing the revised dose rates in the various areas adjacent to the IFTS. The applicant also provided a listing of shielding assumptions used for the revised calculations and described the access

controls for areas adjacent to the IFTS. The applicant evaluated other applications of the shielding code used to determine the dose rates around the IFTS and verified that incorrect use of this code had been an isolated incident limited to these calculations.

In follow-up supplements to RAI 12.4-19 (S03 – S05), the staff asked the applicant to describe the controls implemented for the two areas where the IFTS could be accessed for maintenance purposes and to clarify its shielding assumptions and changes to the geometry of the IFTS. The staff also requested that the applicant describe any shielding thickness changes that it may have made adjacent to the IFTS, as a result of the identification of the shielding error, in order to maintain the existing dose rate designations in the area. In its responses to these additional supplemental RAIs, the applicant provided the requested information. The staff performed confirmatory shielding calculations to verify the estimated dose rates provided by the applicant for accessible areas adjacent to the IFTS. The staff finds the applicant's response to RAI 12.4-19 and the follow-up supplemental RAIs acceptable because the plant design provides sufficient shielding for accessible areas adjacent to the IFTS to permit the necessary access to these areas during normal operations (including those periods when irradiated fuel is being transferred through the IFTS). In addition, the staff finds that the applicant's proposed access controls and postings for these areas (listed in DCD Tier 2, Revision 3, Section 12.3.1.4.4) will comply with the requirements of 10 CFR 20.1601 and 10 CFR 20.1902. RAI 12.4-19 was being tracked as an open item in the SER with open items. Based on the applicant's response, RAI 12.4-19 is resolved.

Potentially lethal radiation exposures could occur in the vicinity of any unshielded portions of the fuel transfer tube when a spent fuel assembly passes through this tube during refueling operations. Rooms 18P2 and 1702 provide access to the unshielded portions of the IFTS for periodic inspections. A system of physical controls, interlocks, and annunciators controls personnel access to these rooms. The interlock system between the door locks, the main operation panel, and the control room prevents activation of the IFTS while the rooms are accessible. Audible alarms and flashing red lights located inside and outside any IFTS maintenance area warn personnel of IFTS operation and the potential radiation hazard. In addition, radiation monitors that enunciate alarms both inside and outside each room provide continuous indication of the actual radiological conditions.

Generic Safety Issue (GSI) 137, "Refueling Cavity Seal Failure," and NRC Bulletin 84-03, "Refueling Cavity Water Seal," dated August 24, 1984, called for reactor licensees to address the potential for inadvertent reactor cavity drain down via the cavity water seal, as well as the associated potential for uncovering spent fuel, either stored or in transit. In accordance with a resolution proposed for GSI 137, as documented in NUREG–0933, "Resolution of Generic Safety Issues," dated August 2008, DCD Tier 2, Revision 6, Section 6.2.1.1 states that the ESBWR design incorporates a permanent reactor cavity seal. However, the applicant did not address the potential for an inadvertent reactor cavity drain down and the resulting potential for uncovering spent fuel. To obtain this information, the staff issued RAI 12.3-15.

In RAI 12.3-15, the staff requested that the applicant describe the location where a spent fuel assembly being transferred from the reactor vessel could be safely lowered into a safe storage area to minimize the potential for high radiation levels in the event of a rapid inadvertent reactor cavity drain down. In addition, the staff requested that the applicant provide assurance that the volume of water in the safe storage area would be sufficient to completely cover the fuel bundle for the 30 minutes allotted for ensuring containment closure. In response, the applicant described several ESBWR design features that provide margin or mitigate the potential for a drain down and discussed why a rapid drain down of the reactor vessel cavity during refueling

activities is not a credible event for the ESBWR. The applicant stated that, if a fuel assembly were to be in transit when a leak in the refueling bellows is discovered, the fuel assembly could either be returned to its former location in the core or it could be placed in the deep pit of the buffer pool where sufficient water volume exists to maintain complete coverage of the active fuel. Finally, the applicant stated that the dose consequences associated with exposure of components stored in the pools during a refueling are not considered, since these components will not become uncovered in the event of a slow loss of reactor cavity water. In response to RAI 12.3-15, the applicant referred to its response to RAI 9.1-128, which addressed the potential for inadvertent reactor cavity drain down. In responding to RAI 12.3-15, the applicant committed to amend DCD Tier 2, Revision 7, Section 12.4.4, to address the staff's concerns, as described above. The staff confirmed that DCD Tier 2, Revision 7, Section 12.4.4 was amended. The staff finds the applicant's response acceptable because, as described above, the ESBWR has no potential reactor cavity drain down paths that would result in a rapid drain down of the reactor cavity and result in an uncovering of either the fuel in the core or fuel in transit, thereby resulting in potentially high radiation levels and/or releases of radioactive material. In addition, in the event that there was a leak in the refueling bellows which could result in a slow loss of reactor cavity water, any fuel in transit to or from the reactor core could either be placed in the core or in the deep pit of the buffer pool. There is sufficient water volume in the deep pit of the buffer pool to maintain complete coverage of the active fuel. Based on the applicant's response, RAI 12.3-15 is resolved.

In evaluating several of the applicant's shielding calculational packages (which form the basis for some of the ESBWR shielding design reviews), the staff noted that several of these shielding packages referenced the use of shielding codes which DCD Tier 2, Revision 6, did not describe. To obtain additional information on these codes, the staff issued RAI 12.3-14, which requested that the applicant provide the following information with respect to the use of these shielding codes: (1) a description of the function of these codes, (2) a comparison of the capabilities of these codes with the codes that were currently referenced in the DCD, and (3) a justification as to why these codes represent acceptable alternatives to the comparable NRC-approved shielding codes. In response, the applicant amended DCD Tier 2, Revision 7, to provide a detailed description of the functions and capabilities of each of these codes and how they were used in the ESBWR shielding design. The applicant provided a comparison of the capabilities and limitations of these codes with the capabilities of the codes described in the DCD. The staff finds that the applicant's modifications to DCD Tier 2, Revision 7, in response to RAI 12.3-14, provided a suitable comparison of the capabilities of these codes with the codes that are currently referenced in the DCD, and because these capabilities are comparable, the staff agrees with the applicant's justification as to why these codes represent acceptable alternatives to the comparable NRC-approved shielding codes. In addition, the applicant has complied with the guidance contained in Section 12.3.2 of RG 1.70 by adding a reference to these additional shielding codes in the DCD. Based on the applicant's response, RAI 12.3-14 is resolved.

In Section 12.3.2 of the SRP it states that the applicant must describe the methods by which the shielding parameters (including pertinent codes, assumptions, and techniques used in the shielding calculations) were determined. DCD Tier 2, Revision 6, Section 12.3.2.2.2, describes the applicant's shielding design methodology. Table 12.3-1 of DCD Tier 2, Revision 6, describes the shielding codes used to determine the adequacy of the station shielding design. Specifically, the applicant used the point kernel shielding codes PANDORA, QAD, and QAD-CGGP to perform pure gamma dose rate calculations throughout the ESBWR plant. The applicant used PANDORA to determine shield wall thicknesses for those cases that did not involve neutron flux or radiation scattering. The applicant used the QAD codes for equipment

geometries and more complex shields. The applicant used the gamma ray scattering point kernel code GGG to evaluate the adequacy of labyrinth shield designs.

GEH used the discrete ordinate transport code DORT and the Monte Carlo code MCNPX for solving specific radiation transport problems that involved radiation scattering and neutron flux. The applicant used the Monte Carlo skyshine code SKYIII-PC to evaluate the gamma ray dose rate at given detector locations outside of structures (such as the turbine building) housing nitrogen-16 gamma ray sources. The applicant used the codes ORIGEN, EMIR, and NISEIS (specific for nitrogen-16) to prepare the input data (source strength) for the above-described shielding codes. The staff confirmed that the applicant has added a reference to the PANDORA, MCNPX, ORIGEN, NISEIS, and EMIR codes in DCD Tier 2, Revision 7.

The staff finds that the information in DCD Tier 2, Section 12.3.2, adequately addresses the guidance of RG 8.8 and the relevant requirements of 10 CFR Part 20 by establishing shielding requirements that ensure that the exposure of the general public, plant personnel, contractors, and visitors during normal operations, and in the unlikely event of an accident, are limited to levels that are ALARA and within the 10 CFR Part 20 requirements. The staff has verified that several of the codes (i.e., QADF, GGG, DORT) described in this section of the DCD have been approved for use by the NRC. The staff has also confirmed, through performing confirmatory calculations, that other codes described in this section (see RAI 12.3-14) represent acceptable alternatives to the comparable NRC-approved shielding codes. On this basis, the staff finds the information contained in the section acceptable.

12.4.3.3 *Ventilation*

Chapter 9 of the DCD addresses the ESBWR ventilation systems, which are designed to provide adequate heating, cooling, and air supply to areas of the plant. The COL applicant will be responsible for determining the airborne concentrations of radionuclides within the plant serviced by these ventilation systems. DCD Tier 2, Revision 9, Appendix 12A, provides a methodology for determining the airborne concentrations in each room and cubicle. Tier 1 of the DCD provides specific ventilation inspections, tests, analyses, and acceptance criteria (ITAAC).

The ESBWR ventilation systems are designed to protect personnel and equipment from extreme environmental conditions and to ensure that personnel exposure to airborne radioactivity levels is minimized. Furthermore, the design ensures that the dose to control room personnel during accident conditions will not exceed the limits specified in GDC 19. The following design objectives apply to all ESBWR building ventilation systems:

- The systems are designed to make airborne radiation exposures to plant personnel and releases to the environment ALARA. To achieve this objective, the applicant will follow the applicable guidance provided in RG 8.8.

- The concentrations of radionuclides in the air in areas accessible to personnel for normal plant surveillance and maintenance will be below the concentrations that define an airborne surveillance, and maintenance will be below the concentrations that define an airborne radioactivity area, as specified in 10 CFR Part 20, during normal power operation.

The source of airborne radioactivity for a room or area is primarily equipment leakage within the specified area. The ESBWR design incorporates the following features to minimize this leakage and thereby reduce the sources of airborne radioactivity:

- For all areas potentially having airborne radioactivity, the ventilation systems are designed such that, during normal and maintenance operations, airflow between areas is always from an area of low potential contamination to an area of higher potential contamination.

- Negative or positive pressure is used appropriately in plant areas to prevent exfiltration or infiltration of possible airborne radioactive contamination, respectively.

- ESBWR equipment design includes provisions for limiting leaks or controlling the fluid that does leak. This includes piping the released fluid to the sumps and using drip pans with drains piped to the floor drains. For systems containing highly radioactive fluids, drains are hard-piped directly to equipment drain sumps so that contaminated fluid does not flow across the floor to a floor drain.

- Systems containing radioactive fluids are welded, to the most practical extent, to reduce leakage through flanged or screwed connections.

The ESBWR ventilation systems incorporate the following design features to minimize personnel exposures:

- Major HVAC equipment is located in dedicated low radiation areas to minimize exposures to personnel maintaining this equipment.

- HVAC ducting is routed outside pipe chases and does not penetrate pipe chase walls (which would compromise the shielding around the piping).

- HVAC ducting penetrations through walls of shielded cubicles are located to minimize the effects of radiation streaming in adjacent areas.

- HVAC filters are provided with adequate space for maintenance activities, such as servicing and filter changeout. The particulate and HEPA filters can be bagged when being removed from the unit to minimize the spread of contamination. To minimize personnel exposures from radioactivity in the charcoal filters, these filters are allowed to decay to minimum levels and then they are removed by a pneumatic transfer system.

These design criteria adhere to the guidelines of RG 8.8 for maintaining doses ALARA and are acceptable.

DCD Tier 2, Revision 1, did not initially list the maximum radiation source terms in the filter media for those RB and control building ventilation systems designed to operate during accident conditions. To obtain this information, the staff issued RAI 12.4-23. In this RAI, and its supplements, the staff also requested that the applicant describe: (1) the location on plant layout drawings of the RB HVAC filter units and the control building emergency filter unit, (2) the postaccident dose rates from the filter units in areas adjacent to the filter units where personnel may be present, and (3) any design features associated with the filter units to ensure that radiation exposures to personnel resulting from maintenance of these systems is ALARA.

In the response to RAI 12.4-23 and its supplements, the applicant amended DCD Tier 2, Revisions 5 and 6, to describe the locations of the RB HVAC filter units and the control building emergency filter unit and to indicate these locations on the plant layout drawings. The applicant also added tables to the DCD listing the accumulated activities in the HVAC filters during accident conditions and the resulting postaccident dose rates in the various rooms adjacent to

the HVAC filters. The staff had requested that the applicant provide the postaccident dose rates in areas adjacent to the filter units so that the staff could evaluate the resulting potential doses to personnel performing maintenance on these filter units. In the applicant's response to the staff's request, the applicant stated that the rooms in the control building housing the emergency filter units will not be normally occupied during accident conditions. In the RB, the applicant stated that a shield wall located between the RB HVAC filter cubicles will ensure that the dose rate contribution, during normal operation, to personnel performing filter maintenance, from the filter in the adjacent cubicles, will not exceed 250 μSv/h (25 mrem/h). The staff finds that use of this design feature (use of a shield wall between filter units) is consistent with the ALARA guidelines provided in RG 8.8 and is therefore acceptable.

In evaluating the estimated postaccident dose rates provided by the applicant for these HVAC filters, the staff noted that the estimated dose rate in one location under the filter exceeded the dose rate designation for a VHRA (as specified in 10 CFR 20.1602). In RAI 12.4-23 S02, the staff requested that the applicant specify what controls would be implemented in the vicinity of the RB HVAC filters to ensure that postaccident access to this area is restricted. In the response, the applicant modified the radiation zoning designation for the area around these filters to make this a Very High Radiation Zone during postaccident conditions. In addition, the applicant stated that these rooms are not occupied during the design basis LOCA event.

The staff finds the applicant's response to this RAI and its supplements acceptable, because the applicant designated the location of the RB HVAC filter units and the control building emergency filter units and listed the postaccident dose rates from the filter units in areas adjacent to the filter units where personnel may be present, as requested by the staff. In addition, the applicant described those plant design features and access controls associated with the RB HVAC filter units and the control building emergency filter unit that are intended to maintain personnel doses ALARA during maintenance of these systems. The staff finds that the described design and administrative features implemented to maintain radiation exposures to personnel resulting from maintenance of these systems ALARA to be acceptable because these features are consistent with the ALARA guidelines provided in RG 8.8. RAI 12.4-23 was being tracked as an open item in the SER with open items. Based on the applicant's response, RAI 12.4-23 is resolved.

DCD Tier 2, Revision 1, did not initially list the maximum radiation source terms in the filtration units (demineralizers) for the RB, radwaste building, and FB liquid waste systems. To obtain this information, the staff issued RAI 12.4-24. In this RAI and its supplements, the staff also requested that the applicant describe the estimated dose rates from the filtration units in areas adjacent to these units where personnel may be present and list any design features associated with these filtration units to minimize personnel doses. In response to RAI 12.4-24 and its supplements, the applicant amended DCD Tier 2, Revisions 4 and 6, to provide the estimated activities and dose rates associated with these demineralizers. The applicant also described several design features and provisions included in the design to ensure that the radiation exposures resulting from maintenance (filter change out) of these systems is ALARA. Some of these provisions include automated demineralizer filling, draining, backwashing, and resin transfer operations; separation of demineralizers into separate rooms to minimize dose rates from adjacent units; and location of system valves and controls on the outside shield walls of the cubicles containing the demineralizers. These features, designed to minimize personnel doses during the operation of the demineralizers, are consistent with the ALARA guidelines provided in RG 8.8 and are therefore acceptable. RAI 12.4-24 was being tracked as an open item in the SER with open items. Based on the applicant's response, RAI 12.4-24 is resolved.

The staff concludes that the ESBWR ventilation systems are designed to protect personnel and equipment from extreme environmental conditions and to ensure that personnel exposure to airborne radioactivity levels is minimized and maintained ALARA, consistent with the guidance in RG 8.8, and within the applicable limits of 10 CFR Part 20. Furthermore, the design ensures that the dose to control room personnel during accident conditions will not exceed the limits specified in GDC 19. On this basis, the staff finds the design of the ESBWR ventilation systems to be acceptable.

12.4.3.4 *Area Radiation and Airborne Radioactivity Monitoring Instrumentation*

DCD Tier 2, Revision 9, Section 12.3.4 addresses radiation monitoring in the following five categories:

(1) Area radiation monitors needed for accident situations, in accordance with RG 1.97, and area radiation monitors for normal operations to ensure that doses are ALARA which meet the criteria in ANSI/ANS-HPSSC-6.8.1, "Location and Design Criteria for Area Radiation Monitoring Systems for Light Water Nuclear Reactors," May 1981

(2) High-range containment monitors to meet the criteria specified in NUREG–0737, Item II.F.1 (10 CFR 50.34(f)(2)(xvii)(D))

(3) In-plant airborne radioactivity monitors

(4) Effluent radiation monitors

(5) Radiation monitors to monitor for accidental criticality (in accordance with 10 CFR 50.68)

In reviewing DCD Tier 2, Revision 4, Sections 12.3.4.1 and 12.3.4.2, the staff noted that these sections did not specify which RG 1.97 category and accident monitoring type variable each area radiation monitor is required to meet. To obtain this information, the staff issued RAI 12.4-25. In response, the applicant provided the requested information and amended DCD Tier 2, Revision 5, Table 7.1-1. RAI 12.4-25 was being tracked as an open item in the SER with open items. Based on the applicant's response, RAI 12.4-25 is resolved.

The area radiation monitors should comply with the applicable requirements of 10 CFR Part 20, 10 CFR Part 50, and 10 CFR Part 70, as well as the personnel radiation protection guidelines of RGs 1.97, 8.2, and 8.8. The area radiation monitoring system (ARMS) continuously measures, indicates, and records the gamma radiation levels at strategic locations throughout the plant except within the primary containment (which is monitored by the containment monitoring system). Monitor readings, alarm setpoints, and operating status of ARMS are indicated on control room displays. The ARMS is designed to provide early detection and warning for personnel to avoid unnecessary or inadvertent exposure to radiation and to ensure that occupational radiation exposures are maintained ALARA, in accordance with the guidelines stipulated in RGs 8.2 and 8.8. To inform personnel of local dose rates in the area, area radiation monitors include a local readout and audible alarm in addition to readouts and alarms in the main control room. Visible alarms are also located outside each monitored area so that operating personnel can see them before entering the monitored area. Section 12.3-12.4 of the SRP references ANSI/ANS-HPSSC-6.8.1-1981, which provides acceptable guidance on the location and design criteria of ARMS. The location of the area and airborne radioactivity monitors for ESBWR, as described in the DCD, meets the criteria of ANSI/ANS/ HPSSC-6.8.1-1981. Therefore, the staff finds it acceptable.

The provisions of 10 CFR 50.34(f)(2)(xvii)(D) require that each applicant provide instrumentation to measure, record, and read out in the control room the containment radiation intensity (high level). Specifically, Item II.F.1(3) of NUREG–0737 states that the reactor containment should be equipped with two physically separate radiation monitoring systems that are capable of measuring up to 10^5 grays per hour (Gy/h) (10^7 R/h) in the containment following an accident.

DCD Tier 2, Revision 9, Section 12.3.4 states that the ESBWR design includes four gamma sensitive ion chambers within the primary containment to monitor gamma rays during normal, abnormal, and accident conditions. Two redundant sensors are located in the drywell and two in the wetwell. The monitors will be located such that they are widely separated to provide independent measurements, with a large fraction of the containment volume considered in both the wetwell and drywell. In addition, the selection of the location will consider reasonable access for personnel to allow for replacement, maintenance, and calibration of this monitoring equipment. The range of each monitor covers 7 decades from 0.01 Gy/h (1 R/h) to 10^5 Gy/h (10^7 R/h). On, the basis of this information, as well as the information added to Section 7.1.6 of DCD Tier 2 (in the applicant's response to RAI 12.4-28), the staff finds that the ESBWR high-range containment monitors meet the criteria of NUREG–0737, Item II.F.1(3) (consistent with 10 CFR 50.34(f)(2)(xvii)(D)) and follow the guidelines of RG 1.97. The design and qualification of these monitors is consistent with the guidelines of RG 1.97 and NUREG–0737, Item II.F.1(3). The staff, therefore, finds these monitors to be acceptable.

The staff issued RAI 12.4-28 to ascertain whether the two containment high-range monitors listed in DCD Tier 2, Revision 1, Section 12.3.4, meet the criteria of NUREG–0737, Item II.F.1, consistent with 10 CFR 50.34(f)(2)(xvii)(D). In its response, the applicant verified that the containment high-range monitors meet the criteria of NUREG–0737, Item II.F.1, and amended the appropriate plant layout figures in DCD Tier 2, Revision 3, to indicate the location of these monitors in the drywell and wetwell. As a result of the staff's review of the applicant's response, the staff agrees that the two containment high-range monitors meet the criteria of NUREG-0737, Item II.F.1 and therefore finds the applicant's response to this RAI to be acceptable. RAI 12.4-28 was being tracked as an open item in the SER with open items. Based on the applicant's response, RAI 12.4-28 is resolved.

In RAI 12.4-29, the staff requested that the applicant describe the in-plant airborne radiation monitoring system, including the location criteria for and detection sensitivity of the monitors. In response, the applicant amended DCD Tier 2, Revision 3, Section 12.3.4, to state that the in-plant airborne radiation monitoring instrumentation will be located so as to monitor selected local areas and ventilation paths. The applicant also specified the detection sensitivity of these monitors. The staff finds the applicant's response acceptable, because the applicant amended the DCD to provide a description of the in-plant airborne radiation monitoring system, including the location criteria for and detection sensitivity of the monitors, in accordance with the guidance contained in Section 12.3.4 of RG 1.70. RAI 12.4-29 was being tracked as an open item in the SER with open items. Based on the applicant's response, RAI 12.4-29 is resolved.

DCD Tier 2, Revision 7, Section 12.3.4 includes a paragraph which describes the location criteria and sensitivity requirements for the in-plant airborne radiation monitoring instrumentation. As part of RAI 14.3-174 S01, the staff requested that the applicant describe which of the airborne radioactivity monitors meet the sensitivity and location criteria for airborne radioactivity monitors described in DCD Tier 2, Section 12.3.4. The applicant's response to the staff's request did not adequately address which specific airborne radioactivity monitors meet the sensitivity and location criteria for airborne radioactivity monitors described in DCD Tier 2, Section 12.3.4. The staff requested that the applicant modify the DCD to include this

information. In a revised response to RAI 14.3-174 S01, the applicant proposed to modify the subject paragraph in DCD Tier 2, Section 12.3.4 to more accurately describe which airborne radioactivity monitors will be used to measure in-plant airborne radioactivity levels. In this proposed revised DCD paragraph, the applicant stated that portable continuous air monitors (CAMs) will meet the sensitivity and location criteria and that these monitors will therefore be used to provide the airborne radioactivity monitoring to meet requirements for worker protection in the local plant areas. These CAMs will also provide a means to observe trends in airborne radioactivity concentrations. In accordance with the guidance of Section 12.3 of the SRP, the applicant stated that CAMs equipped with local alarms are used in occupied areas, where needed, to alert personnel to sudden changes in airborne radioactivity concentrations. Surveys to assess airborne radioactivity levels are performed by using CAMs and by taking grab samples (using portable low or high volume air samplers) with collection media appropriate to the type of sample being taken. The guidance in Section 12.3 of the SRP also states that airborne radioactivity monitors should be capable of detecting 10 DAC-hours of the most limiting particulate and iodine species equivalent to those concentrations specified in Appendix B of 10 CFR Part 20 from any compartment that has a possibility of containing airborne radioactivity and that normally may be occupied by personnel. In accordance with this guidance, the applicant stated that CAM alarm setpoints will be set at a fraction of the concentration values given in 10 CFR Part 20, Appendix B, Table 1, Column 3, for radionuclides expected to be encountered.

The information contained in the applicant's proposed modification to DCD Tier 2, Section 12.3.4 conforms to the guidance in Section 12.3 of the SRP, and, therefore, the staff finds the use of CAMs to be acceptable to provide the airborne radioactivity monitoring to meet requirements for worker protection in the local plant areas. Based on the applicant's response, RAI 14.3-174 is resolved.

The PRMS continuously samples and monitors airborne radioactivity in effluent releases and ventilation air exhausts for noble gases, air particulates, and halogens. Airborne contamination is sampled and monitored at the stack common discharge, in the off-gas releases, and in the ventilation exhaust from the reactor, radwaste, and turbine buildings. Airborne radioactivity samples will be periodically collected and analyzed for radioactivity. Prior to worker entry, the applicant will utilize portable air samplers to evaluate the airborne radiation levels in those work areas where airborne radiation levels may exceed 10 CFR Part 20 limits. DCD Tier 2, Revision 9, Section 11.5 describes the PRMS in more detail.

SRP Section 12.3 states that the DCD must provide the criteria and methods for obtaining representative in-plant airborne radioactivity concentrations in all work areas. Furthermore, 10 CFR 50.34(f)(2)(xxvii)(D) requires that each applicant provide for the monitoring of in-plant radiation and airborne radioactivity, as appropriate, for a broad range of routine and accident conditions. Specifically, Item III.D.3.3 of NUREG–0737 states that each applicant should provide equipment and associated training and procedures for accurately determining the airborne iodine concentrations in areas within the facility where personnel may be present during an accident. The applicant stated, in DCD Tier 2, Revision 7, Section 12.3.4, that the COL applicant will address the criteria and methods for obtaining representative measurements of radiological conditions, including airborne radioactivity concentrations in work areas (Item III.D.3.3 of NUREG-0737). The COL applicant will also address the use of portable instruments and the associated training and procedures to accurately determine the airborne concentrations in areas within the facility where plant personnel may be present during an accident. The applicant identified these issues as COL Information Item 12.3-2-A.

Both the process radiation monitors and area radiation monitors are located in the fuel storage and associated handling areas in order to detect excessive radiation levels. Process radiation monitors monitor ventilation paths from the fuel storage area and, in addition to isolating the appropriate ventilation path upon receipt of an indication of high radiation, provide indication and alarms to the operator. Area radiation monitors are provided in fuel storage areas to detect high radiation levels and provide visual and audible indication to operating personnel. The staff finds that the use and location of these radiation monitors satisfy the radiation monitoring requirements of 10 CFR 50.68(b)(6); therefore, they are acceptable.

The staff concludes that the area radiation and airborne radioactivity monitors described in DCD Tier 2, Revision 9, comply with the applicable requirements of 10 CFR Part 20, 10 CFR Part 50, and 10 CFR Part 70, as well as the personnel radiation protection guidelines of RGs 1.97, 8.2, and 8.8. These monitors are designed to monitor both area and airborne radioactivity levels in the plant to ensure that doses to plant personnel are maintained ALARA. Therefore, the staff finds that these monitoring systems are acceptable.

12.4.3.5 *Postaccident Access*

DCD Tier 2, Revision 9, Section 12.3.5 lists the areas of the plant that may require access to aid in the mitigation of, or recovery from, the consequences of an accident (referred to as vital areas in NUREG–0737, Item II.B.2). DCD Tier 2, Revision 9, Figures 12.3-43 through 12.3-51, also indicate these vital areas, along with their postaccident radiation zone designations.

As stated in 10 CFR 50.34(f)(2)(vii), an applicant must fulfill the following requirements:

- Perform radiation and shielding design reviews of spaces around systems that may, as a result of an accident, contain accident source term radioactive materials.

- Ensure that the plant design provides adequate access to important areas and protects safety equipment from the radiation environment.

Item II.B.2 of NUREG–0737 provides additional guidance on how an applicant can meet these requirements. Item II.B.2 states that an operator should be able to access any vital area, perform the necessary functions to aid in the mitigation or recovery from an accident, and exit the area without exceeding 5×10^{-2} Sv (5 rem) to the whole body or 5×10^{-1} Sv (50 rem) to the extremities (see GDC 19). The dose rate in areas requiring continuous occupancy should be less than 15×10^{-5} Sv/h (15 mrem/h) averaged over 30 days. DCD Tier 2, Section 12.3.5, states that the doses to access all vital areas following an accident are within regulatory guidelines.

The staff noted, after reviewing the postaccident radiation zone maps provided in the radiation zone layout drawings in DCD Tier 2, Revision 1, Section 12.3, that the postaccident plant layout drawings did not contain all of the information that the staff needed to evaluate these drawings in accordance with the requirements of 10 CFR 50.34 (f)(2)(vii) and the criteria in NUREG-0737, Item II.B.2. In RAI 12.4-31, the staff asked the applicant to amend the DCD to provide a complete set of postaccident drawings and to indicate on the drawings the location of those systems and components that contain postaccident materials outside of the primary containment, each specific area (not just the general room) requiring access to mitigate the consequences of an accident listed under Item II.B.2 of DCD Tier 2, Revision 1, Table 1A-1 (including the technical support center and health physics facilities), and the personnel access routes to, and egress routes from, these areas.

As part of this RAI, the staff also requested that GEH provide a detailed description of personnel actions to be taken in each area, the significant radiation sources associated with each, and an analysis of the radiation "mission" dose received (including dose received from area access and egress). In RAIs 12.4-32 and 12.4-33, the staff asked that the applicant describe the criteria used in establishing the postaccident radiation zones and vital areas shown on the postaccident zone maps. The applicant incorporated the responses for RAIs 12.4-32 and 12.4-33 into its response for RAI 12.4-31. In response to RAI 12.4-31, the applicant modified DCD Tier 2, Revision 5, to add the requested information. The applicant modified DCD Tier 2, Revision 6, to describe the postaccident access requirements and provide a listing of the areas requiring postaccident access, as specified in NUREG–0737, Item II.B.2. The applicant added Tables 12.3-12 and 12.3-13 to DCD Tier 2, Revision 6, providing a detailed room-by-room listing of the estimated dose rates for access to and egress from each of the designated plant areas requiring postaccident access. The applicant also added Tables 12.3-14 to 12.3-17 to DCD Tier 2, Revision 6, listing the estimated radiation mission doses that would be received to access each of these areas following an accident. In addition, the applicant modified DCD Tier 2, Revision 6, Section 12.3, to add a series of plant layout drawings depicting the expected postaccident radiation zones, the location of the plant areas requiring postaccident access, and a depiction of the personnel access and egress routes to and from these areas. The staff finds that the changes made to the DCD in response to RAI 12.4-31 address the requirements of 10 CFR 50.34 (f)(2)(vii), as they apply to plant shielding of vital areas, and the criteria in NUREG–0737, Item II.B.2. The staff finds the applicant's response to be acceptable because, in accordance with the criteria in NUREG–0737, Item II.B.2, the applicant has identified all the plant vital areas, and has shown that the plant shielding is sufficient to permit an operator to access each of these vital areas following an accident without the mission dose exceeding the dose criteria established in GDC 19. RAIs 12.4-31, 12.4-32 and 12.4-33 were being tracked as open items in the SER with open items. Based on the applicant's response, RAIs 12.4-31, 12.4-32, and 12.4-33 are resolved.

DCD Tier 2, Revision 9, Section 12.3, contains plant radiation zone maps which reflect maximum radiation fields over the course of an accident. The applicant performed analyses that confirmed that the individual exposure limits following an accident would not exceed the applicable requirements of GDC 19. The staff reviewed the applicant's analyses for establishing the mission dose for each of the vital areas and finds these analyses acceptable. The staff finds that the listing of the plant vital areas, along with these analyses, satisfies the requirements of 10 CFR 50.34(f)(2)(vii) as they apply to plant shielding of vital areas.

The information contained in DCD Tier 2, Revision 9, Section 12.3.5, adequately addresses the relevant requirements of 10 CFR Part 20 and 10 CFR 50.34(f)(2)(vii). Therefore, the staff finds the information contained in this section to be acceptable.

12.4.4 Conclusions

Based on its review of the information on radiation protection design (including facility design features, minimization of contamination, shielding, ventilation, and area radiation and airborne radioactivity monitoring instrumentation) supplied by the applicant in DCD Tier 2, Revision 9, as described above, the staff concludes that the applicant has committed to follow the guidelines of the RGs and staff positions outlined in the applicable portions of Section 12.3-12.4 of the SRP. Because the DCD adheres to these RGs, the staff concludes that the design meets the relevant requirements of 10 CFR Part 20, 10 CFR Part 50, and 10 CFR Part 70. The staff finds it acceptable for the material addressed by COL Information Items 12.3-2-A and 12.3-4-A to be

addressed by the COL applicant as part of the COL review. The staff, therefore, finds the material contained in Section 12.3 of DCD Tier 2, Revision 9, acceptable.

12.5 Dose Assessment

12.5.1 Regulatory Criteria

The applicable regulatory criteria and guidance include the following:

- 10 CFR 20.1201

- RG 1.70

- RG 8.19, Revision 1 "Occupational Radiation Dose Assessment in Light-Water Reactor Power Plants—Design Stage Man-Rem Estimates," June 1979.

The staff compared the SRP (Section 12.3-12.4, 1981 version) used during the review of the DCD with the 2007 version of the SRP and incorporated any additional guidance from the 2007 SRP during the staff's subsequent review of Section 12.3-12.4 of the DCD. Therefore, the staff concludes that the version of the SRP used, in combination with the staff's additional review, is appropriate for this review.

12.5.2 Summary of Technical Information

The staff reviewed the completeness of the applicant's dose assessment for the ESBWR facility contained in DCD Tier 2, Revision 9, Section 12.4, using the guidelines in RG 1.70 and the criteria set forth in Section 12.3-12.4 of the SRP. The staff ensured that the applicant had either committed to follow the criteria of the applicable RG and staff positions outlined in the applicable portions of Section 12.3-12.4 of the SRP or provided acceptable alternatives. For cases in which the DCD adheres to these RGs and staff positions, the staff can conclude that the design meets the relevant requirements of 10 CFR Part 20. In addition, the staff selectively compared the applicant's dose assessment for specific functions and activities against the experience of operating BWRs. Radiation exposures to operating personnel must not exceed the occupational dose limits specified in 10 CFR 20.1201.

12.5.3 Staff Evaluation

SRP Section 12.3-12.4 states that the applicant should describe any dose-reducing measures taken as a result of the dose assessment process for specific functions or activities. SRP Section 12.3-12.4 also states that the dose assessment will be acceptable if it documents in appropriate detail (including assumptions made and calculations used) the numbers and types of workers for each work activity, expected dose rates, and projected person-sievert (person-rem) doses, in accordance with RG 8.19.

Initially, the applicant's dose estimates for plant workers contained in DCD Tier 2, Revision 1, Section 12.4 were not consistent with the guidance in RG 8.19. Therefore, the staff issued RAI 12.5-1, which requested that the applicant provide a complete tabulated dose assessment with a scope and detail consistent with the guidance in RG 8.19. The staff stated that this analysis should clearly indicate the basis (i.e., based on recent BWR experience or calculated based on similar tasks in other industries) for the staff-hour and dose-rate estimates assumed, and show how each was adjusted to account for specific ESBWR design features. In

responding to this RAI, the applicant rewrote DCD Tier 2, Revision 5, Section 12.4 in its entirety to address the staff's concerns contained in RAI 12.5-1.

In addition to revising the dose assessment tables in DCD Tier 2, Section 12.4 to be consistent with the guidance of RG 8.19, the applicant revised the text of this DCD section to provide a more thorough analysis of the basis for the dose assessment. The staff finds that the applicant's response to RAI 12.5-1 is acceptable, because the revised dose assessment tables provide a more detailed and thorough analysis of the estimated collective annual plant doses. This revised dose assessment is broken down by job and work function, in accordance with the guidance in RG 8.19, as opposed to the previous less detailed dose assessment, which was broken down by plant area. RAI 12.5 -1 was being tracked as an open item in the SER with open items. Based on the applicant's response, RAI 12.5-1 is resolved.

DCD Tier 2, Revision 9, Section 12.4 describes several dose-reducing measures and design modifications intended to reduce occupational exposures of plant personnel. The DCD describes these measures and modifications as they pertain to each of the following dose assessment categories, as discussed in RG 8.19: (1) reactor operations and surveillance, (2) routine maintenance, (3) waste processing, (4) refueling operations, (5) ISI, and (6) special maintenance.

The RCS in the ESBWR is less complex than the RCSs in current BWR designs. The ESBWR design eliminates reactor coolant recirculation piping and pumps. Since the recirculation lines are the most significant source of radiation in the drywell during shutdown, removing the reactor coolant recirculation piping and pumps will have a significant effect on reducing the dose rates in the drywell outside the primary shield. The use of a steel cylindrical shield around the reactor vessel also serves to further reduce drywell radiation fields. These design features will reduce expected dose rates to personnel performing maintenance work in the drywell. The simplification of systems in the drywell has also resulted in a significant reduction in the total number of valves and instrumentation in the drywell with a resulting expected decrease in maintenance time required for these components.

DCD Tier 2, Revision 9, Section 12.4.1 provides examples of several routine operation and surveillance activities which may be conducted in the RB, FB, radwaste building, or turbine building. The applicant stated that significant reductions in component and instrumentation requirements that result from the emphasis on passive safety systems in lieu of active systems used in current BWRs and the elimination of systems such as the traversing in-core probe (TIP) system lead to a significant reduction in surveillance, monitoring, and testing work.

Consequently, the resulting doses from these activities should be lower than that for a typical BWR. The ESBWR is also expected to have lower general radiation levels during operation, as compared to the typical BWR, because of the use of more stringent water chemistry controls, redundant reactor water cleanup capacity, and the use of low-cobalt materials.

Maintenance time for the ESBWR has been reduced through the simplification of systems in the drywell and RB, which has resulted in a significant reduction in the total number of valves and instrumentation in these areas. Live-load valve packings are used to control valve stem leakage, thereby reducing valve maintenance and worker radiation exposures for valve repairs. Some of the features in the RB designed to facilitate the ease of maintenance include the use of overhead lifts to shorten maintenance times, the provision of ample space around components to allow greater equipment accessibility and permit maintenance of equipment in place, and the ability to remove most of the equipment in the RB for maintenance or replacement, if necessary.

Either the LWMS or the solid waste management system, which are both housed in the radwaste building, process radioactive waste other than spent fuel. More of the ESBWR radwaste operations involve remote handling than in a typical BWR. Generally, much of the radwaste operations are performed remotely and are controlled by operators from the radwaste control room, which is located in a low dose area. In accordance with the guidelines of RG 8.19, DCD Tier 2, Revision 9, Section 12.4.3 describes the applicant's assumptions in estimating the doses associated with the collection, packaging, and shipment of radwaste quantities. The staff has reviewed the applicant's assumptions and finds them to be reasonable because they incorporate ESBWR radiation zoning levels and are consistent with industry experience.

DCD Tier 2, Revision 9, Section 12.4.4 describes several ESBWR features which serve to reduce the personnel time necessary to perform refueling operations. The use of a special stud tensioner for the 84 RPV head bolts, coupled with the use of other automated equipment available, will reduce the drywell access and RPV disassembly/reassembly exposure times from the 4,500 person-hours typically required at conventional BWRs to approximately 1,200 person-hours for the ESBWR design. During refueling operations, the dryer, chimney/partitions, and chimney head/separator will be transferred underwater to decrease personnel exposures during refueling operations. In accordance with the guidelines of RG 8.19, DCD Tier 2, Revision 9, Section 12.4.4, describes the applicant's assumptions in estimating the doses associated with the refueling operations and transfer of spent fuel assemblies into storage casks. The staff has reviewed the applicant's assumptions and finds them to be reasonable because they incorporate ESBWR radiation zoning levels and are consistent with industry experience.

DCD Tier 2, Revision 9, Section 12.4.5 describes several ESBWR features which serve to facilitate ISI. Some of the ESBWR improvements over the BWR/6 product line to facilitate ISI in the drywell include elimination of 14 nozzle inspections, elimination of shield penetration and shield plug removal, and allowance for specific access past insulation areas into inspection areas. The use of natural circulation in the ESBWR simplifies the design within the drywell by eliminating the recirculation loops, pumps, pipe supports, hangers, and shock suppressors. The required inspections of the piping and valve systems normally associated with active safety systems, such as high-pressure coolant injection, low-pressure coolant injection, residual heat removal, and reactor core isolation cooling, are not necessary for the ESBWR, since the ESBWR design has eliminated these active safety systems. The total reactor vessel weld length inspection for the ESBWR may increase by up to 40 percent as compared to a conventional BWR because of the larger reactor vessel used in the ESBWR. However, the use of modern robotic methods for vessel ISI should result in lowered effective dose rates for this inspection. Some of the additional features incorporated in the ESBWR design to reduce ISI time and lower personnel doses include use of standoff mirror-type insulation around the reactor vessel, use of remote-operated mechanical devices for inspection of the RPV body and nozzle welds, removable pipe insulation, and provision for additional ISI operations laydown space. Overall, the applicant estimates that the person-hours expended for ISI for the ESBWR will be reduced by almost a factor of 2 (or approximately 1,500 hours) from those hours expended at conventional BWRs.

DCD Tier 2, Revision 9, Section 12.4.6 describes the primary special maintenance jobs that will be performed for the ESBWR design, as well as several design features incorporated in the ESBWR design to minimize the doses associated with the performance of these special maintenance jobs. The applicant defines special maintenance as maintenance that goes beyond routine scheduled maintenance or maintenance that cannot be performed without significant expenditure of resources in nonnegligible radiation fields. The applicant has

estimated that the doses attributed to special maintenance work for the ESBWR will account for approximately 40 percent of the total estimated annual ESBWR occupational dose. Since the dose associated with special maintenance work accounts for such a large percentage of the total estimated dose at the ESBWR, this section of the DCD contains a detailed description, by plant area, of the ESBWR features designed to reduce the doses associated with special maintenance work.

As stated earlier, the deletion of the recirculation pumps and associated piping will have a major effect on reducing the dose rates in the drywell. In addition, because the drywell is inaccessible during normal operation, special maintenance activities are primarily conducted in the drywell during refueling outages. The applicant estimates that main steam isolation valve (MSIV) maintenance times will be reduced by approximately 50 percent as compared to conventional BWRs through the use of MSIV overhauling devices, use of main steamline plugs, and an improved MSIV seat grinding system. The use of fine motion control rod drives (FMCRDs) in the ESBWR, as opposed to the hydraulic systems used in most BWRs, simplifies component maintenance and results in lower dose rates associated with this maintenance. The ESBWR design replaces the conventional TIP system with fixed in-core detectors for calibrating the local power range monitors. This design eliminates maintenance, and the resulting radiation exposure, on the complex TIP drive and indexer mechanisms currently in use. In addition the potential radiation exposure associated with the TIP “backing out” events (i.e., the complete withdrawal from the reactor core of the freshly irradiated TIP probe into the drive housing) is eliminated.

The RB has been arranged to take advantage of the reduced quantity of equipment associated with the simpler ESBWR systems by making equipment more accessible, thereby facilitating improved access control and maintenance. Laydown space is provided for periodic inspections. Lifting points, monorails, and other installed devices are provided to facilitate equipment handling and minimize the need for rerigging individual equipment movements. A special low-dose area of the RB has been designated for rebuilding of the FMCRD drive units.

Any significant turbine maintenance work in the turbine building will be conducted when the unit is shut down due to the radiation levels from nitrogen–16 in this building during unit operation. Although the turbine and generator systems for the ESBWR are larger than those at conventional BWRs, the applicant estimated that, because of the use of improved valves, maintenance jigs, automated devices, as well as the benefits of using titanium or stainless steel condenser tubes, the total hours required for special maintenance on the pumps and valves for these systems will not increase.

As discussed above, the ESBWR design incorporates several improvements over current operating BWR designs. These improvements are intended to significantly reduce the personnel exposure associated with operational and maintenance activities. The occupational radiation exposure resulting from unscheduled repairs on valves, pumps, and other components will also be lower for the ESBWR than for current plant designs because of the reduced radiation fields, increased equipment reliability, reduced number of components relative to currently operating plants, improved water chemistry controls, and low cobalt usage.

During the staff’s review of DCD Tier 2, Revision 1, Table 12.4-1, the staff noted that the average dose rate assumed in this table for activities in the radwaste building appeared to be low for the typically high-dose jobs listed. In RAI 12.5-6, the staff requested that the applicant provide a basis for these numbers. In response to this RAI and RAI 12.5-1, the applicant submitted an entirely revised DCD Tier 2, Revision 5, Section 12.4, including a revised dose

assessment contained in Tables 12.4-1 through 12.4-7. After reviewing the revised DCD Tier 2, Section 12.4 the staff finds the revised dose assessment table to be acceptable because this dose assessment provides a more detailed and thorough analysis of the estimated collective annual plant doses for the various job and work functions, which conforms to the guidance provided in RG 8.19. In addition, the staff finds this dose assessment to be reasonable because it incorporates ESBWR radiation zoning levels and is consistent with industry experience. RAI 12.5-6 was being tracked as an open item in the SER with open items. Based on the applicant's response, RAI 12.5-6 is resolved.

In reviewing the information contained in DCD Tier 2, Revision 4, Table 12.4-1, the staff requested that the applicant provide a basis for the annual job time estimates (in person-hours) shown. The staff issued this request to the applicant as RAI 12.5-8. However, in the applicant's response to related RAIs 12.5-1 and 12.5-6, described above, the applicant (in Revision 5 to the DCD), submitted a completely revised DCD Tier 2, Section 12.4. Since Table 12.4-1 (containing the original dose assessment) was also revised as part of this DCD revision, the annual job time estimates contained in the initial version of Table 12.4-1 (on which RAI 12.5-8 was based) were also revised as part of the revised dose assessment (now contained in Tables 12.4-1 through 12.4-7). The revised dose assessment provides a more detailed and thorough analysis of the estimated collective annual plant doses for the various job and work functions. The applicant stated that this revised dose assessment derived the worker time estimates listed in the revised DCD Tier 2, Revision 5 Tables 12.4-1 through 12.4-7 from an estimate of the number of persons necessary to perform each task, the time required to perform the task, and the frequency at which the task might be performed. The applicant stated that it had obtained these data from a number of sources, including operating plants, General Electric maintenance recommendations, design lifetimes for some components, arrangement drawings, and engineering judgment based on discussions with personnel experienced in nuclear plant maintenance and refueling. The applicant then compared these estimates with work permit databases obtained from operating BWR units to confirm that the manpower estimates were reasonable and that typical tasks were not omitted.

The staff finds the applicant's response is consistent with the guidance of RG 8.19, which states that such estimates should be based on operating experience at similar plants and, to the extent possible, should include consideration of the design of the proposed plant, taking into account the effect of any dose-reducing design changes. The staff has evaluated the applicant's revised dose assessment and finds it to be reasonable because it incorporates ESBWR radiation zoning levels which take into account the ESBWR's dose reduction features and is based on dose and worker hour data that are consistent with industry experience. RAI 12.5-8 was being tracked as an open item in the SER with open items. Based on the applicant's response, RAI 12.5-8 is resolved.

Based on all of the design improvements and dose reduction features described above, GEH estimated that the cumulative annual dose for operating an ESBWR plant will be 0.845 person-Sv (84.5 person-rem). This estimate is consistent with the Electric Power Research Institute design guideline of 1.0 person-Sv (100 person-rem) per year and compares favorably with the average current BWR experience (the 2008 average collective dose for U.S. BWRs was 1.29 person-Sv (129 person-rem)).

12.5.4 Conclusions

The staff finds that the dose assessment for the ESBWR complies with the guidelines in RGs 1.70 and RG 8.19, as well as the criteria in the applicable portions of Section 12.3-12.4 of

the SRP. This dose assessment also meets the intent of RG 8.19. By addressing the anticipated occupational radiation exposures resulting from normal and anticipated inspection and maintenance, and by incorporating design features to reduce occupational radiation exposures, the applicant has shown that the ESBWR is designed to operate within the occupational dose limits specified in 10 CFR 20.1201. The staff, therefore, finds the material contained in DCD Tier 2, Revision 9, Section 12.4 to be acceptable.

12.6 Operational Radiation Protection Program

12.6.1 Regulatory Criteria

The applicable regulatory criteria and guidance include the following:

- 10 CFR 20.1101
- 10 CFR 50.34(f)(2)(xxvii)
- NUREG–0737
- RG 1.206

The staff performed a comparison of the SRP (Section 12.5, 1981 version) used during the review of the DCD with the 2007 version of the SRP. For Section 12.5, the staff considered the additional 2007 guidance in its review of the DCD. Therefore, the staff concludes that the version of the SRP used, in combination with its additional review, is adequate for this review.

12.6.2 Summary of Technical Information

RG 1.206 states that Section 12.5 of the DCD should contain a description of the applicant's operational radiation protection program. The applicant has stated that the COL applicant will be responsible for describing the operational radiation protection program. The staff will perform a detailed review of the applicant's operational radiation protection program against the criteria set forth in Section 12.5 of the SRP when it is provided by the COL applicant.

The applicant has identified the following COL information items to ensure that license applicants referencing the ESBWR design will address these issues:

- (COL 12.5-1-A) Equipment, Instrumentation, and Facilities—The COL applicant will provide a description of plant health physics equipment, instrumentation, and facilities to the level of detail given in RG 1.206.

- (COL 12.5-2-A) Compliance with 10 CFR 50.34(f)(2)(xxvii) and NUREG–0737, Item III.D.3.3—The COL applicant will provide a description of the portable instruments that accurately measure radioiodine concentrations in plant areas under accident conditions and training and procedures on the use of these instruments.

- (COL 12.5-3-A) Radiation Protection Program—The COL applicant will provide a description of the operational radiation protection program to the level of detail given in RG 1.206. The radiation protection program will consider special shielding features (e.g., lead blankets, lead curtains) and include a description of access controls to VHRAs. The COL applicant will provide a milestone for full program implementation.

12.6.3 Staff Evaluation

The requirements in 10 CFR 20.1101 state that each licensee must develop, document, and implement a radiation protection program commensurate with the scope and extent of licensed activities and sufficient to ensure compliance with the provisions of 10 CFR Part 20. Section 12.5 of RG 1.206 and the SRP state that the operational aspects of an acceptable radiation protection program should address the following three areas:

(1) Organization
(2) Equipment, instrumentation, and facilities
(3) Procedures

DCD Tier 2, Revision 9, Section 12.5, addresses the objectives and design of the ESBWR health physics facilities. The stated objectives of the ESBWR design include health physics facilities and features that provide the capability for the administrative control of the following:

- Activities of plant personnel to maintain personnel exposure to radiation and radioactive materials ALARA and within the guidelines of 10 CFR Part 20

- Effluent releases from the plant to maintain the releases ALARA and within the limits of 10 CFR Part 20 and the plant technical specifications

- Waste shipments from the plant to meet applicable requirements for the shipment and receipt of the material at the storage or burial site

In its review of the plant layout drawings in DCD Tier 2, Revision 1, the staff noted that these layout drawings did not include layouts for the ESBWR service building housing the health physics offices and plant personnel access and egress control points. For the staff to be able to perform a more detailed review of these facilities and to evaluate the adequacy of the radiation control access points, the staff issued RAI 12.6-1 requesting that the applicant add these layout drawings to Chapter 12 of DCD Tier 2. As part of this RAI, the staff requested that the applicant indicate on the layout drawings the health physics offices, control points, contamination control/monitoring stations, changing rooms (men's and women's), and decontamination stations/showers. The staff also requested that the service building layout drawings indicate the personnel paths to access and egress the radiation controlled area by way of the access control points in this facility. In response, the applicant provided the requested information in DCD Tier 2, Revision 5. Based on its review of the applicant's response to this RAI, the staff finds that the information provided conforms to the guidelines contained in RG 8.8 with respect to access control of radiation areas. Therefore, the information is acceptable. RAI 12.6-1 was being tracked as an open item in the SER with open items. Based on the applicant's response, RAI 12.6-1 is resolved.

In the staff's review of DCD Tier 2, Revision 1, Section 12.5.2, the staff noted that shielded rooms will be provided for radioactivity analysis and instrument calibration. In RAI 12.6-2, the staff asked the applicant to describe the radiation sources that these facilities were designed to contain, the shielding provided, and any other protective considerations in the design. In response, the applicant described the types of radiation sources, including calibration standards and calibration irradiators, which will be used in these areas. The applicant stated that routine radiochemical analyses of samples collected from various process streams will also be performed in these rooms. In addition to these radiation sources, radiography sources may be used by the maintenance organization for nondestructive weld testing procedures. Also,

depending on the requirements of the radiation protection organization, personnel bioassay equipment may either be located on-site or this service may be provided by off-site facilities or commercial vendors. The applicant stated that sufficient shielding will be provided for each of these types of sources to ensure that doses to personnel will be maintained well below the applicable dose limits contained in 10 CFR Part 20. The specific shielding configurations chosen, however, will be dependent on the types of instrumentation and number of sources selected by the plant operator. The staff finds the applicant's description of the expected radiation sources to be used in these rooms to be consistent with industry practice and therefore acceptable. The cognizant plant health physics organization will oversee control of these sources and will determine the needed shielding for the specific radiation sources chosen. Because the selection of these radiation sources and the shielding associated with them will be the responsibility of the COL applicant, GEH modified DCD Tier 2, Revision 5, Section 12.5.3, to reference COL Information Item 12.5-3-A. This COL information item states that it will be the responsibility of the COL applicant to provide a listing of any additional radiation sources (including sources for instrumentation and radiography) to a level of detail provided in RG 1.206. This COL information item further states that the COL applicant, through the operational radiation protection program, will be responsible for the use of special shielding or any other protective considerations for the sources that the facility is designed to contain. The staff finds that the information provided in response to RAI 12.6-2 conforms to the applicable guidance contained in RG 1.206 and is therefore acceptable. RAI 12.6-2 was being tracked as an open item in the SER with open items. Based on the applicant's response, RAI 12.6-2 is resolved.

DCD Tier 2, Revision 9, Section 12.5 states that the health physics facilities are located in the service building. Access to radiologically controlled areas of the reactor, fuel, turbine, and radwaste buildings is normally through the entry/exit area of the health physics facilities. The health physics area contains the personnel contamination monitoring equipment, portable radiation survey instrumentation, decontamination shower facilities, and personnel changing rooms.

The applicant stated that the COL applicant is responsible for fully describing the operational radiation protection program. The applicant identified the following three COL information items to describe the additional information to be provided by the COL applicant:

(1) COL Information Item 12.5-3-A, "Radiation Protection Program"

(2) COL Information Item 12.5-1-A, "Equipment, Instrumentation, and Facilities"

(3) COL Information Item 12.5-2-A, "Compliance with 10 CFR 50.34(f)(2)(xxvii) and NUREG-0737, Item III.D.3.3"

12.6.4 Conclusions

As stated in DCD Tier 2, Revision 9, Section 12.6.3, the COL applicant will be responsible for describing the operational radiation protection program (in accordance with COL Information Items 12.5-1-A, 12.5-2-A, and 12.5-3-A) and will present the program for staff's review as part of the COL application. The staff finds it acceptable for the material addressed by these COL information items to be addressed by the COL applicant. When the COL applicant submits the operational radiation protection program, the staff will review it against the guidelines of the RGs and staff positions outlined in Section 12.5 of the SRP. The staff, therefore, finds the material contained in DCD Tier 2, Revision 9, Section 12.5 acceptable.

13.0 CONDUCT OF OPERATIONS

13.1 Organizational Structure of Applicant

The staff of the U.S. Nuclear Regulatory Commission (NRC) reviewed the economic simplified boiling-water reactor (ESBWR) design control document (DCD), Tier 2, Revision 9, Section 13.1, in accordance with NUREG–0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants (LWR Edition)," issued March 2007 (SRP).

In ESBWR DCD Tier 2, Revision 9, Section 13.1, the applicant stated that the combined license (COL) applicant is responsible for describing the organizational structure applicable to conduct of operations. The staff finds this approach to describing and documenting the management and technical support organization acceptable. The COL applicant will describe and document the management and technical support organization. This is identified as COL Information Item 13.1-1-A in DCD Tier 2, Revision 9, Section 13.1.

13.2 Training

The staff reviewed DCD Tier 2, Section 13.2, in accordance with the 2007 version of the SRP. In DCD Tier 2, Revision 9, Section 13.2, the applicant stated that the COL applicant will do the following:

- Provide a description of, and the schedule for, the training program for reactor operators, senior reactor operators, and the licensed operator requalification program.

- Provide a description of, and the schedule for, the training program for nonlicensed plant staff.

- Incorporate the results of reviews of operating experience into training and retraining programs in accordance with the provisions of Three Mile Island (TMI) Action Plan Item (hereafter referred to as TMI Item) I.C.5, in NUREG–0737, "Clarification of TMI Action Plan Requirements."

- Identify the organization responsible for incorporating the results of these reviews into the training and retraining programs.

- Develop a plant staff training program, to cover all phases of plant operation, including preoperational testing and low-power operation, in accordance with the provisions of TMI Item I.G.1.

Based on the above, the staff finds the approach to the development of training programs acceptable. The COL applicant will address the development of the training programs. This is identified as COL Information Item 13.2-1-A and COL Information Item 13-2-2-A in DCD Tier 2, Revision 9, Section 13.2.

13.3 Emergency Planning

13.3.1 Regulatory Criteria

In its review of the ESBWR DCD Tier 2, Revision 9, the staff considered the regulations in Title 10 of the *Code of Federal Regulations* (10 CFR) 52.48, which require, in part, that the application for a standard design be reviewed for compliance with the standards set out in 10 CFR Part 50, “Domestic Licensing of Production and Utilization Facilities,” and its appendices. Specifically, the staff reviewed the design-related information in DCD Tier 2, Revision 9, Section 13.3, against the applicable requirements in 10 CFR 50.34(f), 10 CFR 50.47(b), and Section IV.E of Appendix E to 10 CFR Part 50. In addition, the staff considered the requirements in 10 CFR 52.47(a)(8) and 10 CFR 52.47(a)(21) regarding generic safety issues (GSIs) that are technically relevant to the ESBWR design.

The staff determined compliance with these regulations by using the guidance in 2007 version of SRP Section 13.3 and Section 14.3.10. In addition, the staff used Regulatory Guide (RG) 1.101, “Emergency Planning and Preparedness for Nuclear Power Reactors” (Revision 4), which endorses NUREG–0654, “Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants” (Revision 1); and through it, NUREG–0696, “Functional Criteria for Emergency Response Facilities.” The staff also used Generic Letter (GL) 82-33, “Supplement 1 to NUREG–0737—Requirements for Emergency Response Capability (Generic Letter No. 82-33).”

13.3.2 Summary of Technical Information

In DCD Tier 2, Section 13.3, the applicant stated that emergency planning is not within the scope of the ESBWR design and that the COL applicant will provide the emergency plan. The applicant further stated that the design basis of the standard plant does consider certain design features, facilities, functions, and equipment necessary for emergency planning, and it provided the information that is described below.

The technical support center (TSC) is located in the electrical building and is environmentally controlled to provide room air temperature, humidity, and cleanliness appropriate for personnel and equipment. The TSC will support 26 people and will provide the radiological protection and monitoring equipment necessary to ensure that radiation exposure to any person working in the TSC will not exceed 0.05 sievert (Sv) (5 roentgen equivalent in man [rem]) total effective dose equivalent (TEDE) for the duration of an accident. The TSC will have reliable voice and data communications with the control room and the emergency operations facility (EOF), and reliable voice communications with the operational support center (OSC), NRC operations centers, and state and local operations centers. The TSC will also have a workstation that is capable of displaying safety parameter display system (SPDS) parameters and control room communication of Emergency Response Data System (ERDS) data with the NRC operations center.

The OSC communications system will have at least one dedicated telephone extension to the main control room (MCR), one dedicated telephone extension to the TSC, and one telephone capable of reaching onsite and offsite locations, as a minimum. In DCD Tier 2, Revision 9, Section 13.3 and DCD Tier 2, Table 1.10-1, the applicant identified the following three COL information items relating to emergency planning:

- COL Information Item 13.3-1-A: The COL applicant is responsible for identifying the OSC and the communication interfaces for inclusion in the detailed design of the control room and TSC.

- COL Information Item 13.3-2-A: The COL applicant is responsible for the design of the communication system located in the EOF, in accordance with NUREG–0696. (See TMI Items III.A.1.2(1) and (2) in safety evaluation report [SER, hereafter referred to as this report] Section 13.3.3.1 below.)

- COL Information Item 13.3-3-A: The COL applicant will provide supplies at the site to decontaminate onsite individuals in the service building adjacent to the main change rooms.

The applicant also identified the following six additional COL information items in DCD Tier 2, Table 1.10-1 and the respective DCD sections, which relate to emergency planning:

- COL Information Item 1C.1-2-A: The COL applicant will address the requirements of Bulletin (BL) 2005-02, "Emergency Preparedness and Response Actions for Security-Based Events," regarding emergency preparedness and response actions for security-based events. (See DCD Tier 2, Section 1C.2, and the discussion about BL 2005-02 in Section 13.3.3.1 of this report.)

- COL Information Item 9.5.2.5-1-A: The COL applicant will describe the emergency notification system provisions required by 10 CFR 50.47(b)(6) and will address the recommendations described in IE BL 80-15, "Possible Loss of Emergency Notification System (ENS) with Loss of Offsite Power." (See DCD Tier 2, Revision 9, Section 9.5.2.5, and Section 9.5.2. of this report)

- COL Information Item 9.5.2.5-3-A: The COL applicant will describe the means of communication between the control room, TSC, EOF, state and local emergency operation centers, and radiological field personnel, in accordance with NUREG–0696 and NUREG–0654. (See DCD Tier 2, Revision 9, Section 9.5.2.5, and Section 9.5.2 of this report.)

- COL Information Item 9.5.2.5-4-A: The COL applicant will describe the communication methods from the control room, TSC, and EOF to NRC Headquarters, including establishment of the ERDS, in accordance with NUREG–0696. (See DCD Tier 2, Revision 9, Section 9.5.2.5 and Section 9.5.2 of this report.)

- COL Information Item 9.5.2.5-5-A: The COL applicant will describe the fire brigade radio system, in accordance with Regulatory Position 4.1.7 in RG 1.189, "Fire Protection for Nuclear Power Plants." (See DCD Tier 2, Revision 9, Section 9.5.2.5 and Section 9.5.2 of this report.)

- COL Information Item 14.3-1-A: The COL applicant shall provide emergency planning (EP) inspections, tests, analyses, and acceptance criteria (ITAAC), based on industry guidance. (See DCD Tier 2, Revision 9, Section 14.3.10 and Section 14.3 of this report.)

13.3.3 Staff Evaluation

The staff reviewed ESBWR DCD Tier 2, Revision 9, Section 13.3 and other relevant DCD sections against the applicable requirements and guidance identified above in Section 13.3.1 of this report. The applicant provided certain design-related features and functions of the TSC and

OSC as described in DCD Tier 2, Revision 9, Section 13.3, and identified several COL information items relating to emergency planning in DCD Tier 2, Revision 9, Section 13.3 and elsewhere in the DCD. As part of its review, the staff requested additional information from the applicant.

In request for additional information (RAI) 13.3-1, the staff asked the applicant to provide more detail regarding how it determined the TSC staffing size of 26, in relation to the facility's ability to support additional people during an emergency. In response to RAI 13.3-1, the applicant provided additional information regarding the physical size and areas of the TSC and stated that the available floor space of the TSC is more than sufficient to accommodate a significant increase of personnel without overcrowding. The applicant further stated that the existing staffing assumption of the TSC should be 27, rather than 26 (i.e., 22 utility personnel plus 5 NRC personnel). The staff finds this acceptable, because it provides a TSC that is sufficient to accommodate and support licensee and NRC pre-designated personnel, consistent with the applicable criteria in NUREG–0696. Therefore, RAI 13.3-1 is resolved.

In RAI 13.3-2, the staff asked the applicant to provide more detail concerning the proximity of the TSC to the control room, including whether any security barriers exist between the two facilities. In response to RAI 13.3-2, the applicant stated that the TSC is housed in the electrical building at grade elevation and that access to the TSC from the control room is through an underground personnel tunnel. The entrance to the TSC is approximately 120 meters (131.23 yards) walking distance from the control room—including a stair climb of 6.65 meters (7.27 yards)—which can be easily covered in less than two minutes. There are no security control points along the pathway, except for card reader controlled doors at the control room exit and TSC entrance. The staff finds this acceptable because it identifies a TSC location that is consistent with the applicable criteria in NUREG–0696. Therefore, RAI 13.3-2 is resolved.

In RAI 13.3-3, the staff asked the applicant to provide more detail relating to the displays and instrumentation that will be available in the TSC. In response to RAI 13.3-3, the applicant stated that plant parameters are collected by the essential distributed control and information system (E-DCIS) and the nonessential distributed control and information system (NE-DCIS). Safety-related data is transmitted through the E-DCIS and NE-DCIS. The NE-DCIS has many different functions, including archiving and manipulating data, and can be accessed by TSC computers with connections to the plant NE-DCIS network. The plant does not have a dedicated SPDS; instead, the applicant has incorporated this capability into the NE-DCIS design. The staff finds this acceptable because it adequately describes the availability of technical data systems in the TSC, consistent with the applicable criteria in NUREG–0696. Therefore, RAI 13.3-3 is resolved. Subsequent to the resolution of RAI 13.3-3, the applicant renamed the E-DCIS and the NE-DCIS as the qualified distributed control and instrumentation system (Q-DCIS) and nonsafety-related distributed control and instrumentation (N-DCIS), respectively.

In RAI 13.3-4, the staff asked the applicant to provide more detail regarding the backup power capabilities of the TSC. In response to RAI 13.3-4, the applicant stated that the TSC obtains power from two-hour uninterruptible power supply (UPS) feeds from both trains A and B, which are part of the plant investment protection (PIP) loads. The PIP trains obtain backup power through the nonsafety onsite diesel generators, and the UPS feeds provide emergency lighting and informational display. In addition, a 72-hour UPS powers the E-DCIS systems and a two-hour UPS powers the NE-DCIS. In all cases, either offsite power or either of the two nonsafety onsite diesels can power and recharge these UPS sources. The staff finds this acceptable because it identifies sufficient alternate or backup TSC power sources, consistent with the applicable criteria in NUREG–0696. Therefore, RAI 13.3-4 is resolved.

In RAI 13.3-5, the staff asked the applicant to provide more detail regarding the level of radiological protection provided by the TSC ventilation system. In response to RAI 13.3-5, the applicant stated that each of the 100-percent capacity redundant heating, ventilation, and air conditioning (HVAC) trains has a 100-percent capacity filter train, consisting of high-efficiency particulate air and charcoal filtration, to provide radiological protection to TSC occupants. The TSC HVAC subsystem automatically transfers from its normal operation mode to its radiological mode upon detection of radioactivity at the outside air intakes to limit the introduction of airborne radiation into the TSC, such that the radiation exposure to any person in the TSC will not exceed 0.05 Sv (5 rem) TEDE for the duration of an accident.

In RAI 13.3-5 S01, the staff asked the applicant to provide additional information concerning the level of radiological protection to TSC communications personnel. In its response, the applicant provided additional information relating to the habitability and occupancy of the TSC. The staff found the applicant's response inadequate because it did not clearly address whether the TSC communication personnel would perform their duties in the TSC, or in Communications Room 5189. In DCD Tier 2, Figure 1.2-26, Communications Room 5189 is located outside, and across the hall from, the TSC. (Figure 1.2-26 contains security-related information that is withheld from public disclosure under 10 CFR 2.390.) As such, Communications Room 5189 would not have adequate radiological protection because it is not within the TSC ventilation system.

In RAI 13.3-5 S02, the staff asked the applicant to: (1) clarify an apparent contradiction relating to the ESBWR's compliance with the requirements of NUREG–0696, (2) address concerns relating to the location of Communications Room 5189 outside the TSC, and (3) clarify or retract the statement that TSC functions would be transferred to the EOF if the TSC facility becomes uninhabitable. In its response, the applicant deleted the contradictory language relating to compliance with NUREG–0696 and retracted the statement addressing the transfer of TSC functions to the EOF. In addition, the applicant explained that personnel would perform communications tasks in the TSC offices (where communications devices would be located) and not in Communications Room 5189. In DCD Tier 2, Figure 1.2-26, the applicant also redesignated Communications Room 5189 as "Communications Equipment Room 5189." The staff finds this response acceptable because it provides the appropriate level of radiological protection to TSC personnel, consistent with the applicable criteria in NUREG–0696. Therefore, RAI 13.3-5 is resolved.

During its review, the staff found that the applicant had included information in DCD Tier 2, Section 13.3.3.2, associated with the EOF requirements in 10 CFR 73.55(e) and (f), that addressed the backup power supply for non-portable communications equipment. The staff believed that this information should be addressed in DCD Tier 2, Section 13.6, rather than in Section 13.3.3.2. In RAI 13.3-7, the staff asked the applicant to revise the DCD Tier 2, accordingly. In its response to RAI 13.3-7, the applicant stated that it will address the information in revised Sections 13.6.1.1.4 and 13.6.1.1.7, and it deleted the EOF requirements in DCD Tier 2, Section 13.3.3.2 associated with 10 CFR 73.55(e) and (f). The staff finds this acceptable. Therefore, RAI 13.3-7 is resolved.

In RAI 13.3-8, the staff asked the applicant to address the structural characteristics of the electrical building regarding the wind and flood design criteria in NUREG–0696 for buildings that house the TSC complex. In its earlier response to RAI 13.3-1, the applicant stated that the TSC is housed in the electrical building (at grade elevation), which is constructed of reinforced concrete and classified as nonsafety-related and non-seismic Category (NS). The applicant did not address the electrical building's structural characteristics in terms of winds and floods with a

recurrence frequency of 100 years. In its subsequent response to RAI 13.3-8, the applicant modified the relevant design bases in DCD Tier 2, Table 2.0-1 and other DCD sections, and provided specific wind, flood, and seismic design criteria for the electrical building. The applicant stated that the maximum wind speed used for the electrical building's design exceeds the standard plant site parameter 100-year wind speed, and the ground floor elevation is above the maximum flood level. The staff finds this response acceptable because it adequately describes the TSC structural characteristics, consistent with the applicable criteria in NUREG–0696. Therefore, RAI 13.3-8 is resolved.

In accordance with the requirements in 10 CFR 52.47(b)(1), a design certification application (DCA) must contain proposed ITAAC. During its review, the staff found that the applicant did not propose EP ITAAC. In RAI 14.3-150, the staff asked the applicant to provide ITAAC for those design features, facilities, functions, and equipment necessary for EP, for which the applicant is seeking design certification. In its response to RAI 14.3-150, the applicant stated that site-specific differences make it impossible to develop generic (i.e., applicable to all sites) EP ITAAC; and that the site-specific COL applications will provide the EP ITAAC. In DCD Tier 2, Section 14.3.10, the applicant included COL Information Item 14.3-1-A, which states that the COL applicant shall provide EP ITAAC, based on industry guidance (see Section 13.3.2 of this report).

At the COL stage, 10 CFR 52.80(a) requires a COL applicant that references the certified design to include in the COL application the proposed ITAAC, "including those applicable to emergency planning." Consistent with the 2007 version of SRP, Section 14.3.10, a COL applicant must include in its application any necessary EP ITAAC associated with the proposed emergency response facilities that are not identified in the standard design application. Furthermore, Section C.III.7 of RG 1.206, "Combined License Applications for Nuclear Power Plants (LWR Edition)," states, in part, that the COL applicant must propose a complete set of ITAAC that addresses the entire facility, including EP ITAAC. This is addressed by COL Information Item 14.3-1-A, which states that the COL applicant shall provide EP ITAAC, based on industry guidance. If a referenced standard design application includes EP ITAAC, they would carry forward into the COL application. The staff's evaluation of emergency planning at the standard design application stage allows for, but does not require, EP ITAAC. Therefore, the staff finds that the absence of proposed EP ITAAC in the DCD is acceptable.

13.3.3.1 *Generic Issues*

The 2007 version of SRP Section 13.3 states that the majority of emergency planning requirements associated with new reactor applications are programmatic in nature and supplement physical facilities and equipment. Although the COL applicant must address all aspects of emergency planning, the standard design may address design-related features in support of emergency preparedness and response. Emergency planning features addressed in an application for a standard design certification must be technically relevant to the design, not site specific, and usable for a multiple number of units or at a multiple number of sites.

As required by 10 CFR 52.47(a)(8), an applicant for a standard DC must demonstrate compliance with any technically relevant portions of the TMI requirements set forth in 10 CFR 50.34(f), except paragraphs (f)(1)(xii), (f)(2)(ix), and (f)(3)(v). In DCD Tier 2, Appendix 1A, Table 1A-1, the applicant addressed all TMI action plan items listed in 10 CFR 50.34(f), including those associated with emergency planning, and described the respective resolution for the ESBWR standard design.

In addition, 10 CFR 52.47(a)(21) requires the applicant to propose resolutions of unresolved safety issues (USIs) and medium- and high-priority GSIs identified in NUREG–0933, "Resolution of Generic Safety Issues." These issues must be technically relevant to the design, and identified in the version of NUREG–0933 current on the date six months prior to the application docket date. Consistent with 10 CFR 52.47(a)(21), in DCD Tier 2, Section 1.11, Table 1.11-1, the applicant identified issues that are technically relevant to the ESBWR design. DCD Tier 2, Revision 9, Section 1.11 references NUREG–0933 and its supplements through Supplement 30.

As part of its program to disseminate information on operational reactor experience to the nuclear industry, the NRC issues generic communications when it believes a significant safety-related event or condition at one or more facilities potentially applies to other facilities. The staff typically issues a GL or BL that requires licensees to inform the NRC regarding what actions they have taken or will take to address an event, condition, or circumstance that is both potentially significant to safety and generic. Potential safety issues highlighted in NRC generic communications have resulted in the establishment of a USI or GSI and have also been incorporated into formal regulatory requirements. As required by 10 CFR 52.47(a)(22), an applicant for a standard design certification must demonstrate how operating experience insights have been incorporated into the plant design. In DCD Tier 2, Revision 9, Appendix 1C, in Table 1C-1, and Table 1C-2, the applicant identified GLs and BLs that are potentially applicable to the ESBWR design or operations.

The staff reviewed NUREG–0933 (through Supplement 30) and the generic issues (GIs) identified by the applicant in DCD Tier 2, Tables 1A-1, 1.11-1, 1C-1, and 1C-2. The following provides the staff's evaluation and resolution of the TMI requirement and other GSIs that are applicable to emergency planning and technically relevant to the design basis of the ESBWR standard plant.

TMI Items III.A.1.2(1) and (2): Upgrade Licensee Emergency Support Facilities—TSC and OSC

As discussed in NUREG–0933, TMI Item III.A.1.2 addresses the requirement for licensees to upgrade emergency support facilities by establishing a TSC, an OSC, and a near-site EOF for command and control, support, and coordination of onsite and offsite functions during reactor accident situations. TMI Item III.A.1.2 was resolved through its clarification in GL 82-33 (Supplement 1 to NUREG–0737) (discussed below), and the NRC's issuance of new requirements in 10 CFR 50.34(f)(2)(xxv). Additional requirements associated with emergency facilities appear in 10 CFR 50.47(b) and Section IV.E of Appendix E to 10 CFR Part 50. Specific guidance relating to emergency facilities also appears in NUREG–0696, which is referenced in GL 82-33. NUREG–0696 includes the TSC and OSC requirements in TMI Items III.A.1.2(1) and (2), respectively, such that compliance with NUREG–0696 will resolve TMI Items III.A.1.2(1) and (2).

In DCD Tier 2, Revision 9, Section 13.3, the applicant states that the EOF is not within the scope of the ESBWR standard plant. The staff agrees that the EOF is not within the scope of the ESBWR standard plant because it is an offsite facility (independent of the reactor design) that supports the reactor site during an emergency. As such, the EOF is site specific and not technically relevant to the ESBWR design. Therefore, only the TSC and OSC are technically relevant to the staff's review of the ESBWR design.

While DCD Tier 2, Section 13.3 states that emergency planning is not within the scope of the ESBWR design, the applicant described certain characteristics of the TSC and OSC, indicating that the design basis of the ESBWR standard plant includes design features, facilities, functions, and equipment necessary for emergency planning. In DCD Tier 2, Table 1A-1, the applicant addressed the resolution of TMI Item III.A.1.2(1) and (2) by stating that space for the TSC is included in the ESBWR standard design on the ground floor of the electrical building, and that the space provided is in conformance with NUREG–0696. DCD Tier 2, Table 1A-1 also identifies DCD Tier 2, Figure 1.2-26, which shows the TSC location in the electrical building. In addition, Table 1A-1 states that provisions for an onsite OSC are discussed in DCD Tier 2, Revision 9, Section 13.3, which states (in COL Information Item 13.3-1.A) that the COL applicant is responsible for identifying the OSC.

The staff finds that the applicant has adequately addressed the requirements in 10 CFR 50.34(f)(2)(xxv), consistent with the applicable criteria in NUREG–0696, for an onsite TSC and OSC. Therefore, TMI Items III.A.1.2(1) and (2) are resolved for the ESBWR design.

GL 80-34: Clarification of NRC Requirements for Emergency Response Facilities

GL 80-34, "Clarification of NRC Requirements for Emergency Response Facilities at Each Site," provides NRC requirements for the TSC, OSC, and EOF. The NRC has finalized requirements in GL 80-34 and incorporated them into NUREG–0696, which provides more detailed design and functional criteria.

In DCD Tier 2, Revision 9, Table 1C-1, the applicant states that the ESBWR design includes provisions for a TSC and that DCD Tier 2, Revision 9, Section 13.3 describes the OSC and EOF. DCD Tier 2, Section 13.3, states that the COL applicant is responsible for identifying the OSC and certain other communications interfaces and that the EOF is not within the scope of the ESBWR standard plant design. In addition to addressing certain aspects of the TSC and OSC, the applicant further stated that NUREG–0696 contains the detailed guidance for these facilities and that complying with it is the responsibility of the COL applicant.

The staff agrees that the EOF is not within the scope of the ESBWR design because it is an offsite facility—independent of the reactor design—that supports the reactor site during an emergency. As such, the EOF is site specific and not technically relevant to the ESBWR design. The staff finds that the limited extent to which the applicant has described certain design-related aspects of the TSC and OSC in DCD Tier 2, Revision 9, Section 13.3, is acceptable because the relevant requirements in GL 80-34 are addressed and this level of information is consistent with the applicable regulations and guidance. Therefore, GL 80-34 is resolved for the ESBWR design.

GL 81-10: Post-TMI Requirements for the EOF

GL 81-10, "Post-TMI Requirements for the Emergency Operations Facility," clarifies NRC requirements for emergency support facilities, including TMI Item III.A.1.2. In addition, GL 81-10 states that NUREG–0696 will provide more detailed design and functional criteria for emergency facilities than previously prescribed.

In DCD Tier 2, Table 1C-1, the applicant refers to its response to GL 80-34 (see above) for an evaluation of its compliance with GL 81-10. The staff finds this acceptable because GL 81-10 and GL 80-34 basically address the same emergency facility requirements, all of which have been finalized and incorporated into NUREG–0696. Furthermore, the resolution of TMI

Item III.A.1.2 (discussed above) also addresses the incorporation of emergency facility requirements into NUREG–0696. Therefore, GL 81-10 is resolved for the ESBWR design.

GL 82-33: Supplement 1 to NUREG–0737—Requirements for Emergency Response Capability

GL 82-33 (Supplement 1 to NUREG–0737) clarifies the certain post-TMI requirements for emergency response capability, including TMI Item III.A.1.2 criteria associated with emergency support facilities (i.e., TSC, OSC, and EOF). GL 82-33 also addresses accident monitoring instrumentation, discussed in Sections 7.1.1.3.4 and 7.5.2.3 of this report, and human factors considerations associated with emergency facilities, discussed in Section 18.8 of this report.

GL 82-33 references, and includes the basic facility requirements from, NUREG–0696, which is intended to be used as a source of guidance and information. NUREG–0696 provides the detailed design and functional criteria relating to emergency support facilities and includes the comparable requirements in GL 82-33. As discussed above, in TMI Item III.A.1.2 and GL 80-34, the guidance in NUREG–0696 applies to the TSC and OSC, while the EOF is not within the scope of the ESBWR design.

Since NUREG–0696 includes the TSC and OSC requirements in GL 82-33, the applicability of NUREG–0696 to the TSC and OSC (described in DCD Tier 2, Revision 9, Section 13.3) resolves the comparable (TMI Item III.A.1.2) requirements in GL 82-33. Furthermore, in DCD Tier 2, Revision 9, Table 1C-1, the applicant references Appendix 1A regarding the resolution of GL 82-33. In Table 1A-1 of Appendix 1A, the applicant addressed the resolution of TMI Item III.A.1.2 for the TSC and OSC. Therefore, GL 82-33 is resolved for the ESBWR design, to the extent it relates to the TSC and OSC in TMI Item III.A.1.2.

BL 2005-02: Emergency Preparedness and Response Actions for Security-Based Events

In DCD Tier 2, Appendix 1C, Table 1C-2, the applicant identified BL 2005-02, "Emergency Preparedness and Response Actions for Security-Based Events". The corresponding evaluation result in Table 1C-2 states that BL 2005-02 is site specific and, therefore, not within the scope of a design certification. Furthermore, the COL applicant will address the requirements of BL 2005-02 regarding emergency preparedness and response actions for security-based events. The applicant identified COL Information Item 1C.1-2-A in Appendix 1C and in DCD Tier 2, Table 1.10-1, which states that the COL applicant will address the requirements of BL 2005-02 regarding emergency preparedness and response actions for security-based events.

The staff agrees that the issues concerning emergency preparedness in BL 2005-02 are site specific, because they cover program areas that are not related to the ESBWR design certification, and the COL applicant will address them. These program areas include: (1) security-based emergency classification levels and emergency action levels, (2) NRC notifications, (3) onsite protective measures and actions, (4) emergency response organization augmentation, and (5) the drill and exercise program. Therefore, BL 2005-02 is resolved for the ESBWR design, to the extent it relates to the emergency preparedness program.

13.3.4 Conclusion

On the basis of its review, as described above, the staff concludes that the applicant has adequately addressed the emergency planning design-related features and generic issues for

the ESBWR standard plant. Therefore, the information is acceptable and meets the applicable requirements in 10 CFR 50.34(f), 10 CFR 50.47(b), Section IV.E of Appendix E to 10 CFR Part 50, 10 CFR 52.47(a)(8) and (a)(21), and 10 CFR 52.48.

13.4 Operational Program Implementation

The staff reviewed DCD Tier 2, Revision 9, Section 13.4, in accordance with the 2007 version of the SRP.

RG 1.206 lists operational programs required by regulations that should be included in a COL application. SECY-05-0197, "Review of Operational Programs in a Combined License Application and Generic Emergency Planning Inspections, Tests, Analysis, and Acceptance Criteria," states that COL applications should fully describe these operational programs.

In DCD Tier 2, Revision 9, Section 13.4, the applicant states that the COL applicant should fully describe operational programs, as defined in SECY-05-0197 and RG 1.206. In DCD Tier 2, Revision 6, Section 13.4.1, the applicant identified COL Information Item 13.4-1-A, which states, "The COL Applicant will develop a description of the Operational Programs." RG 1.206 also states that COL applicants, in accordance with Commission direction in the staff requirement memorandum associated with SECY-05-0197, should provide schedules for implementation of operational programs. In DCD Tier 2, Revision 6, Section 13.4.1, the applicant identified COL Information Item 13.4-2-A, which states, "The COL Applicant will provide implementation milestones for Operational Programs that are required by NRC Regulation."

On the basis of its review of DCD Tier 2, Revision 9, Section 13.4, the staff finds that the two COL information items identified in this section commit the COL applicant to developing a description of, and providing implementation milestones for, the operational programs specified in RG 1.206. RG 1.206 contains the operational programs and lists the specific regulations that relate to each program. Therefore, the staff finds the applicant's operational program implementation, as described in DCD Tier 2, Revision 9, Section 13.4, is acceptable.

13.5 Plant Procedures

The staff reviewed DCD Tier 2, Revision 9, Section 13.5, in accordance with the 2007 version of the SRP.

In DCD Tier 2, Revision 9, Section 13.5, the applicant stated the COL applicant is responsible for the following:

- Developing procedures that describe the administrative controls over activities that are important to safety for the operation of the facility

- Developing operating and maintenance procedures

- Developing operating procedures that direct operator actions during normal, abnormal, and emergency operations and that will include plant operations during periods when plant systems and equipment are undergoing test, maintenance, or inspection

- Describing the different classifications of procedures operators will use and the general format and content of the different classifications of procedures

- Describing the program for developing and implementing operating procedures
- Describing the program for developing and implementing emergency operating procedures
- Describing the classifications of maintenance and other operating procedures and the general objectives and character of each class and subclass of procedure

Six COL information items were identified in DCD Tier 2, Revision 9, Section 13.5 to address the plant procedure development plan identified as follows: COL Information Item 13.5-1-A, COL Information Item 13.5-2-A, COL Information Item 13.5-3-A, COL Information Item 13.5-4-A, COL Information Item 13.5-5-A, and COL Information Item 13.5-6-A. The staff finds the approach to procedure development, as described in SER Section 13.5, is acceptable.

13.6 Physical Security

13.6.1 Introduction/Overview/General

This section of the report documents the review of the physical security aspects of the ESBWR DCA submitted to the NRC by GE-Hitachi Nuclear Energy (GEH). DCD Tier 2, Revision 9, Section 13.6 describes the plant's physical security program, including those elements of physical protection and mitigative measures identified as being within the scope of the applicant's design. The description includes the required physical security elements of a DCA and references safeguards topical reports on physical protection and mitigative measures. It describes the design for protecting the plant against acts of radiological sabotage; specifically, the plant layout and protection of vital equipment are in accordance with 10 CFR 73.55, and applicable regulatory guidance. This section of the report incorporates the staff reviews of DCD Tier 2, Revision 9, Section 13.6; DCD Tier 1, Revision 9, Section 2.19, and Table 2.19-1; and the ITAAC for the physical security hardware and referenced safeguards topical reports.

The applicant responded to a total of 162 RAIs and 36 open items. The staff found that all responses regarding regulatory requirements are acceptable. All RAIs and open items are resolved and discussed in a comprehensive physical protection SER containing safeguards information. The DCD and topical reports identify vital equipment and vital areas; describe armed responder positions, physical security attributes (e.g., delay barrier[s] within the ESBWR design scope), and their characteristics; and analyze adversarial scenarios for design-basis threats (DBTs). Because this information is security sensitive, the comprehensive physical protection SER contains safeguards information and is not available for public disclosure. Those persons with the correct access authorization and need-to-know may view the safeguards information version of the physical protection SER, hereafter referred to as the safeguards information (SGI) SER, of the ESBWR which is located in the NRC's Secure Local Area Network, document number ES100016191.

13.6.2 Summary of Application

GEH provided the design description and information related to physical security in DCD Tier 1, Revision 9, Section 13.6, and Section 2.19, and referenced safeguards topical reports.

DCD Tier 1, Revision 9, Section 2.19, describes the design features and ITAAC for physical security hardware for the ESBWR design, and Table 2.19-1, describes the design commitments for physical security hardware that are within the scope of the ESBWR design.

DCD Tier 2, Revision 9, Section 13.6.1, states that the comprehensive security plan is the responsibility of the COL applicant and that the ESBWR design supports compliance with portions of 10 CFR Part 73, because all vital equipment is located in vital areas.

DCD Tier 2, Revision 9, Section 13.6.1.1, describes the composition of a site's physical protection program.

DCD Tier 2, Revision 9, Section 13.6.1.1.1, describes the design of the isolation zone, protected area (PA), controlled access points, and other physical barriers.

DCD Tier 2, Revision 9, Section 13.6.1.1.3, describes the design of aids capable of detecting and alarming attempted unauthorized entry into the PA or any vital area.

DCD Tier 2, Revision 9, Section 13.6.1.1.4, describes the design of security communications systems enabling continuous communication among the continuously manned alarm stations, on-duty security force personnel, and the MCR. Additional systems enable communications between the continuously manned alarm stations and local law enforcement agencies.

DCD Tier 2, Revision 9, Section 13.6.1.1.5, identifies the means to control access of personnel, vehicles, and materials into the PA. Access control measures ensure the positive identification and authorization of personnel and search of personnel, vehicles, and materials before entry into the PA. Additional controls limit access to vital areas to authorized personnel only. These controls include the use of numbered picture badges.

DCD Tier 2, Revision 9, Section 13.6.1.1.6, states that the lighting of the PA meets the requirements of Institute of Electrical and Electronics Engineers (IEEE) Standard (Std) 692, "IEEE Standard Criteria for Security Systems for Nuclear Power Generating Stations," and that the lighting design will comply with the lighting levels required by 10 CFR 73.55(i)(6) or an alternative approach to meeting illumination criteria.

DCD Tier 2, Revision 9, Section 13.6.1.1.7, states that site security systems will be powered from a reliable power supply meeting the requirements of IEEE Std 692 and that the design includes an UPS to power the security systems and nonportable security communications equipment. The security related secondary UPS is located in a vital area.

DCD Tier 2, Revision 9, Section 13.6.1.1.8, states that surveillance test procedures and frequencies include the frequencies needed to self check the safety-related distributed control and information system, as well as surveillance tests that are less frequent but more comprehensive. The COL applicant will identify a milestone for developing these surveillance test procedures and frequencies (COL Information Item 13.6-10-A). Other testing and maintenance procedures for security systems include those for physical barriers. The COL applicant will identify a milestone for developing these other testing and maintenance procedures (COL Information Item 13.6-11-A).

In DCD Tier 2, Revision 9, Section 13.6.2, GEH states that it is submitting a security plan, in accordance with 10 CFR 52.79(a)(35)(i), as a separate licensing document, which contains safeguards information and is protected against unauthorized disclosure in accordance with 10 CFR 73.21.

In DCD Tier 2, Revision 9, Section 13.6.3, GEH refers to its responses to RAIs 13.6-1 and 13.6-2. In response to RAI 13.6-1, the applicant provided a list of vital areas and components,

including location information and the locations of the central alarm station (CAS), secondary alarm station, and security-related emergency power supplies. The secondary alarm station was identified as a site-specific item. Additional details concerning the secondary alarm station are found in the ESBWR security assessment. The list of vital areas is SGI.

In its response to RAI 13.6-2, which asked for the ESBWR security ITAAC, the applicant stated that it would follow the industry-recommended generic ITAAC being developed through the industry's New Plant Security Task Force and approved by the NRC. The ESBWR security ITAAC captures the items that must be addressed by a DC applicant for the final 10 CFR Part 73 rulemaking. COL applicants that reference the ESBWR design ITAAC are responsible for addressing the security-related hardware ITAAC that are not within the scope of the ESBWR design.

13.6.3 Regulatory Basis

The NRC regulations for protecting nuclear power reactors in 10 CFR Part 73 include specific security and performance requirements that, when implemented, are designed to protect nuclear power reactors against acts of radiological sabotage, prevent the theft or diversion of special nuclear material, and protect safeguards information against unauthorized release.

Regulations in 10 CFR 73.1(a)(1) require the establishment of physical protection systems to protect special nuclear material against the DBT for radiological sabotage, and 10 CFR 73.55 describes the required physical protection for licensed activities. Pursuant to 10 CFR 50.34(c)(2) and 10 CFR 52.79(a)(35) and (36), licensees must prepare and maintain security plans that describe the security-related actions they will take to protect their facilities against acts of radiological sabotage. In the case of an applicant who describes the use of mixed-oxide plutonium fuel, the DCA also describes the protection of unirradiated mixed-oxide fuel assemblies.

Specifically, Subpart B of 10 CFR 52.47 requires that information submitted for a DC include performance requirements and design information sufficiently detailed to permit an applicant to prepare procurement specifications and construction and installation specifications. According to 10 CFR 52.48, the NRC will review applications filed under 10 CFR Part 52 for compliance with the standards set forth in 10 CFR Part 73.

The ESBWR design descriptions, commitments, and acceptance criteria for the security features, including the plant's layout and protection of vital equipment, as described in the DCA, are based on meeting the relevant requirements of the following Commission regulations:

- 10 CFR Part 50
- 10 CFR Part 52
- 10 CFR 73.1(a)(1)
- 10 CFR 73.55 Appendix B; Appendix C; Appendix G; and Appendix H
- 10 CFR 73.70(f)
- 10 CFR Part 74
- 10 CFR 100.21(f)

The 2010 version of SRP Section 13.6.2, Revision 1 was used by the staff to complete the physical security design certification review. The following regulations in 10 CFR 73.55 contain specific acceptance criteria:

- Section (e): The licensee shall locate vital equipment only within a vital area, which, in turn, shall be located within a PA, such that access to vital equipment requires passage through at least two physical barriers (as defined in 10 CFR 73.2) that perform their required function in support of the licensee's physical protection program. The physical barriers at the perimeter shall be separated from any other barrier designated as a physical barrier for a vital area within the PA. Isolation zones in outdoor areas adjacent to the physical barrier at the perimeter of the PA permit observation. An intrusion detection system detects penetration or attempted penetration of the PA barrier. Isolation zones and appropriate exterior areas within the PA are illuminated. The MCR has bullet-resistant external walls, doors, ceiling, and floors. Vehicle control measures, which include vehicle barrier systems, protect against the threat of assault by land vehicles.

- Section (g): The licensee shall control all points of personnel and vehicle access into a PA; this includes providing equipment capable of detecting firearms, explosives, incendiary devices, or other items that could be used to commit radiological sabotage, or a visual and physical search, or both. Unoccupied vital areas are locked and alarmed with activated detection systems that annunciate in both the CAS and secondary alarm station upon intrusion into a vital area. The individual responsible for the last access control function (controlling admission to the PA) must be isolated within a bullet-resisting structure.

- Section (i): All alarms required pursuant to this part must annunciate and display concurrently in at least two continuously staffed onsite alarm stations, at least one of which must be protected in accordance with the requirements of the CAS. The CAS must be inside the PA, and the interior must not be visible from the perimeter of the PA. The applicant must design and equip the continuously staffed CAS and secondary alarm station so that a single act cannot disable both. At least one alarm station must maintain the ability to detect and assess alarms, initiate and coordinate an adequate response to an alarm, summon offsite assistance, and provide command and control. The CAS shall be considered a vital area and be bullet-resistant, and associated onsite secondary power supplies for alarm annunciators and nonportable communication equipment must be located within vital areas. Alarm devices and transmission lines must be tamper indicating and be self-checking. Alarm annunciation on CAS/secondary alarm station computer monitoring stations shall indicate the type of alarm and its location. All emergency exits from protected and vital areas shall be alarmed and secured by locking devices.

- Section (j): Each security officer or armed-response individual shall be capable of maintaining constant communications with an individual in each continuously manned alarm station. Conventional telephone and radio- or microwave-transmitted two-way voice communications shall be established with local law enforcement authorities.

- Section (n): Each applicant shall develop test and maintenance provisions for intrusion alarms, emergency alarms, communications equipment, access-control equipment, physical barriers, and other security-related devices or equipment.

The NRC may apply the following regulatory guidance documents:

- RG 1.91, "Evaluations of Explosions Postulated To Occur on Transportation Routes Near Nuclear Power Plants."

- RG 4.7, "General Site Suitability Criteria for Nuclear Power Stations."

- RG 5.12, “General Use of Locks in the Protection and Control of Facilities and Special Nuclear Materials.”

- RG 5.65, “Vital Area Access Controls, Protection of Physical Security Equipment, and Key and Lock Controls.”

- RG 5.7, “Entry/Exit Control for Protected Areas, Vital Areas, and Material Access Areas,” Revision 1.

- RG 5.44, “Perimeter Intrusion Alarm Systems,” Revision 3.

- Information Notice 86-83, “Underground Pathways into Protected Vital Areas, Material Access Areas, and Controlled Access Areas.”

- “Nuclear Power Plant Security Assessment Format and Content Guide,” Information Systems Laboratories.

- SAND 2007-5591, “Nuclear Power Plant Security Assessment Technical Manual,” Sandia National Laboratory.

Section 14.3.12 of this report evaluates the ITAAC acceptance criteria pertaining to physical security that are derived from the following regulations:

- 10 CFR 73.1, as it relates to the prescribed requirements for the establishment and maintenance of a physical protection system and for protection against the DBT of radiological sabotage

- 10 CFR 73.55, as it relates to the requirements for physical protection against radiological sabotage of licensed activities in nuclear power reactors

- 10 CFR 73.70(f), as it relates to the requirements specific to alarm annunciation records

- 10 CFR 52.47(b)(1), which requires that a DCA contain the proposed ITAAC that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the DC will be built and operated in accordance with the DC; the provisions of the Atomic Energy Act of 1954, as amended (the Act); and NRC regulations

The COL applicant referencing a certified design is responsible for the site specific security operational programs to meet the requirements in 10 CFR 50.34(c)(2) or 10 CFR 52.79(a)(35)(i) and 10 CFR 52.79(a)(36)(i), (ii), and (iii). This is satisfied, in part, by describing a physical protection system and administrative programs and procedures for implementing a site specific protective strategy that demonstrates high assurance that the plant is protected against a DBT. The site specific physical protection system must be reliable and available and must implement defense-in-depth to provide a high assurance of protection. The following specific and performance requirements describe the security operational programs and the physical protection system: 10 CFR Part 26; 10 CFR 73.55; 10 CFR 73.56; 10 CFR 73.57; 10 CFR 73.58; and 10 CFR Part 74. Regulations in 10 CFR 52.79(a)(36)(i) or 10 CFR 50.34(d) and Appendix C to 10 CFR Part 73 require COL applicants to submit the security program and planning for a safeguards contingency. The performance and specific requirements in

Appendix B to 10 CFR Part 73 requires COL applicants to submit a training and qualification program for readiness of security personnel and responders.

Within this context, the DC applicant must address those elements or portions of physical protection systems that are considered within the scope of the design. However, the DC applicant may include descriptions of security systems or hardware, with supporting technical bases that go beyond the physical configuration for the scope of the design, provided that it is clearly stated that they are within the scope of the DC.

The staff used SRP Section 14.3.12 to review the applicant's ITAAC submittal. Section 14.3.12 of this report documents the staff's evaluation.

13.6.4 Technical Evaluation

The staff reviewed DCD Tier 1, Revision 9, Section 2.19 and Table 2.19-1; and DCD Tier 2, Revision 9, Section 13.6, and referenced safeguards topical reports.

In its review of the referenced safeguards topical reports, the staff identified areas in which it needed additional information to complete the review of the applicant's physical security design. The applicant responded to the staff's RAIs as discussed below.

The staff reviewed applicant submissions to determine if the GEH consideration of physical security in the ESBWR design was acceptable.

Upon the completion of their review, the staff finds that the applicant adequately addressed regulations and the SRP acceptance criteria that were identified as within the scope of their design.

13.6.5 Combined License Information Items

The staff reviewed the ESBWR descriptions and commitments for COL information items that a COL applicant referencing the ESBWR certified design must address.

13.6.5.1 *Acceptance Criteria*

Regulations in 10 CFR 52.47(b)(1), require a DC applicant to submit the proposed ITAAC that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design certification will be built and operated in accordance with the DC, the provisions of the Act, and the NRC's regulations. Section 14.3.12 of this document contains the staff's evaluation of the ITAAC for physical security hardware.

In addition to ITAAC, the staff also reviewed the following information that was submitted by applicants for the physical security design. The following information was provided by the applicant to meet the acceptance criteria identified in Section 13.6.3 of this report for physical security design certification. The details of this information are provided in the SGI SER for the ESBWR, which is stored in the automated database of the NRC's Secure Local Area Network Electronic Safe, document number ES100016191.

As required by 10 CFR 73.55(e)(9)(i), a DC applicant shall identify vital areas and a list of vital equipment[1], by location.

As required by 10 CFR 73.55(e)(9)(v) and (vi), a DC applicant shall identify the control room as a vital area and secondary power supply (for alarm annunciator equipment and nonportable communications) as within a vital area.

As required by 10 CFR 73.55(e)(9)(iii), a DC applicant shall provide the design of the locks and alarms of all unoccupied vital areas.

As required by 10 CFR 73.55(e)(5), a DC applicant shall provide the design describing the bullet resistance of the control room and the CAS.

As required by 10 CFR 73.55(g)(1)(i)(B), a DC applicant should identify locks used to protect the facility and special nuclear material as manipulative resistant.

13.6.5.2 *Technical Evaluation of Combined License Information*

The staff evaluated the COL information items identified in its review of the ESBWR DCA and contained in DCD Tier 2, Revision 9, Section 13.6.3. COL information items are those physical security requirements from the above six acceptance criteria that are either met partially or are not addressed by the DC applicant. The staff's evaluation determines whether the DCA adequately describes those physical security requirements so that a COL applicant would be able to address them during the COL licensing process. The DC applicant need not identify as COL information items those physical security elements required by regulation. However, for physical security elements partially met in the DCA, the DC applicant should explicitly identify which part of the requirement it will meet and which part the COL applicant referencing the design will be required to meet.

In COL Information Item 13.6-6-A, GEH stated that the COL applicant will provide a milestone for developing a program to control the issuance of security keys, as described in DCD Tier 2, Revision 9, Section 13.6.1.1.5.

In COL Information Item 13.6-7-A, GEH stated that the COL applicant shall describe the CAS and secondary alarm station as equal and redundant, such that all functions needed to satisfy the requirements of 10 CFR 73.55(i)(4) can be performed from either alarm stations.

In COL Information Item 13.6-8-A, GEH stated that the COL applicant shall demonstrate that the design of the security system precludes any single postulated security event that results in a degradation of the site security staff's ability to monitor and direct the response to a security event from either the CAS or secondary alarm station; this includes the power supplies to both alarm stations.

In COL Information Item 13.6-9-A, GEH stated that the COL applicant will identify a milestone for incorporating the provisions for alarm response procedures into the applicable procedures as discussed in DCD Tier 2, Revision 9, Section 13.6.1.1.3.

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1 The term "equipment" in 10 CFR 73.55(e) encompasses structures, systems, and components determined to be vital.

In COL Information Item 13.6-10-A, GEH stated that the COL applicant will identify a milestone for developing the surveillance test procedures and frequencies for Q-DCIS as discussed in DCD Tier 2, Revision 9, Section 13.6.1.1.8.

In COL Information Item 13.6-11-A, GEH stated that the COL applicant will identify a milestone for developing the other test and maintenance procedures as discussed in DCD Tier 2, Revision 9, Section 13.6.1.1.8.

In COL Information Item 13.6-12-A, GEH stated that the COL applicant will identify a milestone for developing a site response strategy to a confirmed security event that provides for taking specific actions as defined in DCD Tier 2, Revision 9, Section 13.6.3.

In COL Information Item 13.6-13-A, GEH stated that the COL applicant will identify a milestone for incorporating the provisions for security alarm response procedures into the applicable procedures as discussed in DCD Tier 2, Revision 9, Section 13.6.1.1.3.

In COL Information Item 13.6-14-A, GEH stated that the COL applicant will identify a milestone for incorporating into applicable procedures the administrative controls for work performed in cabinets for specific systems listed in the ESBWR security strategy as described in DCD Tier 2, Revision 9, Section 13.6.1.1.5.

In COL Information Item 13.6-15-A, GEH stated that the COL applicant will identify a milestone for incorporating the administrative controls for work on specific systems in the security assessment into applicable procedures as described in DCD Tier 2, Revision 9, Section 13.6.1.1.5.

In COL Information Item 13.6-16-A, GEH stated that the COL applicant will provide a site arrangement drawing that shows the location of specific bullet-resistant protected positions with engagement capabilities. In addition, the COL applicant will provide a description of the level of protection provided to security personnel stationed in the bullet-resistant protected positions from the effects of the equipment described in the DBT.

In COL Information Item 13.6-17-A, GEH stated that the COL applicant will provide a site arrangement drawing that shows the location of the PA fence, the isolation zone on either side of the PA fence, the vehicle barrier system, any red zone or delay fences, and any buildings or structures inside the PA that are not part of the certified design. In addition, the COL applicant will identify a milestone for demonstrating that the ESBWR security strategy remains valid.

In COL Information Item 13.6-18-A, GEH stated that the COL applicant will identify a milestone for determining if armed responders require ammunition greater than the amount normally carried to provide reasonable assurance of successful engagement of adversaries. This includes the necessary procedures to assure adequate ammunition is available.

In COL Information Item 13.6-19-A, GEH stated that the COL applicant will identify a milestone for updating the ESBWR security strategy to reflect site specific locations of specific bullet-resistant protected position engagement positions. The report will be updated to demonstrate that the security protective strategy can be implemented as described.

In COL Information Item 13.6-20-A, GEH stated that the features of the physical security system are covered, in part, by the standard ESBWR design, while other features are plant and site specific. Accordingly, the ESBWR standard ITAAC cover the physical plant security system and

address those features that are part of the standard design. The COL Applicant shall provide the plant and site-specific Physical Security ITAAC not covered by DCD Tier 1, Revision 9, Section 2.19.

On the basis of its review of identified COL Information Items 13.6-6-A through 13.6-20-A, the staff finds that these items appropriately address interface requirements between the referenced ESBWR physical protection system design and the COL applicant's design.

13.6.6 Conclusion

The staff finds that GEH has considered and prescribed physical security systems or features in the standard ESBWR design that provide protection against acts of radiological sabotage and theft of special nuclear material. The details of this information are provided in the SGI SER for the ESBWR, which is stored in the automated database of the NRC's Secure Local Area Network Electronic Safe, document number ES100016191. GEH has adequately described the plant layout for physical protection and has identified vital equipment and areas, in accordance with the requirements of 10 CFR 73.55. In Section 14.3.12 of this report, the staff evaluated the technical bases and assumptions related to ITAAC for physical security hardware and finds that they are acceptable.

GEH identified the issues in the following documents as being outside the scope of the ESBWR design: GL 89-007, "Power Reactors Safeguards Contingency Planning for Surface Vehicle Bombs"; GL 91-010, "Explosives Searches at Protected Area Portals"; and GL 91-003, "Reporting of Safeguards Events." The staff agrees that the issues in the above documents are outside the scope of the design and finds the GEH approach acceptable. The staff's review of the design addresses Task Action Plan Item A-29, "Nuclear Power Plant Design for the Reduction of Vulnerability to Industrial Sabotage."

For this stage of the licensing process, GEH has provided reasonable assurance that the standard ESBWR design will ensure adequate protection against acts of radiological sabotage and theft of special nuclear material. GEH has provided sufficient security information to support the issuance of a design certificate.

14.0 VERIFICATION PROGRAMS

14.1 Introduction

This chapter of the safety evaluation report (SER or this report) provides the staff's review of the initial test program (ITP) and the inspections, tests, analyses, and acceptance criteria (ITAAC) of the GE-Hitachi Nuclear Energy (GEH) economic simplified boiling-water reactor (ESBWR) as part of the design certification review being conducted by the U.S. Nuclear Regulatory Commission (NRC) under Title 10 of the Code of Federal Regulations (10 CFR), Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants." The staff is conducting this review in accordance with NUREG–0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants (LWR Edition)," March 2007, (SRP), Chapter 14.

14.2 Initial Plant Test Program for Final Safety Analysis Reports

14.2.1 Regulatory Criteria

According to 10 CFR 52.47(a), the information for the design certification must include performance requirements and design information sufficiently detailed to permit the preparation of acceptance and inspection requirements by the NRC. In accordance with the requirements in 10 CFR 50.34(b)(6)(iii) and 10 CFR 52.79(a)(28), an applicant for an operating license or combined license (COL) shall provide information concerning plans for pre-operational testing and initial operations.

Section 14.2 of Regulatory Guide (RG) 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants," Revision 3, requires that applicants describe the technical aspects of the ITP in sufficient detail to show that the test program will adequately verify the functional requirements of plant structures, systems, and components (SSCs). The test program should also provide administrative controls to conduct the test program, describe the organizations involved in testing and staffing activities, describe measures to ensure compliance with test program RGs, provide for the use of operating and testing experience, and provide for the trial use of the plant operating and emergency procedures.

RG 1.68, "Initial Test Programs for Water-Cooled Nuclear Power Plants," Revision 2, describes the general scope and depth of the ITPs that are acceptable to the staff for light-water-cooled nuclear power plants. As stated in the RG, the ITP should provide assurance through testing that the facility has been adequately designed and provide validation, to the extent practical, of the analytical models and assumptions used to predict plant responses to anticipated transients and postulated accidents.

SRP Section 14.2, issued March 2007, provides guidance and acceptance criteria to the staff for the review of a proposed design certification or COL Applicant's ITP. Since the COL Applicants referencing the ESBWR design certification are committed to SRP Section 14.2, the staff used this guidance document as part of its regulatory criteria for review and acceptance of the design certification applicant's list of COL information items. The COL Applicants are also committed to the ITP guidance in RG 1.206, "Combined License Applications for Nuclear Power Plants (LWR Edition)," C.I.IV, "Verification Programs," Section C.I.14.2, "Initial Plant Test Program," for COL information items.

In accordance with SRP Section 14.2, the applicant's ITP should address programmatic aspects including consideration of organization and staffing; administrative controls governing the ITP; preparation, review, and technical content of test procedures; conduct of the ITP; sequencing of testing steps; review, evaluation, and approval of test results; use of reactor operating and testing experiences; and verification by trial use, to the extent practical, of the adequacy of the facility's operating and emergency procedures.

The staff reviewed the applicant's ITP to determine whether it meets the relevant guidance in RG 1.68 and SRP Section 14.2, as they relate to demonstrating the performance capabilities of SSCs and design features that will be used during normal and abnormal operations.

14.2.2 Summary of Technical Information

The applicant provided the technical information associated with the ITP in design control document (DCD) Tier 2, Revision 9, Section 14.2. This information applies to the pre-operational testing phase as well as to the initial startup testing phase.

DCD Tier 2, Revision 9, Section 14.2.1, presents a general description of the ITP that includes: (1) construction test objectives, (2) pre-operational test objectives, (3) startup test objectives, and (4) organization and staffing. Pre-operational testing is normally conducted before fuel load, whereas initial startup testing begins with the initial fuel load and extends to commercial operation. DCD Tier 2, Revision 9, Section 14.2.1.4, presents the responsibilities of the organizational groups that will participate during the various testing phases of the ITP. The applicant states that as the principal designer of the ESBWR plant, it will be on site to direct the work of the constructor and to offer consultation and overall technical direction.

DCD Tier 2, Revision 9, Section 14.2.2 lists the ITP requirements for the startup administrative manual (SAM) test procedures; administrative requirements for conducting the test program; organizational methods used to review, evaluate, and approve test results; and retention periods for test records. DCD Tier 2, Revision 9, Section 14.2.3 lists the RGs used by the applicant for the development of the ITP.

In DCD Tier 2, Revision 9, Section 14.2.4, the applicant states that the ESBWR plant design has the benefit of the operating and testing experience acquired from the construction of previous boiling-water reactor (BWR) plant designs that are still in operation. In addition, the applicant states that it will use the additional operating and testing experience obtained from NRC licensee event reports, Institute of Nuclear Power Operations (INPO) correspondence, and other industry sources in the development of the ITP.

In DCD Tier 2, Revision 9, Section 14.2.5, the applicant states that it will use the plant operating and emergency procedures, to the extent practicable, during the implementation of the ITP. This approach will facilitate the familiarization of the plant's operating and technical staff with facility operating and emergency procedures and will verify, by trial use, the adequacy of such procedures.

In DCD Tier 2, Revision 9, Section 14.2.6, the applicant provides general guidance, including checks and verification requirements that will be applied during initial fuel loading and initial criticality. These activities include prefuel load, initial fuel loading, precriticality testing, and initial criticality.

In DCD Tier 2, Revision 9, Section 14.2.7, the applicant provides the proposed timetable for completing the ITP, including the schedule for completing the pre-operational test phase before fuel load and the startup and power ascension test phases. The Licensee will provide the test program schedule and the sequence for conducting each phase of the ITP, as stated in DCD Tier 2, Revision 9, Section 14.2.7. The applicant includes in its ITP the general guidance for the generation, review, and approval of procedures, as well as the actual testing and analysis of the results.

In DCD Tier 2, Revision 9, Section 14.2.8, the applicant describes the individual test descriptions for the SSCs and the design features anticipated for the ESBWR standard design. For each test, the section presents a general test purpose, prerequisites, general test method, and acceptance criteria.

14.2.3 Staff Evaluation

The staff reviewed the ESBWR ITP in accordance with the review guidance contained in RG 1.68 and SRP Section 14.2. In DCD Tier 2, Section 14.2, the applicant described the ESBWR ITP, which consists of pre-operational and initial startup tests. Pre-operational tests, which are performed after the construction and installation of plant equipment but before initial fuel loading, demonstrate the capability of the plant systems to meet relevant performance requirements. Startup tests, which begin with initial fuel loading, demonstrate the capability of the integrated plant to meet performance requirements. For each phase of the ITP, a design certification applicant needs to define organizational responsibilities, provide administrative controls for the development of the test program, and provide test abstracts, which include the objectives of each test, as well as a summary of prerequisites, test methods, and specific acceptance criteria. These test abstracts should address the criteria outlined in RG 1.68 and SRP Section 14.2. In addition, the applicant needs to describe how it considered the use of reactor operating and testing experience, the trial use of plant operating and emergency procedures, and conformance with applicable RGs. Conformance of a proposed test program to the above guidelines provides reasonable assurance that the facility can be operated in accordance with its design criteria and in a manner that will not endanger the health and safety of the public.

The staff noted that the applicant provided guidance in the areas of organization and staffing, conformance with RGs, test procedure control, utilization of reactor operating and testing experience, use of plant operating and emergency procedures, and test program scheduling and sequencing. In addition, the applicant provided individual test descriptions, test performance requirements, and acceptance criteria for each pre-operational and startup test. The following sections discuss these areas in detail.

14.2.3.1 *Initial Test Program Objectives*

The staff reviewed the pre-operational and initial startup testing objectives, as described in DCD Tier 2, Revision 9, Section 14.2. The staff noted that the applicant's proposed test program provided controls to: (1) ensure that construction was complete and acceptable, (2) demonstrate the capability of SSCs to meet performance requirements, (3) demonstrate, where practical, that the plant is capable of withstanding anticipated transients and postulated accidents, and (4) evaluate and demonstrate, to the extent possible, knowledge of the operating group about the plant and plant operating procedures, thus providing reasonable assurance that the plant can be brought safely to its rated power and can be safely operated during sustained power operations.

In the pre-operational testing phase description, the staff noted that the applicant provided controls to ensure that: (1) the design specifications and test acceptance criteria are met, (2) baseline test and operating data are obtained for future reference, (3) plant systems operate together on an integrated basis to the extent possible, and (4) plant operating staff obtains practical experience in the operation and maintenance of plant equipment and systems. In addition, the applicant stated that it will assist the COL Applicant with the development, implementation, and evaluation of normal, abnormal, and emergency operating procedures to the extent possible; establishment and evaluation of surveillance testing procedures; and demonstration that plant systems are operational in order to continue to fuel loading and initial startup testing.

In the initial startup testing phase description, the staff noted that the applicant provided controls to ensure: (1) a safe core loading, (2) a safe and orderly approach to initial criticality, and (3) the plant's ability to meet test acceptance criteria during low-power and power ascension testing based on sufficient testing.

In request for additional information (RAI) 14.2-81, the staff asked for information about the construction test objectives in DCD Tier 2, Section 14.2.1.1. Specifically, a staff review of DCD Section 14.2.1 indicated that the objectives of construction tests did not consider the possibility of field engineering changes to SSCs, and the section did not identify how such changes would be documented and reflected in the conduct of field tests and test acceptance criteria. Accordingly, the staff asked that the applicant update the DCD to include a description of the process that it will use to address how field engineering design changes to SSCs will be documented and reflected in the conduct of initial tests to ensure that the as-built plant will be built and operated in accordance with the design certification and in compliance with NRC regulations.

In response, the applicant stated the following:

> The process of controlling and resolving problems encountered during plant testing phases is to be controlled by the quality process described in the Quality Assurance Program Document (QAPD) established by the COL Applicant and maintained by the Licensee. Problems uncovered in testing will be tied to the QAPD through a link in the SAM and will be added to the list of the items this manual will provide.

In accordance with this response, the applicant changed DCD Tier 2, Revision 3, by adding a seventh bullet to the content requirements of the SAM. Specifically, the applicant added the following bullet to DCD Tier 2, Revision 4, Section 14.2.2.1:

- Identifies the quality process to be used to control the resolution of test failures, deficiencies and oversights discovered in the ITP. This program will address the control of any plant modifications required to resolve these deficiencies.

DCD Tier 2, Revision 4, Section 14.2.2.1, also stated, in part, that "[a] SAM is developed and made available to the NRC 60 days prior to the scheduled start of the Preoperational Test Program." The applicant also added the following COL information item related to the SAM in DCD Tier 2, Revision 4, Section 14.2.10:

14.2-1-H Per Subsection 14.2.2.1, the COL holder will make available 60 days prior to the scheduled start of the pre-operational test program, the SAM.

In accordance with SRP Section 14.2 and RG 1.206, the COL Applicant is required to provide the administrative controls governing the ITP. The staff determined that the administrative controls governing the ITP should be included in the SAM and the COL Applicant should provide the SAM during the COL application review phase. The staff noted that ESBWR DCD Sections 14.2.2.1 and 14.2.10 were not consistent with SRP Section 14.2 and RG 1.206, in that the DCD requires the Licensee to provide this information. The staff requested in RAI 14.2-81 S01, that the applicant revise ESBWR DCD Sections 14.2.2.1 and 14.2-10 and COL Information Item 14.2-1-H to be consistent with the requirement that the COL Applicant provide the SAM to the NRC for review and approval.

In response, the applicant stated they did not agree with the requested change. However, the applicant did agree to add a new COL information item requiring the COL Applicant to provide a description of how the ITP administration will be developed. The applicant stated that this includes discussions and description of the process, organizational controls, and requirements that are to be included in the SAM. The applicant also stated in its response that it will change the wording of the SAM from “Startup Administration Manual” to “Startup Administrative Manual” to be consistent with the guidance provided in SRP Section 14.2.

The applicant submitted DCD Tier 2, Revision 5, Section 14.2.10, to add COL Information Item 14.2-1-A, “Description—Initial Test Program Administration,” and COL Information Item 14.2-5-A, “Site Specific Tests,” provided below:

> A description of the initial test program administration is developed and made available to the NRC by the COL Applicant (Subsection 14.2.2.1).
>
> The COL Applicant will define any required site specific pre-operational and startup testing (Subsection 14.2.9).

In DCD Tier 2, Revision 5, Section 14.2.10, the applicant also revised the COL information item for the SAM to state that the a SAM is developed and made available by the Licensee to the NRC 60 days prior to the scheduled start of the pre-operational test program. This was designated as COL Information Item 14.2-2-H. Based on GEH’s response and the changes provided in DCD Revisions 5, RAI 14.2-81S01 was resolved. However, in DCD Tier 2, Revision 6, the applicant revised the COL information item for the SAM to state that the COL Applicant will provide a milestone for completing the SAM and for making it available for NRC inspection and it was designated as COL Information Item 14.2-2-A. The staff determined that the revised COL Information Items 14.2-1-A and 14.2.2-A are acceptable because the staff will have the opportunity to review the description of the ITP administration, and the proposed milestone for completing the SAM as part of the COL application review to verify conformance with RG 1.68.

The applicant also revised four COL information items, as noted in Section 14.2.4 of this report. The staff finds that the changes provided in DCD Tier 2, Revision 5, are acceptable. Therefore, the changes resolve RAI 14.2-81 S01.

In RAI 14.2-82, the staff requested additional information regarding the pre-operational test objectives in DCD Tier 2, Section 14.2.1.2. Specifically, a review of DCD Tier 2, Revision 3, Section 14.2.1 determined that the objectives of the pre-operational test program did not consider operational programs and procedures as prerequisites for fuel loading and did not

identify when such programs need to be approved and in place. In the context of controlling and monitoring radioactive effluents, the programs include the radiological effluent technical specifications (TS) or standard radiological effluent controls (SREC), offsite dose calculation manual (ODCM), process control program (PCP), and radiological environmental monitoring program (REMP). Accordingly, the staff requested that the applicant update the DCD to identify these program documents and state when such documents must be approved and operationally ready for the conduct of pre-operational tests for all associated systems, as prerequisites before fuel loading.

In response to RAI 11.5-47, the applicant revised DCD Tier 2 to require the COL Applicant to fully describe the SREC, ODCM, and REMP listed in DCD Tier 2, Section 11.5.7, and the PCP in DCD Tier 2, Section 11.4.6. Furthermore, the COL information item in DCD Section 13.4.1, Revision 3, requires implementation milestones for all operational programs to be made available to the NRC for inspection before fuel load. In addition, COL Information Item 11.5.7.2 states that the COL Applicant will develop an ODCM that will include programs for monitoring and controlling the release of radioactive material into the environment.

The applicant stated in its response to RAI 14.2-82 that it was globally changing the COL holder items to COL Applicant items in DCD Tier 2, Revision 4. The applicant also updated DCD Tier 2, Sections 11.5.4.5, 11.5.4.6, 11.5.4.7, 11.5.4.8, and various paragraphs of Section 11.5.7 to show "COL Applicant." The applicant does not plan to revise DCD Section 14.2.1 to address COL Applicant issues, since DCD Sections 14.2.2 and 14.2.10 already discuss COL information. On this basis, the staff determined that RAI 14.2-82 is resolved.

On the basis of the above review, the staff finds that the applicant provided a set of objectives for the ITP that are consistent with the regulatory positions contained in RG 1.68 and SRP Section 14.2.

14.2.3.2 *Initial Test Program's Conformance with Regulatory Guides*

The staff reviewed the methodology used by the applicant to verify that the ITP meets the guidance in the RGs. SRP Section 14.2 states, in part, that the applicant should establish and describe an ITP that is consistent with the regulatory positions outlined in RG 1.68. SRP Section 14.2 also lists supplemental RGs that provide more detailed information pertaining to the testing. Appendix A to RG 1.68 references a set of supplemental RGs that provide additional guidance for particular tests during the pre-operational and initial startup phases. The supplemental RGs contain additional information to help determine if performance of the tests in the proposed manner will accomplish the objectives of certain plant tests.

In DCD Tier 2, Section 14.2.3, the applicant listed the RGs used in the development of the ESBWR ITP. In addition, DCD Tier 2, Section 1.9, Table 1.9-21 lists the RGs applicable to the ESBWR design. The staff reviewed the tables mentioned above to ensure that the applicable RGs were included in the development of the ITP. For those instances where the applicant determined that RGs were not applicable to the ESBWR design, or where the applicant proposed an exception to the RGs, the staff reviewed the applicant's justification for the exception to ensure that the scope of the test program remains sufficient.

The staff reviewed the list of RGs that the applicant had determined are not applicable to the ESBWR design and the exceptions to regulatory positions in these RGs. The list includes the following:

- RG 1.37, “Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants,” Revision 1

- RG 1.52, “Design, Inspection, and Testing Criteria for Air Filtration and Adsorption Units of Post-Accident Engineered-Safety-Feature Atmosphere Cleanup Systems in Light-Water-Cooled Nuclear Power Plants” Revision 3

- RG 1.79, “Preoperational Testing of Emergency Core Cooling Systems for Pressurized Water Reactors,” Revision 1

- RG 1.95, “Protection of Nuclear Power Plant Control Room Operators Against an Accidental Chlorine Release,” Revision 1

- RG 1.108, “Periodic Testing of Diesel Generator Units Used as Onsite Electric Power Systems at Nuclear Power Plants,” Revision 1

- RG 1.116, “Quality Assurance Requirements for Installation, Inspection, and Testing of Mechanical Equipment and Systems,” Revision 0-R

The staff determined that RGs 1.52 and 1.108 do not apply to the ESBWR design certification because the ESBWR design does not include Class 1E diesel generators (DGs) or safety-related atmospheric cleanup systems. RG 1.79 applies only to pressurized-water reactors, and therefore does not apply to the ESBWR design. The NRC withdrew RG 1.95, and it is therefore, not applicable to a design certification review. Thus, the staff concludes that with the exceptions to the regulatory positions in RG 1.37 and RG 1.116, the other RGs do not apply to the ESBWR design certification.

The staff also reviewed and evaluated proposed exceptions in RG 1.37 and RG 1.116 to verify that the applicant had adequately justified the alternate regulatory positions for testing. The applicant stated that Table 2-1 of NEDO-11209-04a, “GE Nuclear Energy Quality Assurance Program Description”, Revision 8, includes alternate positions to the requirements described in RGs 1.37 and 1.116 that the NRC had previously approved. The staff reviewed the alternate positions for testing described in the approved GEH Quality Assurance Program Description (QAPD) and determined that these exceptions meet the guidance in RG 1.68. Therefore, they remain acceptable for the ESBWR design certification application.

The staff issued RAI 14.2-37 to seek clarification of the applicability of the supplemental RGs in SRP Section 14.2.II (RG 1.56, “Maintenance of Water Purity in Boiling Water Reactors [for Comment],” Revision 1; RG 1.128, “Installation Design and Installation of Vented Lead-Acid Storage Batteries for Nuclear Power Plants,” Revision 2; and RG 1.136, “Design Limits, Loading Combinations, Materials, Construction, and Testing of Concrete Containments,” Revision 3). It appeared that DCD Tier 2, Section 14.2.3, did not include these RGs. In response, the applicant stated that DCD Tier 2 did list RG 1.56 but inadvertently omitted RG 1.128, which would be included in the next revision to Section 14.2.3. The staff confirmed that DCD Tier 2, Revision 3, did list both RG 1.56 and RG 1.128. In addition, the staff no longer recommends RG 1.136 in SRP Section 14.2.II, Revision 3, as a supplemental RG for the ITP. Because RG 1.68 provides more detailed guidelines for the initial tests, the staff determined that DCD Tier 2, Section 14.2.3 does not need to list RG 1.136. The applicant’s response is therefore acceptable, and RAI 14.2-37 is resolved.

On the basis of the above review, the staff finds that the ESBWR ITP adequately conforms to the general scope and depth of test programs, as described in RG 1.68, and also conforms to the test program regulatory positions stated in SRP Section 14.2. In addition, the staff finds that the applicant has adequately justified the noted exceptions.

14.2.3.3 *Organizational and Staffing Responsibilities*

The staff reviewed organizational and staffing responsibilities associated with the conduct of the ITP. SRP Section 14.2 and RG 1.68 state that "the applicant should provide and define the responsibilities of the organizational units that will carry out the ITP. These responsibilities include designated functions of each organizational unit and general steps to be followed in conducting these activities."

The applicant proposed in DCD Tier 2, Section 14.2.1.4, a startup coordinating group (SCG) composed of representatives of the plant owner/operator, GEH, and others, for the conduct of the ITP. This group will be responsible for planning, executing, and documenting pre-operational and initial startup testing activities. In addition, the applicant stated that it will coordinate, in conjunction with the Licensee, overall technical direction to the station staff including shift personnel, in testing and operational activities in accordance with a SAM. The staff noted that the Licensee will define the responsibilities, authorities, and qualifications for normal plant staff that will be consistent with the ESBWR design, as described in DCD Tier 2, Section 13.2.

In RAI 14.2-16, the staff asked the applicant to include a COL information item to provide complete, detailed information regarding the applicant's responsibilities, authorities, and personnel qualifications for conducting the ITP in accordance with RG 1.68. This is to ensure that the plant owner/operator provided the necessary information to be reviewed by the staff at the time of the COL application.

In response, the applicant revised DCD Tier 2, Section 14.2.9 and added a COL information item. The COL information item will have the Licensee describe: (1) the responsibilities of the organization that will carry out the test program, (2) methods and plans for providing the necessary manpower, (3) the staff responsibilities, authorities, and personnel qualifications for conducting the ITP, and (4) how the SAM is used to govern the administrative controls for conducting the ITP. The staff reviewed the applicant's response to this RAI and DCD Tier 2, Revision 3, Section 14.2.9, and determined that the revised text appropriately included three of the four provisions noted above. Therefore RAI 14.2-16 is resolved. However, as discussed in Section 14.2.3.1 of this report, the COL information item was changed and now requires the COL Applicant to provide a milestone for completing the SAM and making it available for NRC inspection. In DCD Revision 6, this has been relabeled COL Information Item 14.2-2-A.

On the basis of the above review, the staff finds that organizational and staffing responsibilities associated with the conduct of the ITP submitted by the applicant provide adequate guidance and meet the regulatory positions in RG 1.68 and SRP Section 14.2.

14.2.3.4 *Initial Test Program Test Procedures*

The staff reviewed the methodology submitted by the applicant that will be used to develop, review, and approve individual test procedures to ensure that the relevant requirements of RG 1.68 and SRP Section 14.2 are met. SRP Section 14.2 and RG 1.68 specify that test procedures should control: (1) the sequencing of testing steps, (2) the preparation, review, and

approval of test procedures, (3) the use of temporary equipment, and (4) test acceptance criteria. RG 1.68 also states that the ITP should be conducted using test procedures developed and reviewed by personnel with appropriate technical backgrounds and experience. Additionally, RG 1.68 states that the principal design organizations should participate in establishing test performance requirements and test acceptance criteria.

In DCD Tier 2, Section 14.2.2 the staff noted that the applicant provided general guidance for the development and review of test specifications and procedures. The applicant stated that the startup group will conduct the ITP in accordance with a SAM. This manual, to be made available by the Licensee, will: (1) define the format of pre-operational and startup test procedures, (2) delineate the qualifications and responsibilities of the different positions within the startup group, (3) define the review and approval process for both initial procedures and subsequent revisions or changes, and (4) specify the process for the review and approval of test results and for the resolution of failures. The staff also noted that the SAM will include measures to provide approved test procedures to NRC inspection personnel approximately 60 days before the scheduled performance of the pre-operational tests and will include measures to provide approved procedures for power ascension tests to NRC inspection personnel 60 days before the scheduled fuel loading date.

In RAI 14.2-17, the staff asked the applicant to include a COL information item to provide complete, detailed information regarding the development, review, and approval of test procedures in accordance with RG 1.68.

In response, the applicant revised DCD Tier 2, Revision 5, Section 14.2.9 and added a COL Information Item 14.2-2-H for the Licensee to provide a SAM that delineates the development, review, and approval of test procedures per Appendix C to RG 1.68 (see RAI 14.2-81). In addition, the applicant stated in DCD Tier 2, Revision 5, Section 14.2.10, in COL Information Item 14.2-3-H, that the Licensee will make the approved test procedures available to the staff approximately 60 days before their intended use. The staff reviewed the applicant's response to this RAI and DCD Tier 2, Revision 3, Section 14.2.9, and determined that the revised text appropriately includes these provisions. Therefore, RAI 14.2-17 is resolved.

However, as discussed in Section 14.2.3.3 of this report, the applicant revised the COL information item for the SAM in Revision 6 to the DCD. In DCD Tier 2, Revision 6, this has been relabeled from COL Information Item 14.2-3-H to COL Information Item 14.2-2-A. The applicant also revised COL Information Item 14.2-3-H for test procedures to state that the COL Applicant will provide milestones for making available to the NRC approved test procedures satisfying the requirements for the ITP. This has been relabeled from COL Information Item 14.2-3-H to COL Information Item 14.2-3-A.

The staff finds that the general test specification and test procedure guidelines specified in DCD Tier 2, Revision 9, Section 14.2.2, are acceptable for the design certification because the guidelines are consistent with RG 1.68 and SRP Section 14.2. However, development of test specifications and test procedures will require detailed plant-specific design information and review and approval by the Licensee. Because plant-specific design information will be needed, the staff concludes that it is acceptable to defer responsibility for the development of detailed pre-operational and startup test specifications and test procedures to the Licensee.

14.2.3.5 *Utilization of Reactor Operating and Testing Experience in the Development of the Initial Test Program*

The staff reviewed the methodology submitted by the applicant to include reactor operating and testing experience in the development of the ITP. SRP Section 14.2 and RG 1.68 state that the applicant should describe how it used the operating and testing experiences of other facilities in the development of the ITP.

In DCD Tier 2, Section 14.2.4, the staff noted that the applicant considered the use of operational and testing experience gained from previous BWR plant designs, as well as operating and testing experience obtained from NRC licensee event reports, INPO correspondence, and other industry sources. The applicant stated that it has factored these experiences into the design and test specifications for the ITP. In DCD Tier 2, Section 14.2.2, the staff noted that the COL Applicant will be responsible for providing test specifications and test procedures for pre-operational and startup tests for review by the NRC and for the preparation of the SAM, which will contain the processes and standards that govern the activities associated with the plant ITP.

In RAI 14.2-18, the staff asked the applicant to include a COL information item to provide complete, detailed information regarding the utilization of reactor operating and testing experience, in accordance with RG 1.68.

In response, the applicant revised DCD Tier 2, Section 14.2.9, and added a COL information item requesting the Licensee to make a SAM available to the staff 60 days before use which delineates the utilization of previous reactor operating and testing experience in the development of the test procedures, in accordance with RG 1.68. The staff reviewed the applicant's response to this RAI and DCD Tier 2, Revision 3, Section 14.2.9, and determined that the revised text appropriately includes these provisions and is acceptable. Therefore, RAI 14.2-18 is resolved. DCD Tier 2, Revision 3, Section 14.2.9 was then moved to DCD Tier 2, Revision 9, Section 14.2.4. In DCD Revision 9, this is identified as COL Information Item 14.2-2-A.

The staff finds that the applicant has provided adequate ITP administrative controls, except as noted above, for the utilization of reactor operating and testing experience as described in RG 1.68 and SRP Section 14.2. However, development of ITP test procedures will require detailed plant-specific design information and review and approval by the Licensee, and thus, the staff concludes that it is acceptable to defer the review of the utilization of operating and testing experience to the Licensee.

14.2.3.6 *Trial Use of Plant Operating and Emergency Procedures*

The staff reviewed the methodology submitted by the applicant to verify plant operating and emergency procedures during the conduct of the ITP. SRP Section 14.2 states that the applicant should incorporate plant operating, emergency, and surveillance procedures into the test program, or otherwise verify these procedures through use, to the extent practicable, during the ITP.

In DCD Tier 2, Section 14.2.5, the staff noted that the applicant also included provisions to ensure that the plant's normal, surveillance, abnormal, and emergency operating procedures will be used, to the extent practical, throughout the pre-operational and initial startup tests. Additionally, the COL Applicant will be responsible for the SAM. In DCD Tier 2, Section 14.2.2,

the staff noted that the Licensee will be responsible for developing test specifications and test procedures for pre-operational and startup tests.

In RAI 14.2-19, the staff asked the applicant to include a COL information item to provide complete, detailed information regarding the trial use of operating and emergency procedures in accordance with RG 1.68. In response, the applicant revised DCD Tier 2, Section 14.2.9 and added a COL information item (In DCD Revision 6, this is identified as COL Information Item 14.2-2-A), for the Licensee, which states that the Licensee will make available a SAM to the staff, 60 days before use. The SAM will require the development of plant operating and emergency procedures before fuel loading and their application during the test program, consistent with Section C.7 of RG 1.68. The staff reviewed the applicant's response to this RAI and DCD Tier 2, Revision 3, Section 14.2.9, and finds that the revised text appropriately includes these provisions and is acceptable. This resolves RAI 14.2-19.

On the basis of the above review, the staff finds that it is acceptable to defer the trial use of operating and emergency procedures to the Licensee because the development of the ITP test procedures will require detailed plant-specific design information and review and approval by the Licensee.

14.2.3.7 *Initial Test Program Schedule and Sequence*

The staff reviewed the methodology submitted by the applicant that will be used to develop the ITP schedule and sequence. RG 1.68 states that sufficient time should be scheduled to perform orderly and comprehensive testing, to provide for a minimum time of about nine months for conducting the pre-operational testing phase, and to provide a minimum time of about three months for conducting the initial startup testing phase.

The staff noted that in DCD Tier 2, Section 14.2.7, the applicant provided measures for conducting each major phase of the ITP relative to the initial fuel load date. The Licensee will provide a schedule showing the timetable for the generation, review, and approval of procedures, as well as the actual testing and analysis of the results. The applicant also stated that approved test procedures will be available to the staff no later than 60 days before their intended use.

The staff reviewed the controls that will be implemented during the pre-operational and initial startup testing phases. The applicant provided general controls to ensure that during the pre-operational testing phase, testing is performed as systems and equipment availability allows, considering the interdependence of the systems. Additionally, the applicant stated that during the startup testing phase, test sequencing will depend on specified power conditions and intersystem prerequisites.

In RAI 14.2-20, the staff asked the applicant to include a COL information item to provide complete, detailed information regarding the development of the test program schedule and sequence in accordance with RG 1.68.

In response, the applicant revised DCD Tier 2, Section 14.2.9 and added a COL information item (eventually labeled COL Information Item 14.2-3-A) for the Licensee, which states that the Licensee will make available a SAM to the staff 60 days before use, which defines the requirements for the test program schedule that is consistent with Section C.5 of RG 1.68 and that the test sequence is consistent with Appendix A to RG 1.68. The staff reviewed the applicant's response to this RAI and DCD Tier 2, Revision 3, Section 14.2.9, and determined

that the revised text appropriately includes these provisions and is acceptable. Therefore, RAI 14.2-20 is resolved.

In DCD Revision 6, the applicant revised the COL information item for the testing schedule to state that the COL Applicant will provide a milestone for completing the detailed testing schedule and will make it available to the NRC. This has been relabeled COL Information Item 14.2-3-A in Revision 6 of the DCD.

On the basis of the above review, the staff finds that the guidance provided by the applicant is consistent with the criteria contained in RG 1.68 and SRP Section 14.2. Since the Licensee is designated as responsible for the test program schedule, the staff finds that it is acceptable to defer the detailed test program schedule and sequence to the Licensee. The COL Applicant will provide a milestone for completing the detailed testing schedule and making it available to the NRC (Subsection 14.2.7) (COL Information Item 14.2-4-A).

14.2.3.8 *First-of-a-Kind Tests*

SRP Section 14.2 and RG 1.68 state, in part, that "if new, unique, or first-of-a-kind (FOAK) principal design features will be used in the facility, the in-plant functional testing requirements necessary to verify their performance need to be identified at an early date to permit these test requirements to be appropriately accounted for in the final design."

In RAI 14.2-95, the staff noted that in DCD Section 14.2.8.1 and Section 14.2.8.2, the applicant did not identify any pre-operational, startup, and power ascension tests that are FOAK tests in the ESBWR design. The staff requested additional information on pre-operational, startup, and power ascension tests that are FOAK tests in the ESBWR design.

In response, the applicant agreed that the ESBWR does have FOAK testing associated with the new design. The applicant identified the following FOAK tests:

- Reactor precritical heatup with reactor water cleanup/shutdown cooling (RWCU/SDC)
- Isolation condenser system (ICS) heatup and steady-state operations
- Power maneuvering in the feedwater temperature operating domain
- Load following

The applicant also added a new description of the power ascension test in DCD Tier 2, Section 14.2.8.2.35, and included this new information in DCD Tier 2, Table 14.2-1. The applicant also identified augmented FOAK tests in DCD Tier 2, Sections 14.2.8.2.7 and 14.2.8.2.11. The applicant added these FOAK tests in DCD Tier 2, Revision 5; therefore, this part of RAI 14.2-95 is resolved.

The staff found that some pre-operational test abstracts on new passive design systems in the ESBWR design, such as the gravity-driven cooling system (GDCS) and the passive containment cooling system (PCCS) are also FOAK tests. RAI 14.2-95 S01 requested the applicant to identify these test abstracts as FOAK tests in the ESBWR design.

In response, the applicant added the following information in DCD Tier 2, Revision 5, Section 14.2.8.1.64:

> The PCCS is a unique ESBWR design for passive containment cooling in post accident conditions. The system consists of multiple loops or trains for

> redundancy. The system will not have any special, one unit only, testing in Subsection 14.2.8.2.35 and will not have any preoperational startup testing in Subsection 14.2.8.2. All plants will perform a preoperational test in accordance to this section.

The applicant also added the following information in DCD Tier 2, Revision 5, Section 14.2.8.1.65:

> The GDCS is a unique ESBWR passive cooling system to provide gravity driven flow into the vessel for emergency core cooling in LOCA conditions. This system will not have any special, one unit only, testing in Subsection 14.2.8.2.35 and will not have any operational startup testing in Subsection 14.2.8.2. All plants will perform a preoperational test in accordance to this section.

The staff finds that the applicant adequately addressed these pre-operational tests as unique FOAK tests for the ESBWR design; therefore, RAI 14.2-95 S01 is resolved.

In RAI 14.2-101, the staff requested the applicant to revise the DCD to classify the following FOAK tests in Section 14.2 as Tier 2*:

- 14.2.8.2.35.1 Reactor Pre Critical Heatup with RWCU/SDC
- 14.2.8.2.35.2 ICS Heatup and Steady State Operation
- 14.2.8.2.35.3 Power Maneuvering In the FW Temperature Operating Domain
- 14.2.8.2.35.4 Load Maneuvering Capability
- 14.2.8.2.35.5 Defense-in-Depth Stability Solution Evaluation Test

In DCD Tier 2, Revision 6, Section 14.2.8.2.35 the DCD applicant bracketed and italicized all of the test abstracts in Section 14.2.8.2.35 to designate them as Tier 2*. Prior NRC approval is required to change Tier 2* information. The staff finds this change acceptable and considers this RAI resolved. See Section 14.2.3.11 of this report for additional details.

14.2.3.9 *Initial Fuel Loading and Initial Criticality*

The staff reviewed the measures provided by the applicant that will be used during initial fuel loading and initial criticality. RG 1.68 and SRP Section 14.2 provide general guidance on the conduct of the ITP after the completion of pre-operational testing. As stated in the regulatory guidance, initial fuel loading and precritical tests ensure that: (1) initial core loading is safe, (2) provisions are in place to maintain a shutdown margin, and (3) the facility is in a final state of readiness to achieve criticality and to perform low-power testing.

In DCD Tier 2, Section 14.2.6, the applicant included provisions for pre-fuel load checks, initial fuel loading, precriticality, and initial criticality in accordance with RG 1.68 and SRP Section 14.2. The staff noted that these provisions included TS compliance, proper verification of water level and chemistry, calibration and response of nuclear instrumentation, shutdown margin verifications at predetermined intervals, and control rod functionality tests. These controls are consistent with the regulatory positions in RG 1.68.

In RAI 14.2-36, the staff requested that the applicant list all tests in the table of contents. In response, the applicant agreed to revise the table of contents to list the pre-operational test procedures in Section 1.2.8.1 and the general description of startup tests in Section 14.2.8.2. Therefore, RAI 14.2-36 is resolved.

On the basis of the above review, the staff concludes that the ITP adequately addresses the initial fuel loading and initial criticality testing and meets the associated guidance in RG 1.68 and SRP Section 14.2. The initial startup testing description in Section 14.2.3.11 of this report offers more detail.

14.2.3.10 *Preoperational Test Descriptions*

In DCD Tier 2, Section 14.2.8.1, the applicant provided 65 test abstracts for the pre-operational testing phase. For each of the pre-operational test abstracts, the staff reviewed the test description, purpose, prerequisites, general test acceptance criteria, and test methods to verify conformance with NRC regulatory guidance. The following is a list of the pre-operational test abstracts described in DCD Tier 2, Revision 9, Section 14.2.8.1:

14.2.8.1.1	Nuclear Boiler System (NBS) Preoperational Test
14.2.8.1.2	Feedwater Control System (FWCS) Preoperational Test
14.2.8.1.3	Standby Liquid Control System (SLCS) Preoperational Test
14.2.8.1.4	Control Rod Drive (CRD) System Preoperational Test
14.2.8.1.5	Rod Control and Information System Preoperational Test
14.2.8.1.6	Safety System Logic and Control Preoperational Test
14.2.8.1.7	Distributed Control and Information System (DCIS) Preoperational Test
14.2.8.1.8	Leak Detection and Isolation System (LD&IS) Preoperational Test
14.2.8.1.9	Reactor Protection System (RPS) Preoperational Test
14.2.8.1.10	Neutron Monitoring System (NMS) Preoperational Test
14.2.8.1.11	Plant Automation System (PAS) Preoperational Test
14.2.8.1.12	Remote Shutdown System Preoperational Test
14.2.8.1.13	RWCU Cooling System Preoperational Test
14.2.8.1.14	Fuel and Auxiliary Pools Cooling System (FAPCS) Preoperational Test
14.2.8.1.15	Process Sampling System Preoperational Test
14.2.8.1.16	Process Radiation Monitoring System Preoperational Test
14.2.8.1.17	Area Radiation Monitoring (ARM) System Preoperational Test
14.2.8.1.18	Containment Monitoring System (CMS) Preoperational Test
14.2.8.1.19	Instrument Air (IA) and Service Air (SA) Systems Preoperational Tests
14.2.8.1.20	High-Pressure Nitrogen Supply System Preoperational Test
14.2.8.1.21	Reactor Component Cooling Water System Preoperational Test
14.2.8.1.22	Makeup Water System Preoperational Test
14.2.8.1.23	Hot Water System Preoperational Test
14.2.8.1.24	Chilled Water System Preoperational Test
14.2.8.1.25	Heating, Ventilation, and Air Conditioning (HVAC) Systems Preoperational Test
14.2.8.1.26	Containment Inerting System Preoperational Test
14.2.8.1.27	Containment Isolation Valve Leakage Rate Tests
14.2.8.1.28	Containment Penetration Leakage Rate Tests
14.2.8.1.29	Containment Airlock Leakage Rate Tests
14.2.8.1.30	Containment Integrated Leakage Rate Test
14.2.8.1.31	Containment Structural Integrity Test
14.2.8.1.32	Pressure Suppression Containment Bypass Leakage Tests
14.2.8.1.33	Containment Isolation Valve Functional and Closure Timing Tests
14.2.8.1.34	Wetwell-to-Drywell Vacuum Breaker System Preoperational Test
14.2.8.1.35	DC Power Supply System Preoperational Test
14.2.8.1.36	AC Power Distribution System Preoperational Test
14.2.8.1.37	Standby Diesel Generator & AC Power System Preoperational Test

14.2.8.1.38 Plant Communications System Preoperational Test
14.2.8.1.39 Fire Protection System Preoperational Test
14.2.8.1.40 Radioactive Liquid Drainage and Transfer Systems Preoperational Tests
14.2.8.1.41 Fuel-Handling and Reactor Servicing Equipment Preoperational Test
14.2.8.1.42 Expansion, Vibration, and Dynamic Effects Preoperational Test
14.2.8.1.44 Condensate and Feedwater Systems (CFS) Preoperational Test
14.2.8.1.45 Condensate Cleanup System Preoperational Test
14.2.8.1.46 Reactor Water Chemistry Control Systems Preoperational Test
14.2.8.1.47 Condenser Air Removal System Preoperational Test
14.2.8.1.48 Off-gas System Preoperational Test
14.2.8.1.49 Condensate Storage and Transfer System Preoperational Test
14.2.8.1.50 Circulating Water System (CIRC) Preoperational Test
14.2.8.1.51 Plant Service Water System (PSWS) Preoperational Test
14.2.8.1.52 Turbine Component Cooling Water System Preoperational Test
14.2.8.1.53 Main Turbine Control System (MTCS) Preoperational Test
14.2.8.1.54 Main Turbine Bypass System Preoperational Test
14.2.8.1.55 Steam Bypass and Pressure Control System Preoperational Test
14.2.8.1.56 Heater, Drain, and Vent System Preoperational Test
14.2.8.1.57 Extraction Steam System Preoperational Test
14.2.8.1.58 Moisture Separator Reheater System Preoperational Test
14.2.8.1.59 Main Turbine and Auxiliaries Preoperational Test
14.2.8.1.60 Main Generator and Auxiliary Systems Preoperational Test
14.2.8.1.61 Seismic Monitoring System Preoperational Test
14.2.8.1.62 Liquid and Solid Radwaste Systems Preoperational Tests
14.2.8.1.63 ICS Preoperational Test
14.2.8.1.64 PCCS Preoperational Test
14.2.8.1.65 GDCS Preoperational Test

In comparing the ESBWR pre-operational test program to the pre-operational testing recommended in Section 1 of Appendix A to RG 1.68, the staff identified several areas where it required additional information to complete its review. The following sections discuss the specific issues.

14.2.3.10.1 Fire Protection System Preoperational Test

In RAI 14.2-4, the staff requested additional information about the fire protection system. The staff noted that fire protection systems are to be designed, fabricated, and installed in accordance with the applicable National Fire Protection Association (NFPA) standards, including requirements for the testing and inspection of installed systems and equipment. The staff noted that DCD Tier 2, Section 14.2.8.1.39 did not reflect these requirements. The staff also noted that the section did not include acceptance criteria. The high-level acceptance criteria appropriate to a DCD should be included. Additionally, the staff noted that the pre-operational tests and inspections should also include the following to verify the proper functioning of fire protection features:

- Verification of the integrity of fire barriers (such as penetration seals, fire doors, etc.).

- Verification of the correct location of fire protection equipment including sprinkler heads, spray nozzles, detectors, hose stations, and portable extinguishers.

In response, the applicant stated it would expand DCD Tier 2, Section 14.2.8.1.39 to include references to DCD Tier 2, Section 9.5.1.1 and Table 9.5-1, which include applicable NFPA standards and criteria. The applicant further expanded Section 14.2.8.1.39 to include verification of the proper installation of fire protection system components, including fire barriers, penetration seals, and fire doors, per the design basis in DCD Tier 2, Section 9.5.1.1. The staff reviewed the test abstract in DCD Tier 2, Revision 3, Section 14.2.8.1.39, and finds the revised text responsive to the staff's concerns and is acceptable. Accordingly, the staff concludes that the fire protection system test description follows the guidance in RG 1.68, so it is acceptable. Therefore, RAI 14.2-4 is resolved.

14.2.3.10.2 Feedwater Control System Preoperational Test

In RAI 14.2-5, the staff requested additional information regarding the FWCS pre-operational test description in DCD Tier 2, Section 14.2.8.1.2. Section 1.J of Appendix A to RG 1.68 recommends the testing of instrumentation and control (I&C) systems that: (1) control normal operation of the facility, (2) provide information and alarms in the control room to monitor the operation and status of the facility, (3) establish that the facility is operating within design and license limits, (4) permit or support the operation of engineered safety features (ESFs), and (5) monitor and record important parameters during and following postulated accidents. In addition, Section 1.J of Appendix A to RG 1.68 includes provisions to verify the redundancy and electrical independence of this I&C system. However, the staff noted that the pre-operational test description of the FWCS did not specifically include testing the fault-tolerant digital controllers (FTDCs), nor did it include verification of electrical independence and redundancy of the FWCS.

In response, the applicant stated that the FTDCs will be tested as part of the FWCS factory acceptance tests (FAT) or pre-operational tests. The applicant also stated that DCD Tier 2, Section 7.7.3.4, details the testing of the FTDCs. The applicant explained that the redundancy and electrical independence of the FWCS will be verified by pre-operational tests, as described in DCD Tier 2, Sections 7.7.3.4 and 7.7.3.5. In addition, the applicant provided the following specification in Section 14.2.8.1.2 to demonstrate the testing of the FTDCs for redundancy and electrical independence of the FWCS. The specification states, "Proper operation of instrumentation and controls in the required combinations of logic and instrument channel trips, including verification of setpoints."

The staff determined that the response did not address the concern that the FTDCs and FWCS electrical independence and redundancy would be included within the scope of pre-operational testing. In its response to RAI 14.2-5 S01, the applicant further addressed the staff's concerns. The applicant's revised response clarified that as a prerequisite to verifying the operation of the FWCS, the FAT of the features and requirements of the FTDCs—as described in Sections 7.7.3.4 and 7.7.3.5—will be successfully completed. The staff reviewed the applicant's response to RAI 14.2-5 and the test abstract in DCD Tier 2, Revision 3, Section 14.2.8.1.2. Based on these reviews, the staff finds the revised text is consistent with RG 1.68 and is acceptable. Accordingly, the staff concludes that the FWCS test description follows the guidance in RG 1.68 and is thus acceptable. Therefore, RAI 14.2-5 is resolved.

14.2.3.10.3 Standby Liquid Control System Preoperational Test

In RAI 14.2-6, the staff requested additional information regarding the SLCS pre-operational test description in DCD Tier 2, Section 14.2.8.1.3. Section 1.B of Appendix A to RG 1.68 recommends verifying the redundancy and electrical independence of the SLCS. Specifically, the staff noted that there was not a pre-operational test describing the verification of electrical

independence and redundancy for the SLCS Class 1E electrical system. Also, the staff noted the lack of information pertaining to testing of a heater installed in the mixing drum.

In response to RAI 14.2-6, the applicant stated that redundancy and electrical independence, as it applies to the ESBWR design, are associated with the squib valves, critical instrumentation, and initiating logic channels and will be verified through inspection, analysis, and/or pre-operational tests detailed in DCD Tier 2, Section 7.4.1.3.3. The applicant also stated that DCD Tier 2, Section 14.2.8.1.3 covers testing to support the above statement, as it calls for "Proper operation of instrumentation and equipment in the required combinations of logic and instrument channel trip." With respect to the testing of the mixing drum heater, the applicant stated that DCD Tier 2, Section 9.3.5.2 provides a detailed system description of the heating requirements for the SLCS. Specifically, the DCD states that electrical heating of the accumulator tank and the injection line is not necessary. The applicant also noted that the SLCS heaters, air spargers, and heat tracing used in previous BWR designs to control and maintain solution temperature have been eliminated.

The staff reviewed the applicant's response to this RAI. The staff determined that the operability testing of heaters, spargers, and heat tracing required in RG 1.68 is not applicable to the ESBWR because these components do not exist in the ESBWR design. Also, the staff determined that verification of redundancy and electrical independence, as described in DCD Tier 2, Section 7.4.1.3.3, meets the intent of RG 1.68 and is therefore adequate.

In the applicant's response to RAIs 14.2-6 S01 and 9.3-21 S01, the staff found that the applicant added the regulatory treatment of nonsafety system power supplies and plant investment protection A and B buses, which supply power to two redundant electrical heaters used to ensure that the common-mode failure for heating the SLCS accumulator rooms does not occur. In addition, the SLCS accumulator room temperature is monitored and alarmed when low.

Since the electrical heaters and the temperature alarms are needed to ensure the operability of the SLCS when the temperature falls below 15.6 degrees Celsius (C) (60 degrees Fahrenheit [F]), the staff requested additional information in DCD Tier 2, Section 14.2.8.1.3 to ensure that the heaters and temperature alarms in both SLCS accumulator rooms are tested to ensure that the SLCS remains operable in cold weather. This is RAI 14.2-6 S01.

In response, the applicant agreed to add test requirements to confirm the existence and functionality of the electrical room heaters for the SLCS accumulator rooms. However, the additional testing of the temperature alarms is deemed unnecessary, because this testing is covered by the third bullet in DCD Tier 2, Section 14.2.8.1.3. The staff finds that because the pre-operational test and the startup test in Section 14.2.8.2.34 for the SLCS follow the guidance in SRP Section 14.2 and RG 1.68, they are therefore acceptable. The applicant added the testing of electrical heaters in DCD Tier 2, Revision 5, Section 14.2.8.1.3; this resolves RAI 14.2-6.

14.2.3.10.4 Control Rod Drive System Preoperational Test

In RAI 14.2-7, the staff requested additional information regarding the description of the pre-operational test for the CRD system in DCD Tier 2, Section 14.2.8.1.4. Section 1.B of Appendix A to RG 1.68 recommends testing to verify the correct failure mode on a loss of power for the CRD system. In reviewing the description of the pre-operational test for the CRD system, the staff noted that DCD Tier 2, Section 14.2.8.1.4 did not include information pertaining to this test.

In its response to RAI 14.2-7, the applicant stated that Section 2.2.2 and Table 2.2.2-1 of the ESBWR DCD Tier 1 describe the verification of the correct failure mode for the CRD system. The correct failure mode will be verified in the normal course of the scram test in which a loss of power to the scram solenoid pilot valves in the hydraulic control units (HCUs) causes the scram. The applicant also stated that Section 14.2.8.1.4 enforces the described test in the specification "Proper operation of HCUs and associated valves."

The staff stated in RAI 14.2-7 S01 that the RAI response did not fully address its concern, because the bulleted item did not provide assurance that the CRD test included testing to verify the correct failure mode upon a loss of power. In addition, the staff determined that the test abstract in DCD Tier 2 did not adequately describe the required testing in accordance with RG 1.68. In response, the applicant included a sentence clarifying that as a prerequisite to verifying the operation of the CRD system, the following will be successfully completed: (1) factory quality control tests, (2) functional tests, and (3) operational tests as described in DCD Tier 2, Section 4.6.3.

The staff reviewed the applicant's response and the test abstract in DCD Tier 2, Revision 3, Section 14.2.8.1.4. Based on these reviews, the staff finds the revised text is consistent with RG 1.68 and is acceptable. Therefore, the staff determined that verification of the correct failure mode upon a loss of power, as described in DCD Tier 1, Section 2.2.2, meets the intent of RG 1.68. In addition, the staff determined that the DCD revision clarifies the CRD system testing. Therefore, RAI 14.2-7 is resolved.

In RAI 14.2-39, the staff noted that the test description of the CRD system did not clearly state that the CRD high-pressure makeup mode of operation will be tested. This mode of operation will be initiated by a low reactor water Level 2 signal and the start of a standby pump, followed by the automatic opening of the injection valves. The staff questioned the applicant about this mode of operation and whether both CRD pumps will be tested.

In response, the applicant stated that the high-pressure makeup mode of operation will be tested, as indicated in the fifth item in DCD Tier 2, Section 14.2.8.1.4.. The item reads, "Proper operation of CRD makeup to reactor pressure vessel (RPV) on reactor low level signal." The applicant also stated that testing this mode includes the simultaneous operation of both CRD pumps to deliver the required high-pressure makeup flow rate to the reactor.

The staff reviewed the applicant's response. On the basis that the CRD system pre-operational test includes the testing of the CRD high-pressure makeup mode of operation, the staff finds that the CRD test description satisfies RG 1.68 requirements and is thus acceptable. Therefore, RAI 14.2-39 is resolved.

14.2.3.10.5 Safety System Logic Control System Preoperational Test

In RAI 14.2-8, the staff requested additional information regarding the safety system logic control (SSLC) system pre-operational test description in DCD Tier 2, Section 14.2.8.1.6. The staff noted that Section 1.C of Appendix A to RG 1.68 recommends testing the response time of each of the protection channels, including the sensors. However, the staff determined that the SSLC pre-operational test description did not clearly explain testing of the channel response time or sensor calibration and testing for the SSLC system channels and sensors.

In response, the applicant stated that the response time and calibration/testing of each of the safety-related channels (including sensors) would be performed as part of the testing of the

system with which they were associated. The applicant further stated that the ESF comprises the GDCS, the automatic depressurization system (ADS), the PCCS, the ICS, the SLCS, and the LD&IS. To that end, ESF channel response times for the ICS, GDCS, and ADS will be tested in accordance with DCD Tier 2, Sections 14.2.8.1.63, 14.2.8.1.65, and 14.2.8.1.1. For clarity, the applicant added to these sections the specification that the tests check for the "[a]cceptability of instrument channel response times, as measured from each applicable process variable input signal to the applicable process actuator confirmation signal."

The applicant stated that ESF channel response times for the LD&IS will be tested in accordance with DCD Tier 2, Section 14.2.8.1.8. For clarity, the applicant added to Section 14.2.8.1.8 the specification that the tests check for the "[a]cceptability of instrument channel response times, as measured from each applicable process variable input signal to the applicable process actuator confirmation signal."

The applicant also stated that ESF channel response times for the SLCS will be tested in accordance with DCD Tier 2, Section 14.2.8.1.3. To clarify that channel response times will be tested, the applicant added to Section 14.2.8.1.3 the specification "[a]cceptability of instrument channel response times, as measured from each applicable process variable input signal to the applicable process actuator confirmation signal."

The applicant stated that the PCCS channel response time test was not applicable because the PCCS does not rely on instrumentation to function. In addition, the applicant provided the requirement for channel response time testing for the RPS in DCD Tier 2, Section 14.2.8.1.9. The applicant's response clarifies the pre-operational testing requirements for response time testing of RPS/ESF systems and is acceptable. For the calibration of sensors, the applicant stated that the RPS pre-operational test description addresses such testing.

Also, the applicant added a new item in the LD&IS pre-operational test description to address the calibration of sensors. The applicant noted that for the ICS, GDCS, and SLCS, the item "proper operation of instrumentation and equipment in all combinations of logic and instrument channel trip," which is cited in DCD Tier 2, Sections 14.2.8.1.63, 14.2.8.1.65, and 14.2.8.1.3, covers the calibration of sensors. The staff determined that this portion of the applicant's response was not responsive to the staff's concern, because the phrase cited above did not specify the calibration of sensors.

In response to RAI 14.2-8 S01, the applicant noted that it had added the phrase "[p]roper calibration of instrumentation" to DCD Tier 2, Sections 14.2.8.1.3, 14.2.8.1.63, and 14.2.8.1.65. The staff reviewed the test abstracts in DCD Tier 2, Revision 3, Sections 14.2.8.1.1, 14.2.8.1.3, 14.2.8.1.8, 14.2.8.1.9, 14.2.8.1.63, and 14.2.8.1.65 and determined that the revised text provides reasonable assurance that the response time testing and sensor calibration will be accomplished in these tests. Therefore, the change is acceptable. Accordingly, the staff concludes that the SSLC system test description follows the guidance in RG 1.68 and is acceptable. Therefore, RAI 14.2-8 is resolved.

In RAI 14.2-68, the staff requested additional information regarding the SSLC pre-operational test description in DCD Tier 2, Section 14.2.8.1.6. The staff requested that the applicant describe the testing of the following design features:

- Bypass interlocks and resulting indication
- "Fail-safe" logic test for the RPS de-energization to trip
- A "fail-as-is" logic test for the ESF energization to trip

The applicant provided the following response to RAI 14.2-68:

> The features suggested in the RAI are part of each individual safety-related system, which are covered by SSLC and they are being verified as a part of those systems. Reactor Protection System logic testing is described in 14.2.8.1.9. Additionally, the following tests will be added to DCD Tier 2, Subsection 14.2.8.1.6:
>
> - Verify proper operation of instrumentation and controls in appropriate design combinations of logic and instrument channel trip;
>
> - Verify bypass logic and bypass indications;
>
> The ITAAC that will demonstrate conformance with "Operating Bypasses" and "Maintenance Bypasses" ([Institute of Electrical and Electronic Engineers Standard] IEEE-603-1991, Safety System Criteria 6.6 and 7.4, and 6.7 and 7.5) have been added to DCD Tier 1, in Subsection 2.2.15, Tables 2.2.15-1, and 2.2.15-2.
>
> The preoperational test descriptions provided are considered appropriate to describe functional testing of logic that may be either fail-safe or fail-as-is. Subsection 14.2.8 discusses the level of detail for the descriptions of each preoperational test and the planned availability of the actual test procedures prior to their intended use.

The applicant added the two bullets noted above in DCD Tier 2, Revision 5, Section 14.2.8.1.6. Therefore, RAI 14.2-68 is resolved.

In RAI 14.2-70, the staff requested additional information regarding DCD Tier 2, Section 14.2.8.1.6. Specifically, the staff asked the applicant to include functional checks of the digital trip logic module (DTLM) and the safety system output logic unit (OLU) as described by the appropriate design specification.

The applicant provided the following response:

> The terms DTLM and OLU are typically used in the NUMAC platform and may not be applicable to the SSLC. Without identifying specific components within an instrument channel and division of logic, guidance will be updated in DCD, Tier 2, Subsection 14.2.8.1.6, to test the instrumentation and controls in the appropriate design combinations of logic and instrument channel trip. Terms such as digital trip modules/DTLM (i.e., signal comparator modules), voting logic units and OLU, etc., are not called out specifically because their use and designation may vary, depending on the logic platform. This level of detail is addressed in the actual test procedures. The factory acceptance test(s) and preoperational tests (inclusive of the tests of individual systems) will thoroughly test that the logic (whether) individual chassis or integrated logic (in a common controller), input and output signals, operator interface and links to Non-Safety-Related Distributed Control and Information System (N-DCIS) are functioning correctly. Subsection 14.2.8 discusses the level of detail for the descriptions of each preoperational test and the planned availability of the actual test procedures prior to their intended use.

On the basis of the above review, the applicant plans to add the following items to DCD Tier 2, Section 14.2.8.1.6, as noted in the applicant's response to RAI 14.2-68:

- Verify proper operation of instrumentation and controls in appropriate design combinations of logic and instrument channel trip.

- Verify bypass logic and bypass indications.

The staff found the applicant's response to be unacceptable, since DCD Tier 2, Section 14.2.8.1.6 should identify the major functions. The identification of this information in the test abstract is necessary to demonstrate that the RPS will perform its intended safety functions.

In a follow-up response to RAI 14.2-70, the applicant stated that it does not plan to add design details, since this is a generic test plan with general test methods described in DCD Tier 2, Section 14.2.8.1.6. As discussed in the response, terms such as digital trip modules/DTLMs, voting logic units, and OLUs are not called out because their use and designation may vary depending on the logic platform. The actual test procedure addresses this level of detail. As previously indicated in the response to RAI 14.2-70, GEH updated DCD Tier 2, Section 14.2.8.1.6 to specify that the test will do the following:

- Verify proper operation of instrumentation and controls in appropriate design combinations of logic and instrument channel trip.

The NRC will have access to the detailed pre-operational tests as part of the design implementation process. Therefore, whether the applicant uses modules or controllers, the associated function is tested. On the basis of the response above and COL Information Item 14.2.3-A, the NRC inspectors will inspect the Licensee's pre-operational test procedures 60 days before their intended use.

However, in RAI 14.2-70 S01, the staff stated that regardless of whether the Licensee uses modules or controllers in the SSLC, the applicant should describe the SSLC major functions that will be tested in DCD Tier 2, Section 14.2.8.1.6. Regardless of the logic platform, the applicant should describe the SSLC sensor calibration and testing. In accordance with RG 1.68 and SRP Section 14.2, the applicant should include testing of the channel response time or sensor calibration and testing for the SSLC system channels and sensors in the SSLC pre-operational test description. RAI 14.2-70 S01 was being tracked as an open item in the SER with open items.

In response, the applicant stated the following:

> The Safety System Logic and Control Engineered Safety Feature (SSLC/ESF) must satisfy Inspections, Tests, Analyses and Acceptance Criteria (ITAAC) and software management planned testing as part of implementation and installation. This means that testing that might otherwise be considered SSLC/ESF preoperational testing is already completed during the implementation and installation phases of the SSLC/ESF construction. Therefore, the only SSLC/ESF preoperational activities remaining involve the clearing of any SSLC/ESF system diagnostic alarms and any other site-specific testing determined to be necessary. Other systems' preoperational testing require a

functional SSLC/ESF and upon their completion further indicate a fully functional SSLC/ESF.

> Detail sufficient to conclude that adequate SSLC/ESF testing has been performed prior to preoperational and startup testing is, therefore, part of the SSLC/ESF ITAAC and software management planned testing documentation. The "General Test Methods and Acceptance Criteria" does not cover details of the SAT because they are part of the Software Quality Assurance Program (SQAP) documentation. The NRC will have access to the detailed test/acceptance records as part of the design implementation process. Subsection 14.2.8.1.6 will be revised to include this detail.

> DCD, Tier 2, Subsection 7.2.1.4.2 is an example of the specific types of SSLC/ESF tests performed during operation that verify proper operation of instrumentation and controls in appropriate design combinations of logic and instrument channel trips, including channel response time or sensor calibration and testing. These types of tests are performed prior to operation in the preoperational test phase also.

The applicant added the following information to DCD Tier 2, Section 14.2.8.1.6:

> The objective of this test is to verify proper operation of the Safety System Logic and Control Engineered Safety Feature (SSLC/ESF) and the safety–related distributed control and information system (Q-DCIS) and N-DCIS plant DCIS indicated in Subsection 14.2.8.1.7. Proper functioning of the DCIS includes those functions utilized for the preoperational testing and the aggregate plant systems.

The applicant also added the following information to DCD Tier 2, Section 14.2.8.1.6:

> Because the SSLC/ESF must be functional for utilization in the preoperational testing of other systems, SSLC/ESF testing is completed during the implementation and installation phases of construction. The SSLC/ESF implementation and installation testing includes adhering to the commitments of the software development process (see Subsection 14.3.3.2). The commitments of the software plans include such testing as FAT and SAT. That which is not tested during the FAT, that which could change in transit, or that which is otherwise determined to need testing at the site is tested during the SAT.

The applicant added the information noted above to DCD Tier 2, Revision 6, Section 14.2.8.1.6. Therefore, RAI 14.2-70 S01 is resolved.

14.2.3.10.6 Distributed Control and Information System (DCIS) Preoperational Test

The staff noticed that in DCD Tier 2, Section 14.2.8.1.7, the description of the pre-operational testing for the DCIS is incomplete in that it does not provide sufficient detail to conclude that adequate system testing will be performed.

In RAI 14.2-99, the staff indicated that in DCD Tier 2, Section 14.2.8.1.7, the DCD should clarify that construction tests that include the DCIS FAT and the ITAAC commitment tests have been successfully completed.

The staff further indicated in RAI 14.2-99 that DCD Tier 2, Section 14.2.8.1.7 should describe the following elements in the "General Test Methods and Acceptance Criteria" after the DCIS installation: (1) Conduct of the site acceptance test (SAT) shall include both Q-DCIS and N-DCIS,and (2) The SAT shall test all DCIS functions and capabilities specified in the Technical Design Specification (major elements identified in the life-cycle phase summary baseline review record) of the DCIS. The following items should be considered during the DCIS pre-operational tests:

(1) Video display unit (VDU) performance

(2) Database capacity

(3) All spare requirements

(4) Cyber security aspects

(5) Redundancy features of controllers

(6) Power supplies

(7) Data communications and interface requirements, etc

(8) The system loop test shall be conducted for each Input/Output (I/O) by connecting all field devices to the DCIS I/O terminals.

(9) The system control logic and man-machine interface design features shall be tested

In its response, the applicant stated the following:

> Chapter 14 of the ESBWR DCD covers preoperational and startup testing. Preoperational testing follows completion of construction (and construction-related) inspections, tests, and acceptance and takes place before fuel is loaded. Startup testing takes place during and after fuel loading. Detail sufficient to conclude that adequate DCIS testing has been performed prior to preoperational and startup testing is, therefore, not included in Chapter 14.
>
> Construction and preoperational testing concepts for the DCIS differ from other systems in that the DCIS must be functional before many other preoperational tests can begin. The DCIS must therefore be installed and shown to be working acceptably during construction, the implementation, and installation phases.
>
> The DCIS must satisfy ITAAC and software management planned testing as part of implementation and installation. This means that testing that might otherwise be considered DCIS preoperational testing is already completed during the implementation and installation phases (of the DCIS construction). Therefore, the only DCIS preoperational tests remaining involve the clearing of any DCIS system diagnostic alarms and any other site-specific testing determined to be necessary. Other systems' preoperational testing turnover packages require a functional DCIS and upon their completion, further indicate a fully functional DCIS.

Detail sufficient to conclude that adequate DCIS testing has been performed prior to preoperational and startup testing is, therefore, part of the ITAAC and software management planned testing documentation. The NRC will have access to the detailed test/acceptance records as part of the design implementation process.

The "General Test Methods and Acceptance Criteria" does not cover details of the SAT because they are part of the SQAP documentation.

The DCIS system control logic and man-machine interface design features are tested as part of the other systems' testing and testing committed to in the software plans. Details on software plan tests will be in test plans developed through implementation of the Software Management Program and SQAP and include, but are not limited to, the following.

VDU performance,
Database capacity,
All spare requirements,
Cyber security aspects,
Redundancy features of controllers,
Power supplies,
Data communications and interface requirements, etc.

The system loop testing is satisfied for each I/O through the testing of each system that makes up the DCIS.

Based on the response above, the applicant committed to making the following revision to DCD Tier 2, Section 14.2.8.1.7:

Purpose
The object of this testing is to verify proper functioning of both the safety-related (Q-DCIS) and non- safety-related (N-DCIS) plant DCIS. Proper functioning of the DCIS includes those functions utilized for the preoperational testing of the aggregate plant systems.

Prerequisites
Since the DCIS must be functional for utilization in the preoperational testing of other systems, DCIS testing is completed during the implementation and installation of phases of construction. The DCIS implementation and installation testing includes adhering to the commitments of the software plans (see Subsection 14.3.3.2). The commitments of the software plans include such testing as FAT, that which could change in transit, or that which is otherwise determined to need testing at the site is tested during the Site Acceptance Test.

DCIS construction tests have been successfully completed and the SCG has both reviewed test procedures and approved the initiation of testing. The required AC and DC electrical power sources shall be operational and the appropriate interfacing systems shall be available as required to support the specified testing.

General Test Methods and Acceptance Criteria
The testing of the following:

- Verify that all DCIS diagnostic alarms have been resolved, cleared, and documented as such or have been documented for later resolution during individual/specific systems preoperational testing.

The applicant added the above information to DCD Tier 2, Revision 6, Section 14.2.8.1.7. Therefore, RAI 14.2-99 is resolved.

14.2.3.10.7 Leak Detection and Isolation System Preoperational Test

In RAI 14.2-9, the staff requested additional information regarding the LD&IS pre-operational test description in DCD Tier 2, Section 14.2.8.1.8. Section 1.J of Appendix A to RG 1.68 recommends testing I&C systems that permit or support the operation of ESFs. In reviewing the LD&IS pre-operational test description, the staff determined that the test description did not address testing for the following manual control functions:

- Actuation of each main steam isolation valve (MSIV) test switch
- MSIV isolation switches
- MSIV logic reset
- RWCU/SDC isolation switch
- Containment isolation manual switch
- Containment isolation logic reset
- Reactor building HVAC isolation

In its response to RAI 14.2-9, the applicant stated that the test description in Section 14.2.8.1.8 included the pre-operational tests of all test switches, manual switches, isolation switches, and logic resets for the LD&IS. This testing is covered in the specification “Proper operation of instrumentation and controls in all combinations of logic and instrument channel trip.” The staff reviewed the applicant’s response to this RAI. On the basis that the LD&IS will be tested in conjunction with the manual control functions detailed above, as part of the overall containment isolation and main steamline isolation initiation logic, the staff finds that the LD&IS test description satisfies RG 1.68 and is adequate. Therefore, RAI 14.2-9 is resolved.

In RAI 14.2-73, the staff requested additional information on DCD Tier 2, Section 14.2.8.1.8, regarding information necessary to identify the interfacing functions and systems that must be available. These include the following:

- Drywell pressure signals, or simulated, from the RPS
- The reactor mode switch signals from the RPS
- The interlock from the RPS bypassing the MSIV isolation when not in the “RUN” mode

In its response to RAI 14.2-73, applicant stated the following:

> ESBWR DCD, Tier 2, Revision 3, Subsection 14.2.8.1.8, 5th bullet requires the LD&IS Preoperational Test to demonstrate “Proper interface with related systems in regard to the input and output of leak detection indications and isolation initiation commands.” These indications include: the Drywell pressure signals, or simulated signals from the RPS; and the reactor mode switch signals from the RPS. Also, the 6 bullet of Subsection 14.2.8.1.8 “Proper operation of bypass switches and related logic” includes the interlock from the RPS bypassing the MSIV when not in “RUN” mode. The LD&IS interfacing diagram is provided in Figure 7.3-3.

However, the applicant does not plan to add information to DCD Tier 2, Section 14.2.8.1.8. In a supplemental RAI, the staff requested that the applicant describe under the LD&IS pre-operational test methods and acceptance criteria, the LD&IS component functions that can be tested during this test phase and the acceptance criteria that must be met to demonstrate that the LD&IS meets its design basis. In addition, the staff requested that the applicant revise DCD Tier 2, Section 14.2.8.1.8 to include the testing of I&C systems for the LD&IS, in accordance with RG 1.68 Appendix A, Item J, "Instrumentation & Control Systems," Items (1) through (25).

In a follow-up response to RAI 14.2-73, the applicant stated the following:

> The operation of the LD&IS functional logic is demonstrated during a series of overlapping preoperational tests. As indicated in the GEH response to RAI 14.2-73, DCD, Subsection 14.2.8.1.8 (5 and 6 bullets) performs the applicable preoperational tests requested by the NRC RAI. LD&IS controls, interlocks and bypasses are also verified through LD&IS ITAAC No. 4, DCD, Tier 1, Table 2.2.12.5. The LD&IS and RPS controls, interlocks and bypasses are described in DCD, Tier 1, Table 2.2.12-4 and 2.2.7-3, respectively.

On the basis of the above response and COL Information Item 14.2.2-A, NRC inspectors will inspect the Licensee LD&IS and RPS pre-operational test procedures 60 days before their intended use.

However, in RAI 14.2-73 S01, the staff asked the applicant to describe the major functions in DCD Preoperational Test Section 14.2.8.1.8, including LD&IS controls, interlocks, and bypasses that are verified in the LD&IS ITAAC. These include the major LD&IS and RPS control, interlock, and bypass functions described in Tables 2.2.7-3, 2.2.12-4, and 2.2.12-5. RAI 14.2-73 S01 was being tracked as an open item in the SER with open items.

To address RAI 14.2-73 S01, the applicant added the following information to DCD Tier 2, Section 14.2.8.1.8:

> The objective of this test is to verify proper response and operation of the LD&IS logic, the safety-related (Q-DCIS) and non-safety-related (N-DCIS) plant DCIS, indicated in Subsection 14.2.8.1.7. Proper functioning of the DCIS includes those functions utilized for the preoperational testing of the aggregate plant systems.

The applicant also added the following information to the "Prerequisites" section:

> Since the RPS and SSLC/ESF must be functional for utilization in the preoperational testing of other systems, LD&IS testing is completed during the implementation and installation phases of construction. The RPS and SSLC/ESF implementation and installation testing includes adhering to the commitments of the software plans (see Subsection 14.3.3.2). The commitments of the software plans include such testing as FAT and Site Acceptance Tests (SAT). That which is not tested during the FAT, that which could change in transit, or that which is otherwise determined to need testing at the site is tested during the SAT.

The applicant added the above information to DCD Tier 2, Revision 6, Section 14.2.8.1.8. Therefore, RAI 14.2-73 S01 is resolved.

14.2.3.10.8 Neutron Monitoring System Preoperational Test

In RAI 14.2-74, the staff requested additional information regarding the NMS pre-operational test description in DCD Tier 2, Section 14.2.8.1.10. Specifically, the staff requested that the applicant describe pre-operational testing of the thermometer system and calibration of any local power range monitors (LPRMs).

In response, the applicant stated following:

> DCD, Tier 2, Subsection 14.2.8.1.10, notes the prerequisite that the Startup Range Neutron Monitor (SRNM) and Power Range Neutron Monitor (PRNM) components have been calibrated per vendor instructions. The “Prerequisites” paragraph also notes that “required interfacing systems shall be available, as needed, to support the specified testing.” The Automated Fixed In-core Probe (AFIP) subsystem is such a required interfacing system. The AFIP and LPRM sensors are contained within the LPRM assemblies, which are part of the PRNM subsystem. This prerequisite ensures that the AFIP detectors (gamma thermometers (GTs)) and the LPRMs will be pre-calibrated prior to in-situ preoperational testing.
>
> Section 14.2.8.1.10, “General Test Methods and Acceptance Criteria” also notes that the following shall be demonstrated:
>
> - Proper operation of detectors and associated cabling, preamplifiers, and power supplies.;
> - Proper operation of system and subsystem self-test diagnostic and calibration functions.; and
> - The ability to communicate and interface between appropriate plant systems and NMS subsystems.
>
> These three items ensure that the AFIP detectors and the LPRMs will be calibrated during preoperational testing, including demonstration of the communications interfaces between the AFIP subsystem and the NMS. The LPRMs cannot be calibrated in-situ without the use of the AFIP subsystem.
>
> The final calibration of the GTs and the application of GT calibration factors to the LPRMs can be accomplished only during reactor operation during startup and power testing.

The applicant did not revise DCD Tier 2, Section 14.2.8.1.10 in response to this RAI. However, the staff finds this response clarifies the testing requirements and is acceptable. Therefore, RAI 14.2-74 is resolved.

In RAI 14.2-75, the staff requested additional information regarding the NMS pre-operational test description contained in DCD Tier 2, Section 14.2.8.1.10. Specifically, the staff asked the applicant to provide additional details on the subsystems and the specific tests involved, such as the following:

- Verification of rod block monitor input matrix and trip output for correct functions.

- Verification of the oscillation power range monitor (OPRM) instrumentation for correct trip, alarm, and bypass functions.

In response, the applicant stated the following:

> DCD, Tier 2, Subsection 14.2.8.1.10, notes the prerequisite that the PRNM “components have been calibrated per vendor instructions.” The “Prerequisites” paragraph also notes that “required interfacing systems shall be available, as needed, to support the specified testing.” The OPRM algorithms and tables are contained completely within the PRNM subsystem, and the Multichannel Rod Block Monitor (MRBM) subsystem is a required interfacing system. This prerequisite ensures that all of the PRNM, including the OPRM functions and the MRBM functions and interfaces, will be subjected to in-situ preoperational testing.
>
> DCD, Tier 2, Subsection 14.2.8.1.10, “General Test Methods and Acceptance Criteria,” also notes that the following shall be demonstrated:
>
> - Proper operation including rod block.
>
> - Proper functioning of instrumentation, displays, alarms, and annunciators used to monitor system operation and status;
>
> - The ability to communicate and interface between appropriate plant systems and NMS subsystems.
>
> These three items ensure that the OPRM and MRBM functions and software tables will be verified prior to and during preoperational testing, including demonstration of the communications interfaces between the MRBM subsystem and the NMS.
>
> In accordance with DCD, Tier 2, Subsections 7.2.2.2.7.4 and 7.2.2.2.7.5, the OPRM alarms and trips are bypassed in all reactor operation modes except run and when operating below the required power level (typically 30 percent). Therefore, the final checks of OPRM functions can be accomplished only during reactor operation during preoperational testing.

The staff finds this response clarifies the testing requirements and is acceptable. Therefore, RAI 14.2-75 is resolved.

14.2.3.10.9 Plant Automation System Preoperational Test

In RAI 14.2-76, the staff requested additional information regarding the PAS pre-operational test description in DCD Tier 2, Section 14.2.8.1.11. The staff asked the applicant to provide additional detail about the tests involved. Examples include the following:

- For redundant controllers, tests would be conducted to confirm a response to simulated controller failures.

- The capability of the PAS to automatically decouple from plant control and revert to plant operation in manual mode.

In response, the applicant indicated that it would make no changes to DCD Tier 2, Section 14.2.8.1.11. The staff determined that without the additional information on these tests in DCD Tier 2, Section 14.2.8.1.11, under the PAS pre-operational test methods and acceptance criteria it was not clear that the PAS testing would include all of the functions required to demonstrate that the system acceptance criteria will be met to satisfy design-basis requirements. DCD Tier 2, Section 14.2.8.1.11 should include testing of the instrumentation and control systems for the PAS in accordance with RG 1.68 Appendix A Item J, items (1) through (25).

The applicant stated that the PAS is a nonsafety-related system that does not perform or ensure any safety-related function and is not required to achieve or maintain a safe shutdown. The PAS is non-safety-related and has no safety design basis.

The applicant also stated that specific testing to be performed and the applicable acceptance criteria for each pre-operational test are documented in test procedures to be made available to the NRC approximately 60 days before their intended use and are in accordance with the system specification and associated equipment specifications. These tests will demonstrate that the installed equipment and systems perform within the limits of these specifications. Therefore, DCD Tier 2, Section 14.2.8.1.11 does not require revision.

The staff verified that the instrumentation and control testing of the PAS meets the guidance in RG 1.68, Appendix A, Item J. On the basis of this response and COL Information Item 14.2.2-A, NRC inspectors will inspect the Licensee's PAS pre-operational test procedures 60 days before their intended use. Therefore, RAI 14.2-76 is resolved.

14.2.3.10.10 Remote Shutdown System Preoperational Test

In RAI 14.2-10, the staff requested additional information regarding the remote shutdown system (RSS) pre-operational test description in DCD Tier 2, Section 14.2.8.1.12. Section 1.J of Appendix A to RG 1.68 recommends the testing of instrumentation and controls used for a shutdown from outside the control room. In reviewing the pre-operational test description of the RSS, the staff determined that DCD Tier 2, Section 14.2.8.1.12 did not clearly describe the testing adequately to demonstrate proper operation of individual systems and equipment when operated from the remote shutdown panel.

In response, the applicant stated that factory and pre-operational tests will be performed to demonstrate the proper functioning of the control and instrumentation associated with the RSS panel. To this end, the applicant revised Section 14.2.8.1.12 to include verification of RSS switches and override of main control room (MCR) functions during the performance of factory and pre-operational tests. The staff reviewed the applicant's response to this RAI. Based on this review, the staff determined that the revised text clarifies the RSS testing requirements in DCD Tier 2, Revision 3, Section 14.2.8.1.12. Accordingly, the staff concludes that the RSS test description follows the guidance in RG 1.68 and is acceptable. Therefore, RAI 14.2-10 is resolved.

14.2.3.10.11 Fuel and Auxiliary Pools Cooling System Preoperational Test

In RAI 14.2-11, the staff requested additional information regarding the FAPCS pre-operational test description in DCD Tier 2, Section 14.2.8.1.14. Section 1.M of Appendix A to RG 1.68 recommends the testing of equipment and components used to handle or cool irradiated and nonirradiated fuel. In accordance with RG 1.68, the pre-operational test description should also

include verification of redundancy and electrical independence. In reviewing the FAPCS test description, the staff determined that DCD Tier 2, Section 14.2.8.1.14 did not have provisions for verifying electrical independence and redundancy. In addition, the staff noted that the FAPCS has eight modes of operation. Each of these modes requires a different flow path to achieve the design pool cleaning and cooling functions of the FAPCS. The FAPCS test description did not include provisions for testing these modes of operation.

In response, the applicant stated that factory and pre-operational tests will be performed to demonstrate the proper functioning of the control and instrumentation associated with the FAPCS and will include verification of redundancy and electrical independence of the safety-related instrumentation. The applicant also stated that tests will be performed for all modes of operation. To that end, the applicant revised DCD Tier 2, Section 14.2.8.1.14 to include the above testing.

The staff reviewed the applicant's response and the test abstract in DCD Tier 2, Revision 3, Section 14.2.8.1.14. Based on this review, the staff finds that the revised text describes the necessary provisions for testing the FAPCS. Accordingly, the staff concludes that the FAPCS test description follows the guidance in RG 1.68 and is acceptable. Therefore, RAI 14.2-11 is resolved.

14.2.3.10.12 Area Radiation Monitoring System Preoperational Test

In RAI 14.2-12, the staff requested additional information regarding the ARM system pre-operational test description in DCD Tier 2, Section 14.2.8.1.17. Section 1.K of Appendix A to RG 1.68 recommends testing the equipment and components used to monitor or measure radiation levels. In accordance with RG 1.68, the pre-operational test description should also include testing to verify redundancy and electrical independence. However, the staff determined that DCD Tier 2, Section 14.2.8.1.17 did not clearly describe the provisions for verifying electrical independence and redundancy during the pre-operational testing of the ARM system.

In response, the applicant stated that DCD Tier 1, Table 2.3.2-1 provided pre-operational testing information for the ARM system. The applicant also stated that redundancy at the monitor level was not required because the ARM system does not have a safety-related function. The applicant noted that the fail-safe design will initiate a local alarm and an alarm in the MCR on interruption of power, component failure, or loss of signal. The applicant revised DCD Tier 2, Section 14.2.8.1.17 to add the following to the ARM pre-operational test description:

> Proper functioning following power interruption to each ARM monitor, including appropriate local and MCR alarms has no affect on the functionality of other ARM monitors.

The staff reviewed the applicant's response to this RAI. The staff verified this change in DCD Tier 2, Revision 3, Section 14.2.8.1.17 and determined that the revised text addresses the staff's concern. Accordingly, the staff concludes that the revised ARM system test description noted above follows the guidance in RG 1.68 and is acceptable. Therefore, RAI 14.2-12 is resolved.

In RAI 14.2-92, the staff requested additional information regarding which ARM monitors listed in DCD Tier 1, Revision 4, and in Table 2.3.2-1 have associated system trips. The staff asked

the applicant to describe, for each radiation monitor that has an associated system trip, the purpose and function of the associated system trip.

In response, the applicant stated that since the ARM system is nonsafety-related and is for alarm and indication purposes only, it does not provide any trip or interlock to external devices. The applicant revised DCD Tier 2, Revision 5, Section 14.2.8.1.17 to replace the word “trips” with the words “indications and alarms are observed.” Therefore, RAI 14.2-92 is resolved.

14.2.3.10.13 Containment Monitoring System Preoperational Test

In RAI 14.2-77, the staff requested additional information regarding the CMS pre-operational test. Specifically, the staff requested that DCD Tier 2, Section 14.2.8.1.18 provide information on the tests involved.

In response, the applicant stated that under “General Test Methods and Acceptance Criteria” in DCD Tier 2, Section 14.2.8.1.18, it will add the following items:

- Proper operation of heat tracing and self-regulating functions used in each H2/02 sample line;.

- Proper operation of logic and bypass functions;.

- Proper operation of oxygen and hydrogen analyzers per manufacturer’s instructions.

The applicant added this information in DCD Tier 2, Revision 5, Section 14.2.8.1.18. Therefore, RAI 14.2-77 is resolved.

In RAI 14.2-93, the staff requested additional information regarding the description of the purpose/function of the system trip associated with the subsystem of the CMS that monitors radiation levels in containment.

In response, the applicant stated that the portion of the CMS subsystem monitoring gamma radiation levels in the containment is nonsafety-related and is provided for alarm and indication only. Therefore, this subsystem does not provide trips or interlocks for external devices. The applicant added this information in DCD Tier 2, Revision 5, Section 14.2.8.18, as it replaced the words “system trips” with the words “indication and alarm” and specified that this acceptance criterion applies to the containment radiation and atmospheric monitoring subsystems. Therefore, RAI 14.2-93 is resolved.

14.2.3.10.14 Instrument Air and Service Air Systems Preoperational Test

In RAI 14.2-13, the staff requested additional information regarding pre-operational test descriptions of the the IA and SA systems in DCD Tier 2, Section 14.2.8.1.19. Section 1.N, “Auxiliary and Miscellaneous Systems,” of Appendix A to RG 1.68 recommends the testing of the compressed gas systems that are used to support the normal operation of the facility or are essential for the operation of standby safety equipment or ESFs. In accordance with RG 1.68, the test program should also include verification of redundancy and electrical independence of the compressed gas system. RG 1.68.3, “Preoperational Testing of Instrument and Control Air Systems,” provides guidance for conducting pre-operational testing of the instrument and control air systems. Specifically, Regulatory Position 9 of RG 1.68.3 calls for tests to

demonstrate that air supplies such as the SA supply would not be inadvertently tied into the IA system. In reviewing the pre-operational test description for the IA and SA systems, the staff noted that the test descriptions did not include provisions for verifying electrical independence and redundancy, nor did they include provisions to demonstrate that the air systems could not be inadvertently interconnected.

In response, the applicant stated that the IA and SA systems are nonsafety-related and, therefore, not required to have redundancy and electrical independence to support the safety design basis of the plant. The applicant added that the IA and SA systems are designed with redundant compressors in each system and are powered from separate buses, thus providing electrical independence. In addition, the applicant stated that pre-operational tests will be performed to ensure that the backup compressors in each system start as expected from their assigned power buses. DCD Tier 2, Section 14.2.8.1.19 reflected this in the following items:

- “Proper operation of instrumentation and equipment in all combinations of logic and instrument channel trip,”

- “Proper operation of compressors and motors in all design operating modes,” and

- “Ability of compressor(s) to maintain receiver at specified pressure(s) and to recharge within specified time under design loading conditions.”

Regarding provisions to demonstrate that both air systems cannot be inadvertently interconnected, the applicant stated that inadvertent interconnection between the IA and SA systems will be verified during pre-operational testing, as described by items in Section 14.2.8.1.19 requiring “Proper operation of instrumentation and equipment in all combinations of logic and instrument channel trip” and “Ability of the SAS to act as backup to the IAS.”

The staff reviewed the applicant’s response to this RAI. On the basis that the IA and SA systems will be tested against the requirements delineated in RG 1.68 and RG 1.68.3, including verification of redundancy, electrical independence, and the inadvertent operation of both systems, the staff finds the pre-operational test description of the IA and SA systems satisfies RG 1.68 and is acceptable. Therefore, RAI 14.2-13 is resolved.

14.2.3.10.15 Expansion, Vibration, and Dynamic Effects Preoperational Test

In RAI 14.2-24, the staff asked the applicant to discuss the expansion, vibration, and dynamic effects for conformance with RG 1.68, RG 1.56, RG 1.128, and RG 1.136 and to justify exceptions to the RG positions. The staff also referred the applicant to RG 1.20, “Comprehensive Vibration Assessment Program for Reactor Internals During Preoperational and Initial Startup Testing,” Revision 3, for vibration assessment program guidance for reactor internals and potential adverse flow effects in steam and feedwater systems.

In response, the applicant stated, in part:

> With regard to compliance with RG 1.68 relative to thermal expansion, vibration and dynamic effects for the preoperational test program, DCD, Tier 2, Subsections 3.9.2.1.1 and 3.9.2.1.2 have been revised to specifically address compliance with this regulation and other industry standards with respect to

safety-related piping. The test program conformance with RG 1.68 is described in DCD, Tier 2, Subsection 14.2.8.1, and Pre-operational Test Procedures. Where applicable, the Test Acceptance Criteria for the thermal expansion, vibration and dynamic effects for the preoperational and/or startup tests will also meet the requirements of the other RGs 1.56 and 1.128. RG 1.136 will not be listed in Chapter 14 of the DCD because it is not applicable and is referenced in Subsection 3.8.1.6. In addition, the development of the test criteria will require consideration of the potential adverse flow effects on piping systems recommended in RG 1.20, and in SRP Section 3.9.2 and SRP Section 3.9.5. RG 1.68, 1.56, and 1.20 have been referenced in DCD, Tier 2, Subsection 14.2.3 (Titled: Test Program's Conformance with Regulatory Guides). No exceptions to the regulatory positions in the applicable RGs are being requested by GEH.

Based on the applicant's response and the changes noted to DCD Tier 2, Revision 4, Section 14.2.3, the staff finds that the applicant provided sufficient information on conformance to RGs. Therefore, RAI 14.2-24 is partially resolved for pre-operational tests. The staff issued RAI 14.2-24 S01, which addresses the staff's concerns regarding vibration tests at power. Section 14.2.3.11.8 of this report discusses RAI 14.2-24 S01.

14.2.3.10.16 Nuclear Boiler System Preoperational Test

In RAI 14.2-40, the staff requested additional information regarding the NBS pre-operational test description in DCD Tier 2, Section 14.2.8.1.1. The staff determined that DCD Tier 2, Section 14.2.8.1.1 did not clearly specify provisions to verify whether the depressurization valve (DPV) tests had been completed.

In response, the applicant stated that the manufacturer of the DPV will perform DPV engineering development and operability tests. In addition, the applicant stated that it will revise the prerequisites portion of Section 14.2.8.1.1 to denote the completion of such testing. GEH confirmed that it will revise the DCD to include the previously completed DPV engineering development and operability tests, in the interest of document completeness. The staff reviewed the test abstract in DCD Tier 2, Revision 3, Section 14.2.8.1.1 and determined that the revised text clarifies DPV testing requirements. Accordingly, the staff concludes that the NBS test description follows the guidance in RG 1.68 and is acceptable. Therefore, RAI 14.2-40 is resolved.

14.2.3.10.17 Gravity-Driven Cooling System Preoperational Test

In RAI 14.2-41, the staff requested additional information regarding the GDCS pre-operational test description in DCD Tier 2, Section 14.2.8.1.65. The staff asked the applicant to provide information on test setup conditions (e.g., vessel and drywell pressures) and limiting conditions that will be considered in the tests. In addition, the staff asked whether GDCS testing will be performed with installed check valves and squib valves.

In response, the applicant stated that DCD Tier 1, Table 2.4.2-1 describes GDCS testing. The applicant stated that the test will be an open reactor vessel test at atmospheric conditions in both the drywell and vessel. In addition, the applicant stated that testing will be conducted with check valves and squib valves installed, using previously activated squib valves. The staff determined that the applicant had clarified that the GDCS tests will be conducted at atmospheric conditions in both the drywell and the vessel. The applicant also confirmed that testing will be conducted with check valves and squib valves installed, and previously activated squib valves

will be used. By design, the GDCS will be activated after reactor system depressurization. Therefore, the staff finds that the initial tests under atmospheric conditions are acceptable. On this basis, the staff concludes that the GDCS test description is acceptable. Therefore, RAI 14.2-41 is resolved.

14.2.3.10.18 Condensate and Feedwater System Preoperational Test

In RAI 14.2-46, the staff requested additional information regarding the CFS pre-operational test description in DCD Tier 2, Section 14.2.8.1.44. The staff asked the applicant to include condensate booster pumps to ensure consistency with Position C.1.a of RG 1.68.1, "Preoperational and Initial Startup Testing of Feedwater and Condensate Systems for Boiling Water Reactor Power Plants," Revision 1.

In response, the applicant stated that the ESBWR does not have condensate booster pumps. The applicant also stated that because the reactor feed pump (RFP) has a booster pump and a main pump on the same shaft and motor, it will revise Section 14.2.8.1.44 to require the demonstration of "Proper operation of pumps and motors in all design operating modes (Condensate and RFP)." The staff reviewed the test abstract in DCD Tier 2, Revision 3, Section 14.2.8.1.44 and determined that the revised text is consistent with RG 1.68. Accordingly, the staff concludes that the CFS test description follows the guidance in RG 1.68 and is acceptable. Therefore, RAI 14.2-46 is resolved.

In RAI 14.2-47, the staff asked the applicant to clarify whether the feedwater flow control valve testing described in DCD Tier 2, Section 14.2.8.1.44 meets Regulatory Position C.1.d of RG 1.68.1, Revision 1. The staff was specifically interested in testing the proper response of the valves for the design operating range and the correct operation of protective features.

In response, the applicant stated that the ESBWR uses valve control for the low-flow control of feedwater flow and feed pump speed control for normal at-power feedwater flow rate control. DCD Tier 2, Section 14.2.8.1.2 describes the pre-operational testing of the FWCS. However, the applicant stated that it will revise the text in DCD Tier 2, Section 14.2.8.1.44 to include the following:

- Proper operation of system valves, including timing, under expected operating conditions, and proper response of flow control valves for the design operating range and correct operation of protective features.

The staff reviewed the test abstract in DCD Tier 2, Revision 3, Section 14.2.8.1.44 and determined that the revised text clarifies that the testing will verify proper valve response over the design operating range with the correct operation of protective features. Accordingly, the staff concludes that the CFS test description is acceptable. Therefore, RAI 14.2-47 is resolved.

In RAI 14.2-48, the staff noted that Section 14.2.8.1.44 does not include a comprehensive FWCS test as described in Regulatory Position C.1.f of RG 1.68.1, Revision 1. The staff asked the applicant to provide a justification or an alternative method for demonstrating the operability of the FWCS.

In response, the applicant stated that DCD Tier 2, Section 14.2.8.1.2 describes the FWCS pre-operational test that addresses the individual components of the FWCS but does not address the overall response of the control system as stipulated in RG 1.68.1. The applicant stated that it will add the following to Section 14.2.8.1.2:

- Proper overall response of the control system including the final control element.

The applicant noted that this will include control system response to simulated control system malfunctions and simulated plant transients at full flow, including MSIV closure and turbine trip without the bypass capability. The staff reviewed the test abstract in DCD Tier 2, Revision 3, Section 14.2.8.1.2 and determined that the revised text clarifies the comprehensive FWCS testing recommended by RG 1.68. Accordingly, the staff concludes that the CFS test description is acceptable. Therefore, RAI 14.2-48 is resolved.

14.2.3.10.19 Circulating Water System Preoperational Test

In RAI 14.2-50, the staff requested additional information regarding the CIRC pre-operational test description in DCD Tier 2,Section 14.2.8.1.50. The staff asked the applicant to confirm whether the ESBWR pre-operational testing of the CIRC included verification of pump net positive suction head (NPSH) and verification of proper system operation, while powered from primary and alternate power sources.

In response, the applicant confirmed that pre-operational activities will include verification of acceptable NPSH under the most limiting design flow conditions, and that it will add a statement to DCD Tier 2 Section 14.2.8.1.50 to indicate such verification. The applicant also stated that the CIRC does not have a backup power supply or redundant power source specific to the system. The power source for the CIRC pumps is the unit auxiliary transformer, which will be backed up by the reserve auxiliary transformer. The staff reviewed the test abstract in DCD Tier 2 Revision 3, Section 14.2.8.1.50 and finds that the revised text clarifies the NPSH and alternate power source testing requirements. Accordingly, the staff concludes that the CIRC test description follows the guidance in RG 1.68 and is thus acceptable. Therefore, RAI 14.2-50 is resolved.

14.2.3.10.20 Main Turbine Control System Preoperational Test

In RAI 14.2-51, the staff requested additional information regarding the MTCS pre-operational test description in DCD Tier 2, Section 14.2.8.1.53. The staff asked the applicant to confirm whether the ESBWR pre-operational testing for the MTCS will verify the proper operation of trip devices for main stop and control valves and combined intermediate valves (CIVs).

In response, the applicant stated that Section 14.2.8.1.53 describes the general test methods and acceptance criteria for the turbine control system, including proper operation of the main stop and control valves and CIVs in response to simulated signals related to turbine speed, load, and pressure. The applicant also stated that the turbine main stop, control, and CIVs will be equipped with fast-acting solenoid valves (i.e., trip devices) to facilitate fast closure in response to an overspeed signal, although this section does not specifically discuss overspeed or trip devices. The applicant stated that DCD Tier 1, Table 2.11.4-1 included testing of the control logic of the as-built overspeed protection system with simulated overspeed signals to verify closure of the valves that supply steam to the turbine upon receipt of an overspeed signal. The applicant also stated that it will revise DCD Tier 2, Section 14.2.8.1.53 to specifically address the verification of proper operation of turbine valve overspeed trip devices. The staff determined that performance of this test makes it possible to verify the proper operation of the trip devices required to prevent a turbine overspeed. The staff reviewed the test abstract in DCD Tier 2, Revision 3, Section 14.2.8.1.53 and finds that the revised text addresses the staff's

concerns and is acceptable. Accordingly, the staff concludes that the MTCS test description follows the guidance in RG 1.68 and is thus acceptable. Therefore, RAI 14.2-51 is resolved.

14.2.3.10.21 Main Turbine and Auxiliaries Preoperational Test

In RAI 14.2-53, the staff requested additional information regarding the main turbine and auxiliaries pre-operational test description in DCD Tier 2 Section 14.2.8.1.59. The staff asked the applicant to include testing of the overspeed trip system, which is consistent with the guidance in RG 1.68.

In response, the applicant indicated that it will add the following text to the DCD in a future revision:

> Proper operation of the turbine overspeed protection system to provide mechanical overspeed trip and electrical backup overspeed trip as specified in Subsection 10.2.2.4 and the manufacturer's technical instruction manual. (During the preoperational test phase, simulated speed signals will be used for these tests.)

The staff reviewed the test abstract in DCD Tier 2 Revision 3, Section 14.2.8.1.59 and finds that the revised text addresses overspeed trip testing. Accordingly, the staff concludes that the main turbine and auxiliaries test description follows the guidance in RG 1.68 and is thus acceptable. Therefore, RAI 14.2-53 is resolved.

14.2.3.10.22 Direct Current Power Supply System Preoperational Test

In RAI 14.2-55, the staff requested additional information on DCD Tier 2, Revision 1, Section 14.2.8.1.35. Specifically, on page 14.2-34, the sixth bullet did not accurately reflect the newly revised DCD for Chapter 8 (i.e., the ESBWR design will utilize only Class 1E batteries with a 72-hour duty cycle).

In DCD Tier 2, Revision 3, Section 14.2.8.1.35, the applicant revised the acceptance criterion in the sixth bullet to read as follows:

- Verify that safety-related batteries have the capacity to support Safety-Related loads for a period of 72 hours.

The staff finds this change clarifies the acceptance criteria in DCD Section 14.2.8.1.35 and follows the guidance in RG 1.68. Therefore, the test description is acceptable and RAI 14.2-55 is resolved.

14.2.3.10.23 Alternating Current Power Distribution System Preoperational Test

In RAI 14.2-57, the staff requested additional information regarding the alternating current (ac) power distribution system pre-operational test description in DCD Tier 2, Section 14.2.8.1.36. Specifically, the staff asked the applicant to describe the system tests that demonstrate proper termination of power and control cables.

In response, the applicant stated that per Appendix A to RG 1.68, construction and preliminary tests, including wiring continuity and separation checks, will be performed before the start of pre-operational testing. These tests will verify the proper termination of power and control and

will include point-to-point continuity, high pot, and fiber optic optical checks as applicable. Therefore, no change to Section 14.2.8.1.36 is needed to address the demonstration that power and control cables will be properly terminated. No DCD change is required in response to this RAI. The staff finds this response sufficiently clarifies ac power distribution testing requirements and is acceptable. Therefore, RAI 14.2-57 is resolved.

In RAI 14.2-98, the staff noted that DCD Tier 2, Revision 5, Section 14.2.8.1.36 states the following:

> Performance shall be observed and recorded during a series of individual component and integrated system tests to demonstrate the following: (1) Proper operation of initiating, transfer, and trip devices; (2) Proper operation of relaying and logic; (3) Proper operation of equipment protective devices, including permissive and prohibit interlocks; (4) Proper operation of instrumentation and alarms used to monitor system and equipment status; (5) Proper operation and load carrying capability of breakers, switchgear, transformers, and cables; (6) The capability of transfer between onsite and offsite power sources as per design; (7) The ability of emergency and vital loads to start in the proper sequence and to operate properly under simulated accident conditions; and (8) The adequacy of the plant emergency lighting system.

The staff asked the applicant to include the following additional items in the ITP or to justify their exclusion: (a) verification of analytically derived voltage values from voltage analyses of the onsite distribution system against actual measurements (Branch Technical Position 8-6), and (b) proper operation of the automatic transfer capability of normal preferred power source to the alternate preferred power source is verified. RAI 14.2-98 was being tracked as an open item in the SER with open items.

In response, the applicant stated the following:

> GEH concurs with Item (a) noted above. GEH considers Item (b) to be satisfied by existing requirements in DCD, Tier 2, Subsection 14.2.8.1.36 as described below:
>
> (a) An item will be added to DCD, Tier 2, Subsection 14.2.8.1.36 to verify the analytical derived voltage values of the onsite distribution system against actual measurements.
>
> (b) The requested verification of the transfer capability from the normal to preferred power source to the alternate preferred power source is satisfied by the existing requirements in DCD, Tier 2, Subsection 14.2.8.1.36, that verifies "Proper operation of initiating, transfer and trip devices." This verification includes proper operation of controls, relays and breakers required for transfer from the normal preferred power source to the alternate preferred power source.

The applicant planned to revise DCD Tier 2, Section 14.2.8.1.36 by adding the following bullet, as noted below:

> - Verify the analytical derived voltage values of the onsite distribution system against actual measurements.

The applicant added this information to DCD Tier 2, Revision 6, Section 14.2.8.1.36. Therefore, RAI 14.2-98 is resolved.

14.2.3.10.24 Standby Diesel Generator and Alternating Current Power System Preoperational Test

In RAI 14.2-59, the staff requested additional information regarding the standby DG and ac power system pre-operational test description in DCD Tier 2, Section 14.2.8.1.37. Specifically, the staff asked the applicant to describe the basis for the phrase “at a load equivalent to the continuous rating” that is used in the following quotation from DCD Tier 2, Revision 1, Section 14.2.8.1.37, on page 14.2-36:

> Fuel Load carrying capability of the DG for a period of not less than 24 hours, of which 22 hours are at a load equivalent to the continuous rating of the DG and 2 hours are at the manufacturer’s 2 hour load rating, including verification that the diesel cooling system functions within design limits, and that the HVAC System maintains the DG room within design limits.

The staff’s understanding is that the continuous rating should include kilovolt-amperes and the power factor.

In DCD Tier 2, Revision 3, Section 14.2.8.1.37, the applicant added the following criterion under “General Test Methods and Acceptance Criteria”:

- The DGs will be tested at full power and rated power factor for a period of 24 hours. This will ensure all diesel cooling and HVAC systems perform their design functions.

The staff finds that the above change is responsive to its question and is acceptable. Therefore, RAI 14.2-59 is resolved.

14.2.3.10.25 Pressure Suppression Containment Bypass Leakage Tests

In DCD Tier 2, Revision 2, Section 14.2.8.1.32 the applicant stated that an objective of the pressure suppression containment bypass leakage tests is to “verify that the suppression pool bypass leakage rate is within limits for high pressure and low pressure tests.” In RAI 14.2-63, the staff asked the applicant to provide the values of the high and low pressures and explain their significance.

In response, the applicant stated the following:

> A review of this RAI and Subsection 14.2.8.1.32 led, by reference, back to DCD, Chapter 6, Subsection 6.2.1.1.5 (Bypass Leakage and Surveillance). Subsection 6.2.1.1.5.4.1 (High Pressure Leak Test) was deleted in DCD, Tier 2, Revision 3. Chapter 14 will be revised to eliminate the description of high and low pressure tests. In addition, subsections under DCD, Chapter 6, Subsection 6.2.1.1.5 will be revised to be in line with the changes made in Chapter 14.
>
> The testing for bypass leakage in Chapters 6 and14 will consist of local leak rate testing at a single pressure plus visual inspections. Therefore, the request to

provide values for high and low pressure testing and their significance is no longer relevant.

In RAI 14.2-63 S01, the staff asked the applicant to measure the total bypass leakage without using unverified assumptions. The applicant subsequently revised DCD Tier 2, Section 14.2.8.1.32 to determine the overall suppression pool bypass leakage effective area and to confirm that the leakage value is within the limits of the low-pressure test acceptance criteria. The test method used will form the basis for leakage tests conducted at the same frequency as the integrated leak rate tests (ILRTs). In addition, the applicant revised the general test methods and acceptance criteria to verify that the calculated value of the overall suppression pool bypass leakage effective area ($A/\sqrt{K}$) is within the design limit specified in Section 6.2.1.1.5.

In response to RAI 14.2-63 S01, GEH proposed to update DCD Tier 2, Section 14.2.8.1.32 to include the statement that the "test method used will form the basis for use during subsequent leakage rate tests conducted at the same frequency as the ILRT."

In RAI 6.2-145 S02, the staff asked GEH to provide additional justification for this proposed change. In RAI 14.2-63 S02, the staff asked that GEH make the responses to RAIs 14.2-63 and 6.2-145 consistent.

In response to RAI 6.2-145 S02 and RAI 14.2-63 S02, the applicant proposed to change the TS suppression pool bypass test frequency from two years to 10 years. The staff approved TS license amendment requests for surveillance test frequencies of 10 years in existing plants but has not approved this test frequency for new plants. RAI 6.2-145 was being tracked as an open item in the Chapter 6 SER with open items. Resolution of RAI 6.2-145 is discussed in Section 6.2 of this report. This issue does not affect the pre-operational pressure suppression containment bypass leakage tests in DCD Tier 2, Section 14.2.8.1.32, since these are one time pre-operational tests to satisfy the requirements of RG 1.68 and SRP Section 14.2 for the ITP. Therefore, RAI 14.2-63 S02 is resolved.

14.2.3.10.26 Feedwater Control System Preoperational Test

In RAI 14.2.-65, the staff noted that as part of DCD Tier 2, Section 14.2.8.1.2, the following tests should be added for attributes of the triplicate FTDCs to be consistent with RG 1.68:

- Single and Three Element control
- Independence of controllers by taking each one, and then all combinations of two, out of service and verifying that the system is functioning properly
- Manual Feedpump Control—verify that each RFP can be fully controlled through the FTDCs

To be consistent with RG 1.68, each parallel processing channel should be tested for various design attributes.

In response, the applicant noted the following:

> Verification of the Single and Three Element controller is already encompassed within the statement to demonstrate the proper overall response of the control

system. This will be done while using simulated signals for inputs. No change to the DCD is required.

The applicant agreed to add a statement to verify, by demonstration, that the loss and then restoration of a single processor in the FTDCs will not cause substantial change to the system output signals, nor require operator action beyond recognition of an alarm when the processor is out of service. However, the simultaneous loss of two processors will not be demonstrated, as that condition goes beyond the fault-tolerant design of the FWCS. This position is consistent with DCD Tier 1, Revision 9, Chapter 2 (see DCD Tier 1, Revision 9, Table 2.2.3-2, Item 2).

The applicant agreed to add a statement to require pre-operational testing of each motor-driven reactor feed pump (MDRFP) using the manual control mode of the controller to the extent practical.

The applicant committed to add the following two bullets to DCD Tier 2, Section 14.2.8.1.2:

- Independence of system functional operation from loss of operation of one of the redundant channels of the FTDCs controllers/processors will be confirmed by test. Testing involves using simulated input signals and removing, then restoring the normal operation of each one of the three channels. During testing, important control system outputs are monitored and their response is used for confirming the system remains properly functional.

- Verification of each MDRFP will be made using the controller’s manual control mode with a flow path through the long path recycle line. Maximum test flow rate to be consistent with the equipment limitations.

The applicant added this information in DCD Tier 2, Revision 5, Section 14.2.8.1.2. Therefore, this addition resolves RAI 14.2-65.

14.2.3.10.27 Rod Control and Information System Preoperational Test

In RAI 14.2-66, the staff requested additional information regarding DCD Tier 2, Section 14.2.8.1.5 where the proper functioning of instrumentation should include status signals from the HCUs and a failure indication of any one position detector for an individual fine motion control rod drive (FMCRD).

In response, the applicant provided the following conclusions and new criteria regarding testing of the HCU and FMCRD:

> RC&IS and the N-DCIS, there are already tests and on-line diagnostics from both the RC&IS and the N-DCIS that provide proper functioning of the status signals from the HCUs and rod position detector failure for an individual FMCRD.

The applicant committed to revise the “General Test Methods and Acceptance Criteria” in Section 14.2.1.8.5 to add a new criterion after the third criterion as follows:

- Proper functioning of instrumentation used to monitor status signals from HCUs and failure indication of any one position detector for an individual FMCRD.

The applicant added this new information in DCD Tier 2, Revision 5, Section 14.2.8.1.5, thereby resolving RAI 14.2-66.

14.2.3.10.28 Radioactive Liquid Drainage and Transfer System Preoperational Test

In RAI 14.2-85, the staff requested additional information regarding the radioactive liquid drainage and transfer system pre-operational test description in DCD Tier 2, Section 14.2.8.1.40. Specifically, the staff asked the applicant to explain why the scope does not describe how the installation and operation of mobile waste processing systems will be integrated in this test.

In response, the applicant stated the following:

a. DCD Subsection 14.2.8.1.62 "Prerequisites", states, the construction tests have been successfully completed. Included in the construction tests are individual component tests. Interfaces between liquid waste management system (LWMS) and mobile systems will be included in these tests. The mobile equipment is designed to the requirements of RG 1.143, which insures all mobile equipment has the same standard of design as the LWMS. As stated in the RAI, the solid and liquid radwaste process relies on both permanently installed plant systems and mobile waste treatment systems. The preoperational testing described in DCD, Subsection 14.2.8.1.62 addresses both liquid and solid radwaste systems. Test requirements include:

   - Acceptable system and component flow paths and flow rates, including pump capacities and tank volumes

   - Proper operation of equipment controls and logic, including prohibit and permissive interlocks

   - Proper functioning of instrumentation and alarms used to monitor system operation and status,

   These tests could not be successfully completed if the plant systems and the mobile waste treatment systems were not interfacing as designed.

b. The mobile systems are designed in accordance with RG 1.143 and installation of the systems will follow quality assurance requirements to ensure that the installation follows the design requirements. Controlling and monitoring effluent release is described in Subsection 14.2.8.1.62 which states proper operation of equipment protective features and automatic isolation functions, including those for ventilation systems and liquid effluent pathways; and proper functioning of instrumentation and alarms used to monitor system operation and status is verified. GEH response to RAI 11.5-23, MFN 07-030, dated April 10, 2007, revised DCD, Subsection 11.5.7.2 to require the COL Applicant to provide programmatic details, ODCM, for monitoring and controlling the release of radioactive material to the environment.

c. The applicant's response to RAI 11.2.3-1, Supplement No. 1, MFN 07-371, dated July 13, 2007, changed DCD Tier 2, Table 11.2-3 to require filtration and adsorbent media meet or exceed the decontamination factors listed.

d. The applicant's response to RAI 11.2.3-1 Supplement No. 1, MFN 07-371, dated July 13, 2007, changed DCD Tier 2, Table 11.2-3 to require filtration and adsorbent media meet or exceed the decontamination factors listed.

In response to this RAI, the applicant made no changes to DCD Section 14.2.8.1.40. On the basis of the preceding information, the staff agrees with the applicant's response, and RAI 14.2-85 is resolved.

14.2.3.10.29 Off-gas System Preoperational Test

In RAI 14.2-86, the staff requested additional information regarding the off-gas system pre-operational test description in DCD Tier 2, Section 14.2.8.1.48. Specifically, the staff requested that the applicant clarify the scope of this pre-operational test. The test does not describe the process that will be used in confirming the proper selection and performance characteristics of the media to treat gaseous process, waste, and effluent streams.

In response, the applicant stated the following:

> DCD, Subsection 11.3.2.1, "Adsorption" provides design criteria for the charcoal media such as vendor tests of charcoal for krypton and xenon adsorption. During the preoperational test phase a prerequisite to off-gas testing is verification that the correct amount of charcoal has been loaded in the absorber beds and that the charcoal that is being used meets the requirements for charcoal described in DCD, Subsection 11.3.2.1. Off-gas performance can only be confirmed during startup testing when there are radionuclides in the waste stream. The startup test for the off-gas system is described in DCD, Subsection 14.2.8.2.29. Subsection 14.2.8.2.1 describes the samples taken to verify off-gas performance.
>
> a. The adsorbent media for the guard and charcoal beds is described in DCD, Subsection 11.3.1, Table 11.3-1. The charcoal mass is no less than 33,000 lbs for the guard beds and 490,000 lbs for the charcoal beds. The guard and charcoal beds are sized to process three times the source term without affecting delay time of the noble gases (30-minute).
>
> b. DCD, Subsection 14.2.8.2.29 describes the startup testing of the off-gas system. The performance of the charcoal absorbers is tested to verify that the radioactivity effluents meet the TS limits. COL Information Item 11.5.-2-A states the COL Applicant will develop an ODCM that will include programs for monitoring and controlling the release of radioactive material to the environment.

The response to this RAI was also tied to the disposition of RAIs 11.5-47 and 12.2-9 S02, which were resolved separately. The applicant did not revise DCD Section 14.2.8.1.48 to address this RAI. The staff agrees with the applicant's response, and RAI 14.2-86 is resolved.

In RAI 14.3-157, the staff requested additional information regarding the off-gas system test abstract in DCD Tier 2, Section 14.2.8.1.48. The staff determined that the acceptance criteria specified in DCD Tier 2, Revision 4, Section 14.2.8.1.48, were inconsistent with DCD Tier 1, Revision 4, Section 2.10.3, and DCD Tier 2, Revision 4, Section 11.5.3.2.2. Specifically, the test methods and acceptance criteria do not identify a test to demonstrate the proper closure of the isolation valve on high-radioactivity levels. Accordingly, the staff asked the applicant to revise the acceptance criteria listed in DCD Tier 2, Revision 4, Section 14.2.8.1.48 to include a confirmation of system isolation on high-radioactivity level signals. This issue is related to an ITAAC under DCD Tier 1, Section 2.10.3.

In response, the applicant added the following information in DCD Tier 2, Revision 5, Section 14.2.8.1.48, in the fourth bullet under "General Test Methods and Acceptance Criteria":

- Proper operation of system valves, including isolation features, under expected operating conditions, including isolation of the off-gas system discharge valve upon receipt of high radioactivity level signals:

Since this addressed the operation of off-gas system isolation on a high-radioactivity level, the staff finds that this response is acceptable. Therefore, RAI 14.3-157 is resolved.

14.2.3.10.30 Nuclear Boiler System, Standby Liquid Control System, and Gravity Driven Cooling System Preoperational Tests

In RAI 14.2-64, the staff requested additional information regarding equipment or components that cannot be actuated without damage or without upsetting the plant. In the response to RAI 14.12-64, the applicant stated, in part, that actuation of equipment or components during either pre-operational or startup test programs should not cause damage or upset the plant to an extent that there would be damage. The applicant recognized that some components are designed for single-use actuation (e.g., squib valves). The applicant also agreed that it should acknowledge the acceptability of isolating these devices to prevent them from being actuated during pre-operational tests.

The applicant also stated that the ESBWR utilizes single-use squib valves in the ADS, GDCS, and SLCS. The applicant will add a statement allowing the isolation of these single-use components before the pre-operational tests of these three systems. Accordingly, the applicant will revise the "Prerequisite" sections of DCD Tier 2, Sections 14.2.8.1.1, 14.2.8.1.3, and 14.2.8.1.65 by adding the following statement:

- To prevent actuation of single use squib valves during the logic portion of this testing process, the valve(s) may be isolated electrically to prevent actuation. This isolation, verification of the firing signal during the test, and reconnection process must be controlled within the test document.

The applicant added this information to the sections mentioned above in DCD Tier 2, Revision 5, thus resolving RAI 14.2-64.

14.2.3.10.31 Preoperational Test Descriptions Conclusions

On the basis of its review of DCD Tier 2, Revision 9, Section 14.2.8.1, the staff finds that the test abstracts provided by the applicant are generally consistent with the pre-operational test criteria in RG 1.68 and SRP Section 14.2 and therefore acceptable. However, since the

Licensee will be responsible for the development of detailed test specifications and test procedures, the staff finds that it is acceptable to defer the development of these documents until the post COL phase.

14.2.3.11 *Initial Startup Test Descriptions*

In DCD Tier 2, Revision 6, Section 14.2.8.2 the applicant provided the following 38 test abstracts for the initial startup testing phase:

(1)	14.2.8.2.1	Chemical and Radiochemical Measurements Test
(2)	14.2.8.2.2	Radiation Measurements Test
(3)	14.2.8.2.3	Fuel Loading Test
(4)	14.2.8.2.4	Full Core Shutdown Margin Demonstration Test
(5)	14.2.8.2.5	CRD System Performance Test
(6)	14.2.8.2.6	NMS Performance Test
(7)	14.2.8.2.7	Core Performance Test
(8)	14.2.8.2.8	Nuclear Boiler Process Monitoring Test
(9)	14.2.8.2.9	System Expansion Test
(10)	14.2.8.2.10	System Vibration Test
(11)	14.2.8.2.11	Reactor Internals Vibration Test (Initial Startup Flow-Induced Vibration (FIV) Testing)
(12)	14.2.8.2.12	Feedwater Control Test
(13)	14.2.8.2.13	Pressure Control Test
(14)	14.2.8.2.14	Plant Automation and Control Test
(15)	14.2.8.2.15	Feedwater System Performance Test
(16)	14.2.8.2.16	Main Steam System Performance Test
(17)	14.2.8.2.17	RWCU Cooling System Performance Test
(18)	14.2.8.2.18	PSWS Performance Test
(19)	14.2.8.2.19	HVAC System Performance Test
(20)	14.2.8.2.20	Turbine Valve Performance Test
(21)	14.2.8.2.21	MSIV Performance Test
(22)	14.2.8.2.22	SRV [Safety/Relief Valve] Performance Test
(23)	14.2.8.2.23	Loss of Feedwater Heating Test
(24)	14.2.8.2.24	Feedwater Pump Trip Test
(25)	14.2.8.2.25	Shutdown from Outside the MCR Test
(26)	14.2.8.2.26	Loss of Turbine Generator and Offsite Power Test
(27)	14.2.8.2.27	Turbine Trip and Generator Load Rejection Test
(28)	14.2.8.2.28	Reactor Full Isolation Test
(29)	14.2.8.2.29	Off-gas System Test
(30)	14.2.8.2.31	Concrete Penetration Temperature Surveys Test
(31)	14.2.8.2.32	Liquid Radwaste System Performance Test
(32)	14.2.8.2.33	Steam and Power Conversion System Performance Test
(33)	14.2.8.2.34	Isolation Condenser (IC) Performance Test
(34)	14.2.8.2.35.1	Reactor Pre Critical Heatup with RWCU/SDC
(35)	14.2.8.2.35.2	ICS Heatup and Steady State Operation
(36)	14.2.8.2.35.3	Power Maneuvering in the FW Temperature Operating Domain
(37)	14.2.8.2.35.4	Load Maneuvering Capability
(38)	14.2.8.2.35.5	Defense-in-Depth Stability Solution Evaluation Test

In RAI 14.2-101, the staff identified the five FOAK tests in the final safety analysis report (FSAR), Section 14.2.8.2.35, as Tier 2* information that is subject to NRC review and approval.

The staff requested the applicant to identify these FOAK tests as Tier 2* information in DCD Tier 2, Section 14.2.8.3.35.

In response, the applicant revised Section 14.2.8.2.7, "Core Performance Test," Description Section to italicize the second paragraph and bracket Tier 2* information for a FOAK test observation of reactor stability. In addition, DC applicant revised all of Section 14.2.8.2.35, "ESBWR First of A Kind Tests," to italicize and bracket all Tier 2* information in this section.

The applicant added this change to DCD Tier 2, Revision 6, Section 14.2.8.2.35. Therefore, RAI 14.2-101 is resolved.

SRP Section 14.2 and RG 1.68 provide general guidance on the conduct of the ITP after the completion of pre-operational testing. Following verification of SSC functional capability during pre-operational testing, the ITP transitions to initial fuel loading, pre-critical testing, initial startup, low-power testing, and power ascension testing. After core loading, sufficient tests and checks will be performed to ensure that the facility will be in a final state of readiness to achieve criticality and perform low-power testing.

As described in RG 1.68, after the initial reactor startup, low-power testing will be conducted to: (1) confirm the design, (2) validate analytical models and verify, to the extent practical, that the assumptions used in the safety analysis are conservative, and (3) confirm the operability of plant systems and design features that could not be completely tested during the pre-operational test phase because of the lack of an adequate heat source for the reactor coolant system (RCS) and the main steam system. Power ascension testing will be conducted to demonstrate that the facility can be operated in accordance with the design during normal steady-state conditions, and, to the extent practical, during and following anticipated transients. SRP Section 14.2 contains criteria for startup and power ascension testing to ensure that test abstracts include objectives, prerequisites, test methods, and acceptance criteria to establish the functional adequacy of SSCs and design features.

The staff reviewed the initial startup test abstracts in DCD Tier 2, Section 14.2.8.2. In comparing the ESBWR initial startup testing to the testing recommended in RG 1.68, Appendix A Section 2, "Initial Fuel Loading and Precritical Tests"; Section 3, "Initial Criticality"; Section 4, "Low-Power Testing"; and Section 5, "Power-Ascension Tests"; the staff identified several areas where it required additional information to complete its review. Descriptions of the specific issues follow.

14.2.3.11.1 Chemical and Radiochemical Measurement Test

In RAI 14.2-90, the staff noted that DCD Tier 2, Revision 4, Section 14.2.8.2.1 provides an incomplete description of the criteria for radioactivity that are present in gaseous and liquid effluents. Specifically, Section 14.2.8.2.1 limits the criteria to "licensee limitations" and does not include the NRC effluent concentration limits in Table 2 of Appendix B, "Annual Limits on Intake (ALIs) and Derived Air Concentrations (DACs) of Radionuclides for Occupational Exposure; Effluent Concentrations; Concentrations for Release to Sewerage," to 10 CFR Part 20, "Standards for Protection Against Radiation." Accordingly, the staff asked the applicant to revise Section 14.2.8.2.1 (criteria) to include Table 2 of Appendix B to 10 CFR Part 20 as one set of criteria, and to change "license limitations" to "license conditions." This RAI also applied to the criteria identified for the off-gas system test (DCD Tier 2, Section 14.2.8.2.29) and the

LRT performance test (DCD Tier 2, Section 14.2.8.2.32). The staff requested that the applicant revise these sections accordingly.

In response, the applicant agreed to revise Sections 14.2.8.2.1, 14.2.8.2.29, and 14.2.8.2.32 to provide a complete description of the criteria for radioactivity that are present in gaseous and liquid effluents. The applicant added this information to these DCD sections in Revision 5, thereby resolving RAI 14.2-90.

In RAI 14.2-96, the staff asked the applicant to describe the scope of filter performance associated with radiochemical measurements. Specifically, the staff noted that the description should include charcoal media and should clarify that the filters include high-efficiency particulate air (HEPA) filters used for the purpose of controlling airborne radioactive effluent discharges. In addition, the staff noted that the description should include filters and strainers and the reverse osmosis subprocessing system used to process liquid effluents. Accordingly, the staff requested that the applicant revise DCD Tier 2, Section 14.2.8.2.1 (under “Description”) to include HEPA filters, charcoal media, filters and strainers, and reverse osmosis subsystems.

In response, the applicant agreed that the “Purpose” description should be extended to include gaseous process streams, so that the Licensee could assess fuel performance for evidence of fission product leakage into the RCS. The applicant also stated that the testing of HEPA and charcoal filters is periodically performed as part of the plant TS. Therefore, it is not appropriate to add this detail to Section 14.2.8.2.1. The applicant also agreed to add carbon filters and reverse osmosis treatment units to Section 14.2.8.1.62. The applicant added the information discussed in the above response to DCD Tier 2, Revision 5, Sections 14.2.8.2.1 and 14.2.8.1.62, thus resolving RAI 14.2-96.

14.2.3.11.2 Radiation Measurements Test

DCD Tier 2, Revision 4, Section 14.2.8.2.2 includes the test descriptions of radiation measurements tests. To verify that the established radiation zones (which determine plant area accessibility) will be accurate, the staff requested in RAI 14.2-94 that the applicant perform radiation surveys throughout the plant for all accessible areas, including all potentially high and very high radiation areas.

In response, the applicant updated DCD Tier 2, Revision 5, Section 14.2.8.2.2 to state that radiation surveys will be performed in all potentially high and very high radiation areas, thus resolving RAI 14.2-94.

14.2.3.11.3 Fuel Loading Test

In RAI 14.2-42, the staff requested additional information regarding the fuel loading test description in DCD Tier 2, Section 14.2.8.2.3. Section 2 of Appendix A to RG 1.68 recommends tests to be performed after the core is fully loaded. Specifically, Item C in Section 2 of Appendix A to RG 1.68 recommends “final functional testing of the RPS to demonstrate proper trip points, logic, and operability of scram breakers and valves.” It also recommends that testing “demonstrate operability of manual scram functions.” However, in DCD Tier 2, Section 14.2.8.2.3, the testing recommended by RG 1.68 was planned to be conducted before (instead of after) commencing fuel loading. The staff asked the applicant to discuss whether the tests listed above will be conducted after the core is fully loaded or to justify the lack of such a plan.

In response, the applicant stated that Section 2 of Appendix A to RG 1.68 recommends a list of tests and verifications that should be conducted during or following initial fuel loading. The applicant stated that it would remove the bulleted item under Section 14.2.8.2.3 that describes the guidance recommended in Section 2 of Appendix A to RG 1.68 and add it to Section 14.2.8.1.9. The applicant agreed to move the subject tests to the RPS pre-operational test description and stated explicitly that those tests will be conducted during or following initial fuel loading. The staff reviewed the test abstract in DCD Tier 2, Revision 3, Sections 14.2.8.2.3 and 14.2.8.1.9, and finds that the revised text is consistent with RG 1.68 and is acceptable. Accordingly, the staff concludes that the fuel loading test description follows the guidance in RG 1.68 and is therefore acceptable. Thus, RAI 14.2-42 is resolved.

In RAI 14.2-43, the staff requested additional information regarding the fuel loading test description. Section 2 of Appendix A to RG 1.68 recommends that a "prediction of core reactivity should be prepared in advance to aid in evaluating the measured responses to specified loading increments." The staff asked the applicant to clarify whether it will be prepared to provide predictions of core reactivity and what actions it would take if the measured results deviate from expected values.

In response, the applicant stated that shutdown margin tests provide the greatest assurance of core subcriticality. To that end, the Licensee will make predictions of shutdown margin before initial fuel loading. In addition, the applicant stated that to comply with the requirements of Section 2 of Appendix A to RG 1.68, it will add to the description under DCD Tier 2, Section 14.2.8.2.3, the statement that "Criteria for and actions required to address any deviations from expected results will be delineated in the fuel loading procedures as described in Section 14.2.2." The staff agreed that shutdown margin tests will provide the assurance of core subcriticality. The staff reviewed the test abstract in DCD Tier 2, Revision 3, Section 14.2.8.2.3 and finds that the revised text clarifies subcriticality prediction criteria and is acceptable. Accordingly, the staff concludes that the fuel loading test description follows the guidance in RG 1.68 and is therefore acceptable. Thus, RAI 14.2-43 is resolved.

14.2.3.11.4 Neutron Monitoring System Performance Test

In RAI 14.2-78, the staff requested that the applicant provide additional information on the NMS performance test. The staff asked the applicant to include the GT system verification of the NMS performance test in DCD Tier 2, Section 14.2.8.2.6. Specifically, in the section titled "Criteria," a sentence states, "The LPRMs shall be calibrated consistent with design specifications." However, this statement does not specify how the LPRMs, including the GT system, will be calibrated.

In response, the applicant stated that in accordance with DCD Tier 2, Sections 7.2.2 and 7.7.6, the LPRMs will be calibrated based upon calibration factors provided by the AFIP GT subsystem. The accuracy of this calibration shall be consistent with the GT Licensing Technical Report in NEDE-33197P, GT System for LPRM Calibration and Power Shape Monitoring.

The applicant added this information in DCD Tier 2, Revision 5, Section 14.2.8.2.6. The staff finds this change acceptable. Therefore, RAI 14.2-78 is resolved.

In RAI 14.2-79, the staff requested that the applicant provide additional information in DCD Tier 2, Section 14.2.8.2.6. Specifically, the staff asked the applicant to clarify the criteria for the SRNM count rates under "design requirements" and the overlapping neutron flux indications under "design specification," with regard to the criteria found in the TS.

The applicant provided the following response to RAI 14.2-79:

> The ESBWR TSs do not specify numerical values for count rates, only “count rates indicative of neutron flux levels within the core.” (Reference DCD, Tier 2, Chapter 16, TS Basis B3.3.1.6, SRP Section 3.3.1.6.4)
>
> DCD Tier 2, Table 7.2-2 provides specific count values required during the SRNM operation.
>
> The ESBWR TSs do not specifically require that the SRNM and LPRM ranges overlap. However, the TS Bases do note the following requirements:
>
> - “The SRNM cover the range of plant operation from source range through startup range (i.e., more than 10 percent of reactor rated power).”
>
> - “The APRM cover the range of plant operation from a few percent to greater than rated power.”
>
> (Reference DCD Tier 2, Chapter 16, TS Basis B3.3.1.6)
>
> - Because “a few percent” APRM is less than “more than 10 percent” SRNM, then an overlap of the two instrument ranges does occur.
>
> - A description of the SRNM and APRM LPRM overlapping ranges is provided in DCD Tier 2, Subsection 7.2.2.1 and on DCD Tier 2, Figure 7.2-3.

Revision 5 to the “Criteria” discussion in DCD Tier 2, Section 14.2.8.2.6, cross-references to DCD Tier 2, Section 7.2.2. On the basis of these changes to DCD Tier 2, Section 14.2.8.2.6, the staff finds that this response is acceptable, and RAI 14.2-79 is resolved.

14.2.3.11.5 Core Performance Test

In RAI 14.2-44, the staff requested additional information regarding the core performance test description in DCD Tier 2, Section 14.2.8.2.7. Specifically, the staff requested that the applicant describe the specific methods for calculating core flows and core power, including the variables that will be obtained from the in-vessel measurement to calculate core flows and core power. The staff also asked the applicant to provide a detailed test plan for testing vessel natural circulation at various power levels after fuel loading and during startup testing.

In response, the applicant provided derivations for the core mass flow rate and core power from mass and energy balance equations. The applicant clearly explained variables in the equations and noted that they will be obtained from in-vessel measurements or from measurements on the coolant systems connected to the reactor, or they will be evaluated based on correlations. This response answered the staff’s question concerning how to calculate core flows and core power, and the staff finds it acceptable.

For the test plan, the applicant clarified that a detailed startup test procedure will be written during the procedure preparation phase, in accordance with the description in DCD Tier 2, Section 14.2.8.2.7. This RAI response identifies the power range for the tests. The applicant will present the written startup test procedure to the NRC for formal review, in accordance with

the SAM preparation scheduling. This response resolves RAI 14.2-44. However, this startup test procedure will be developed as part of COL Information Items 14.2-2-A and 14.2-3-A.

In RAI 14.2-89, the staff asked the applicant to provide a startup testing plan to identify the impacts, if any, of operating at reduced power levels where flow-transition-induced flow oscillations may be possible. In response, the applicant stated that it will add the following information to DCD Tier 2, Section 14.2.8.2.7 under the "Description" section:

> A FOAK test will be conducted for observation of reactor stability. The objective of this test is to characterize the stability performance during power ascension, where chimney partition may experience flow-regime-transition-induced flow oscillation. The test will begin at 20 percent thermal power and the first time the reactor achieves a new 5 percent power increment above that point. The test will collect pertinent LPRM data to identify stability performance characteristics and determine a decay ratio during the ascension to rated reactor power. The monitoring LPRM signals are filtered to remove noise components with frequencies above the range of stability related to power oscillation. This data will be collected at sufficient instances to capture the development of instability pattern (if any) that may occur during the ascent to rated power.

With this change in DCD Tier 2, Revision 5, Section 14.2.8.2.7, the staff finds that RAI 14.2-89 is resolved.

14.2.3.11.6 System Expansion Test

The purpose of the thermal expansion test is to confirm that the pipe suspension system is working as designed and the piping is free of obstructions during power changes. Upon completion of the thermal expansion test, the measured and observed pipe expansion should be in accordance with the design, and the piping should return to its approximate cold condition after cooldown. The staff could not determine whether the applicant's testing program would achieve this objective.

In RAI 14.2-26, the staff asked the applicant to provide the type and source of design performance information that will be used in the development of detailed test procedures for system expansion testing. The staff found that DCD Tier 2, Section 14.2.8.2.9 did not contain sufficient information about to the design performance and test procedures for the staff to assess the adequacy of the development of the system expansion test procedures.

In response, the applicant stated the following:

> DCD, Tier 2, Subsection 14.2.8.2.9 describes the prerequisites and the acceptance criteria conditions for the thermal expansion testing. Additional detail and special requirements for a thermal expansion test will be performed in accordance with the test procedure that would be developed and evaluated against acceptance criteria.

In DCD Tier 2, Revision 3, Section 14.2.8.2.9, the applicant amended the section to include the test procedure requirements. Based on its review of the revised version of the DCD, the staff finds DCD Tier 2, Section 14.2.8.2.9 acceptable because the applicant provided the test procedure requirements, as requested. Thus, the staff determined that RAI 14.2-26 is resolved.

This is addressed by COL Information Item 14.2-3-A in DCD Tier 2, Revision 6, Section 14.2.2.2.

In RAI 14.2-29, the staff asked the applicant to provide additional information about the system expansion test program schedule and the sequence for conducting the tests planned for the startup test phase. Also, the staff requested that the applicant state the time available between the approval of testing procedures and their intended use.

In response, the applicant stated the following:

> Table 14.2-1 provides the test matrix for various systems. DCD, Tier 2, Subsection 14.8.2.2 states that the power ascension test phase procedures will be made available to the NRC 60 days prior to the fuel loading. In addition, to insure the tests are conducted in accordance with the established methods and acceptance criteria, the associated plant testing specification(s) is made available to the NRC.

The staff finds the applicant's response acceptable. The Licensee will develop plant test specifications, test procedures, and acceptance criteria before the fuel loading and will make them available to the NRC. Therefore, the concerns related to RAI 14.2-29 are resolved. This is addressed by COL Information Item 14.2-2-A in DCD Tier 2, Revision 6, Section 14.2.2.1.

In RAI 14.2-30, the staff requested additional information regarding the special test of the effects of thermal stratification in the feedwater discharge piping. Specifically, the staff asked that the applicant address the staff''s concern about DCD Tier 2, Section 14.2.8.2.9. The staff found that the section did not contain sufficient information regarding the special tests that will be conducted to monitor the effects of thermal stratification in the feedwater discharge piping to establish the functional adequacy of this piping.

In response, the applicant stated that it will revise DCD Tier 2, Section 14.2.8.2.9 to add requirements to include the acceptance criteria for the effects of thermal stratification in the test procedure for the feedwater discharge piping. In addition, the applicant stated that it will include requirements for thermal expansion testing as requested. The staff reviewed the test abstract in DCD Tier 2, Revision 3, Section 14.2.8.2.9 and finds that the revised text is acceptable. Accordingly, the staff concludes that the system expansion test description addresses the staff's concern, meets the guidance of RG 1.68, and is therefore acceptable. RAI 14.2-30 is resolved.

14.2.3.11.7 System Vibration Test

In RAI 14.2-32, the staff asked the applicant to provide the type and source of design performance information that will be used in the development of detailed system vibration test procedures. In response, the applicant stated that DCD Tier 2, Section 14.2.8.2.10 identifies the critical systems that would require vibration testing. In addition, the applicant revised DCD Tier 2, Section 14.2.8.2.10 to add requirements to the test procedure to include past experience with vibration testing of earlier BWR piping systems as guidance for developing a test procedure description and acceptance criteria. The staff finds that the bases for the development of detailed system vibration test procedures are reasonable and acceptable. Therefore, RAI 14.2-32 is resolved.

In RAI 14.2-35, the staff asked the applicant to provide information about the vibration test program schedule and sequence for the system vibration test phase. The applicant provided

this information in DCD Tier 2, Revision 4, Section 14.2.7, which states (in part) that nine months is allowed for conducting the pre-operational test phase before the fuel loading date, and three months is allowed for conducting the startup and power ascension that commences fuel loading. Test procedure preparations are scheduled such that approved procedures are available to the NRC 60 days before their intended use or 60 days before fuel load for power ascension test procedures. On the basis of this information, RAI 14.2-35 is resolved. This is addressed by COL Information Items 14.2-2-A and 14.2.4-A in DCD Tier 2, Revision 6, Sections 14.2.2.1 and 14.2.7.

14.2.3.11.8 Reactor Internals Vibrations Test (Initial Startup Flow-Induced Vibration Testing)

In RAI 14.2-24, the staff asked the applicant to discuss the expansion, vibration, and dynamic effects test programs for conformance with applicable RGs, including RG 1.20. In response to this RAI, the applicant stated that the development of the test criteria will require consideration of the potential adverse flow effects on piping systems, as recommended in RG 1.20 and in SRP Sections 3.9.2 and 3.9.5. The applicant did not request any exceptions to the regulatory positions recommended in the applicable RGs. In addition, nuclear power plant operating experience has revealed the potential for adverse flow effects from vibration caused by hydrodynamic loads and acoustic resonance within reactor coolant, steam, and feedwater systems, as well as reactor internal components such as steam dryers. However, the system vibration test for the piping systems discussed in DCD Tier 2, Revision 5, Section 14.2.8.2.10 does not address these potential adverse flow effects. Therefore, the staff asked the applicant to describe the implementation of the program to address potential adverse flow effects on safety-related piping and components in these systems. RAI 14.2-24 S01 was being tracked as an open item in the SER with open items.

In its response to RAI 14.2-24 S01, the applicant stated the following:

> The following startup measurements, instrumentations and analyses address the potential adverse flow effects on safety-related piping and components in these systems applicable to RG 1.20 requirements:
>
> The details of main steam line acoustic monitoring testing were provided in the response to RAI 3.9-134.
>
> Vibration sensors on susceptible valve operators provide on-line condition monitoring to alert potential valve operator failure due to acoustic resonance. Normally sensors are installed at locations where higher acceleration responses due to dynamic loads, such as seismic and other building filtered loads are expected. The measured values will be compared with manufacturer's or IEEE allowable limits.
>
> Instrumentation inside the safety-related flow systems are evaluated for responses due to vortex shedding and other potential acoustic effect. The thermal well, velocity and pressure sensors in the feedwater and main steam pipes are examples. Similarly, for components in non-safety-related systems where damage of such instrumentations might be carried into safety-related systems, and impact the performance of components such as isolation or check valves, an evaluation will also be performed. The analysis will be performed in

accordance with ASME Appendix N. The calculated stresses will meet American National Standard, ANSI/ASME OM-S/G criteria.

> The preoperational and startup test requirements have been provided in the response to RAI 3.9-70. The test hold points are described in the response to RAI 3.9-68.

In accordance with the guidance for flow-induced vibration testing in RG 1.2, the staff reviewed the response to RAI 14.2-24 S01 and finds it acceptable. Therefore, RAI 14.2-24 S01 is resolved.

In RAI 14.2-97, the staff expressed concerns that the discussions of the test description and acceptance criteria for the reactor internals vibration test program (Initial Startup Flow Induced Vibration Testing) in ESBWR Revision 5, Section 14.2.8.2.11, are too broad and general. The staff also indicated that there is no reference to the GEH Licensing Topical Report NEDE-33259P, Revision 1, "Reactor Internals Flow Induced Vibration Program," which contains an item-by-item discussion of the components requiring testing during the startup test program of the first ESBWR, as well as the types and locations of the sensors for monitoring FIV behavior. The applicant should revise the test description in ESBWR Section 14.2.8.2.11 to include a discussion demonstrating conformance with this topical report and other applicable references in the ESBWR DCD. The applicant's current approach to steam dryer load definition is identified as the plant-based load evaluation method, which is discussed in Licensing Topical Report NEDC-33408P, "ESBWR Steam Dryer-Plant Based Load Evaluation Methodology." The development of the FIV loads, as described in this report, is in accordance with RG 1.20, Revision 3. The FIV loads will be used in combination with other design loads in qualifying the steam dryer as described in Licensing Topical Report NEDE-33313P, "ESBWR Steam Dryer Structural Evaluation." The staff requested that the applicant discuss conformance with these licensing topical reports in DCD Tier 2, Section 14.2.8.2.11.

RAI 14.2-97 was being tracked as an open item in the SER with open items.

The applicant provided the following response:

> A description of the Flow Induced vibration program and associated startup testing is provided in DCD, Tier 2, Section 3L. Section 3L includes references to topical reports NEDE-33259P, Revision 1, NEDC-33408P, and NEDE-33313P. A reference to DCD Tier 2, Subsection 3.9.2.4 will be added to Subsection 14.2.8.2.11.

The applicant plans to revise DCD Tier 2, Section 14.2.8.2.11 to state:

> A complete description of the reactor internals vibration test program is provided in Subsection 3.9.2.4.

The applicant added this information to DCD Tier 2, Revision 6, Section 14.2.8.2.11. Therefore, RAI 14.2-97 is resolved.

14.2.3.11.9 Feedwater Control Test

In RAI 14.2-80, the staff requested additional information regarding DCD Tier 2, Section 14.2.8.2.12. Specifically, the staff asked that the criteria section be expanded to include

open and closed loop testing to check the dynamic flow response of the main feedwater actuators and the dynamic response of the master level controller, respectively.

In response, the applicant stated the following:

> During the preoperational test, FWCS open loop and closed loop testing will be performed.
>
> In control system open loop testing, the demand of the low flow controller or the Adjustable Speed Drive feedwater pump speed controller will be adjusted and the feedwater flow will be monitored to check the dynamic response of the feedwater low flow control valve actuator position or variable frequency drive pump speed.
>
> In control system closed loop testing, the master level controller's set point will be adjusted and the feedwater flow and reactor water level will be monitored to check the dynamic response of the FWCS.

In accordance with this response, the applicant plans to revise the "Criteria" description in DCD Tier 2, Section 14.2.8.2.12, as shown below:

> The FWCS performance shall be stable such that any type of divergent response is avoided. Through the Open and Closed Loop testing, the response shall be sufficiently fast but with any oscillatory modes of response well damped, usually with decay ratios less than 0.25.

On the basis of this change in DCD Tier 2, Revision 5, Section 14.2.8.2.12, the staff finds this response acceptable. Therefore, RAI 14.2-80 is resolved.

14.2.3.11.10 Plant Service Water System Performance Test

In RAI 9.2-24, the staff requested that the applicant describe in Section 14.2.8.2.18 the automatic actuation of the PSWS standby loop or the actuation of both loops following a loss of power. The applicant should describe how this test will not result in a significant water-hammer event, with the PSWS return aligned to either the natural draft or mechanical draft cooling towers.

In response, the applicant modified a bullet in DCD Tier 2, Revision 6, Section 14.2.8.1.51 to state:

> Proper operation of system valves, including automatic air release/vacuum valves, including timing, under expected operating conditions.

The staff finds that this addresses mitigation of water hammer while performing pre-operational testing of the PSWS. The air release/vacuum valves remove any air in the service water system to prevent water hammer before pre-operational testing begins (e.g., starting the service water pumps). The staff finds that the modified bullet in DCD Tier 2, Revision 6, Section 14.2.8.1.51 is acceptable, and it is unnecessary to modify DCD Tier 2, Section 14.2.8.2.18; therefore, this portion of RAI 9.2-24 is resolved.

14.2.3.11.11 Liquid Radwaste System Performance Test

The staff identified an inconsistency in DCD Tier 2, Revision 4, Section 14.2.9 and Table 14.2-1, with respect to the scope of the test matrix assigned during power ascension for the liquid radwaste system (LRS). Specifically, Table 14.2-1 did not include midpower as a testing plateau in confirming the performance of the LRS. This omission was inconsistent with the design objective of the liquid radwaste processing system of DCD Tier 2, Revision 4, Section 11.2, which stated that the system was designed to control, collect, process, handle, store, and dispose of liquid wastes generated during normal operation and anticipated occurrences, without making any distinctions among the various phases of power ascension or operation.

In RAI 14.2-91, the staff requested that in accordance with RG 1.68, the applicant revise DCD Tier 2, Table 14.2-1 to include midpower as a testing phase during reactor power ascension. This change to the LWMS test matrix would make it consistent with the text matrix assigned for the gas waste management system/off-gas system.

In response, the applicant agreed to identify performance testing of the LRS in the midpower plateau as a point to conduct LRS. The applicant added this information to DCD Tier 2, Revision 5, Table 14.2-1 thereby resolving RAI 14.2-91.

14.2.3.11.12 Steam and Power Conversion System Performance Test

In RAI 14.2-54, the staff requested additional information in DCD Tier 2, Section 14.2.8.2.33. Specifically, the staff asked the applicant to provide acceptance criteria for each of the power conversion systems and components, similar to the descriptions of Level 2 acceptance criteria in Section 14.2.12.2.39 of the advanced BWR DCD to ensure that all power conversion systems and components meet their design criteria.

In DCD Tier 2, Revision 3, Section 14.2.8.2.33 the applicant added the following information to the “Criteria” section:

> Performance characteristics (such as pressures, flows, temperatures, voltage, amps) of the various systems in the power conversion systems and related subsystems will be monitored and the data obtained will be evaluated against the systems process flow diagrams or equivalent design basis information. Any deviations observed will be evaluated to determine the cause and significance of the deviation.

In addition, in its response to RAI 14.2-54, the applicant stated that the test specifications and test procedures to be created for each plant will provide the detailed test criteria, including the level of the criteria that defines the actions required if the test criteria are not met. This is COL Information Item 14.2-3-A, which will be available to the NRC 60 days before its intended use. The staff finds these changes to DCD Tier 2, Revision 3, Section 14.2.8.2.33 and the COL information item acceptable. Therefore, RAI 14.2-54 is resolved.

14.2.3.11.13 Turbine Trip and Generator Load Rejection Test

DCD Tier 2, Section 8.3.1.1 states that the unit auxiliary transformers provide normal preferred offsite power or generator island mode power to each of the plant's two power generation and plant investment protection load groups.

The applicant did not include a demonstration of the generator island mode operation. In RAI 14.2-100, the staff asked the applicant to include in DCD Tier 2, Section 14.2.8.2 the main generator island mode operation test or provide justification for not including this test in the startup test program.

In response, the applicant stated the following:

> In DCD Tier 2, Subsection 14.2.8.2.27, the method of testing the turbine trip and generator load rejection will be clarified by adding a statement that delineates which breaker (generator output breaker or switchyard breaker) is open in which test. In addition, the test success criteria section that the plant shall not SCRAM following a turbine trip or generator load rejection testing will be removed.

The applicant revised DCD Tier 2, Section 14.2.8.2.27 to state:

> From an initial power level of 100%, the main generator is tripped (generator output breaker is open for the turbine trip test and the switchyard breaker is opened for the generator load rejection test) in order to verify the proper reactor and integrated plant response.

The applicant also revised DCD Tier 2, Section 14.2.8.2.27 to state:

> For high power turbine or generator trips, reactor dynamic response shall be consistent with predictions based on expected system characteristics and shall be conservative relative to analysis results based on design assumptions.

The applicant added this information to DCD Tier 2, Revision 6, Section 14.2.8.2.27,. Therefore, RAI 14.2-100 is resolved.

14.2.3.11.14 Isolation Condenser System Performance Test

In RAI 14.2-3, the staff requested additional information regarding the IC performance test description in DCD Tier 2, Section 14.2.8.1.63. The staff had concerns about the structural integrity and design of the ICS. The specific concern was leakage in the ICS during testing at the PANTHER-IC facility, which the staff considered an issue of ICS structural integrity that needed to be resolved for the ESBWR design certification. The applicant stated that the O-ring design has been changed to a Helicoflex self-energizing O-ring design that will be more resilient to distortion. The applicant further stated that closing the condensate return valve will be controlled to limit the gradients associated with the shutdown and cooldown of the ICS heat exchanger.

Further, in DCD Tier 2, Table 14.2-1 indicates that the ICS performance test will be conducted at the medium-power (MP) level but not at the high-power (HP) level. Since one of the objectives of the power ascension test should be to demonstrate ICS structural integrity, the staff believes that an ICS performance test at HP is more appropriate because the operating conditions at HP are expected to be more challenging to the structural integrity of the ICS. The staff, therefore, requested that the IC system performance test be conducted at the HP rather than MP level.

In response, the applicant stated that the ascension test matrix (DCD Tier 2, Table 14.2-1) proposes that the ICS be tested at medium (up to about 75 percent rated) power. The applicant

further stated that pressure and temperature, not the reactor power level, affect the structural integrity of the ICS. When the reactor startup begins, the reactor is brought to the rated pressure and temperature at approximately five percent power, as stated in DCD Tier 2, Section 14.2.1.3. As the power level increases, the same rated pressure and temperature will be maintained; therefore, conducting the ICS test at MP will be sufficient. The applicant also stated that testing at HP would not be more challenging from the viewpoint of structural integrity of the ICS. In addition, testing the ICS at MP, instead of HP, would avoid a potential HP transient resulting from IC system cold water injection into the RPV that could challenge thermal limits on the reactor core; therefore, testing the ICS at MP would be more appropriate. On the basis of this information, the staff finds that the applicant's response is acceptable. Therefore, RAI 14.2-3 is resolved.

14.2.3.11.15 Initial Test Program Test Abstract Conclusions

On the basis of its review, the staff determined that the test abstracts provided by the applicant in DCD Tier 2, Revision 9, Section 14.2.8, are consistent with the criteria in RG 1.68 and SRP Section 14.2. Further, since the Licensee will be responsible for the development of detailed test specifications and test procedures, the staff finds that it is acceptable to defer development of the test specifications and test procedures until the COL phase. COL Information Item 14.2-2-A encompasses this issue.

14.2.4 Site Specific Preoperational and Start Up Tests

In DCD Tier 2, Section 14.2.9, the applicant stated the COL Applicant will define any required site-specific pre-operational and startup testing. This is identified in DCD Revision 6 as COL Information Item 14.2-5-A.

In RAI 14.2-15, the staff requested additional information regarding the SSCs and design features listed in Section 14.2.9 of ESBWR DCD Tier 2, which the applicant identified as candidates for exemptions from operating license conditions requiring prior NRC approval for major test changes. The staff asked the applicant to provide the basis for an exemption for each of the listed SSCs.

In response, the applicant deleted the list of specific systems in DCD Tier 2, Section 14.2.9, and revised the section to denote that the COL Applicant will list any tests to be performed as part of the power ascension test phase that are proposed to be exempt from operating license conditions requiring prior NRC approval for major test changes and the basis for the exemption. The applicant included a list of systems that are related to site-specific aspects of the plant that need testing to demonstrate their capability to meet performance requirements and acceptance criteria. Below are the systems that may require such testing:

- Electrical switchyard and equipment
- Site security plan
- Personnel monitors and radiation survey instruments
- Automatic dispatcher control system (if applicable)

The applicant also stated that if tests are identified as requiring an exemption from operating license conditions after the COL application has been submitted, the Licensee will identify the tests requiring an exemption and the basis for the exemption.

The staff reviewed the applicant's response to this RAI. Regulatory Position C.1 of RG 1.68 specifies criteria for determining which SSCs and design features must be tested. Certain tests during the initial startup test phase may be subject to license conditions requiring prior NRC approval for major test changes. For such instances, the applicant deferred this responsibility to the COL Applicant. The staff finds that this was consistent with RG 1.68 and therefore acceptable. The staff also reviewed DCD Tier 2, Revision 3, Section 14.2.9, and finds that the revised text appropriately addresses the staff concern and is acceptable. Therefore, RAI 14.2-15 is resolved.

14.2.5 Summary of COL Information Items

The staff finds that all ITP COL information items are in accordance with RG 1.68 and SRP Section 14.2; therefore, they are acceptable.

14.2.1-A Description - Initial Test Program Administration
A description of the initial test program administration is developed and made available to the NRC by the COL Applicant (Section 14.2.2.1).

14.2.2-A Startup Administrative Manual
The COL Applicant will provide milestones for completing the SAM and will make it available for NRC inspection (Section 14.2.2.1).

14.2-3-A Test Procedures
The COL Applicant will provide milestones for making available to the NRC approved test procedures that satisfy the requirements for the ITP (Section 14.2.2.2).

14.2-4-A Test Program Schedule and Sequence
The COL Applicant will provide a milestone for completing the detailed testing schedule and for making it available to the NRC (Section 14.2.7).

14.2-5-A Site-Specific Tests
The COL Applicant will define any required site-specific pre-operational and startup testing (Section 14.2.9).

14.2-6-A Site-Specific Test Procedures
The COL Applicant will provide milestones for making available to the NRC approved test procedures that satisfy the requirements for the ITP (Section 14.2.9).

14.2.6 Conclusions

The staff reviewed the RAIs noted below and finds that these RAIs: (1) were outside the scope of RG 1.68 and SRP Section 14.2, (2) resulted in a change to another DCD section and were addressed in those sections of this report, (3) did not result in any change or were very minor editorial comments on DCD Tier 2, Section 14.2, or (4) were already discussed in another RAI in Section 14.2 of this report. Therefore, Section 14.2 of this report does not discuss RAIs 14.2-1, 14.2-2, 14.2-14, 14.2-22, 14.2-23, 14.2-25, 14.2-27, 14.2-28, 14.2-31, 14.2-33, 14.2-34, 14.2-45, 14.2-49, 14.2-52, 14.2-56, 14.2-58, 14.2-60, 14.2-61, 14.2-62, 14.2-67, 14.2-69, 14.2-71, 14.2-72, 14.2-83,14.2-84, 14.2-87, and 14.2-88. The staff considers the RAIs listed above to be resolved.

The staff completed its review of the ESBWR ITP in accordance with the requirements of 10 CFR 52.47; 10 CFR 50.34; 10 CFR 52.79; and Criterion XI, of Appendix B to 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities". The staff concludes that the applicant has provided sufficient information in the ITP to test all SSCs important to safety and adequately addressed the methods and guidance in SRP Section 14.2 and in RG 1.68. The staff concludes that the applicant has resolved all open items related to the ITP; therefore, the applicant's ITP is acceptable.

14.3 <u>Inspections, Tests, Analyses, and Acceptance Criteria</u>

This section provides the selection criteria and processes used to develop the ESBWR ITAAC. This section addresses ESBWR DCD Tier 2, Revision 9, Section 14.3 and ESBWR DCD Tier 1, Revision 9.

14.3.1 Selection Criteria and Methodology for Tier 1

Summary of Application

DCD Tier 2, Revision 9, Section 14.3, discusses the criteria and methodology for selecting the SSCs to be included in the ITAAC. This section includes definitions and general provisions, design descriptions, ITAAC, significant site parameters, and significant interface requirements. It specifically addresses the ITAAC for the SSCs within the scope of the ESBWR DCD. In addition, this section addresses the proposed ESBWR design acceptance criteria (DAC) for specific areas for which a design process has been prescribed to produce predictable and acceptable designs. DCD Tier 2, Revision 9, Section 14.3 also includes a proposed approach for completing the design-related ITAAC (i.e., DAC).

DCD Tier 1, Revision 9, provides the results of the implementation of DCD Tier 2, Section 14.3, selection criteria and methodology for determining the SSCs described throughout DCD Tier 2. These need to be included in the ESBWR DCD Tier 1 verification program to ensure that an ESBWR facility has been constructed and will operate in accordance with the design certification.

Regulatory Basis

10 CFR 52.47(b)(1) requires that the design certification application contain the proposed ITAAC that are necessary and sufficient to provide reasonable assurance that if the inspections, tests, and analyses are performed and the acceptance criteria are met, a facility that incorporates the design certification has been constructed and will be operated in conformity with the design certification, the provisions of the Atomic Energy Act, and the rules and regulations of the NRC.

Office of the Secretary of the Commission (SECY)-90-377, "Requirements for Design Certification under 10 CFR Part 52," dated November 8, 1990 and its associated staff requirements memorandum (SRM) dated February 15, 1991 provide Commission guidance on the level of detail that a design certification application should reflect. In addition, SECY-90-241, "Level of Detail Required for Design Certification under Part 52," and its associated SRM; SECY-91-178, "Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC) for Design Certifications and Combined Licenses;" SECY-91-210, "Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC) Requirements for Design Review and Issuance of a Final Design Approval (FDA)," dated July 16, 1991, and SECY-92-214, "Development of Inspections, Tests,

Analyses, and Acceptance Criteria (ITAAC) for Design Certifications," dated June 11, 1992, provide Commission guidance on the development and use of ITAAC included in the licensing process described in 10 CFR Part 52, "Early Site Permits; Standard Design Certifications; and Combined Licenses for Nuclear Power Plants." In SECY-92-053, "Use of Design Acceptance Criteria During 10 CFR Part 52 Design Certification Process," dated February 19, 1992, the staff discussed a method for using the DAC, together with detailed design information, during the 10 CFR Part 52 process for reviewing and approving designs. The NRC intended the DAC to be used for applications that do not provide design and engineering information at a level of detail customarily considered by the staff in reaching a final safety decision, and primarily for areas of design that are subject to rapidly changing technologies. Finally, SECY-05-0197, "Review of Operational Programs in a Combined License Application and Generic Emergency Planning Inspections, Tests, Analyses, and Acceptance Criteria," dated October 28, 2005, discussed the use of programmatic emergency planning ITAAC.

Section 14.3 of NUREG–0800, except as noted in this report, establishes the regulatory basis for acceptance of the ITAAC associated with a design certification application and, specifically in this case, the ESBWR DCD. RG 1.206 gives COL Applicants guidance for the development of site-specific ITAAC and the use of ITAAC contained in a certified design.

In DCD Tier 2, Revision 9, Section 14.3, the applicant provided the selection criteria and processes used to develop the DCD Tier 1 ITAAC. The DCD Tier 1, Revision 9, information provides the principal design bases and design characteristics that are certified by the 10 CFR Part 52 rulemaking process and that would be included in the ESBWR design certification rule.

Technical Evaluation

The staff reviewed DCD Tier 2, Revision 9, Section 14.3 for conformance with the guidance contained in SRP Section 14.3.

In DCD Tier 2, Sections 14.3.1 and 14.3.2, the applicant discussed the NRC regulatory guidance it used to develop the selection methodology for DCD Tier 1 information. These sections describe the content and format of the DCD Tier 1 information and include a table of contents; lists of tables, illustrations, abbreviations, and acronyms; an introduction section; design descriptions and ITAAC; non-system-based material; interface material; and site parameters. The sections also discuss in detail the selection criteria and DCD Tier 2 review methodology for including the SSCs in DCD Tier 1, Section 2.0. These sections discuss the format and content of the ITAAC; the criteria for developing and selecting the design commitments; the inspections, tests, and analyses that are prescribed to verify that the design commitment was met; and the acceptance criteria for determining the successful completion of the verification method. The applicant also discussed the interface between the verification performed under DCD Tier 1 and the initial plant test program. The staff reviewed the information provided by the applicant in DCD Tier 2, Sections 14.3.1 and 14.3.2 in accordance with SRP Section 14.3. The staff finds it consistent with the staff review guidance and concludes that it is acceptable. As a result, the staff concludes that the applicant's implementation of the selection criteria and methodology will result in the design descriptions and ITAAC necessary to demonstrate that the facility has been constructed and will operate in accordance with the certified design.

In DCD Tier 2, Section 14.3.3, the applicant discussed non-system-based material included in DCD Tier 1, Section 3.0, whose design descriptions and associated ITAAC for design and

construction activities apply to more than one system. This section includes the basis for using the DAC and discusses the limited use of DAC for piping systems and components, software development for I&C, and human factors engineering (HFE). In addition, this section provides summary discussions of DCD Tier 1 information associated with radiation protection, the ITP, the design reliability assurance program (D-RAP), post-accident monitoring instrumentation, and environmental qualification (EQ) of mechanical and electrical equipment. The staff reviewed the information provided by the applicant in DCD Tier 2, Section 14.3.3 in accordance with the guidance contained in SRP Section 14.3, as well as the Commission policy on the use of DAC contained in SECY-90-241, SECY-91-178, SECY-91-210, SECY-92-053, and SECY-92-214 and their associated SRMs. The staff finds the applicant's use of the DAC to be consistent with NRC guidance and established NRC policy. As a result, the staff concludes that the information provided by the applicant in DCD Tier 2, Section 14.3.3 is acceptable.

In DCD Tier 2, Section 14.3.4, the applicant discussed the interface material included in DCD Tier 1, Section 4.0. This section explains the regulatory basis for the interface requirements, the scope of these requirements with respect to the use of site-specific designs to support the ESBWR system designs, and the selection criteria and methodology for the interface requirements. This section specifies that applicants for a license that references the ESBWR standard design are responsible for ensuring that their applications include site-specific designs that comply with these interface requirements, along with any necessary verification requirements included in site-specific ITAAC. In DCD Tier 2, Section 14.3.5, the applicant discussed the site parameters included in DCD Tier 1, Section 5.0. This section describes the site parameters as the basis for the ESBWR standard design and represents them as a bounding envelope of site conditions for any license application referencing the ESBWR design. The discussion provides the regulatory basis for the inclusion of site parameters in DCD Tier 1 and requires any license applicant that references the ESBWR standard to demonstrate that the characteristics for the selected site are within the ESBWR certification envelope. The staff reviewed the information provided by the applicant in DCD Tier 2, Sections 14.3.4 and 14.3.5, in accordance with SRP Section 14.3. The staff finds it consistent with the staff's review guidance and concludes that it is acceptable. As a result, the staff finds that the applicant's criteria for establishing interface requirements and site parameters are acceptable.

In DCD Tier 2, Section 14.3.6, the applicant summarized the application of its selection criteria and methodology for generating DCD Tier 1 information and presenting DCD Tier 1 results. In DCD Tier 2, Section 14.3.7, the applicant discussed the regulatory basis and evaluation process for changing the design descriptions and the ITAAC provided in DCD Tier 1 for the ESBWR design and for determining site-specific ITAAC. The applicant provided specific criteria for determining the appropriate level of detail and content for general DCD Tier 1 content, design descriptions, and ITAAC. In DCD Tier 2, Section 14.3.8, the applicant described the regulatory basis and the overall ITAAC content for COL applications. This section reiterates the guidance provided in RG 1.206 and specifies that the overall ITAAC content for a COL application must include site-specific ITAAC as well as ITAAC for design certification, emergency planning, and physical security hardware. In DCD Tier 2, Section 14.3.9, the applicant provided a more detailed discussion of the site-specific ITAAC, along with references to the appropriate guidance contained in RG 1.206. DCD Tier 2, Section 14.3.10 contains a consolidated list of the information items that any COL application referencing the ESBWR standard design must contain. This section includes COL information items for emergency planning ITAAC and site-specific ITAAC for systems not included in the scope of an ESBWR design certification. The applicant provided references in DCD Tier 2, Section 14.3.11. The staff reviewed the information provided by the applicant in DCD Tier 2, Sections 14.3.6 through 14.3.11, in

accordance with SRP Section 14.3. The staff finds the information consistent with the staff's review guidance contained in SRP Section 14.3. The staff concludes that it is acceptable.

The applicant provided a tabulation summarizing the types of systems described in DCD Tier 2 and their graded treatment for inclusion in the design descriptions and ITAAC in DCD Tier 1. A separate tabulation summarizes the test, inspection, or analysis approach used to verify ITAAC design commitments and the application of this approach for complying with the ITAAC acceptance criteria. The staff reviewed these tables in accordance with SRP Section 14.3. The staff finds the information consistent with the staff's review guidance and concludes that it is acceptable.

The applicant provided selection criteria and a process for including SSCs in DCD Tier 1 at an appropriate level of detail, in accordance with a graded approach commensurate with the safety significance of the SSCs for the ESBWR design. The applicant selected this top-level information from the design descriptions provided in DCD Tier 2 of the ESBWR DCD, identified the principal performance characteristics and safety functions of the SSCs to be verified appropriately by the ITAAC, and included design-specific and unique features of the ESBWR, as appropriate. The ITAAC included those SSCs that were determined to be risk-significant in the probabilistic risk assessment (PRA), including SSCs that were selected for special treatment in accordance with the regulatory treatment of nonsafety systems (RTNSS). In addition, the selection criteria and process included important insights and assumptions from the PRA; integrated plant safety analyses such as those for fires, floods, and severe accidents; and shutdown risk. Based on its review of the applicant's selection criteria and process for identifying DCD Tier 1 information contained in DCD Tier 2, Section 14.3, the staff finds that the applicant's process is consistent with the guidance contained in SRP Section 14.3 and is therefore acceptable. The applicant did not, however, provide cross-references in DCD Tier 2, Section 14.3 showing where key parameters from the analyses discussed above are addressed in the DCD Tier 1 information. These analyses include the safety analyses of design-basis accidents, severe accidents, flooding, overpressure protection, containment, core cooling, fire protection, transients, shutdown risk, anticipated transient without scram (ATWS), Three Mile Island action plan items, PRAs, and RTNSS. The staff asked GEH in RAI 14.3-405 to provide these cross-references. RAI 14.3-405 was being tracked as an open item in the SER with open items.

In response, GEH added Tables 14.3-1A, 14.3-1B, and 14.3-1C to DCD Tier 2, Section 14.3. Table 14.3-1A includes a list of DCD Tier 1 contents, with an indication of which systems have ITAAC. In Table 14.3-1B, GEH included those design features that are related to a specific transient and accident analysis such as ATWS, overpressure protection, containment, and emergency core cooling. Table 14.3-1C addresses the design features key to the PRA and severe accident insights, including core cooling, flooding, fire, management of molten debris, and RTNSS. In its response, GEH also indicated that during the development of Table 14.3-1C, it determined that additional ITAAC were needed and existing ITAAC needed to be changed to address the key design features. The staff issued RAI 14.3-405 S01, identifying minor changes that GEH needed to make to the tables for clarity and consistency. In response, GEH revised the tables to address the staff's comments. Based on the above discussion, therefore, RAI 14.3-405 and the associated open items are resolved.

During its review of the criteria provided in DCD Tier 2, Section 14.3, the staff identified concerns with criteria for identifying and depicting the basic configuration of the portions of systems that are safety significant, including any components located in those portions of the systems, ensuring consistency between the information provided in the introductory section of

Tier 1 and the criteria provided in DCD Tier 2, Section 14.3, and ensuring that references to NRC guidance included final guidance instead of draft guidance. The applicant adequately addressed these issues in its responses to RAIs 14.3-338, 14.3-339, and 14.3-340. Therefore, these RAIs are resolved.

In DCD Tier 2, Appendix 14.3A, the applicant proposed a DAC closure process for the ITAAC. For a design certification application, NRC regulations neither require nor prohibit such a closure process. However, 10 CFR 52.99 describes a general ITAAC closure process that requires a Licensee to submit an initial schedule for completing the ITAAC and to provide periodic updates throughout construction. The Licensee must submit the initial schedule within one year of the COL issuance or at the start of construction, whichever is later. The applicant discussed the options for closing the DAC following certification of the design through an amendment of the design certification rule, through the COL application review process, and through the DAC after the COL issuance. The applicant chose the latter option after COL issuance and its proposed approach to achieve closure of the DAC will apply to not only the first standard ESBWR plant but to all subsequent ESBWR plants as well.

This standard approach is voluntary on the part of each Licensee referencing the standard ESBWR design. The process envisions an NRC review, inspection, or audit of the DAC completion that applies the “one issue, one review, one position” concept as discussed in RG 1.206, Section C.III.5, to the DAC resolution for the first and subsequent ESBWR plants. A COL Applicant can apply this standard approach to each of the ESBWR design areas that include DAC (i.e., piping design, digital I&C design, and HFE design). The staff finds that this standard approach is consistent with the NRC policy of a design-centered-review approach and is therefore acceptable. In addition, the applicant included a COL information item (COL Information Item 14.3A-1-1), whereby each COL Applicant must provide a DAC closure schedule in the COL application and identify whether the standard approach will be used. Inclusion of this COL information item resolves RAI 14.3-210 and will provide the staff with the information necessary to facilitate its review, inspection, or audit of the DAC resolution.

In addition to its review in accordance with SRP Section 14.3, the staff reviewed ESBWR DCD Tier 1 in accordance with the following SRP Section 14.3 subsections:

- 14.3.2, “Structural and Systems Engineering”
- 14.3.3, “Piping Systems and Components”
- 14.3.4, “Reactor Systems”
- 14.3.5, “Instrumentation and Controls”
- 14.3.6, “Electrical Systems”
- 14.3.7,” Plant Systems”
- 14.3.8, “Radiation Protection”
- 14.3.9, “Human Factors Engineering”
- 14.3.10, “Emergency Planning”
- 14.3.11, “Containment Systems”
- 14.3.12, “Physical Security Hardware”

Organization of Safety Evaluation Report and Reference to Appendix A

The applicant’s DCD Tier 1 document, which was organized based on SSCs, does not provide for direct correlation to the SRP staff review guidance shown above. However, the applicant’s organization of DCD Tier 1 information is acceptable, because it is consistent with previous

design certification applications to the NRC and it facilitates a more efficient staff review of DCD Tier 1 information in conjunction with the DCD Tier 2 information from which it is derived. The information in DCD Tier 1 is cross-cutting in nature and required several staff technical review branches to provide a comprehensive review. To facilitate this comprehensive review of the DCD Tier 1 information, the staff developed a review matrix and included it as Appendix A to this section. Appendix A identifies the SRP sections used to evaluate the SSCs covered in DCD Tier 1, Revision 9, Sections 2 and 3 and the associated sections of this report in which the evaluation is documented.

In DCD Tier 1, the applicant provided the results of its implementation of the selection criteria and methodology used to develop DCD Tier 1 information and ITAAC, as described in DCD Tier 2, Section 14.3. The applicant provided the following information in DCD Tier 1:

- A table of contents and a list of tables, figures, abbreviations, and acronyms

- An introduction that provides definitions of terms used in the DCD Tier 1 information and discusses the treatment of individual items, the implementation of the ITAAC, matters related to operation, the interpretation of figures and a figure legend, and the rated reactor core thermal power

- A section containing the design descriptions including associated tables and figures, and the ITAAC necessary to demonstrate that the facility referencing the ESBWR standard design has been constructed and will operate in accordance with the design certification

- A section containing non-system-based material that discusses the use of the DAC for piping systems and components; digital I&C software development; HFE (including the necessary design completion ITAAC and installation verification ITAAC for these areas); and areas of the ESBWR standard design that are applicable to more than one system including radiation protection, ITP, D-RAP, post-accident monitoring instrumentation, and the EQ of mechanical and electrical equipment

- A section containing the provisions and/or specifications for interface material that license applicants referencing the ESBWR standard design must provide in their applications

- A section containing the site parameters upon which the ESBWR standard design is based and that applicants must demonstrate are parameters enveloping the site characteristics for the locations where they have chosen to build and operate the ESBWR design

The staff’s review of the DCD Tier 1 information resulted in a large number of RAIs that included requests for clarification, completeness, and consistency as well as format issues. These requests are summarized as follows:

- Provide complete and correct lists of acronyms.

- Provide clarifications of definitions included in the DCD Tier 1 introduction by expanding and/or adding definitions.

- Ensure accuracy and consistency of the scope of ITAAC verification activities that reference tables and figures in the design description.

- Ensure identification of all DAC within the appropriate ITAAC tables.

- Clarify changes to DCD Tier 1 from previous revisions of the DCD.

- Ensure consistency between the design descriptions in DCD Tier 1 and the design information in DCD Tier 2.

- Ensure consistency of terminology and language between the design descriptions and the “design commitment” entries in the ITAAC table.

- Ensure clear and consistent use of numbering schemes for ITAAC entries that will allow for greater clarity when documenting successful ITAAC completion.

- Ensure consistency of terminology and language across the entries in the “design commitment”; “inspections, tests, analyses”; and “acceptance criteria” columns of ITAAC.

- Clarify ambiguities in the design descriptions, design commitments, and acceptance criteria to facilitate objectivity and to avoid subjective interpretations about whether compliance with the acceptance criteria has been achieved.

- Ensure that cross-references between ITAAC tables are consistent and accurate (e.g., for functions such as minimum inventory of alarms, displays, controls, and status indications in the main control room [MCR]; equipment qualification; digital instrumentation; and control software development).

- Clarify the use of simulated signals versus actual signals for verifying the proper functioning of I&C items such as alarms and detectors (e.g., radiation source calibration versus simulated signal calibration).

- Clarify whether testing of components is in-situ or via a test facility (i.e., shop testing or type-testing).

- Provide consistency in the use of terminology related to regulatory requirements, industry standards, and guidance (e.g., “complies with” versus “conforms to”; “retains its pressure boundary integrity at its design pressure” versus “retains its pressure boundary integrity at internal pressures that will be experienced during service”).

- Clarify measurements of timing and/or other performance values where measurement tolerances, minimums, maximums, or ranges of values are necessary to clarify acceptance criteria.

For the sake of brevity, the individual RAI numbers are not provided here. However, the RAIs that were tracked as open items in the SER with open items are discussed in the following sections of this report. The following discussions consider other areas of staff inquiry and evaluation. In Sections 14.3.2 through 14.3.12 of this report, the staff discusses its review of DCD Tier 1 in accordance with SRP Sections 14.3.2 through 14.3.12 and focuses its discussions primarily on the RAIs that dealt with specific SSC performance requirements.

ITAAC for ASME Code Systems

The staff identified several issues during its review of ITAAC for systems designed to the meet the requirements in Section III of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (referred to as the ASME Code). The staff requested greater clarity,

consistency, and organizational separation of the design completion and installation verification activities in the ITAAC tables. In addition, the staff requested that the acceptance criteria clearly identify requirements applicable to design completion and installation. In particular, the staff requested that the applicant included specific reference to the requirements of the ASME Code, such as design reports, ASME Code reconciliations, and data reports. These staff requests applied to all ASME Code systems included in the ITAAC. In addition, the staff requested that the associated definitions for "reports" and ASME Code Reports be clearly articulated in the definition section for Tier 1 information. Several RAIs addressed format, content, and consistency of the ITAAC for ASME Code systems and structures. The following sections of this report discusses these RAIS, which include RAIs 14.3-131, 14.3-213, 14.3-351, 14.3-352, 14.3-353, 14.3-354, 14.3-368, and 14.3-384. The staff discussed with the applicant its efforts to refine these ITAAC. The staff was tracking these RAIs as open items in the SER with open items.

Technical Evaluation for "No Entry" Systems

The applicant included a number of systems in DCD Tier 1 that have no safety-related, risk-significant, or regulatory compliance function. The applicant identified these systems in DCD Tier 1 by title only and indicated that no ITAAC are necessary. The staff identified these "no entry" systems in Appendix A to this report and reviewed them in accordance with the guidance contained in SRP Section 14.3. The staff finds the inclusion of these "no entry" systems in DCD Tier 1, without any associated ITAAC, to be in conformance with SRP Section 14.3 and therefore acceptable.

Technical Evaluation of Initial Test Program

In DCD Tier 1, Section 3.5, the applicant provided an overview of the ITP and a commitment that states that COL Applicants referencing the certified design will implement an ITP that meets the objectives presented in DCD Tier 2, Section 14.2. As stated by the applicant, ITAAC intended to verify ITP implementation are neither necessary nor required.

The staff reviewed the DCD Tier 1 information in accordance with the guidance provided in SRP Section 14.3. The staff noted that in DCD Tier 1, the applicant made a high-level commitment to an ITP in accordance with SRP Section 14.3.10. In addition, the applicant provided a general description of its pre-operational and power ascension test programs and the administrative controls that will govern the conduct of the ITP. The applicant also provided adequate justification in DCD Tier 2, Section 14.3 for not including an ITAAC for the ITP in DCD Tier 1.

Current regulations do not require ITAAC for the ITP for several reasons:

- The system-specific ITAAC delineate the specific testing necessary to verify design features and performance aspects of the design. DCD Tier 1 certified design material (CDM), when applied to the ITP, should consist of a high-level commitment to an ITP and a description of the program and major program documents that constitute an acceptable ITP (i.e., a site-specific SAM, test specifications, and test procedures).

- The ITP covers a broader spectrum of time than the ITAAC cover. Although the applicant must complete pre-operational testing before fuel load, it will conduct the ITP startup and power ascension testing after fuel load. As the ITP involves testing after fuel load, it is not appropriate to define associated ITAAC entries, as 10 CFR Part 52 specifies that the ITAAC will be completed before fuel load.

On the basis of the staff's review of the material in ESBWR DCD Tier 1 and a review of the selected methodology and criteria for the development of DCD Tier 1 contained in ESBWR DCD Tier 2, Section 14.3, the staff concludes that the ITP is appropriately described in DCD Tier 1 and is therefore acceptable.

Technical Evaluation Design Reliability Assurance Program

The staff's review of the D-RAP information in DCD Tier 1 was being tracked as an open item in the SER with open items.

In DCD Tier 1, Revision 4, Section 3.6 the applicant provided the design description and associated ITAAC for the D-RAP. Section 3.6 specified a design commitment that the D-RAP will provide reasonable assurance that the design of risk-significant SSCs is consistent with their risk analysis assumptions. The associated D-RAP ITAAC acceptance criteria ensure that the reliability of each as-built risk-significant SSC is consistent with the reliability assumed in the ESBWR Design PRA.

The staff reviewed the information provided in DCD Tier 1, Section 3.6, in accordance with the guidance provided in SRP Section 14.3. The staff noted that the D-RAP ITAAC should not solely be based on numerical values, because some numerical estimates (e.g., estimated reliability, assumed reliability) may not be available, and additional aspects of the D-RAP are needed in the D-RAP ITAAC in order to address other key assumptions and risk insights. Therefore, the applicant's D-RAP ITAAC may not be practical or effective in providing reasonable assurance that the plant is designed and constructed in a manner that is consistent with the key assumptions and risk insights for the SSCs within the scope of D-RAP. It is important to have a process that would control reliability/availability of risk-significant SSCs. The staff requested in RAI 14.3-437 that the applicant consider revising the D-RAP ITAAC in DCD Tier 1, Section 3.6 taking into consideration the staff's comments provided above. In response, the applicant revised DCD Tier 1, Section 3.6 and associated D-RAP ITAAC, where the D-RAP ensures that the design of the SSCs within the scope of the reliability assurance program (RAP) is consistent with the risk insights and key assumptions. In the associated D-RAP ITAAC acceptance criteria, all RAP SSCs are designed in accordance with the applicable reliability assurance activities for the D-RAP. The staff concludes that this proposed revision is consistent with the recommendations in Item E of SECY-95-132, "Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety Systems (RTNSS) in Passive Plant Designs, SECY-94-084," and is therefore acceptable. The staff confirmed that this revision was incorporated into DCD Tier 1, Revision 7, Section 3.6. Therefore, RAI 14.3-437 is resolved.

Technical Evaluation of Interface Material

In DCD Tier 1, Section 4.0, regarding interface material, the applicant discussed the requirement in 10 CFR 52.79(c) (now 52.79[d]) for a COL Applicant that references the ESBWR design to provide design features or characteristics that comply with the interface requirements for the ESBWR plant design and to provide ITAAC for the site-specific portion of the facility design. In accordance with 10 CFR 52.47(a)(26), the applicant provided site interface requirements for the PSWS, since this system is necessary to support the post-72-hour cooling requirements of the ESBWR plant. The applicant specified the heat removal requirements for the PSWS design as 2.02×10^7 megajoules (MJ) (1.92×10^{10} British thermal units [BTU]) over a period of 7 days without active makeup. The staff reviewed the interface material proposed by the applicant for the PSWS design, in accordance with the guidance provided in SRP Section 14.3 and the applicable SRP Section 14.3 subsections and concludes that it is acceptable.

During the review, the staff identified the lack of interface requirements for the offsite power system as a concern in RAI 14.3-394. Section 14.3.6 of this report discusses this issue.

Technical Evaluation of Site Parameters

Every COL Applicant referencing the ESBWR standard plant design must demonstrate that the site characteristics specific to its COL application fall within the site parameters contained in DCD Tier 1, Section 5.0, which are intended to apply to a wide range of sites for the construction and operation of this plant. The tabulation in DCD Tier 1, Section 5.0, is a consolidation of the site parameters contained in ESBWR DCD Tier 2, Chapter 2. The staff reviewed the tabulation of the ESBWR site parameters provided by the applicant in DCD Tier 1, Section 5.0 for conformance with SRP Section 14.3 and the applicable SRP Section 14.3 subsections. The staff concludes that they are acceptable and consistent with those parameters contained in ESBWR DCD Tier 2, Chapter 2 and are evaluated in Section 2.0 of this report.

14.3.2 Structural and Systems Engineering

The applicant provided design-basis information, including associated tables and figures, in accordance with the selection criteria and methodology for developing DCD Tier 1 information described in DCD Tier 2, Section 14.3 to support the ITAAC for the ESBWR SSCs. The applicant organized its DCD Tier 1 information in the systems, structures, and topical areas format shown in the DCD Tier 1 table of contents. The staff reviewed the DCD Tier 1 information provided by the applicant, in accordance with the review matrix provided in Appendix 14.3A and in accordance with SRP Section 14.3.2.

To facilitate completion of its review, the staff issued a number of RAIs (discussed below) that describe some of its concerns.

RAI 14.3-97: In RAI 14.3-97, the staff requested that the applicant include Types B and C leak rate tests as part of the ITAAC associated with DCD Tier 1, Table 2.15.1. In response, GEH stated, in part, that Type B and C local leak rate testing, as required by Appendix J to 10 CFR Part 50, would be added to DCD Tier 1, Revision 4, Table 2.15.1-1 as ITAAC Items 12 and 13, respectively. In reviewing Tables 2.15.1-1 and 2.15.1-2 of DCD Tier 1, Revision 4, the staff could not confirm that GEH had added ITAAC Items 12 and 13, as indicated in the response. However, the staff noted that ITAAC Item 7 of Table 2.15.1-2 appeared to address the same issue discussed in the GEH response to the RAI. In RAI 14.3-97 S01, the staff requested the applicant to clarify and resolve the above noted inconsistency and omission of ITAAC Items 12 and 13 in Table 2.15.1-2 and confirm, as appropriate, its intent with respect to ITAAC Item 7 in Table 2.15.1-2.

GEH responded that the original response to this RAI indicated that two ITAAC items would be added to Table 2.15.1-1: one for Type B testing and one for Type C testing. In finalizing Revision 4 of the DCD, two changes were made relative to its response to RAI 14.3-97. First, Table 2.15.1-1 was renumbered to become Table 2.15.1-2. Second, instead of adding two new ITAAC items (Item 12 for Type B testing and Item 13 for Type C testing), GEH revised ITAAC Item 7 (Item 4 in DCD Revision 3) to include all Appendix J containment leakage testing. This approach was used because it was recognized that the "design commitment" associated with containment leakage testing is that the containment provides a barrier against the release of fission products, and the performance of all Type A, B, and C tests are necessary to verify the design commitment. Hence, the revision of the DCD was intended to keep the inspections,

tests, and analyses the same, as indicated in the initial response, but to more accurately state the design commitment associated with Appendix J leak testing.

The staff reviewed ITAAC Item 7 in DCD Tier 2, Revision 4, Table 2.15.1-2 and determined that it included the appropriate Type B and C local leak rate testing information. On the basis of the above information, RAI 14.3-97 is resolved.

<u>RAI 14.3-178</u>: In RAI 14.3-178, the staff requested that GEH clarify its intent for verification of diaphragm floor and vent wall structures. Specifically, DCD Tier 1, Revision 4, Section 2.15.3 (page 2.15-24) states, in part, that "(5) The diaphragm floor and vent wall structures that separate the DW (drywell) and WW (wetwell) retain their integrity when subject to pressure at or above design pressure." The staff was not clear as to the exact meaning and intent of the phrase: "...when subject to pressure at or above design pressure." The staff asked GEH to clearly define the meaning of the term "above design pressure" and justify its use. GEH responded that the design commitment for the diaphragm floor and vent wall structures is to retain their integrity when subjected to the design differential pressures as defined in DCD Tier 2, Table 6.2-1. The associated inspections, tests, and analyses describe testing as part of the structural integrity test, which is specified for the containment system boundary in DCD Tier 1, Section 2.15.1, in accordance with Article CC-6000 of ASME Code, Section III, Division 2. The code specifies a test pressure of at least 1.15 times the design pressure for the containment structure and a differential test pressure for the internal structures to be at least 1.0 times the design differential pressure. GEH committed to revise DCD Tier 1, Section 2.15.3, Item (5) and Table 2.15.3-2 to clarify these requirements.

The staff reviewed the response to RAI 14.3-178, including GEH's markup of DCD Tier 1, Section 2.15.3, Item 5, and Table 2.15.3-2, and determined that the applicant provided adequate clarification. The staff checked DCD Tier 1, Revision 9, Section 2.15.3 item (5) and Table 2.15.3-2 to confirm that the modified text properly reflected the GEH response. Based on the above information, RAI 14.3-178 is resolved.

<u>RAI 14.3-179</u>: In RAI 14.3-179, the staff requested GEH to address an ambiguous statement related to the decay of fission products in the reactor building (RB). Specifically, DCD Tier 1, Revision 4, Section 2.16.5, RB, states, in part, that "(4) The RB offers some holdup and decay of fission products that may leak from the containment after an accident. Assuming a loss-of-coolant accident (LOCA), the offsite dose limits and the control room dose limits are met based on a 50 wt percent per day leakage rate from the RB." The staff determined that the sentence "The RB offers some holdup and decay of fission products..." stated above was ambiguous and needed additional clarification regarding item (4) above. GEH responded that the first sentence in item (4) would be revised to read as follows: "The RB provides holdup which allows time for radioactive decay of fission products that may leak from the containment after an accident."

The staff reviewed the above response, including the GEH markup of DCD Tier 1, Section 2.16.5, and determined that the applicant had provided adequate clarification. The staff checked DCD Tier 1, Revision 9, Section 2.16.5 item (4) to confirm that the modified text properly reflected the GEH response. Based on the above information, RAI 14.3-179 is resolved.

RAIs 14.3-296, 14.3-297, 14.3-383 and 14.3-386 discussed staff requests for the applicant to clarify its terminology in order to achieve greater specificity and reduce the potential for misinterpretation of whether or not the ITAAC have been met:

With respect to ITAAC Table 2.1.2-3, RAI 14.3-296 requested the applicant either to provide a definition for "Nuclear Island" or revise it to refer to the RB or another seismic Category I structure, as applicable. The staff noted that the use of the term "Nuclear Island" is typical throughout the ITAAC and indicated that the applicant should ensure that all other applicable ITAAC are appropriately revised.

With respect to ITAAC Table 2.1.2-3, RAI 14.3-297 requested that the applicant clarify the meaning of "a seismic structure" or refer to a specific building (e.g., the RB or other building, as appropriate, which has its own ITAAC to verify its seismic pedigree) in the acceptance criteria.

With respect to ITAAC Table 2.15.4-2, RAI 14.3-383 requested that the applicant either provide a definition for "Nuclear Island" or replace it with a reference to the appropriate seismic Category I structure (e.g., the RB) for which another ITAAC is provided to verify its seismic pedigree.

With respect to ITAAC Table 2.15.7-2, RAI 14.3-386 requested that the applicant either provide a definition for "Nuclear Island" or replace it with a reference to the appropriate seismic Category I structure (e.g., the RB) for which another ITAAC is provided to verify its seismic pedigree.

GEH provided responses to RAIs 14.3-296, 14.3-297, 14.3-383, and 14.3-386 and indicated that it would modify DCD Tier 1 to replace the terms "Nuclear Island" or "seismic structure" with the term "seismic Category I structure" throughout the ITAAC Tables. GEH also clarified that the seismic Category I structures discussed in DCD Tier 2, Section 3.8 include the concrete containment, RB, control building (CB), fuel building (FB) and fire water service complex.

The staff reviewed the above responses, including the GEH markup of DCD Tier 1, Tables 2.1.1-3, 2.1.2-3, 2.2.4-6, 2.3.1-2, 2.4.1-3, 2.4.2-3, 2.5.10-1, 2.6.1-2, 2.15.4-2 and 2.15.7-2, and determined that the applicant provided adequate clarification. Based on the above information, RAIs 14.3-296, 14.3-297, 14.3-383, and 14.3-386 are resolved.

RAI 14.3-358: For ITAAC Item 3 in Table 2.5.5-1, the staff requested that the applicant provide clear criteria for successful performance of a load test. In response, the applicant stated that the test should be performed in accordance with American National Standards Institute (ANSI) N14.6, "Radioactive Materials—Standard for Special Lifting Devices for Shipping Containers Weighing 10,000 Pounds (4,500 kg) or More," issued January 1993. The staff finds the applicant's response acceptable, and RAI 14.3-358 is resolved.

RAI 14.3-360: For ITAAC Item 7 in Table 2.5.5-1, the staff requested that the applicant provide clear criteria for successful performance of a load test, such as those provided by an industry standard. The applicant responded by stating that in accordance with its practice, it would load test the fuel handling machine auxiliary hoist(s) to 125 percent of rated capacity. The staff finds the applicant's response acceptable, and this RAI 14.3-360 is resolved.

RAI 14.3-380: For ITAAC Item 8 in Table 2.15.1-2, the staff requested that the applicant include the specific design pressure in the acceptance criteria to demonstrate compliance with the ASME Code and requested that the design commitment include a reference to ASME Code Section III Division 2, design and construction requirements. In response, the applicant changed the design commitment to state that the containment system pressure boundary retains its integrity at a design pressure of 310 kPa gauge (45 psig). The applicant also changed the acceptance criteria to state that a test pressure at or above 310 kPa gauge

(45 psig) does not affect containment integrity. In RAI 14.3-380 S01, the staff asked the applicant whether the design and test pressures were the same, since ASME does not treat them as such. The staff also asked the applicant to include a reference to the ASME Code that governs the requirements of concrete and steel containments. In response, the applicant made the appropriate revisions by using the term “design pressure” in both the design commitment and the acceptance criteria and by changing the acceptance criteria to state, “Test report documents that the containment system pressure boundary retains its structural integrity when tested and evaluated in accordance with ASME Code, Section III, Division 2, at a test pressure of at least 115 percent of the design pressure of 310 kPa gauge (45 psig).” The staff finds this modification acceptable, because it clarifies the relationship between the design and test pressures and references the correct ASME Code section. Therefore, RAI 14.3-380 is resolved.

<u>RAI 14.3-381</u>: For ITAAC Item 2 in Table 2.15.3-2, the staff requested that the applicant provide a reference to the containment internal structures identified in Table 2.15.3-1. The applicant should have both “inspection and analyses” performed in the inspections, tests, and analyses and should delete the phrase “as documented in the design reports.” In addition, the applicant should clarify the acceptance criteria to state that inspection reports and analyses document the fact that the as-built components of the containment internal structures comply with the requirements in ANSI-AISC N690-06, “Specification for Safety-Related Steel Structures for Nuclear Facilities” The applicant made the requested changes. The staff finds the applicant’s response acceptable. Therefore, RAI 14.3-381 is resolved.

<u>RAI 14.3-382</u>: For ITAAC Item 3i in Table 2.15.3-2, the staff requested that the applicant provide a reference to the containment internal structures identified in Table 2.15.3-1 in the inspections, tests, and analyses and acceptance criteria. In addition, the applicant should revise the inspections, tests, and analyses to state, “analyses will be performed on the containment internal structures identified in Table 2.15.3-1 to ensure they meet seismic Category I requirements and can withstand seismic design-basis loads and suppression pool hydrodynamic loads without loss of structural integrity and safety function.” In response, the applicant made the requested revisions but also stated that the containment internal structures can withstand loads generated by design-basis LOCAs, hydrodynamic loads, and annulus pressurization loads in the design commitment and the ITAAC without losing the structural integrity and safety function. The applicant made the other requested changes. The staff finds the applicant’s response acceptable. Therefore, RAI 14.3-382 is resolved.

Based on the staff’s review as set forth above, as well as on the applicant’s implementation of the selection criteria and methodology for the development of the DCD Tier 1 information in DCD Tier 2, Section 14.3 the staff concludes that DCD Tier 1 appropriately describes the top-level design features and performance characteristics of the SSCs; the DCD Tier 1 design descriptions associated with the scope of SRP Section 14.3.2 can be verified adequately by the ITAAC; and the DCD Tier 1 information associated with the scope of SRP Section 14.3.2 is acceptable.

Furthermore, the staff concludes that the DCD Tier 1 design descriptions within the scope of SRP Section 14.3.2 can be verified adequately by the ITAAC. Therefore, the staff concludes that the ESBWR ITAAC within the scope of SRP Section 14.3.2 are necessary and sufficient to assure that with respect to these ITAAC, if the inspections, tests, and analyses are performed and the acceptance criteria are met, a facility referencing the certified ESBWR design has been constructed and will be operated in compliance with the design certification and applicable regulations.

14.3.3 Piping Systems and Components

The applicant provided design-basis information, including associated tables and figures, in accordance with the selection methodology for DCD Tier 1 as described in Tier 2, Section 14.3, to support the ITAAC for the ESBWR SSCs. The applicant organized its Tier 1 information in the systems, structures, and topical areas format shown in the Tier 1 table of contents. The staff reviewed the DCD Tier 1, Revision 9, information provided by the applicant using the review matrix provided in Appendix 14.3A, in accordance with the SRP Section 14.3.3.

In SECY-92-053, the staff provided the Commission with a method for using the DAC, together with detailed design information, during the 10 CFR Part 52 process for reviewing and approving designs. The staff used this method for design certification applications that did not provide design and engineering information at a level of detail customarily considered by the staff in reaching a final safety decision on the design. The Commission previously issued guidance on the level of design detail required for design certification. The SRM to SECY-90-377 provided the level of detail that the design should reflect.

14.3.3.1 *Generic Piping Design*

Section 3.12 of this report evaluates the piping design aspects of the ESBWR design provided in DCD Tier 2, Chapter 3. GEH did not provide the complete design information in this design area before design certification because the piping design is dependent upon as-built and as-procured information. Instead, GEH provided the processes and acceptance criteria by which it would develop, design, and evaluate the details of the piping design. GEH provided amplifying information regarding the processes in this area in DCD Tier 2, Section 14.3.3.1. The material in DCD Tier 1, Section 3.1 applies to ESBWR piping systems classified as safety-related systems, as specified in the DCD Tier 1 material for the individual systems in DCD Tier 1, Section 2.

The staff used the SRP guidelines to evaluate the piping design information in DCD Tier 1 and DCD Tier 2 and performed a detailed audit of the piping design criteria, including sample calculations. The staff evaluated the adequacy of the structural integrity and functional capability of safety-related piping systems. The review was not limited to ASME Code Class 1, 2, and 3 piping and supports but included buried piping, instrumentation lines, the interaction of nonseismic Category I piping with seismic Category I piping, and any safety-related piping designed to industry standards other than the ASME Code. The staff's evaluation included the methods of analysis, design procedures, acceptance criteria, and related ITAAC that are to be used for the completion and verification of the ESBWR piping design. The staff's evaluation included both methods to be used for completing the piping design, modeling techniques, pipe stress analysis criteria, pipe support design criteria, and high-energy line break criteria. The staff discussed the development of the DAC in this area in a memorandum to the Commission entitled, "Evaluation of Potential Recommendations to Reduce the Future Use of Design Acceptance Criteria," dated May 6, 2008.

During a public meeting held on October 18, 2007, the staff asked how the piping DAC would be implemented. In response, GEH reworked the ITAAC used to address the piping design and documented a process in DCD Tier 2, Section 14.3.A.1. This section describes the design and the implementation of the process, which is the responsibility of the COL Applicant or Licensee. In multiple RAIs, the staff questioned the process and the terminology used in the ITAAC, including the meaning of the terms used, the documents specified, how they compared with those specified in the ASME Code, and how the piping DAC were differentiated from the other

ITAAC. Through multiple responses, GEH modified the ITAAC so that the actions to be taken were better defined and the ITAAC were more uniform throughout the document. DCD, Tier 1, Revision 5, Section 2 provides component ITAAC on a system basis. In reviewing these ITAAC, the staff identified a number of errors, both in the text and in the tables. In RAI 14.3-414, the staff requested GEH to correct the identified errors in the following six items, as well as others that may have existed in the component ITAAC programs throughout the entire Section 2:

(1) In Table 2.2.2-7 Item 2.a2, the description of inspections, tests, and analyses for as-built ITAAC was not consistent with the intended revision. In Item 2.a3, the description of the entire fabrication and installation ITAAC wais missing.

(2) In Section 2.4.1 Item 2.a2, the description of the as-built ITAAC was not consistent with the intended revision.

(3) In Table 2.6.2-2 Item 2.a3, the description of the entire fabrication and installation portion of the ITAAC was missing.

(4) Section 2.11.1 provided no description of the component ITAAC.

(5) Table 2.11.1-1 included no component ITAAC.

(6) In Table 2.15.1-2 Item 2.c1, descriptions of the inspections, tests, and analyses and acceptance criteria for the fabrication and installation portion of the ITAAC were not consistent with the intended revisions.

RAI 14.3-414 was being tracked as an open item in the SER with open items.

The applicant provided a response that addressed all six items in Tier 1 stated above. The applicant revised the ITAAC to include the missing information and to address the identified concerns, and the staff verified that the modifications were made correctly. Accordingly, the staff finds the applicant’s response acceptable. Therefore, RAI 14.3-414 is resolved.

The material in DCD Tier 1, Section 3.1 describes the process to develop the piping and component designs for the nuclear safety-related (seismic Category I) systems of the ESBWR design and provides a list of the specific Tier 1 sections that contain ITAAC relevant to the piping and component design. Piping systems that must remain functional during and following a safe-shutdown earthquake (SSE) are designated as seismic Category I and are further classified as ASME Code Class 1, 2, or 3. The piping systems and their components are designed and constructed in accordance with the ASME Code requirements identified in the individual systems of the ESBWR design. DCD Tier 1 ensures that the applicant will design the piping systems to perform their safety-related functions under all postulated combinations of normal operating conditions, system operating transients, postulated pipe breaks, and seismic events. The material in this DCD Tier 1 section also addresses the consequential effects of pipe ruptures such as jet impingement, potential missile generation, and pressure and temperature effects.

GEH has specified six ITAAC in DCD Tier 1 to ensure that the design description includes the design process for piping systems. Four of the ITAAC were placed in each of the systems specified in DCD Tier 1, Section 3.1 as applicable. The first ITAAC requires an ASME Code design report to ensure that components identified as ASME Code, Section III are designed in

accordance with ASME Code, Section III requirements and seismic Category I requirements. The second ITAAC requires that piping identified as ASME Code, Section III be designed in accordance with Section III and seismic Category I requirements. The ASME Code gives the specific contents and requirements of the certified design report. As used in this report, an ASME-certified design report is the design document required by ASME Code, Section III, Subarticle NCA-3550. A certified design report provides assurance that requirements of ASME Code, Section III for design, fabrication, installation, examination, and testing have been met and that the design complies with the design specifications. The third and fourth ITAAC require that the as-installed components and piping identified as ASME Code, Section III be reconciled with the design requirements.

Two of the ITAAC remain in DCD Tier 1, Section 3.1. In the third ITAAC in that section (the first two were moved to the systems sections), SSCs that are required to be functional during and following an SSE are to be protected against or qualified to withstand the dynamic and environmental effects associated with the analyses of postulated failures in seismic Category I and non-safety-related piping systems. In the sixth ITAAC (the fourth and fifth were moved to the systems sections), on an individual component and/or system basis, the as-built SSCs are to be reconciled with the analysis results of the postulated failures in seismic Category I and non-safety-related piping systems.

The staff generated RAIs on the issues discussed below. They were being tracked as open items in the SER with open items.

<u>RAIs 14.3-212 and 14.3-131</u>: In DCD Tier 2, Revision 4, the applicant revised Sections 3.6.2.5 and 3.6.5-1-A. Specifically, DCD Tier 2, Revision 4, Section 3.6.5-1-A stated that the COL Applicant shall provide the information identified in Section 3.6.2.5, while Section 3.6.2.5 lists the information that will be included in the pipe break evaluation report. The applicant also stated that the pipe break evaluation report will be completed in conjunction with closure of ITAAC Item 3 in Table 3.1-1. Furthermore, in response to RAI 14.3-131 S01, the applicant proposed to delete Section 3.6.5-1-A regarding the COL information item, which would have required that the COL Applicant provide details of pipe break analysis results and protection methods. Based on its review of the above information, the staff, in RAI 14.3-212, requested the applicant to provide additional information concerning the pipe break evaluation report. Specifically, the staff noted that ITAAC Item 3 in Table 3.1-1 would require inspection of the as-built pipe break analysis report as opposed to the as-designed pipe break hazards analysis. In RAI 14.3-131 S01, the staff also requested the applicant to address similar pipe break-related issues as contained in ITAAC Table 3.1-1. In response, GEH modified ITAAC Item 3 in Table 3.1-1 to apply to the “as-designed” rather than the “as-built” pipe analysis. The write-up referred to ITAAC 1 through 6, but only five ITAAC were included.

Based on its review of the information provided by the applicant, the staff requested in RAI 14.3-131 S02, Question 4, that the applicant provide additional modifications to ITAAC Table 3.1-1. Specifically, the staff requested that the ITAAC should be revised to state, “as-designed pipe break analysis results” as opposed to “pipe analysis.” The staff requested that this change should also be made under “Inspections, Tests, and Analyses” to refer to the report specified in Section 3.6.2.5 of the DCD. Further, the staff requested that Item 6 should remain and be modified to address the reconciliation with the report specified in Section 3.6.2.5.

GEH provided its response to RAI 14.3-131 S02. Based on its review of that RAI response, as well as on the information provided in Revision 5 of the DCD, the staff found that the “as-built” wording was changed to “as-designed” in Revision 5 of DCD ITAAC Table 3.1-1. In addition,

the staff found that Item 6 was included in that table. However, the staff determined that GEH did not address the staff's concern pertaining to the wording "pipe analysis" of the ITAAC table, and the wording of the ITAAC failed to fully address the COL information item. As written, the new ITAAC called for a report to document the conclusions of the as-designed pipe analysis: (1) that, for each postulated piping failure, the reactor can be shut down safely, and (2) that the reports document the results of the analyses to determine where protection features are necessary to mitigate the consequences of a pipe break. The COL information item would have required the applicant to provide the information identified in DCD Section 3.6.2.5. This section called for a pipe break evaluation report to be completed in conjunction with the closure of ITAAC 3.1-1. The report was to include the following:

- A summary of the dynamic analyses applicable to high-energy piping systems in accordance with Section 3.6.2.5 of RG 1.70, Revision 3. These include sketches of applicable piping systems showing the location, size, and orientation of postulated pipe breaks; the location of pipe whip restraints and jet impingement barriers; and a summary of the data developed to select postulated break locations, including calculated stress intensities, cumulative usage factors, and stress ranges as delineated in the Branch Technical Position 3-4, Revision 2, "Postulated Rupture Locations in Fluid System Piping Inside and Outside Containment." issued March 2007

- For failure in the moderate-energy piping systems, descriptions showing how safety-related systems are protected from the resulting jets, flooding, and other adverse environmental effects.

- Identification of protective measures provided against the effects of postulated pipe failures for protection of each of the systems listed in Tables 3.6-1 and 3.6-2.

- The details of how the functional capability of the MSIV is protected against the effects of postulated pipe failures.

- Typical examples, if any, where protection for safety-related systems and components against the dynamic effects of pipe failures include their enclosure in suitably designed structures or compartments (including any additional drainage system or equipment EQ needs).

- The details of how the functional capabilities of the feedwater line check and feedwater isolation valves are protected against the effects of postulated pipe failures.

Since the information discussed above is associated with the completion of the DAC and the deleted COL information item was previously found acceptable to the staff, the staff requested that the ITAAC require the same design information as previously discussed in the deleted COL information item. RAI 14.3-131 S02, Question 4, was being tracked as an open item in the SER with open items.

In RAI 14.3-131 S03, the staff requested the applicant to modify the ITAAC table to address the above concern of the staff. In response, GEH provided two marked-up pages of DCD Tier 1, ITAAC Table 3.1-1. Specifically, the applicant changed the wording "as-designed pipe analysis report" of Items 3 and 6 of the ITAAC table into "as-designed pipe break analysis results report." The applicant further stated that DCD Tier 2, Section 14.3A states that the content of the pipe break analysis results report referred to in ITAAC Table 3.1-1 is discussed in DCD Tier 2, Section 3.6.2.5, which provides the details of the information required in the pipe break analysis

results report. Based on its review of the information provided by the applicant, the staff determined that the applicant's proposed changes to ITAAC Table 3.1-1 adequately addresses the staff's concerns, because the revised wording in ITAAC Table 3.1-1 is now consistent with the title of the pipe break analysis report included in DCD Tier 2, Section 3.6.2.5. In addition, the revised DCD Tier 2, Section 14.3.A now clearly refers to DCD Tier 2, Section 3.6.2.5. The staff concludes that RAI 14.3-131 S03, RAI 14.3-212, and the associated open items are resolved.

The following RAIs were being tracked as open items in the SER with open items:

RAI 14.3-368: For ITAAC Item 5a in Table 2.6.2-2, the staff requested the applicant to revise the design commitment to use the term "equipment" to be in agreement with the referenced table(s). The staff requested that the safety-related equipment be stated to be seismic Category I and be able to withstand seismic design-basis loads without a loss of safety function. RAI 14.3-368 was being tracked as an open item in the SER with open items.

The applicant's response includes the changes suggested by the staff. The definition of "equipment" in Revision 6 of the DCD also applies to this ITAAC and to any changes in it. The Design Commitment states that the safety-related equipment as listed in Table 2.6.2-1 withstands seismic Category I loads without a loss of safety function.

The three inspections, tests, and analyses respectively, verify: (1) by inspection, that the equipment, including the piping in Table 2.6.2-1, is located in a seismic Category I structure, (2) that the type tests, analyses, and/or a combination of them are performed on the seismic Category 1 equipment using analytical assumptions or stated conditions that bound the seismic Category I design requirements, and (3) by inspection and analyses that the seismic response of the installed equipment, including piping and anchorages, is bounded by the tested or analyzed conditions.

These three acceptance criteria respectively require that: (i) the equipment in Table 2.6.2-1 is located in a seismic Category I structure, (ii) the seismic Category I equipment, including associated piping, can withstand seismic design-basis loads without a loss of safety function, and (iii) the as-installed equipment, including anchorages, have been tested or analyzed under conditions necessary to ensure compliance with seismic Category I design requirements. The staff agrees with the applicant's response and the revisions made to this ITAAC. Therefore, RAI 14.3-368 and the associated open items are resolved.

RAI 14.3-387: For ITAAC Item 3b in Table 2.16.2-2, the staff asked the applicant to clarify in the inspections, tests, and analyses that the "testing or analyzed conditions bound the seismic Category I design requirements."

The inspections, tests, and analyses verify: (1) by inspection that the components in Table 2.16.2-2 are located in a seismic Category I structure, not just a seismic structure, (2) that the type tests, analyses, and/or a combination of both are performed using analytical assumptions or stated conditions that bound the seismic Category I design requirements, and (3) by inspection and analyses that the installed components, including anchorage, are seismically bounded by the tested or analyzed conditions.

Also, the staff requested that the applicant have reports for the three acceptance criteria that conclude: (1) by inspection, that the components are located in a seismic Category I structure, not just a seismic structure, (2) by type tests and/or analysis that the seismic Category I

components can withstand seismic design-basis loads without a loss of safety function, and (3) by inspection and analysis, that the installed components, including anchorage, are seismically bounded by tested or analyzed conditions. RAI 14.3-387 was being tracked as an open item in the SER with open items.

The applicant in its response revised this ITAAC and other ITAAC tables to address the staff's concerns. The staff agrees with the applicant's response and the revisions made to this ITAAC. Therefore, RAI 14.3-387 and the associated open items are resolved.

Additionally, the following RAIs were being tracked as open items in the SER with open items and are similar in nature to RAIs 14.3-368 and 14.3-387 but were associated with different ITAAC tables:

RAI 14.3-352 (associated with ITAAC Item 13 in Table 2.2.4-6)
RAI 14.3-353 (associated with ITAAC Item 5 in Table 2.4.1-3)
RAI 14.3-354 (associated with ITAAC Item 5 in Table 2.4.2-3)
RAI 14.3-384 (associated with ITAAC Item 5 in Table 2.15.4-2)

As with the responses to RAIs 14.3-368 and 14.3-387, the staff agrees with the applicant's response to these RAIs. The above RAIs and associated open items are resolved.

RAI 14.3-349: For ITAAC Item 1 in Table 2.2.2-7, the staff requested that the applicant revise the acceptance criteria for the CRD system to address the results of inspections, tests, and type tests, not just inspections. The applicant revised the acceptance criteria to state, "A report exists that documents the results of inspection(s), test(s), and type test(s) that confirm the as-built CRD system conforms with the functional arrangement defined in Table 2.2.2-1 and as shown in Figure 2.2.2-1." The staff finds the applicant's response acceptable. Therefore, RAI 14.3-349 is resolved.

RAI 14.3-357: For ITAAC Item 2 in Table 2.5.5-1, the staff requested that it include both inspections and analyses. When the applicant made this change, the staff asked if this ITAAC was consistent with the Tier 2 material. The applicant then changed Tier 2 to state that the refueling machine in the RB was seismic Category I. The staff finds these responses to be acceptable. Therefore, RAI 14.3-357 is resolved.

RAI 14.3-359: For ITAAC Item 6 in Table 2.5.5-1, the staff asked the applicant to include a design commitment for the seismic qualification of the fuel handling machine in the FB. In addition, the staff requested that the applicant modify the inspections, tests, and analyses portion of the ITAAC to clearly state that "inspections and analyses… will both be performed." The applicant stated that it would revise the design commitments for ITAAC Items 2 and 6 in Table 2.5.5-1 to show that both the FB fuel handling machine and the RB refueling machine are seismic Category I. The applicant also revised the inspections, tests, and analyses for Item 6 to include both inspections and analyses. The staff asked if this ITAAC was consistent with Tier 2 material. The applicant later changed Tier 2 to state that the RB refueling machine was seismic Category I. The staff finds these responses to be acceptable. Therefore, RAI 14.3-359 is resolved.

RAI 14.3-366: For ITAAC Item 2 in Table 2.6.2-2, the applicant should (1) verify design completion in an ASME design report; (2) reconcile the as-built installation with the design documents in an ASME design report; and (3) verify in an ASME data report that the SSCs are fabricated, constructed, and installed in accordance with the design documents. The applicant,

in response to RAI 14.3-131 S02, provided a revision that added the previously missing steps. The staff finds the applicant's response acceptable. Therefore, RAI 14.3-366 is resolved.

<u>RAI 14.3-377</u>: For ITAAC Item 2 in Table 2.15.1-2, the applicant should ensure that the ITAAC includes the performance of the following three steps for each of the components and piping: (1) verifies the design and documents that are in an ASME design report, (2) verifies the reconciliation of the design with as-built installation and documents that are in an ASME design report, and (3) verifies that the SSC is fabricated, constructed, and installed per the design and documents that are in an ASME data report. The applicant provided a response that addressed all three elements stated above. The staff finds the applicant's response to be acceptable. Therefore, RAI 14.3-377 is resolved.

<u>RAI 14.3-378</u>: For ITAAC Item 4i in Table 2.15.1-2, the design commitment referred to "components and piping," while the inspections, tests, and analyses and acceptance criteria referred only to "components." The staff requested that the applicant make the design commitment, inspections, tests, and analyses, and acceptance criteria consistent in scope. Also, the inspections, tests, and analyses referred to a "hydrostatic or pressure test," while the acceptance criteria referred only to a "pressure test." The staff requested that the applicant ensure consistency between the inspections, tests, and analyses and acceptance criteria and noted that this ITAAC involves ASME equipment, whereas the hydrostatic test requirements (not pressure testing) would normally be the applicable requirement. The applicant revised the inspections, tests, and analyses and acceptance criteria to use the phrase "components and piping" and changed the acceptance criteria to refer to "hydrostatic testing." The staff finds the applicant's response to be acceptable. Therefore, RAI 14.3-378 is resolved.

<u>RAI 14.3-388</u>: For ITAAC Item 1 in Table 3.1-1, the staff requested that the applicant (1) provide a reference table that would list all of the safety-related piping for which this ITAAC is applicable, (2) clarify or provide a distinction between design commitment and as-built verification (the ASME Code Certified Stress Report only provides verification of the design of the system), and (3) provide an ITAAC to verify the as-built system was constructed in accordance with the ASME Code. The applicant provided a list of systems and components in Section 3.1 of its DCD that are subject to ASME Code, Section III requirements. The applicant also specified in the respective ITAAC for the other sections of the DCD that ASME Code Design Reports will be used to close the DAC ITAAC and to verify that the as-built piping, components, and/or structures subject to Section III of the ASME Code meet the design requirements. In addition, separate ITAAC will be used to verify that those same piping, components, and structures are fabricated, installed, and inspected based on the results recorded in ASME Code Data Reports. The applicant deleted ITAAC Items 1 and 2 in Table 3.1-1 and developed ITAAC in other sections of the DCD to address the applicant's responses for this RAI. The staff finds the applicant's responses to be acceptable. Therefore, RAI 14.3-388 is resolved.

Based on the staff's review as set forth above,, as well as on the applicant's implementation of the selection criteria and methodology for the development of the DCD Tier 1 information in Section 14.3 of DCD Tier 2, the staff concludes that the top-level design features and performance characteristics of the SSCs are appropriately described in Tier 1, and the Tier 1 information associated with the scope of SRP Section 14.3.3 is acceptable.

Furthermore, the staff concludes that the Tier 1 design descriptions within the scope of SRP Section 14.3.3 can be verified adequately by the ITAAC. Therefore, the staff concludes that the ESBWR ITAAC within the scope of SRP Section 14.3.3 are necessary and sufficient to assure

that with respect to these ITAAC, if the inspections, tests, and analyses are performed and the acceptance criteria are met, then a facility referencing the certified ESBWR design has been constructed and will be operated in compliance with the design certification and applicable regulations.

14.3.3.2 *Verifications of Components and Systems*

In addition to addressing piping and component design, the staff confirmed that DCD Tier 1 addresses the verification of piping and component classification, fabrication, dynamic and seismic qualification, and selected testing and performance requirements through specific ITAAC in the individual DCD Tier 1 systems.

In RAI 14.3-180, the staff questioned ASME Code applicability to the chimney and partitions, the chimney head and steam separator assembly, and the steam dryer assembly. In its response to RAI 14.3-180, GEH modified the design description, equipment list, and ITAAC to include the internal structures of concern. The staff finds that the changes addressed its concerns. Therefore, RAI 14.3-180 is resolved.

The following examples discuss some of the other concerns that the staff identified during its review that were resolved as a result of RAI responses:

In RAI 14.3-210, the staff requested that GEH provide a COL information item requiring the applicant to provide a closure schedule for the DAC in its COL application. In its response to RAI 14.3-210, GEH included a COL information item to address the staff's concerns. COL Information Item 14.3A-1-1 requires each applicant to provide a DAC ITAAC closure schedule in the COL application and identify whether the standard approach will be used. The staff finds the applicant's response acceptable. Therefore, RAI 14.3-210 is resolved.

RAI 14.3-367: For ITAAC Item 4 in Table 2.6.2-2, the design commitment referred to "piping and components;" however, the inspections, tests, and analyses and acceptance criteria referred only to "components." The staff requested that the applicant ensure consistency among the associated design commitment, inspections, tests, and analyses, and acceptance criteria. In addition, the staff requested clarification of the phrase "a hydrostatic or pressure test" used in the inspections, tests, and analyses. The staff discerned no need for a distinction when ASME Code Section III requirements are applied. Likewise, the use of the term "pressure test" in the acceptance criteria should be clarified or modified to be consistent with the inspections, tests, and analyses. The applicant made some of the requested modifications. However, in its response, the applicant used the term "pressure test" instead of the more acceptable term "hydrostatic test," which is the preferred test of the ASME Code. The applicant made the requested modifications in the DCD for the ESBWR in Revision 5 by modifying the inspections, tests, and analyses and acceptance criteria to use the term "hydrostatic test." The staff finds the applicant's responses to be acceptable. Therefore, RAI 14.3-367 is resolved.

RAI 14.3-371: For ITAAC Item 2 in Table 2.10.1-2, the staff requested that the applicant revise the acceptance criteria to (1) identify the components omitted from the test, and (2) document the reason the component was omitted from hydrostatic testing and indicate whether an alternative test (alternative to hydrostatic testing) was conducted to verify pressure boundary integrity. The applicant revised the acceptance criteria to explain that the LWMS piping systems will be hydrostatically pressure tested in conformance with the requirements in the American Petroleum Institute or ASME Codes and in accordance with RG 1.143, Revision 2, "Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed

in Light-Water-Cooled Nuclear Power Plants," issued November 2001. The ITAAC meets the recommendations of RG 1.143, Section 4.4. The applicant stated it would make an assessment of any components that might be omitted from the hydrostatic test, when developing the test procedure for hydro-testing the system, since the determination of appropriate alternate testing could only be made based on the specific system design configuration. Pneumatic or manufacturer type testing are examples of alternative testing that could be used to demonstrate system leak integrity.

The staff did not find the applicant's response completely acceptable and requested the following revisions. The inspections, tests, and analyses should include a hydrostatic test on the LWMS piping systems with exceptions that are in accordance with RG 1.143, Revision 2, and the applicant should revise the acceptance criteria to document: (1) that the results of the hydrostatic test of the LWMS piping systems was in accordance with ASME/ANSI B31.3, (2) that it conformed to the requirements of the ASME Code and RG 1.143, Revision 2, and (3) that no unacceptable pressure boundary leakage occurred. The applicant made these revisions in ESBWR DCD Tier 2, Revision 5. The staff finds the applicant's responses to be acceptable. Therefore, RAI 14.3-371 is resolved.

RAI 14.3-372: For ITAAC Item 4b in Table 2.10.3-1, the staff requested that the applicant modify the acceptance criteria to specifically define "treat mode alignment" to mean that a MCR alarm will sound and gas will flow through the charcoal beds. An alternative was to define "treat mode alignment" in the design description for the LWMS. The applicant chose to define the term "treat mode alignment" within the ITAAC itself. The staff finds the applicant's response to be acceptable. Therefore, RAI 14.3-372 is resolved.

RAI 14.3-373: For ITAAC Item 1 in Table 2.12.1-1, the staff asked the applicant to provide a list of the makeup water system penetrations and isolation valves referred to in the ITAAC as being in Section 2.15.1 or provide a suitable justification for not including such a list. The applicant included a new Table 2.15.1-1a that lists the valves and penetrations. The staff finds the applicant's response to be acceptable. Therefore, RAI 14.3-373 is resolved.

Based on the staff's review as set forth above, as well as on the applicant's implementation of the selection criteria and methodology for the development of the DCD Tier 1 information in DCD Tier 2, Section 14.3, the staff concludes that the top-level design features and performance characteristics of the SSCs are appropriately described in Tier 1, and the Tier 1 information associated with the scope of SRP Section 14.3.3 is acceptable.

Furthermore, the staff concludes that the DCD Tier 1, Revision 9, design descriptions within the scope of SRP Section 14.3.3 can be verified adequately by the ITAAC. Therefore, the staff concludes that the ESBWR ITAAC within the scope of SRP Section 14.3.3 are necessary and sufficient to assure that with respect to these ITAAC, if the inspections, tests, and analyses are performed and the acceptance criteria met, then a facility referencing the certified (ESBWR) design has been constructed and will be operated in compliance with the design certification and applicable regulations.

14.3.4 Reactor Systems

The applicant provided design-basis information, including associated tables and figures, in accordance with the selection criteria and methodology for developing DCD Tier 1, Revision 9, information, as described in Tier 2, Revision 9, Section 14.3, to support the ITAAC for the ESBWR SSCs. The applicant organized the Tier 1 information using the systems, structures,

and topical areas format shown in the DCD Tier 1, Table of Contents. The staff reviewed the DCD Tier 1 information provided by the applicant, in accordance with the review matrix provided in Appendix 14.3A and in accordance with SRP Section 14.3.4.

The staff found that many of the systems within the scope of review of SRP Section 14.3.4 were classified as safety-related, and thus many of the characteristics and features of these systems were judged to have safety significance. This is reflected in a higher level of detail in the ITAAC for these systems. The staff reviewed the ITAAC to verify that plant safety analyses, such as for core cooling, transients, overpressure protection, and anticipated transients without scram, were adequately addressed. The staff used the tables contained in DCD Tier 2, Sections 6.3, 15.2, and 15.3 to determine if the important input parameters used in the transient and accident analyses were verified by the ITAAC. The staff also interacted with specialists in PRA and severe accident analyses to ensure that the ITAAC incorporated the important insights and design features from these analyses. For the severe accident analyses, in particular, the basis for the staff's review was the Commission guidance in SECY-90-016, “Evolutionary Light-Water Reactor (LWR) Certification Issues and Their Relationship to Current Regulatory Requirements,” dated January 12, 1990, and SECY-93-087, “Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs.” dated April 2, 1993.

For both PRA and severe accident analyses, design features important for severe accident prevention and mitigation resulting from these analyses were selected for treatment in the ITAAC. The supporting information regarding the detailed design and analyses remain in DCD Tier 2. The staff determined that the detailed supporting information in DCD Tier 2 for the nuclear fuel, fuel channel, and control rod CDM, if considered for a change by a COL Applicant or Licensee that references the certified ESBWR design, would require prior staff review under the criteria of 10 CFR 50.59. Thus, the staff has concluded that the fuel cycle and control rod design criteria in DCD Sections 4B and 4C, the first cycle fuel, control rod and core design, and the methods used to analyze these components, may not be changed without prior NRC review and approval. (This information is designated as Tier 2* in the DCD.) The specific fuel, control rod, and core designs presented in DCD Chapter 4 will constitute, based on the staff’s review and approval, an approved design that may be used for the COL first cycle core loading without further NRC review. If any other core design is requested for the first cycle, the COL Applicant or Licensee will be required to submit, for the staff’s review, specific fuel, control rod, and core design analyses as described in DCD Tier 2, Revision 9, Chapters 4, 6, and 15.

Based on the guidance provided in SRP Section 14.3, Tier 2* information is information that is generally not appropriate for treatment in Tier 1 because it is subject to change. As such, the staff believes that no ITAAC are required for the CDM information in the areas discussed above. In addition, the ITAAC must be performed prior to fuel load. Therefore, verification that the actual core performs in accordance with the analyzed core design are addressed in post-fuel-load testing programs (e.g., startup testing and power ascension testing).

As a result of its review, the staff identified a number of RAIs involved in requests for the applicant to provide additional definitions in DCD Tier 1, Sections 1 and 2 and to further clarify ITAAC Items 1, 3, 4, 5, and 7 for RPV systems in DCD Tier 1, Table 2.1.1-3. The requested information was to improve overall understanding of the design commitment and ITAAC stated in Table 2.1.1-3. The applicant incorporated the definitions in the appropriate sections and modified Table 2.1.1-3 accordingly. The staff finds that these changes are acceptable.

The applicant resolved several staff concerns through the RAI process, including the following examples:

RAI 14.3-356: For ITAAC Item 8a in Table 2.4.2-3, the staff asked the applicant to modify the inspections, tests, and analyses to include “analysis” and to modify the acceptance criteria to include test results, in addition to the analysis results. The staff also asked the applicant to provide specific acceptance criteria to determine acceptability. The applicant made the revisions in the inspections, tests, and analyses and the acceptance criteria, except for including specific acceptance criteria for acceptability. The applicant stated that existing acceptance criteria for acceptability were appropriate. The staff finds the applicant’s response acceptable. Therefore, RAI 14.3-356 is resolved.

RAI 14.3-370: For ITAAC Item 7b in Table 2.6.2-2, the inspections, tests, and analyses specifies the performance of a test for both the flow path and capacity, while the acceptance criteria only refers to flow path. The staff requested that the applicant modify the acceptance criteria to include the flow rate criteria for acceptance. The applicant modified the acceptance criteria to include the flow rate. The staff finds the applicant’s response acceptable. Therefore, RAI 14.3-370 is resolved.

In response to RAI 14.3-180, GEH added Chimney and partitions, Chimney head, steam separator assembly, and steam dryer assembly to Table 2.1.1-1. The staff finds the applicant’s response acceptable. Therefore, RAI 14.3-180 is resolved.

In response to RAI 14.3-189, GEH added the pressure loss coefficient in Table 2.1.2-3 for the following components: Steam Separator, Fuel Bundle, Fuel support piece orifice, Control Rod Guide Tubes, and Shroud Support. The staff finds the applicant’s response acceptable. Therefore, RAI 14.3-189 is resolved.

DCD Tier 1, Revision 9, Section 2.2.2 includes the design description and associated ITAAC for the CRD system (Table 2.2.2-1) and a detailed system drawing (Figure 2.2.2-1). In its response to RAI 4.6-21 regarding an unspecified electric-motor drive speed, GEH revised ITAAC No. 3 to verify the specified motor speed. The staff considers the applicant’s response to be acceptable. Therefore, RAI 4.6-21 is resolved. Since the “electric scram” is a backup to the hydraulic scram, the motor speed is not safety significant. Hence, the ITAAC for the motor speed was deleted.

In its response to RAI 4.6-24 regarding the clarification of scram insertion requirements, GEH revised the DCD to require verification testing of the “maximum allowable scram insertion times for each FMCRD,” instead of the “average of all FMCRDs.” The applicant revised DCD Tier 1, Section 2.2.2 and Table 2.2.2-1 and DCD Tier 2, Section 4.6.1.2.4 to clarify the scram insertion requirements. The staff considers the RAI response acceptable, since defining the maximum scram insertion time for each FMCRD is consistent with the same requirements in the standard TS for the current fleet of operating BWRs. Therefore, RAI 4.6-24 is resolved.

DCD Tier 1, Section 2.2.2 states that each HCU “also provides the flow path for purge water to the associated drives during normal operation.” In response to RAI 4.6-24 regarding this mode and any other CRD system lineup and their potential impact on scram insertion, the applicant stated that “as long as the scram accumulator remains charged, there is no operating mode of the CRD system that can impact the scram insertion mode.” The response subsequently described the features of the system that maintain the capability of the scram function. The CRD system design description provided in DCD Tier 1, Section 2.2.2, along with the ITAAC in Table 2.2.2-1, provide sufficient design specification and validation testing to ensure that the

ESBWR CRD system will satisfy the applicable regulatory criteria. DCD Tier 2, Section 4.6.3.2 offers additional details on the CRD system design. Based on this information, RAI 4.6-25 is resolved.

The staff reviewed DCD Tier 1, Section 2.2.4, SLCS, including the ITAAC presented in Table 2.2.4-6. In RAI 9.3-15, the staff requested that GEH add an ITAAC in Table 2.2.4-6, to verify that the initial SLC injection flow rate is consistent with the assumptions in the safety analysis. In DCD Tier 1, Revision 5, the acceptance criteria for ITAAC Item 7 in Table 2.2.4-6 specified that the first and second 5.4 m^3 of boron solution injects in less than or equal to 519 seconds during the ATWS. In addition, in DCD Tier 1, Revision 5, the acceptance criteria for ITAAC Item 8 in Table 2.2.4-6 specified that test and analysis reports exist and conclude that the as-built SLC system (both accumulators) injects a total volume of 15.6 m^3 boron solution in response to a LOCA. Based on the revisions provided in DCD, Revision 5, as described above, the staff finds the response to RAI 9.3-15 acceptable.

In response to RAI 14.3-202, GEH revised the acceptance criteria for ITAAC Item 13 in Table 2.4.1-3 to include the ICS condensate return valve opening time of no less than 7.5 seconds and no greater than 31 seconds. The staff finds the applicant's response acceptable. Therefore, RAI 14.3-202 is resolved.

In response to RAI 14.3-149, GEH added GDCS deluge system functions in the design description and included associated ITAAC Items 22, 25, 26, and 27 to verify these functions in Table 2.4.2-3. The staff finds the applicant's response acceptable. Therefore, RAI 14.3-149 is resolved.

The following RAIs were being tracked as open items in the SER with open items:

<u>RAI 14.3-397</u>: In RAI 14.3-397 regarding DCD Tier 1, Figure 2.1.1-2, the staff requested the applicant to indicate the relative locations of the startup range neutron monitors, low-power range monitors, neutron sources, and spare source locations in a manner similar to DCD Tier 2, Figure 4.1-1. In addition, the staff requested the applicant to include the quantities in the figure legend. In response, GEH revised DCD Tier 1, Figure 2.1.1-2 indicating the relative locations of the NMS and the quantities. Therefore, RAI 14.3-397 and the associated open item are resolved.

<u>RAI 14.3-398</u>: In DCD Tier 1, Table 2.1.1-3, Item 9, the ITAAC acceptance criteria for flow-induced vibration testing of fuel bundles is given as one order of magnitude. The staff did not believe that this value was well supported and asked the applicant to provide supporting information in DCD Tier 2 to justify the value. The staff sent supplementary RAIs 4.8-7 S01 to S04 to GEH requesting additional information. In response to the RAIs, GE provided additional information. The staff SER for NEDC-33240P, "GE 14E Mechanical Design Report," includes the discussion for the resolution of this issue. DCD Tier 1, Table 2.1.1-3, Item 9 was revised to delete the criterion of "one order of magnitude." The revised acceptance criterion included in Revision 6 of the DCD is acceptable. Therefore, RAI 14.3-398 and the associated open item are resolved.

<u>RAI 14.3-399</u>: In RAI 14.3-399 regarding DCD Tier 1, Table 2.1.1-3, Item 11, the staff requested the applicant to revise the ITAAC acceptance criteria for the RPV to state… "A report exists and concludes that the as-built reactor system fuel bundle, control rod, instrumentation, and neutron source locations conform to the locations shown on Figure 2.1.1-2." In response, GEH revised the acceptance criteria in Revision 6 of the DCD to verify the as built arrangement.

Since the revised acceptance criteria provides that the installed equipment location conform to the design, it is acceptable. Therefore, RAI 14.3-399 and the associated open item are resolved.

RAI 14.3-400: In DCD Tier 1, Revision 5, Table 2.2.4-5 and Table 2.4.1-2, the "Active Function" column was deleted. The response to RAI 14.3-354 was given as the basis for the deletion. However, the response to RAI 14.3-354 is related to mechanical equipment rather than electrical equipment. Therefore, the staff requested a clarification on the applicability of the response to electrical equipment. As a comparison, the active safety function column was maintained in Table 2.1.2-2. The staff requested that the applicant explain this inconsistency. In response, GEH clarified that the "Active Safety Function" is provided in Table 2.1.2-1 (valve safety-related position). Therefore, RAI 14.3-400 and the associated open item are resolved.

RAI 14.3-401: In RAI 14.3-401 regarding DCD Tier 1, Section 2.4.1, the staff requested that GEH include a statement to indicate that the ICS minimum inventory of alarms, displays, controls, and status indications in the MCR are addressed in DCD Tier 1, Table 3.3-1a, which includes the ICS minimum inventory of MCR alarms, displays and controls. The staff determined the response is acceptable. Therefore, RAI 14.3-401 and the associated open item are resolved.

RAI 14.3-350: For ITAAC Item 9 in Table 2.2.2-7, the staff requested that the applicant modify the inspections, tests, and analyses and acceptance criteria to include verification that the associated interfacing systems specified in Table 2.2.2-3 were functional, based on other ITAAC, and that the list of interfacing systems was complete. The initial change only addressed; (1) the conformance of the CRD system in regard to automatic initiators, functions, and associated interfacing systems, and (2) the use of tests and type tests to generate simulated signals from all interfacing systems. The staff found the content of the initial change to be acceptable, as far as it goes, but requested that this ITAAC be reformatted to resemble the ITAAC for RAI 14.3-348. The applicant could do this by using tests and type tests to generate simulated signals of the initiators to perform the automatic functions of the CRD system listed in Table 2.2.2-3.

In response, the applicant indicated that the two steps in the inspections, tests, and analyses and in the acceptance criteria indicate: (i) that tests and type tests show that the CRD is capable of performing the functions defined in Table 2.2.2-3 using simulated signals initiated from all the interfacing systems specified in Table 2.2.2-3, and (ii) that inspections show that the as-installed CRD system conforms with all the automatic initiators, functions, and associated interfacing systems defined in Table 2.2.2-3. The staff agrees with the applicant's response and the revisions made to this ITAAC. Therefore, RAI 14.3-350 and the associated open item are resolved.

RAI 14.3-351: For ITAAC Item 12 in Table 2.2.4-6, the staff requested that the applicant identify the type of ASME report that is required (i.e., whether it is a design report or a data report). The staff requested that the applicant review all of its ITAAC associated with ASME systems and components and make the same change, as appropriate.

The applicant revised these ITAAC and similar ones in its response by changing their Design Commitments as requested. The staff also requested the applicant identify the type of ASME report referred to in the ITAAC. The applicant revised the ITAAC to refer to an ASME Code Data Report. The staff agrees with the applicant's responses and the revisions made to these

ITAAC and other similar ones. Therefore, RAI 14.3-351 and the associated open item are resolved.

Based on the staff's review as set forth, as well as on the applicant's implementation of the selection criteria and methodology for the development of the DCD Tier 1 Revision 9, information in DCD Tier 2, Revision 9, Section 14,3, the staff concludes that DCD Tier 1, Revision 8 appropriately describes the top-level design features and performance characteristics of the SSCs and that the DCD Tier 1, Revision 9, information associated with the scope of SRP Section 14.3.4 is acceptable.

Furthermore, the staff concludes that the DCD Tier 1, Revision 9, design descriptions within the scope of SRP Section 14.3.4 can be adequately verified by the ITAAC. Therefore, the staff concludes that the ITAAC within the scope of SRP Section 14.3.4 are necessary and sufficient to assure that with respect to these ITAAC, if the inspections, tests, and analyses are performed and the acceptance criteria met, a facility referencing the certified ESBWR design has been constructed and will be operated in compliance with the design certification and applicable regulations.

14.3.5 Instrumentation and Controls

The applicant provided design-basis information, including associated tables and figures, in accordance with the selection methodology for DCD Tier 1 as described in DCD Tier 2, Section 14.3, to support the ITAAC for the ESBWR SSCs. The applicant organized the DCD Tier 1 information in the systems, structures, and topical areas format shown in the DCD Tier 1, "Table of Contents." The staff reviewed the DCD Tier 1 information provided by the applicant using the review matrix provided in Appendix 14.3A, in accordance with SRP Section 14.3.5.

In SECY-92-053, the staff provided the Commission with a method for using the DAC, together with detailed design information, during the 10 CFR Part 52 process for reviewing and approving designs. The staff used this method for design certification applications that did not provide design and engineering information at a level of detail customarily considered by the staff in reaching a final safety decision on the design. The Commission previously issued guidance on the level of design detail required for the design certification. The SRM to SECY-90-377 provided the level of detail that the design should reflect.

The applicant may provide the DAC in lieu of detailed system design information in areas where the technology is rapidly changing, such as I&C. The COL licensee must verify the implementation of the DAC as part of the ITAAC performed to demonstrate that the as-built facility conforms to the certified design. In this case, the DAC should be sufficiently detailed to provide an adequate basis for the staff to make a final safety determination regarding the design, subject only to the satisfactory verification of completion of the design (i.e., verification of the DAC) and installation of the completed design by the COL Applicant or Licensee.

The specific areas of review are as follows:

- DCD Tier 1 information on I&C systems involving reactor protection and control, ESF actuation, and other systems using I&C equipment

- DCD Tier 1 information related to the design process of digital computers in I&C systems

- Selected interface requirements related to I&C issues

- Functional requirements of IEEE Std 603 and the GDC when implementing the safety system

The staff reviewed the ITAAC associated with I&C found in ESWBR DCD Tier 1, Sections 2.2.1 through 2.2.15, 2.15.7, 2.3.1, 2.3.2 and 3.2. Other DCD Tier 1 sections and specific ITAAC entries refer to DCD Tier 1, Section 2.2.15 and 3.2 for I&C quality requirements. The staff also considered additional information provided in DCD Tier 1, Sections 3.3, 3.7, and 3.8.

As a result of the staff's review and the RAI process, the applicant refocused on conformance with IEEE Std 603 requirements, as documented in DCD Tier 1, Section 2.2.15.

The following paragraphs provide examples of staff concerns that were resolved through the RAI process.

<u>RAI 14.3-251</u>: In DCD Tier 1, Revision 4, Section 2.2.3 it stated that the FWCS is nonsafety-related and the FWCS is a triple-redundant, fault-tolerant digital controller. The staff asked the applicant to discuss the mode(s) that can control that function and to include the mode(s) in DCD Tier 1, Table 2.2.3-3. To address the staff's concerns, the applicant clarified the intent of DCD Tier 1, Table 2.2.3-1, and in Revision 5 of the DCD, the applicant changed the title of Table 2.2.3-1 to "FWCS Functional Arrangement." The staff finds this change to the table to be an acceptable and satisfactory response to the RAI. Therefore, RAI 14.3-251 is resolved.

<u>RAI 14.3-252</u>: In RAI 14.3-252 regarding DCD Tier 1, Revision 4, Table 2.2.3-1, the staff requested clarification on whether the applicant is taking credit for the triple redundant characteristic in the accident analysis. If so, then an adequate design description should be provided in the DCD Tier 1 information. To address the staff's concern, the applicant clarified the intent of the triple redundant characteristic, as requested. Therefore, RAI 14.3-252 is resolved.

<u>RAI 14.3-247</u>: The staff requested that the electrical separation criterion discussed in the Tier 1 information and the associated ITAAC be specific. In Revision 5 of the DCD, the applicant updated the Tier 1 information and the associated ITAAC to specify that the electrical separation criterion complies with the separation requirements in RG 1.75, Revision 3, "Criteria for Independence of Electrical Safety Systems," issued February 2005. The staff finds the response acceptable. Therefore, RAI 14.3-247 is resolved.

<u>RAI 14.3-260</u>: The staff requested the applicant to include the requirement for the controllers in the steam bypass and pressure control (SB&PC) system to be fault tolerant in the design description and associated ITAAC provided in DCD Tier 1. In DCD Tier 1, Revision 5, Section 2.2.9 the applicant included a design commitment for the SB&PC controllers to be fault tolerant in the design description and associated ITAAC table. The staff considers the response to be satisfactory. Therefore, RAI 14.3-260 is resolved.

<u>RAI 14.3-265</u>: The staff requested that all IEEE Std 603 requirements and the method of compliance be addressed in DCD Tier 2, and the ITAAC to verify these compliances should be documented in DCD Tier 1. In Revision 6 of the DCD, the applicant rewrote Section 2.2.15 entirely in response to the concerns raised in RAI 14.3-265. Based on the staff's review of DCD Tier 1, Section 2.2.15, RAI 14.3-265 is resolved.

The following provides a summary of the staff's evaluation of each portion of DCD Tier 1, Section 2.2:

(1) In DCD Tier 1, Section 2.2.1, the applicant provided design-basis information, including associated tables, in accordance with the selection criteria and methodology for DCD Tier 1 information as described in DCD Tier 2, Section 14.3 to support the ITAAC for the ESBWR RC&IS. DCD Tier1, Section 2.2.1 includes the following:

- Table of functional arrangement
- Table of major functional groups
- Table of automatic functions, initiators, and associated interfacing systems
- Table of rod block functions
- Table of controls, interlocks, and bypasses

As an example of the staff's review, the staff requested the following in RAI 14.3-348:

For ITAAC Item 3 in Table 2.2.1-6, the staff requested that the applicant modify the inspections, tests, and analyses and acceptance criteria to include verifications that the associated interfacing systems specified in DCD Tier 1, Table 2.2.1-3 are functional, based on other ITAAC, and that the list of interfacing systems is complete. The applicant's initial change only addressed the conformance of the RC&IS with regard to automatic initiators, functions, and associated interfacing systems and tests and type tests used to generate simulated signals from all interfacing systems. The staff did not find this initial change acceptable, because the simulated signals from the interfacing systems were not shown to generate the stated RC&IS functions in Table 2.2.1-3. In Revision 5 of the DCD, the applicant updated the inspections, tests, and analyses to specify tests and type tests that generate simulated signals for the initiators to perform the automatic functions of the RC&IS listed in DCD Tier 1, Table 2.2.1-3. The staff finds this change acceptable and the issue is resolved.

In summary, the staff reviewed the information in DCD Tier 1, Section 2.2.1 for consistency with the information provided in DCD Tier 2, Sections 7.7.2.2.5, 7.7.2.2.6, 7.7.2.2.7, 7.7.2.3, 7.7.2.4 and 7.7.2.5. The staff finds that the design description provided in DCD Tier 1, Section 2.2.1 and the associated ITAAC specified in DCD Tier 1, Table 2.2.1-6 are sufficient to verify the design of the RC&IS.

(2) In DCD Tier 1, Section 2.2.2, the applicant provided design-basis information, including associated tables, in accordance with the selection criteria and methodology for DCD Tier 1 information as described in DCD Tier 2, Section 14.3 to support the ITAAC for the ESBWR CRD system. DCD Tier 1, Section 2.2.2 includes the following:

- Table of functional arrangement
- Table of CRD maximum allowable scram times
- Table of automatic functions, initiators, and associated interfacing systems
- Table of controls and interlocks
- Tables of mechanical and electrical equipment, including design bases

The staff reviewed the information provided in DCD Tier 1, Section 2.2.2 for consistency with the information provided in DCD Tier 2, Sections 3.9.4, 7.7.2.2, and 7.8.1. The staff finds that the design description provided in Section 2.2.2 and the associated ITAAC specified in Table 2.2.2-7 are sufficient to verify the design of the CRD system.

(3) In DCD Tier 1, Section 2.2.3, the applicant provided design-basis information including associated tables, in accordance with the selection criteria and methodology for DCD

Tier 1 information, as described in DCD Tier 2, Section 14.3, to support the ITAAC for the ESBWR FWCS. DCD Tier 1, Section 2.2.3 includes the following:

- Table of functional arrangement
- Table of automatic functions, initiators, and associated interfacing systems
- Table of FWCS controls

The staff reviewed the information provided in DCD Tier 1, Section 2.2.3 for consistency with the information provided in DCD Tier 2, Section 7.7.3. The staff finds that the design description provided in DCD Tier 1, Section 2.2.3 and the associated ITAAC specified in DCD Tier 1, Table 2.2.3-4 are sufficient to verify the design of the FWCS.

(4) In DCD Tier 1, Section 2.2.4, the applicant provided design-basis information including associated tables, in accordance with the selection criteria and methodology for DCD Tier 1 information as described in DCD Tier 2, Section 14.3 to support the ITAAC for the ESBWR SLC system. DCD Tier 1, Section 2.2.4 includes the following:

- Table of SLC system automatic functions, initiators, and associated interfacing systems

- Table of SLC system controls and interlocks

- Tables of SLC system mechanical and electrical equipment including design bases

The staff reviewed the information provided in DCD Tier 1, Section 2.2.4 for consistency with the information provided in DCD Tier 2, Sections 7.4.1 and 7.8.1. The staff finds that the design description provided in Section 2.2.4 and the associated ITAAC specified in Table 2.2.4-6 are sufficient to verify the design of the SLC system.

(5) In DCD Tier 1, Section 2.2.5, the applicant provided design-basis information including associated tables, in accordance with the selection criteria and methodology for DCD Tier 1 information as described in DCD Tier 2, Section 14.3 to support the ITAAC for the ESBWR NMS. DCD Tier 1, Section 2.2.5 includes the following:

- Table of NMS functional arrangements
- Table of NMS functions, initiators, and associated interfacing systems
- Table of NMS controls, interlocks, and bypasses

The staff reviewed the information provided in DCD Tier 1, Section 2.2.5 for consistency with the information provided in DCD Tier 2, Section 7.2.2. The staff finds that the design description provided in DCD Tier 1, Section 2.2.5 and the associated ITAAC specified in DCD Tier 1, Table 2.2.5-4 are sufficient to verify the design of the NMS.

(6) In DCD Tier 1, Section 2.2.6 the applicant provided design-basis information including associated tables, in accordance with the selection criteria and methodology for DCD Tier 1 information as described in DCD Tier 2, Section 14.3, to support the ITAAC for the ESBWR RSS. DCD Tier 1, Section 2.2.6 includes the following:

- Table of RSS functional arrangement
- Table of RSS controls

The staff reviewed the information provided in DCD Tier 1, Section 2.2.6, for consistency with the information provided in DCD Tier 2, Section 7.4.2. The staff finds that the design description provided in DCD Tier 1, Section 2.2.6 and the associated ITAAC specified in DCD Tier 1, Table 2.2.6-3 are sufficient to verify the design of the RSS.

(7) In DCD Tier 1, Section 2.2.7 the applicant provided design-basis information including associated tables, in accordance with the selection criteria and methodology for DCD Tier 1 information as described in DCD Tier 2, Section 14.3, to support the ITAAC for the ESBWR RPS. DCD Tier 1, Section 2.2.7 includes the following:

- Table of RPS functional arrangement
- Table of RPS automatic functions, initiators, and associated interfacing systems
- Table of RPS controls, interlocks (system interfaces), and bypasses

In DCD Tier 1, Revision 4, Figure 2.2.7-1 was removed. The staff review guidance in SRP Section 14.3 states the following:

> The amount of design information is proportional to the safety-significance of the structures and systems of the design. The level of detail in DCD Tier 1 is governed by a graded approach to the SSCs of the design, based on the safety significance of the functions they perform. The design descriptions include the figures associated with the systems.

The staff's explained this guidance in RAI 14.3-259 and the staff asked the applicant to include a figure depicting the RPS basic configuration block diagram and associated information necessary to verify the functional arrangement of the RPS. RAI 14.3-259 was being tracked as an open item in the SER with open items.

In response to RAI 14.3-259 S01, the applicant stated that it had determined that detailed information, such as in DCD Tier 1, Revision 3, Figure 2.2.7-1, is not appropriate for Tier 1 content and rulemaking based on NRC guidance in NUREG–0800, Section 14.3, regarding the items that are subject to change. After further review, the staff agrees with the applicant's determination. Therefore, RAI 14.3-259 and the associated open item are resolved.

In DCD Tier 2, Revision 5, the applicant modified Section 7.2.1.3.1, to include an anticipatory reactor trip to comply with the requirement of 10 CFR 50.34(f)(2)(xxiii)[II.K.2.10], which states that the reactor will trip in response to the loss of main feedwater. In the ESBWR, this feature is designed as an anticipatory trip actuated upon the loss of power to two of the four main feedwater pumps. This design feature was not included in DCD Tier 1, Section 2.2.7 or in Table 2.2.7-2. In RAI 14.3-403, the staff requested the applicant to include this anticipatory trip in the design description and ITAAC for the RPS. Therefore, RAI 14.3-403 was being tracked as an open item in the SER with open items.

In DCD Tier 1, Revision 6, Section 2.2-7 and Table 2.2.7-2 were revised to add the phrase "Loss of all feedwater event" in parentheses after the "Power Generation Bus Loss" scram initiator. The loss of feedwater flow event is detected by the loss of the power generation bus. This revision clarifies that the loss of all feedwater event is the same as the power generation bus loss scram initiator. DCD Tier 2, Sections 7.2.1.2.4.2, 7.2.1.3, 7.2.1.5.4, 7.3.5.3.1, and 7.4.4.3.1 document this clarification. The staff considers this issue

resolved. The staff finds that the design description provided in Section 2.2.7 and the associated ITAAC specified in Table 2.2.7-4 are sufficient to verify the design of the RPS.

(8) In DCD Tier 1, Section 2.2.8, the applicant did not provide a design description and associated ITAAC. The staff finds this acceptable because no credit is taken for the PAS in the safety analyses nor does failure of the system affect any safety function. The expected design basis accidents analyzed in Chapter 15 envelope the failure modes associated with the PAS digital controls.

(9) In DCD Tier 1, Section 2.2.9 the applicant provided design-basis information, including associated tables, in accordance with the selection criteria and methodology for DCD Tier 1 information as described in DCD Tier 2, Section 14.3, to support the ITAAC for the ESBWR SB&PC system. DCD Tier 1, Section 2.2.9 includes the following:

- Table of SB&PC functional arrangement
- Table of SB&PC functions and initiating conditions

The staff reviewed the information provided in DCD Tier 1, Section 2.2.9 for consistency with the information provided in DCD Tier 2, Section 7.7.5. The staff finds that the design description provided in DCD Tier 1, Section 2.2.9 and the associated ITAAC specified in DCD Tier 1,Table 2.2.9-3 are sufficient to verify the design of the SB&PC system.

(10) In DCD Tier 1, Section 2.2.10 the Q-DCIS is the designation given to the collection of hardware and software that comprises the safety-related portion of the following systems and the associated ITAAC specified in the corresponding DCD Tier 1 sections:

- Platform for the RTIF/NMS

- Platform for the SSLC/ESFs

- Independent Control Platform for Vacuum Breaker Isolation Function, ATWS/SLC, and HP CRD Isolation Bypass Function

In its response to RAI 14.3-241, the applicant included a crosswalk to connect the DCD Tier 1 system I&C ITAAC with the software development program described in DCD Tier 1, Section 3.2. With this cross-reference, the staff finds that the design description in DCD Tier 1, Section 2.2.10 and the associated ITAAC are sufficient to verify the Q-DCIS design.

(11) In DCD Tier 1, Section 2.2.11, the N-DCIS is the designation given to the collection of hardware and software that comprises the non-safety-related I&C of the following systems and the associated ITAAC specified in the corresponding DCD Tier 1 sections:

- N-DCIS Network Segment for diverse protection system (DPS)

- N-DCIS Network Segment of plant investment protection (PIP) A and PIP B for FAPCS and supporting systems

- N-DCIS Network Segment of PIP A and PIP B for reactor water cleanup Suction Backup Isolation

In response to RAI 14.3-241, the applicant included a crosswalk to connect the DCD Tier 1 system I&C ITAAC with the software development program described in DCD Tier 1, Section 3.2. With this cross-reference, the staff finds that the design description in DCD Tier 1, Section 2.2.11 and the associated ITAAC are sufficient to verify the N-DCIS design.

(12) In DCD Tier 1, Section 2.2.12, the applicant provided design-basis information, including associated tables, in accordance with the selection criteria and methodology for DCD Tier 1 information as described in DCD Tier 2, Section 14.3 to support the ITAAC for the ESBWR LD&IS. DCD Tier 1, Section 2.2.12, includes the following:

- Table of LD&IS isolation function monitored variables
- Table of LD&IS leakage source monitored variables
- Table of LD&IS controls, interlocks, and bypasses

The staff reviewed the information provided in DCD Tier 1, Section 2.2.12, for consistency with the information provided in DCD Tier 2, Section 7.3.3. The staff finds that the design description provided in DCD Tier 1, Section 2.2.12, and the associated ITAAC specified in DCD Tier 1, Table 2.2.12-5 are sufficient to verify the design of the LD&IS.

(13) In DCD Tier 1, Section 2.2.13, the applicant provided design-basis information including associated tables, in accordance with the selection criteria and methodology for DCD Tier 1 information as described in DCD Tier 2, Section 14.3 to support the ITAAC for the ESBWR SSLC/EFS. DCD Tier 1, Section 2.2.13 includes the following:

- Table of SSLC/ESF functional arrangement
- Table of SSLC/ESF automatic functions, initiators, and associated interfacing systems
- Table of SSLC/ESF controls, interlocks, and bypasses

The staff reviewed the information provided in DCD Tier 1, Section 2.2.13 for consistency with the information provided in DCD Tier 2, Section 7.3.5. The staff finds that the design description provided in DCD Tier 1, Section 2.2.13 and the associated ITAAC specified in DCD Tier 1, Table 2.2.13-4 are sufficient to verify the design of the SSLC/ESF.

(14) In DCD Tier 1, Section 2.2.14 the applicant provided design-basis information, including associated tables, in accordance with selection criteria and methodology for DCD Tier 1 information as described in DCD Tier 2, Section 14.3 to support the ITAAC for the ESBWR diverse instrumentation and control systems (DICS). DCD Tier 1, Section 2.2.14 includes the following:

- Table of DICS functional arrangement
- Table of DICS functions, initiators, and associated interfacing systems
- Table of DICS controls, interlocks, and bypasses

As a result of its review, the staff noted a concern with the last line in Table 2.2.14-1 that stated, "DPS [diverse protection system] uses hardware and software that is separate and independent from that used by the RPS and SSLC/ESF." The staff stated that it believed that the hardware and software should be diverse in addition to being separate and independent from the RPS and SSLC/ESF. DCD Tier 1, Section 2.2.14, Table 2.2.14-1 did not document the "diverse" design feature. In RAI 14.3-404, the request that the applicant identify the design requirement of diversity in Table 2.2.14-1, RAI 14.3-404 was being tracked as an open item in the SER with open items. In DCD Tier 1, Revision 6,

Section 2.2.14, Item (18) documents that the DPS network segment uses hardware and software diverse from that used by the RPS and SSLC/ESF. The staff finds the clarification acceptable. Therefore, RAI 14.3-404 and the associated open item are resolved.

The staff reviewed the information provided in DCD Tier 1, Section 2.2.14 for consistency with information provided in DCD Tier 2, Section 7.8. The staff finds that the design description provided in DCD Tier 1, Section 2.2.14 and the associated ITAAC specified in DCD Tier 1, Table 2.2.14-4 are sufficient to verify the design of the ESBWR DCIS.

(15) In DCD Tier 1, Section 2.2.15 the applicant provided design-basis information, including associated tables, in accordance with selection criteria and methodology for DCD Tier 1 information as described in DCD Tier 2, Section 14.3 to support the ITAAC for ESBWR I&C compliance with IEEE Std 603. Section 7.1.1.3.10 of this report addresses the detailed evaluation of the compliance of the ESBWR I&C design with IEEE Std 603.

The staff followed the guidance provided in SRP Chapter 7, Appendix 7.1-C and Appendix 7.1-D to verify that the ESBWR I&C design has addressed all of the criteria listed in IEEE Std 603, as required by 10 CFR 50.55a(h). Because the I&C design has not yet been completed, the applicant is unable to demonstrate conformance to the IEEE Std 603 criteria. The applicant provided the DAC in the ITAAC specified for the systems referenced in DCD Tier 1, Section 2.2.15. The staff reviewed DCD Tier 1, Section 2.2.15, design completion commitments documented as DAC, and determined that the DAC and ITAAC specified in DCD Tier 1, Table 2.2.15-2 are sufficient to verify conformance to the IEEE Std 603 criteria when the design is complete.

(16) In DCD Tier 1, Section 2.2.16 the applicant provided design-basis information, including associated tables, in accordance with selection criteria and methodology for DCD Tier 1 information as described in DCD Tier 2, Section 14.3, to support the ITAAC for the ESBWR HP CRD isolation bypass function (IBF). DCD Tier 1, Section 2.2.16 includes the following:

- Table of HP CRD IBF functional arrangement
- Table of HP CRD IBF automatic functions, initiators, and associated interfacing systems
- Table of controls, interlocks, and bypasses

The staff reviewed the information provided in DCD Tier 1, Section 2.2.16 for consistency with the information provided in DCD Tier 2, Section 7.4.5. The staff finds that the design description provided in DCD Tier 1, Section 2.2.16 and the associated ITAAC specified in DCD Tier 1, Table 2.2.16-4 are sufficient to verify the design of the HP CRD IBF.

(17) In DCD Tier 1, Section 3.2 the applicant provided design-basis information including associated tables, in accordance with the selection criteria and methodology for DCD Tier 1 information as described in DCD Tier 2, Section 14.3 to support the ITAAC for the ESBWR I&C software development. Section 7.1.2 of this report addresses the detailed evaluation of ESBWR I&C software development. The staff follows the guidance provided in SRP Chapter 7, Branch Technical Position 7-14, “Guidance on Software Reviews for Digital Computer-based Instrumentation and Control Systems.” Because the I&C design

and the associated software development have not yet been completed, the applicant is unable to demonstrate the detailed life cycle design process. The applicant provided the DAC in the ITAAC specified for the systems referenced in DCD Tier 1, Section 3.2, Table 3.2-1. The staff reviewed the DCD Tier 1, Section 3.2 design completion commitments documented as the DAC and ITAAC specified in DCD Tier 1, Table 3.2-1 and finds that the specified DAC are sufficient to verify conformance with design requirements and the SRP review guidance when the design is complete.

(18) In DCD Tier 1, Section 2.15.7, the applicant provided design-basis information, including associated tables, in accordance with the selection criteria and methodology for DCD Tier 1 information as described in DCD Tier 2, Section 14.3 to support the ITAAC for the ESBWR CMS. DCD Tier 1, Section 2.15.7 includes a table of CMS electrical equipment design bases.

The staff reviewed the information provided in DCD Tier 1, Section 2.15.7 for consistency with the information provided in DCD Tier 2, Sections 7.3, 7.7 and 7.8. The staff finds that the design description provided in DCD Tier 1, Section 2.15.7 and the associated ITAAC specified in DCD Tier 1, Table 2.15.7-2 are sufficient to verify the design of the CMS.

(19) In DCD Tier 1, Section 2.3.1 the applicant provided design-basis information, including associated tables, in accordance with the selection criteria and methodology for DCD Tier 1 information as described in DCD Tier 2, Section 14.3 to support the ITAAC for the ESBWR process radiation monitoring system (PRMS). DCD Tier 1, Section 2.3.1 includes a table and figure of the PRMS functional arrangement.

The staff reviewed the information provided in DCD Tier 1, Section 2.3.1 for consistency with the DCD information provided in Section 7.5. The staff finds that the design description provided in DCD Tier 1, Section 2.3.1 and the associated ITAAC specified in DCD Tier 1, Table 2.3.1 are sufficient to verify the design of the PRMS.

(20) In DCD Tier 1, Section 2.3.2, the applicant provided design-basis information, including associated tables, in accordance with the selection criteria and methodology for DCD Tier 1 information as described in DCD Tier 2, Section 14.3, to support the ITAAC for the ESBWR ARM system. DCD Tier 1, Section 2.3.2 includes a table of ARM system locations.

The staff reviewed the information provided in DCD Tier 1, Section 2.3.2 for consistency with the DCD information provided in DCD Tier 2, Section 7.5. The staff finds that the design description provided in DCD Tier 1, Section 2.3.2 and the associated ITAAC specified in DCD Tier 1, Table 2.3.2 are sufficient to verify the design of the ARMS.

Based on the staff's review as set forth above, as well as on the applicant's application of the selection methodology and criteria for the development of DCD Tier 1 information in Section 14.3 of DCD Tier 2, the staff concludes that the top-level design features and performance characteristics of the SSCs are appropriately described in DCD Tier 1, Revision 9, and the DCD Tier 1, Revision 9, information associated with the scope of SRP Section 14.3.5 is acceptable.

Furthermore, the staff concludes that the DCD Tier 1, Revision 9, design descriptions within the scope of SRP Section 14.3.5 can be verified adequately by the ITAAC. Therefore, the staff concludes that the ESBWR ITAAC within the scope of SRP Section 14.3.5 are necessary and

sufficient to assure that with respect to these ITAAC, if the inspections, tests, and analyses are performed and the acceptance criteria are met, then a facility referencing the certified ESBWR design has been constructed and will be operated in compliance with the design certification and applicable regulations.

14.3.6 Electrical Systems

The applicant provided design-basis information, including associated tables and figures, in accordance with the selection criteria and methodology for developing DCD Tier 1 information, as described in DCD Tier 2, Section 14.3 to support the ITAAC for ESBWR SSCs. The applicant organized the DCD Tier 1 information in the systems, structures, and topical areas format shown in the DCD Tier 1, Table of Contents. The staff reviewed the DCD Tier 1 information provided by the applicant, in accordance with the review matrix provided in Appendix 14.3A of this report and SRP Section 14.3.6.

The staff's review generated a number of RAIs which the applicant resolved satisfactorily. The following paragraphs provide examples of several RAIs which have been resolved.

RAI 14.3-129: In RAI 14.3-129, the staff requested that the applicant add design commitments and ITAAC to address the seismic design of the mounting of the components of the four safety-related divisions of the direct current (dc) systems by including the following recommended wording: "Design Commitment - The mounting of the components of the four safety-related divisions of the dc system (batteries, battery chargers, inverters, buses, etc.) conform to seismic Category I requirements; Inspections, Tests and Analyses - An inspection will be performed of the mounting of the components of the four safety-related divisions of the direct current system (batteries, battery chargers, inverters, buses, etc.) to verify that the installed equipment including anchorage is seismically bounded by the tested and/or analyzed condition; acceptance criteria - A report exists and concludes that the as-installed equipment including anchorage is seismically bounded by the tested and/or analyzed conditions." The applicant added a new ITAAC item to address seismic requirements for mounting. The staff finds the applicant's response to be acceptable. Therefore, RAI 14.3-129 is resolved.

RAI 14.3-376: In RAI 14.3-376, the staff requested that the applicant modify the acceptance criteria for ITAAC Item 1 in DCD Tier 1, Table 2.13.3-3 to be consistent with the design certification, either by adding conformance to DCD Tier 1, Table 2.13.3-1 in the acceptance criteria or by revising Table 2.13.3-1 to Section 2.13.3 in the design certification. The applicant modified the acceptance criteria to be consistent with the design certification by adding conformance to DCD Tier 1, Table 2.13.3-1 in the acceptance criteria. The staff finds the applicant's response to be acceptable. Therefore, RAI 14.3-376 is resolved.

RAI 9.5-60 S02: In response to RAI 9.5-60 S01, GEH stated that emergency lighting in the remote shutdown area is fed from the safety-related uninterruptible power supply (UPS) for 72 hours, similar to the power supply arrangement for the MCR emergency lighting. As a result, the staff responded that the ITAAC for the lighting power supply (Section 2.13.8) should be revised to indicate that emergency lighting in the RSS is fed from the safety-related UPS for 72 hours. Specifically, ITAAC Table 2.13.8-1 Items 1 thru 4 should be modified to include the RSS emergency lighting; the design description of Section 2.13.8 should be modified to indicate control room and RSS emergency lighting; and an ITAAC item for electrical isolation between the safety-related power supply and the nonsafety-related emergency lighting in the MCR and RSS should be provided. In response, GEH stated that the emergency lighting in the RSS and MCR is fed from the safety-related UPS. As a result, the applicant committed to the following:

ITAAC for the lighting power supply (DCD Tier 1, Section 2.13.8) would be revised to state the source of emergency lighting power as safety-related UPS; the design description of Section 2.13.8 and the description of ITAAC Table 2.13.8 Items 1 thru 4 would be updated to include MCR and RSS emergency lighting; and, a new item number 6 would be added in ITAAC DCD Tier 1, Table 2.13.8-1 to state that the electrical isolation between the nonsafety-related control room and the RSS emergency lighting circuits from the safety-related UPS is accomplished by using two isolation devices in series. The staff finds that DCD Tier 1, Revision 5, Section 2.13.8 was revised in accordance with the GEH response to RAI 9.5-60 S02. Therefore, RAI 9.5-60 S02 is resolved.

RAI 14.3-206: The ITAAC for the EQ of Mechanical and Electrical Equipment in DCD Tier 1, Section 3.8 includes safety-related mechanical, electrical and digital I&C equipment. In DCD Tier 2, Section 3.11, GEH stated that electrical equipment within the scope of this section includes all three categories of 10 CFR 50.49(b). Staff review determined the ITAAC did not include 10 CFR 50.49(b)(2) and (b)(3) equipment. As a result, the staff requested in RAI 14.3-206 that the applicant include 10 CFR 50.49(b)(2) and (b)(3) equipment in the ITAAC or provide justification for not including that equipment in the ITAAC. In response the applicant stated that DCD Tier 1, Section 3.8 is consistent with SRP Section 14.3 and included safety-related equipment in harsh environments and digital I&C. The staff found that this response to RAI 14.3-206 was not adequate and requested the applicant to ensure that DCD Tier 1, Section 3.8 includes: (1) safety-related electrical equipment, (2) safety-related mechanical equipment, and (3) safety-related digital I&C equipment governed by EQ requirements in 10 CFR 50.49(b)(1), (b)(2), and (b)(3). SRP Section 14.3.6 provides guidance on the EQ of electrical equipment important to safety and states that applicants must ensure that safety-related, certain nonsafety-related, and certain post-accident monitoring equipment can perform their functions in various anticipated environments. The applicant subsequently provided Revision 5 of the ESBWR DCD. Based upon the staff's review of DCD Tier 1, Revision 5, Section 3.8 the staff finds that the applicant had revised Section 3.8 accordingly, and the RAI is closed.

RAI 14.3-345: In RAI 14.3-345, the staff requested the applicant to clarify ITAAC Item 21 in Table 2.2.15-2 to indicate that: each mechanical/electrical division for the systems listed in Table 2.2.15-1 receives power from safety-related power supplies in the same division, and that the means for verification should be tests of each mechanical/electrical division one at a time, along with inspections to verify that the electrical one-line diagrams indicate the correct power sources. The staff also requested that the acceptance criteria for ITAAC Item 5 in Table 2.1.2-3 be revised to indicate that the required reports exist. In addition, the staff requested that ITAAC Item 6b in Table 2.1.2-3 be revised to verify that both physical and electrical independence are provided between the divisions of the NBS and other mechanical systems and between the safety-related equipment of the NBS system and nonsafety-related equipment. Therefore, RAI 14.3-345 was being tracked as an open item in the SER with open items.

The applicant's response provided a DAC and an ITAAC that performed a function similar to that of the original ITAAC. ITAAC Item 21 in Table 2.2.15-2 and the new DAC and ITAAC became Items 22a and 22b in Table 2.2-15-2, respectively. The subject matter of both items is I&C "software projects" (a specific type of microprocessor-based digital architecture unique to one vendor) and not each I&C system. The staff deemed DAC Item 22a acceptable because the Licensee will verify by the review of a design phase summary baseline review record that the vendor had incorporated into design of the software projects the capability to supply the electrical components of each division of the software projects by separate power supplies, and that design aspect was to be further verified by the implementation of ITAAC 22b. The applicant's response also indicated that ITAAC Item 22b was added for which actual tests will

be performed during the installation phase on the as-built software project's electrical components, by providing test signals in one safety-related division at a time to verify that the components receive power from their respective, divisional, safety-related power supplies. The staff accepts the applicant's response for this revised ITAAC because the test provides a direct and visible means for verifying the design capability stated in the Design Commitment of this ITAAC that only the electrical components connected to a power supply in the same division receive the designated test signal, when it is applied to each division of the software projects one at a time. RAI 14.3-345 and the associated open item are resolved.

RAI 14.3-379: In RAI 14.3-379, the staff requested that the applicant clarify the inspections, tests, and analyses and acceptance criteria because there was no clear correlation between the subject matter of the design commitment, which was concerned with the sources of electrical power for the safety-related components listed in DCD Tier 1, Table 2.15.1-1, and the oblique references provided in the inspections, tests, and analyses and the acceptance criteria to just "Tier 1, Section 2.13" of the DCD. The references in the inspections, tests, and analyses and acceptance criteria did not indicate the actions to be taken or the conditions to be met in order to implement or perform this ITAAC. Originally, this ITAAC was Item 6b in DCD Tier 1, Table 2.15.1-2 instead of item 6a. Therefore, RAI 14.3-379 was being tracked as an open item in the SER with open items.

In response, the applicant changed the following: (1) Item 6b became Item 6a, (2) Table 2.15.1-1 was divided into different sections, (3) the design commitment now referred to the "safety-related components associated with actuation and status monitoring of the final control elements of the Containment System components listed in DCD Tier 1, Table 2.15.1-1," (4) the inspections, tests, and analyses was modified to require that tests be performed by providing a test signal in only one safety-related division at a time, and (5) the acceptance criteria stated that test reports indicate that the test signal exists only in the single, safety-related division (or at the equipment powered by the safety-related division) under test. The staff requested GEH to revise the design commitment, inspections, tests, and analyses, and acceptance criteria to refer to 'electrical safety-related components" instead of just "safety-related components." The intent of the ITAAC remained the same. In its response, the applicant made the requested change. The staff agrees with the applicant's response because it correlates to the design commitment with the ITAAC and identifies the actions and conditions needed to implement the ITAAC. Therefore, RAI 14.3-379 and the associated open item are resolved.

RAI 14.3-407: In RAI 14.3-407, the staff asked the applicant to include containment electrical penetrations in Tables 2.15.1-1. In response, the applicant stated that the purpose of DCD Tier 1, Tables 2.15.1-1a, 1b, and 1c is to list the containment isolation valves and summarize their functions and positions. The containment electrical penetrations are not operated and do not isolate or reposition on a containment isolation signal. Their containment isolation function is to passively maintain pressure boundary. Because of this, the requirements apply equally to all of the containment electrical penetration assemblies, and they can be addressed on a generic basis. DCD Tier 2, Table 8.1-1 summarizes the applicable design criteria for the design of ESBWR electrical systems. This table indicates that RG-1.63 and IEEE Std 317 are applicable to the ESBWR design. IEEE Std 317, among other things, requires the mechanical design, materials, fabrication, examinations, and testing of the pressure boundary of the electrical penetration assembly to be in accordance with the requirements of ASME Boiler and Pressure Vessel Code Division 1, Section III, Subsection NE for Class MC Components. The applicant stated that a new ITAAC will be added to DCD Tier 1, Section 2.15.1 and Table 2.15.1-2 to verify the ASME pressure boundary and seismic Category I requirements as they apply to the containment electrical penetration assemblies. The staff finds the applicant's response

acceptable, and therefore, RAI 14.3-407 is resolved. The staff confirmed that DCD Revision 6, incorporated the changes as discussed above.

RAI 14.3-408: In RAI 14.3-408, the staff asked the applicant to include the following items in DCD Tier 1, Table 2.13.1-1 or provide justification for not including them: (1) breaker to regulating transformer and relay (degraded voltage and under-frequency) compartments, and (2) ancillary diesel buses. Therefore, RAI 14.3-408 was being tracked as an open item in the SER with open items.

In response, the applicant stated that DCD Tier 1, Table 2.13.1-1 will be revised to add the isolation power centers' protective relaying and isolation power center breakers to the ancillary diesel buses. The applicant further stated that the regulating transformer and the breaker to the regulating transformer are deleted in response to RAI 8.2-14 S01. The staff finds the applicant's response to RAI 8.2-14 S01 acceptable because the applicant made a design change to delete the regulating transformers, thus eliminating the potential for disruptive voltages and frequencies to reach the safety-related loads. The staff's evaluation of the response to RAI 8.2-14 S01 is discussed in Section 8.3.1.3 of this report. Based on the above and the staff's evaluation in Section 8.3.1.3 of this report, RAI 14.3-408 is resolved. The staff confirmed that DCD, Revision 6, incorporated the changes discussed above.

RAI 14.3-409: In RAI 14.3-409, the staff requested the applicant to update Sheet 2 of Figure 2.13.1-1 to correct the following: (a) the ancillary diesel bus is missing, (b) 480V buses do not include all loads (e.g., UPS rectifiers, regulating transformers, etc.), (c) PIP bus A feeds to Isolation Power Center Bus A alternate feed is incorrect, and (d) PIP bus B feeds to Isolation Power Center Bus D alternate feed is incorrect. Therefore, RAI 14.3-409 was being tracked as an open item in the SER with open items.

In response, the applicant stated that all the items except the regulating transformers are included in Figure 2.13.1-1 Sheet. 2. The applicant deleted the regulating transformers in response to RAI 8.2-14 S01. The staff finds the applicant's response acceptable, and therefore, RAI 14.3-409 is resolved. The staff confirmed that DCD, Revision 6, incorporated the changes as discussed above.

RAI 14.3-410: In RAI 14.3-410, the staff requested the applicant to include the ITAAC for the following in DCD Tier 1, Table 2.13.4-2 or provide justification for not including them: (1) verification of automatic load sequencing, (2) verification that controls exist in the MCR to start and stop each standby diesel generator (SDG), and (3) verification that the ancillary DGs and associated auxiliaries (e.g., controls, electrical buses, fuel tanks, etc.) are seismic Category II. Therefore, RAI 14.3-410 was being tracked as an open item in the SER with open items.

In response, the applicant stated that DCD Tier 1, Section 2.13.4 and ITAAC Item 2.a of Table 2.13.4-2 would be modified to include the verification of SDG load sequencing. Additionally, GEH would add ITAAC to verify the existence of control in the MCR to start and stop each SDG and to verify that each ancillary DG and associated auxiliaries, buses, fuel tanks, and fuel transfer pumps are seismic Category II. The applicant provided a revised DCD Tier 1, Section 2.13.4. The staff determined that the response was inadequate because DCD Tier 1, Table 2.13.4-2, Item 2.a did not include testing of automatic load sequencer and load stepping intervals.

In RAI 14.3-410 S1, the staff asked the applicant to add the following language under "Inspections, Tests, Analyses": "An actual or simulated signal is initiated to start the load

sequencer operation. Output signals will be monitored to determine the operability of the load sequencer. Time measurements are taken to determine the load stepping intervals." Additionally, the staff asked the applicant to add the following language under "Acceptance Criteria": "The load sequencer initiates a closure signal with ±5 seconds of the set intervals to connect the load." In response, the applicant stated that sequencing of the nonsafety-related SDG-backed PIP buses will be controlled by N-DCIS logic. Upon receiving a DG ready-to-load signal, auto loading would be initiated by the auto load sequencing logic (N-DCIS) signaling each system load to close into the bus in its predetermined order, with feedback as a precondition to move to the next load. This feedback could consist of further ready-to-load signals based on DG and PIP bus voltage and frequency returning to desired levels. This logic for monitoring voltage and frequency and enabling the next load closure would also be within the N-DCIS. Therefore, signals from N-DCIS controllers to sequence loads onto the SDG would not be based solely on programmed time intervals, but instead would be based on the DG being ready to accept the next load before signaling. Once N-DCIS logic allows the closure of the next predetermined load, the only delay in sending the closure signal would be that of the N-DCIS response time, which is expected to be on the order of tens of milliseconds. Alarms will be generated if sequencing does not occur as expected. Sequencing of the ESBWR SDG need not follow the procedures typically applied to traditional safety-related emergency diesel generators, because the ESBWR design does not require ac power to achieve and maintain a safe shutdown for 72 hours. Therefore, the requested additions to the existing ITAAC to specifically test automatic load sequencers and load starting intervals are not necessary. The applicant will add a clarification to the inspections, tests, and analyses of DCD Tier 1, Revision 6, Section 2.13.4, ITAAC Item 2.a to state that subsequently generated signals will start load sequencing. The staff finds the applicant's response acceptable. Therefore, RAI 14.3-410 and the associated open item are resolved. The staff confirmed that DCD, Revision 6, incorporated the changes as discussed above.

RAI 14.3-411: In RAI 14.3-411, the staff requested that the applicant should make corrections to the ITAAC by including the following or provide justification for their exclusion: (1) CB and RB distribution panels are missing from DCD Tier 1, Table 2.13.5-1, (2) in Table 2.13.5-2, Item 6 should include maximum and minimum battery terminal voltages in the design commitment, and the associated acceptance criteria for Item 6 should specify the voltage and frequency tolerances,and (3) an item should be added for the regulating transformers, since the regulating transformer and other inverter will supply power in the case of one inverter problem. The staff also asked the applicant to include the synchronization scheme to be used for this case and add it as another ITAAC item. Therefore, RAI 14.3-411 was being tracked as an open item in the SER with open items.

In response, the applicant stated that CB and RB distribution panels should not be added to the table since the exact number and location of the distribution panels will not be finalized until completion of the final design. The safety-related loads are shown as "typical." The applicant stated that it will revise DCD Tier 1, Table 2.13.5-2, Item 6 to state that each safety-related inverter can supply its ac load at both minimum and maximum battery terminal voltages in the design commitment. GEH will revise the acceptance criteria to specify that the inverter will supply its rated load, while maintaining its rated voltage at its rated frequency, within tolerances acceptable for its ac loads. Additionally, the applicant stated that regulating transformers have been deleted and thus, ITAAC to address regulating transformers are not necessary. The staff finds the applicant's response acceptable; and therefore, RAI 14.3-411 and the associated open item are resolved. The staff confirmed that DCD, Revision 6, incorporated the changes as discussed above.

RAI 14.3-413: In response to RAI 8.2-14 regarding the effects of a voltage spike on the electrical distribution system components after a loss of the electrical grid during islanding, GEH stated that fast transients on the ac input to the UPS input rectifiers and battery chargers can result in high dc voltages and, if the rectifiers and inverter trips are not coordinated, subsequent inverter trips and the loss of power to safety-related loads can occur. Since trip coordination of battery chargers and UPS input rectifiers with inverters is critical for the proper operation of the UPS under excessive ac input voltage conditions during the islanding mode, an ITAAC is necessary to verify the trip coordination of safety-related battery chargers and UPS input rectifiers with inverters. The staff reviewed this RAI response and requested that the applicant provide an ITAAC to address the proper operation of the above devices. Therefore, RAI 14.3-413 was being tracked as an open item in the SER with open items.

In response, the applicant stated that it will revise DCD Tier 1, Section 2.13.5 and Table 2.13.5-2 to include the requirement to verify trip coordination of the safety-related battery chargers and UPS input rectifiers with the inverters. This new DCD Tier 1 ITAAC is based on new information to be added to DCD Tier 2, Section 8.3.1.1.3, which discusses coordination of the rectifier and inverter high dc voltage trips. The applicant stated that the safety-related battery chargers and UPS input rectifier high dc voltage trips are coordinated such that the associated inverters do not trip on high dc input voltage during voltage transients on the ac distribution system. The trips are coordinated such that the inverter high dc input voltage trip setpoint is greater than the associated battery charger and the UPS input rectifier high dc output trip setpoints. In addition, the time delay for the inverter high dc input voltage trip is greater than the time delay for the battery charger and the UPS input rectifier high dc output voltage trips. In this way, the high dc voltage protection is coordinated in both magnitude and time so the battery charger and UPS input rectifier always trip before their dc output voltage reaches the level that would cause an inverter trip on high dc input voltage. The actual trip magnitude and time margins are a function of the vendor-specific equipment design. The staff finds the applicant’s response acceptable because the operation of the protective devices will be coordinated and the UPS inverter will be available for its operation. Therefore, RAI 14.3-413 and the associated open item are resolved. The staff confirmed that DCD, Revision 6, incorporated the changes as discussed above.

RAI 14.3-394 S01: DCD Tier 1, Revision 4, Section 4 stated that an applicant for a COL that references the ESBWR certified design must provide design features or characteristics that comply with the interface requirements for the plant design and ITAAC for the site-specific portion of the facility design, in accordance with 10 CFR 52.79(c) (now 52.79(d)). However, the applicant identified no interface requirements for the offsite power system in the certified design. RG 1.206, CIII.7.2, “Site-Specific ITAAC,” recommends that applicants develop ITAAC for the site-specific systems that are designed to meet the significant interface requirements of the standard certified design; that is, the site-specific systems that are needed for operation of the plant (e.g., offsite power).

As indicated in DCD Tier 2, Section 8.1.5.2.4, the ESBWR standard design complies with the requirements of GDC 17, “Electric power systems,” with respect to two independent and separate offsite power sources. Therefore, an ITAAC to verify that the required circuits from the transmission network satisfy the requirements of GDC 17 is needed in regard to offsite power source capacity and capability, regardless of its low-risk significance in the ESBWR design. The staff requested that the applicant revise DCD Tier 1, Section 4 to include interface requirements for the offsite power system. COL Applicants should provide site-specific ITAAC for offsite power to satisfy the interface requirements. In response, GEH revised DCD Tier 1, Section 4 to include a new Section 4.2 that included requirements for the COL Applicant to develop an ITAAC to verify, by inspection, that two physically independent circuits will supply

electric power from the transmission network to the onsite electrical distribution system. However, GEH did not add an interface requirement demonstrating the capacity and capability of the offsite power system. In RAI 14.3-394 S01, the staff requested that GEH modify the DCD to add this interface requirement. Therefore, RAI 14.3-394 was being tracked as an open item in the SER with open items.

In response, the applicant stated that it had added new ITAAC for demonstrating the capacity and capability of the normal and alternate preferred power supplies to DCD Tier 1, Section 2.13.1. GEH also added interface requirements for demonstrating the capacity and capability of the site-specific portions of the normal and alternate preferred power supplies to DCD Tier 1, new Section 4.2. The interface requirements for offsite power system will include the following:

(1) At least two independent circuits supply power from the transmission network to the interface with the onsite portions of the preferred power supply (PPS).

(2) Each offsite circuit interfacing with the onsite portions of the PPS is adequately rated to supply the load requirements during design-basis operating modes.

(3) During steady-state operation, the offsite portions of the PPS are capable of supplying voltage at the interface with the onsite portions of the PPS that will support the operation of safety-related loads during design-basis operating modes.

(4) During steady-state operation, the offsite portion of the PPS is capable of supplying the required frequency at the interface with the onsite portions of the PPS that will support the operation of safety-related loads during design-basis operating modes.

(5) The fault current contribution of the offsite portion of the PPS is compatible with the interrupting capability of the onsite fault current interrupting devices.

Additional supporting information has been added to DCD Tier 2, Chapter 8. On the basis of its review, the staff finds that the interface requirements specified above will provide assurance that the offsite power system has adequate capacity and capability to satisfy GDC 17. Therefore, RAI 14.3-394 and the associated open item are resolved. The staff confirmed that DCD, Revision 6, incorporated the changes as discussed above.

RAI 14.3-424: In RAI 14.3-424, staff asked the applicant to include an ITAAC to address the fault current withstand capability of cables for: (a) onsite ac power, (b) dc power, (c) DG power, and (d) uninterruptible ac power. In response, the applicant stated that it addressed the cables for the applicable portions of the onsite ac power supply, specifically the preferred power supply in response to RAI 14.3-394 S01 and included the ITAAC in response to that RAI.

GEH stated that it will add ITAAC Item 12 to DCD Tier 1, Table 2.13.3-3 to address the fault current withstand capability of cables for the safety-related portions of the dc power supply system.

Portions of the onsite DG power supply systems capable of supporting the safety-related loads are covered by ITAAC provided for the onsite ac power supply. DCD Tier 1, Table 2.13.1-2, ITAAC Item 10 addresses the fault current withstand capability.

Furthermore, the applicant will add ITAAC Item 12, to DCD Tier 1, Table 2.13.5-2 to address the fault current withstand capability of cables for the safety-related portions of the UPS system.

The staff finds the applicant's response acceptable. Therefore, RAI 14.3-424 is resolved. The staff confirmed that DCD, Revision 6, incorporated the changes as discussed above.

RAI 14.3-425: In RAI 14.3-425, the staff asked the applicant to include ITAAC to address equipment protective devices for: (a) onsite ac, (b) dc power, (c) DG power, and (d) uninterruptible power. In response, the applicant stated that it had addressed the protective devices for the applicable portions of the onsite ac power supply, specifically the preferred power supply, in response to RAI 14.3-394 S01 and had provided ITAAC in response to the RAI.

The applicant will add ITAAC Item 13 in DCD Tier 1, Table 2.13.3-3 to address the fault withstand capability of protective devices for the safety-related portions of the dc power supply system.

Portions of the onsite DG power supply systems capable of supporting the safety-related loads are covered by ITAAC provided for the onsite ac power supply. DCD Tier 1, Table 2.13.1-2, ITAAC Item 10, addresses the fault current interrupting capability.

Furthermore, the applicant will add ITAAC Item 13 in DCD Tier 1, Table 2.13.5-2 to address the fault current withstand capability of protective devices for the safety-related portions of the UPS system.

The staff finds the applicant's response acceptable. Therefore, RAI 14.3-425 is resolved. The staff confirmed that DCD, Revision 6, incorporated the changes as discussed above.

RAI 14.3-427: In RAI 14.3-27, the staff asked the applicant to include ITAAC to address the grounding and lightning protection system. In response, the applicant stated that it will add DCD Tier 1, Section 2.13.9 and Table 2.13.9-1 to address the design description and ITAAC for the lightning protection and grounding system. In addition, GEH will revise DCD Tier 2, Appendix 8A.1.1 to delete the statement that lightning protection ground rods would be separate from the normal grounding system. The ITAAC will verify that a connection exists between the lightning protection system and the station ground grid. This change, and allowing the lightning protection ground rods to tie to the ground grid, will make the lightning protection system more robust by providing additional volume to adequately dissipate lightning strikes. The staff finds the applicant's response acceptable and therefore, RAI 14.3-427 is resolved. The staff confirmed that DCD, Revision 6, incorporated the changes as discussed above.

RAI 14.3-429: In RAI 14.3-429, the staff asked the applicant to include ITAAC to address cable tray loading. In response, the applicant stated that it will add ITAAC Item 14 to DCD Tier 1, Table 2.13.3-3 to address the raceway sizing and loading for the safety-related portions of the dc power supply system. The applicant will also add ITAAC Item 14 to DCD Tier 1, Table 2.13.5-2 to address the raceway sizing and loading for the safety-related portions of the UPS system. The staff finds the applicant's response acceptable. Therefore, RAI 14.3-429 is resolved. The staff confirmed that DCD, Revision 6, incorporated the changes as discussed above.

RAI 14.3-431: In RAI 14.3-431, the staff asked the applicant to include ITAAC to address utilization voltage adequacy. In response, the applicant stated that it will add an ITAAC to DCD Tier 1, Section 2.13.5 to address utilization voltage adequacy for loads on the safety-related UPS 120 volt buses. The as-built safety-related UPS 120 volt distribution system will be analyzed to confirm that the voltage at the terminals of the loads is within the utilization

equipment voltage tolerance limits. Factory testing will document that the utilization equipment functions properly at the established maximum and minimum terminal voltage. The staff finds the applicant's response acceptable. Therefore, RAI 14.3-431 is resolved. The staff confirmed that DCD, Revision 6, incorporated the changes as discussed above.

RAI 14.3-448: In RAI 14.3-448, the staff asked the applicant to provide an ITAAC associated with the coordination of interrupting devices so that the circuit interrupter closest to the fault opens before other devices. The coordination study should include all voltage levels. In response, the applicant stated that it had added an ITAAC to DCD Tier 1, Revision 6, Section 2.13.1 for the coordination of interrupting devices in response to RAI 14.3-443. The applicant further stated that interrupting devices at all voltage levels will be coordinated to ensure that the interrupter closest to a fault opens before other devices, as described in DCD Tier 2 Section 8.3.1.1.6. Additionally, the applicant revised the ITAAC for both Sections 2.13.3 and 2.13.5 and the design description for Item 13 to state, "Protective devices for the safety-related 250 V dc (or UPS) system are rated to interrupt analyzed fault currents and are coordinated to only trip the protective device closest to the fault," as is appropriate for both the inverter ac loads and the single dc load. The applicant stated that its response to RAI 14.3-448, Revision 1, supersedes its response to RAI 14.3-425. The staff finds the applicant's response acceptable. Therefore, RAI 14.3-448 is resolved. The staff confirmed that DCD, Revision 6, incorporated the changes as discussed above.

Based on the staff's review as set forth above, as well as on the applicant's implementation of the selection criteria and methodology for the development of the Tier 1 information in DCD Tier 2, Section 14.3, the staff concludes that DCD Tier 1, Revision 9, appropriately describes the top-level design features and performance characteristics of the SSCs, and the DCD Tier 1, Revision 9, information associated with the scope of SRP Section 14.3.6 is acceptable.

Furthermore, the staff concludes that the DCD Tier 1, Revision 9, design descriptions within the scope of SRP Section 14.3.6 can be verified adequately by the ITAAC. Therefore, the staff concludes that the ESBWR ITAAC within the scope of SRP Section 14.3.6 are necessary and sufficient to assure that with respect to these ITAAC, if the inspections, tests, and analyses are performed and the acceptance criteria are met, a facility referencing the certified ESBWR design has been constructed and will be operated in compliance with the design certification and applicable regulations.

14.3.7 Plant Systems

The applicant provided design-basis information including associated tables and figures, in accordance with the selection criteria and methodology for developing DCD Tier 1, Revision 9, information, as described in DCD Tier 2, Revision 9, Section 14.3 to support the ITAAC for the ESBWR SSCs. The applicant organized its Tier 1 information in the systems, structures, and topical areas format shown in the DCD Tier 1, Revision 9, Table of Contents. The staff reviewed the DCD Tier 1 information provided by the applicant, in accordance with the review matrix provided in Appendix 14.3A of this report and in accordance with SRP Section 14.3.7.

The staff's review of the plant systems generated several RAIs regarding the regulatory treatment of the standby DG support systems and their inclusion in the ITAAC. These included RAIs 19.1.0-2, 22.5.4, 14.3-151, and their supplements and RAI 14.3-177. The applicant included all DG supporting systems as RTNSS; Chapter 19 of this report discusses these systems further. The applicant committed to providing ITAAC for each of the DG supporting systems and included ITAAC entries for each DG supporting system, including the DG cooling

water system, lubrication system, and combustion air and exhaust system in DCD Revision 5. The staff finds the applicant's response to be acceptable, and this RAI is resolved.

The staff reviewed DCD Tier 1, Sections 2.12.3, 2.12.5, and 2.12.7 for the ITAAC. In its review, the staff requested additional information in RAI 14.3-69, RAI 22.5-1, RAI 22.5-1 S01, and RAI 9.2-24. All of these RAIs were resolved. Sections 9.2.1.3.4, 9.2.2.3.4, and 9.2.7.3.4 of this report document the staff's detailed evaluations of these RAIs.

RAI 14.3-369: For ITAAC Item 7a in DCD Tier 1, Table 2.6.2-2, the staff requested that the applicant appropriately modify the item so that both the FAPCS flow path and the capacity are verified in the inspections, tests, and analyses and confirmed in the acceptance criteria. The applicant's response addressed flow path and capacity in both the inspections, tests, and analyses and acceptance criteria. In addition, the acceptance criteria had the actual flow rate in both cubic meters per hour and gallons per minute. The staff finds the applicant's response to be acceptable, and therefore RAI 14.3-369 is resolved.

The staff generated several RAIs to complete its review of the fire protection systems. The following paragraphs discuss RAIs associated with fire protection that have been resolved and are considered significant to the conclusions of the safety evaluation of the ESBWR fire protection program:

RAI 14.3-7: In this RAI, the staff directed GEH to include ITAAC for the fire barriers. The staff based this request on the requirement for new reactor fire protection programs to provide fire barrier separation between redundant trains (except inside the containment and in the MCR), as well as verification that all fire barriers and barrier penetration seals and other closure devices are constructed in accordance with the applicable approved designs, including verification that the design-basis integrity of each barrier is provided. RAI 14.3-7 S01 directed GEH to include ITAAC to verify that the area in which the fire occurs is separated by a fire barrier from any circuits for which a fire-induced failure could cause a spurious actuation that would prevent the protected train (the train outside the area in which the fire occurs) from performing its required post fire, safe-shutdown function. In response, the applicant revised DCD Tier 1, Section 2.16.3.1 and Table 2.16.3.1-1 to include an acceptable design description and related ITAAC for fire barriers. The staff finds the applicant's response to be acceptable, and RAI 14.3-7 is resolved.

RAI 14.3-11: This RAI directed GEH to include an ITAAC to verify that the appropriate seismic analyses had been performed to demonstrate that the SSE function is provided and that the piping and equipment have been installed in accordance with the design. The staff based this request on the fire protection acceptance criteria in RG 1.189, which require a seismically qualified (i.e., must remain functional during and following an SSE) source of fire water supply to standpipes and hose stations in areas with safe-shutdown equipment. The staff finds the applicant's response to be acceptable, and RAI 14.3-11 is resolved.

RAI 14.3-393: This RAI directed GEH to include an ITAAC to verify that the fire-proofing of exposed structural steel in safety-related areas is installed in accordance with the approved design. GEH stated that it would not respond specifically to this RAI but would incorporate the request in Revision 5 of the DCD. The staff finds the associated change to Revision 5 of the DCD to be acceptable and therefore, RAI 14.3-393 is resolved.

RAI 14.3-395: The staff requested that GEH address the impact of a nonseismic failure during an SSE on the ability to ensure that adequate water flow and pressure reach areas containing

equipment relied upon for a safe plant shutdown in the event of an SSE. The staff describes it concern as follows: DCD Tier 1, Table 2.16.3-1 provides the seismic classification of all fire protection pumps, with the exception of the standpipe booster pumps. DCD Tier 2, Section 9.5.1.4 states that booster pumps maintain minimum standpipe pressure. If these pumps are relied upon to meet the post-SSE requirement for hose station protection, equipment such as the pumps, motors, and power supply should be seismic Category I to ensure that they will function following an SSE. The booster pumps should be included in Table 2.16.3-1, with the appropriate seismic category indicated. If the pumps are not seismic Category I, a justification should be provided that including any provisions for bypassing the pumps, if required. This RAI was being tracked as an open item in the SER with open items.

GEH agreed to add the following to DCD Tier 2, Revision 6, Section 9.5.1.4:

> THE ESBWR design does not require the use of booster pumps to maintain minimum standpipe pressure for the post-SSE requirements for hose station protection. Booster pump installation will be limited to the secondary circuit to ensure failure will not impact areas containing equipment performing any safe shutdown function

The staff finds this change acceptable since the booster pumps are not needed to maintain minimum standpipe pressure. Therefore, RAI 14.3-395 and the associated open item are resolved.

<u>RAI 14.3-396</u>: The change in DCD Tier 1, Revision 5, Table 2.16.3-2, Item 3, called for the applicant to verify that hose station protection will be provided for locations outside the containment that contain or could present a hazard to <u>safe-shutdown</u> equipment. GDC 3, "Fire protection," requires that the fire protection program provide protection for SSCs <u>important to safety</u>. Safe-shutdown equipment is a subset of equipment important to safety. Consequently, this ITAAC does not adequately verify compliance with the GDC 3 requirements. In RG 1.189, Revision 1, "Fire Protection for Nuclear Power Plants," issued March 2007, Regulatory Position 3.4.1 states, "Interior manual hose installations should be able to reach any location that contains, or could present a fire exposure hazard to, equipment important to safety…." While RG 1.189 contains some specific guidance for the protection of safe-shutdown SSCs (e.g., hose station coverage following an SSE), the fire protection program must protect SSCs important to safety to ensure compliance with GDC 3. This RAI was being tracked as an open item in the SER with open items.

GEH revised the DCD Tier 1, Sections 2.16.3, 2.16.3.1, as well as Tables 2.16.3-2 and 2.16.3.1-1 to change "safe shutdown" to "safety-related." The staff finds this change acceptable. Therefore, RAI 14.3-396 and the associated open item are resolved.

DCD Tier 1, Section 2.3.1 for the PRMS and Section 2.10 for the Radioactive Waste Management System (RWMS) contain the supporting information for verification of the RWMS design aspects of the ESBWR standard design. The RWMS includes the LWMS, the gaseous waste management system (GWMS), and the solid waste management system (SWMS). These systems are involved in the management of radioactive wastes (liquid, gas, and wet and dry solids) produced during normal operation and anticipated operational occurrences. The PRMS includes subsystems used to collect process and effluent samples during normal operation, during anticipated operational occurrences, and under post-accident conditions.

Areas of the staff's review included implementation of the selection criteria and methodology for developing DCD Tier 1 information, as discussed in DCD Tier 2, Section 14.3, and the resultant DCD Tier 1 information associated with the RWMS. The areas of review included design objectives, design criteria, identification of all expected releases of radioactive effluents, methods of treatment, and operational programs for controlling and monitoring effluent releases and for assessing associated doses to members of the public. In addition, the review included an evaluation of the PRMS, which is used to monitor liquid and gaseous process streams and effluents and the solid wastes generated by these systems. The staff generated a number of RAIs (not listed here for the sake of brevity) during its review of the design certification application. In summary, the RAIs involved requests for the applicant to (1) provide clarifications for technical completeness, (2) provide details supporting the design descriptions and functional arrangements for demonstrating compliance with regulatory requirements, (3) revise and update tables and drawings for consistency with DCD Tier 2 system descriptions, (4) revise technical and regulatory references, and (5) provide information to enable the staff to conduct further evaluations of supporting topics presented in DCD Tier 2 to support DCD Tier 1 design descriptions and the associated ITAAC. The RAIs addressed the following major technical and regulatory topics:

- Descriptions, functional arrangements, application, and scope of ITAAC for the LWMS, GWMS, SWMS, and PRMS

- Design descriptions and ITAAC addressing the initiation and closure of valves and isolation of systems in controlling and limiting releases of radioactive liquid and gaseous effluents into the environment

- Scope of tests and acceptance criteria to confirm that radiation monitors would alarm and initiate valve closures or isolation of systems upon receipt of a high-radiation signal, exceeding a setpoint value, from a radiation detector

- Basis of criteria for the inclusion and application of ITAAC that, although for non-safety-related systems, are required to demonstrate compliance with 10 CFR Part 20 effluent concentration limits for members of the public and the design objectives in Appendix I to 10 CFR Part 50

- Criteria for verifying the nominal capacities of the major processing tanks of the SWMS, including the high- and low-activity resin holdup tanks, the condensate resin holdup tank, the phase separator tanks, and the concentrated waste tank

- Criteria for installing steel liners in cubicles housing LWMS tanks and vessels to ensure that, in the event of a tank rupture, the effluent concentration limits of Table 2 (Column 2) in Appendix B to 10 CFR Part 20 will not be exceeded at offsite locations

- Initial installation of appropriate types and amounts of absorbent and filtration media in LWMS (demineralizers) and GWMS/off-gas system charcoal beds (guard and main beds) in demonstrating compliance with 10 CFR Part 20 effluent concentrations and dose limits for members of the public and with the design objectives in Appendix I, "Numerical Guides for Design Objectives and Limiting Conditions for Operation to Meet the Criterion "As Low As Reasonably Achievable" for Radioactive Material in Light-Water-Cooled Nuclear Power Reactor Effluents," to 10 CFR Part 50

- Correction of internal inconsistencies in DCD Tier 1 design descriptions and the design commitments specified in the associated ITAAC

The applicant resolved 16 RAIs, including RAIs 14.3-138 through 14.3-143, RAI 14.3-145, RAIs 14.3-154 through 14.3-161, and RAI 14.3-391. An example of one such resolved RAI is RAI 14.3-161, described below.

RAI 14.3-161: The staff noted that DCD Tier 1, Revision 4, Section 2.3.1 did not include ITAAC assigned to PRMS subsystems that are used to monitor compliance with the liquid and gaseous effluent concentration limits found in 10 CFR Part 20, Appendix B, Table 2. The lack of ITAAC for nonsafety-related yet essential subsystems used in demonstrating compliance with 10 CFR Part 20 is not consistent with the criteria and application process described in DCD Tier 2, Revision 4, Section 14.3.7.3, on design features used to comply with the NRC regulations. Accordingly, the staff requested that the applicant revise DCD Tier 1, Section 2.3.1 to include the necessary ITAAC for all PRMS subsystems that are used to monitor, control, and terminate radioactive effluent releases into the environment. The applicant revised Table 2.3.1-2 to include an ITAAC for nonsafety-related radiation monitors included in the plant to actively/automatically restrict offsite doses to below the limits in 10 CFR Part 20. The staff finds this response acceptable, and RAI 14.3-161 is resolved.

The ITAAC reviewed by the staff in accordance with SRP Section 14.3.7 also include systems that are nonsafety-related but are used to ensure compliance with the regulatory requirements of 10 CFR Part 20, Sections 20.1301 and 20.1302; 10 CFR 50.34a; 10 CFR 50.36a; the dose objectives in Appendix I to 10 CFR Part 50; GDC 60, “Control of releases of radioactive material to the environment”, GDC 63, “Monitoring fuel and waste storage”, and GDC 64, “Monitoring radioactivity releasees” in Appendix A to 10 CFR Part 50; and the waste form characteristics in 10 CFR Part 61, “Licensing Requirements for Land Disposal of Radioactive Waste.” In demonstrating compliance with the above regulatory requirements, the operation of these systems is governed by operational programs that are mandated under license conditions. These operational programs include the ODCM for confirming that instrumentation alarm setpoints are established in limiting radioactive release rates or radionuclide concentrations in the environment; the PCP for ensuring that radioactive wastes meet waste form characteristics for disposal; and the REMP for confirming that liquid and gaseous effluent releases meet the 10 CFR Part 20 dose and effluent concentration limits and the as-low-as-reasonably-achievable (ALARA) design objectives in Appendix I to 10 CFR Part 50. DCD Tier 2, Section 13.4 addresses, as COL commitments, the milestones for the development and implementation of the ODCM, PCP, and REMP. The proposed ITAAC, once performed by a COL Applicant and having met their respective acceptance criteria, provide reasonable assurance that a plant incorporating the requirements of the ESBWR design certification will operate in accordance with the design certification and the provisions of the Atomic Energy Act and the NRC regulations.

Based on the staff’s review as set forth above, as well as on the applicant’s implementation of the selection criteria and methodology for the development of the DCD Tier 1, Revision 9, information in DCD Tier 2, Revision 9, Section 14.3, the staff concludes that DCD Tier 1 appropriately describes the top-level design features and performance characteristics of the SSCs, and the DCD Tier 1, Revision 9, information associated with the scope of SRP Section 14.3.7 is acceptable.

Furthermore, the staff concludes that the ITAAC can adequately verify the DCD Tier 1, Revision 9, design descriptions within the scope of SRP Section 14.3.7. Therefore, the staff concludes

that the ESBWR ITAAC associated with the scope of SRP Section 14.3.7 are necessary and sufficient to assure that with respect to these ITAAC, if the inspections, tests, and analyses are performed and the acceptance criteria are met, a facility referencing the certified ESBWR design has been constructed and will be operated in compliance with the design certification and applicable regulations.

14.3.8 Radiation Protection

The applicant provided design-basis information, including associated tables and figures, in accordance with the selection criteria and methodology for developing DCD Tier 1, Revision 9, information, as described in DCD Tier 2, Revision 9, Section 14.3, to support the ITAAC for ESBWR SSCs. The applicant organized the Tier 1 information in the systems, structures, and topical areas format shown in the DCD Tier 1, Revision 9, Table of Contents. The staff reviewed the DCD Tier 1 information provided by the applicant, in accordance with the review matrix provided in Appendix 14.3A of this report and in accordance with SRP Section 14.3.8.

The documents that contain the supporting information for the verification of the radiation protection aspects of the ESBWR design are DCD Tier 1, Revision 9, Section 2.3.1 for the PRMS; Section 2.3.2 for the ARM system; and Section 3.4 for radiation protection. The PRMS includes a description of the airborne radioactivity system used to monitor airborne radioactivity levels in various areas within the plant. The ARM system continuously monitors the gamma radiation levels within the various areas of the plant and provides an early warning to operating personnel when high radiation levels are detected, so the appropriate action can be taken to minimize occupational exposure. The ITAAC on radiation protection provide a verification of the means by which the plant is designed to maintain radiation exposures ALARA (i.e., through the use of ventilation flow and the containment of airborne radioactive materials, the use of area radiation monitoring to measure radiation levels throughout the plant, and the incorporation of radiation shielding to obtain radiation dose rates in each plant area commensurate with that area’s occupancy requirements).

Areas of the staff’s review included implementation of the selection criteria and methodology for developing DCD Tier 1 information, as discussed in DCD Tier 2, Revision 9, Section 14.3, and the resultant DCD Tier 1 information associated with the ARM systems, airborne radioactivity monitoring systems, and radiation shielding provided by structures and components for normal and emergency conditions. In addition, the review included an evaluation of the PRMS with respect to the airborne radioactivity monitors used to measure airborne radioactivity levels within the plant. The staff generated a number of RAIs requesting the applicant to provide clarifications for technical completeness, revisions and updates to tables for consistency with the DCD Tier 2 system descriptions, and information on which the staff could base further evaluations of supporting topics presented in DCD Tier 2 to support DCD Tier 1 design descriptions and the associated ITAAC. The RAIs generated by the staff addressed the following topics:

- Incorporation of an ITAAC on radiation shielding to be consistent with the guidance in the SRP
- Identification of those ARM systems located in areas where the dose rate increase can exceed 1 millisievert per hour (100 millirem per hour)
- Inconsistencies in the DCD Tier 1 design descriptions and the design commitments specified in the associated ITAAC

- Corrections of inconsistencies between information provided in DCD Tier 1 and Tier 2 tables and text regarding the listing of area radiation monitors

All of these RAIs have been resolved. Some examples of these resolved RAIs are RAIs 14.3-343, 14.3-174 S01, and RAI 14.3-175 S01, which are described below:

RAI 14.3-343: For ITAAC Item 1 in Table 3.4-1, although the design commitment addressed two functions for the system containing airborne radioactive materials and maintaining the concentration of airborne radionuclides at levels consistent with personnel access needs, the inspections, tests, and analyses and acceptance criteria only verified the latter. The applicant revised the inspections, tests, and analyses to address testing of isolation dampers and revised the acceptance criteria to state that "A test report documents that isolation dampers close within the designed time frame and limit leakage to a rate below the design assumed leakage rate." The staff finds the applicant's response acceptable, and RAI 14.3-343 is resolved.

RAI 14.3-174 S01: In this RAI supplement, the staff asked the applicant to provide additional details and clarifications regarding airborne radioactivity monitoring. Specifically, the staff requested the applicant to clearly identify the airborne radioactivity monitors that meet the sensitivity and location criteria to ensure that plant personnel are not inadvertently exposed to airborne contaminants in excess of the limits provided in 10 CFR Part 20. The staff requested that the applicant include a list of these monitors in the ITAAC. In addition, the staff requested that the applicant provide a table in the appropriate part of the DCD specifying the airborne radioactivity monitors that meet the sensitivity and location criteria. The staff also asked the applicant to provide acceptance criteria for the location of these airborne radioactivity monitors. Therefore, RAI 14.3-174 S01 was being tracked as an open item in the SER with open items.

In response to RAI 14.3-174 S01, GEH did not identify the specific airborne radioactivity monitors that meet the sensitivity and location criteria to ensure that plant personnel are not inadvertently exposed to airborne contaminants in excess of the limits provided in 10 CFR Part 20. In GEH's revised response to this supplemental RAI, GEH committed to revise DCD Tier 2, Revision 9, Section 12.3.4 to state that portable airborne radiation monitors will be used to provide the local airborne radioactivity monitoring to meet requirements for worker protection. These portable continuous air monitors (CAMs) will provide a means to observe trends in airborne radioactivity concentrations. CAMs equipped with a local alarm capability are used in occupied areas where needed to alert personnel to sudden changes in airborne radioactivity concentrations. In order to warn personnel of changing airborne conditions, CAM alarm setpoints are set at a fraction of the concentration values given in 10 CFR Part 20 Appendix B, Table 1, Column 3, for radionuclides expected to be encountered. The number of CAMs used, as well as the placement of these portable monitors, will be the responsibility of the COL Applicant. Since the operational considerations and placement of the monitors to be used for airborne radioactivity monitoring will be the responsibility of the COL Applicant (as is specified in COL Information Item 12.3-2-A, Operational Considerations), no ITAAC are necessary for these portable monitors. The staff finds the applicant's response to this supplemental RAI to be acceptable because the applicant stated that portable airborne radiation monitors used to meet requirements for worker protection in the local areas will meet the sensitivity and location criteria specified in SRP Section 12.3-12.4 to ensure that plant personnel are not inadvertently exposed to airborne contaminants in excess of the limits provided in 10 CFR Part 20. Since these matters are not within the scope of the ESBWR design certification as described above, RAI 14.3-174 is resolved.

RAI 14.3-175 S01: In this RAI supplement, the staff requested that the applicant revise the numbering of the area radiation monitors in Figures 12.3-23 through 12.3-42 to provide specific ARM identifiers that are clear and objective and that cannot be misidentified with ARMs in different building locations. The staff also requested the applicant to clarify the acronyms used in Tables 12.3-2 through 12.3-6 that are associated with the monitoring range to ensure they are clear and objective. Therefore, RAI 14.3-175 was being tracked as an open item in the SER with open items.

In response, GEH explained that in the final design, the component numbers are uniquely assigned using GEH design control procedures. The component ID identifies the system and building so that each radiation monitor is uniquely differentiated. GEH provided examples of how the ARMs located in different building will be numbered. GEH further stated in their response that DCD Tier 2, Section 12.3.4.2 identifies the channel monitoring range, and the acronym for each area radiation channel is found in DCD Tier 2, Table 12.3-7, so a clarification in each table is not necessary. On the basis of the applicant's response, RAI 14.3-175 and the associated open item are resolved.

The ITAAC reviewed by the staff in accordance with SRP Section 14.3.8 also included systems that are nonsafety-related but are used to ensure compliance with the regulatory requirements of 10 CFR 20.1101, 10 CFR 20.1201, 10 CFR 50.34a, 10 CFR 50.34(f), and GDC 19, "Control room", in Appendix A to 10 CFR Part 50. Programs that will be mandated by license conditions govern the operation of these systems to demonstrate compliance with the above regulatory requirements. These operational programs include the radiation protection program, which addresses plant management policy, organization, facilities, instrumentation, and equipment and procedures sufficient to ensure that occupational doses and doses to the public areas remain ALARA. DCD Tier 2, Revision 8, Section 13.4 addresses, as COL commitments, the milestones for developing and implementing the operational Radiation Protection Program. The proposed ITAAC, in conjunction with the implementation of these operational programs, once performed by a COL Applicant and having met their respective acceptance criteria, provide reasonable assurance that a plant incorporating the requirements of the ESBWR design certification has been constructed and will operate in accordance with the design certification and the provisions of the Atomic Energy Act and the NRC regulations.

Based on the staff's review as set forth above, as well as on the applicant's implementation of the selection criteria and methodology for the development of the DCD Tier 1 information in DCD Tier 2, Revision 9, Section 14.3 the staff concludes that DCD Tier 1 appropriately describes the top-level design features and performance characteristics of the SSCs, and the DCD Tier 1, Revision 9, information associated with the scope of SRP Section 14.3.8 is acceptable.

Furthermore, the staff concludes that the Tier 1 design descriptions within the scope of SRP Section 14.3.8 can be verified adequately by the ITAAC. Therefore, the staff concludes at this time that the ESBWR ITAAC within the scope of SRP Section 14.3.8 are necessary and sufficient to assure that with respect to these ITAAC, if the inspections, tests, and analyses are performed and the acceptance criteria are met, then a facility referencing the certified ESBWR design has been constructed and will be operated in compliance with the design certification and applicable regulations.

14.3.9 Human Factors Engineering

The applicant provided design-basis information, including associated tables and figures, in accordance with the selection criteria and methodology for developing DCD Tier 1, Revision 9, information, as described in DCD Tier 2, Revision 9, Section 14.3 to support the ITAAC for the ESBWR SSCs. The applicant organized the Tier 1 information in the systems, structures, and topical areas format shown in the DCD Tier 1, Revision 9, Table of Contents. The staff reviewed the Tier 1 information provided by the applicant in accordance with the review matrix provided in Appendix 14.3A of this report and in accordance with SRP Section 14.3.9.

In SECY-92-053, the staff provided the Commission with a method for using the DAC together with detailed design information, during the 10 CFR Part 52 process for reviewing and approving designs. The staff has used this method for design certification applications that did not provide design and engineering information at a level of detail customarily required by the staff to reach a final safety decision on the design. The Commission previously issued guidance on the level of design detail required for the design certification. The SRM to SECY-90-377 provided the level of detail that the design should reflect.

The applicant may provide the DAC in lieu of detailed system design information in areas such as HFE, where technology is rapidly changing and applicants believe it is unwise to prematurely freeze the design. The DAC are a set of prescribed limits, parameters, procedures, and attributes upon which the NRC relies in a limited number of technical areas, in making a final safety determination in support of the design certification. The acceptance criteria for the DAC should be objective; that is, they should be able to be inspected, tested, or subjected to analysis using preapproved methods and should be verified as a part of the ITAAC performed to demonstrate that the as-built facility conforms to the certified design. Thus, the acceptance criteria for the DAC are specified together with the related ITAAC in DCD Tier 1, Revision 9, and both are part of the design certification.

Design certification applicants should provide the design-related processes and associated DAC in DCD Tier 1 that a COL Applicant or Licensee would follow to complete the design. The COL licensees must verify implementation of the DAC as part of the ITAAC performed to demonstrate that the as-built facility conforms to the certified design. In this case, the DAC should be sufficiently detailed to provide an adequate basis for the staff to make a final safety determination regarding the design, subject only to the satisfactory verification of completion of the design (i.e., verification of the DAC) and the installation of the completed design by the COL licensee.

For the control room and the RSS design (human factors), the design descriptions and DAC provided in Tier 1 delineate the process and requirements that a COL Applicant or Licensee must implement to develop the design information required in each area. The ITAAC specifies the acceptance criteria for the development process at various stages of detailed design and subsequent construction and testing. The NRC requires the COL Applicant or Licensee to develop the procedures and test programs necessary to demonstrate that the DAC requirements are met at each stage. Since the DAC are considered to be design-completion ITAAC, the COL Applicant or Licensee must certify to the NRC that the design through that phase is in compliance with the design criteria specified in the certified design. The NRC reviews, audits, and/or inspects the work to confirm that the COL Applicant or Licensee has adequately implemented the commitments of the DAC in these phases.

The staff issued a number of RAIs to facilitate the completion of its review. The RAIs discussed below describe some of the concerns of the staff with respect to its review of the DCD Tier 1 information associated with HFE.

RAI 14.3-85: DCD Tier 1, Table 3.3-1, Item 10.a. calls for a "Procedure Implementation Plan," which the applicant has completed and the NRC is reviewing as part of the ESBWR design certification. Therefore, Item 10.a does not belong in the ITAAC. Item 10.b relates to the implementation of the "Procedure Development Plan" and is appropriate, but it should be modified to perhaps be similar to the HFE ITAAC used for AP1000. In response, GEH stated that "In Revision 2 to DCD, Tier 1, Item 10.a was deleted and Item 10.b (now Item 7) was modified considering the suggested guidance." The staff reviewed the revision to the DCD and, finds that the applicant's response adequately addresses its concern; RAI 14.3-85 is resolved.

RAI 14.3-86: DCD Tier 1 ITAAC for "Training Development," Table 3.3-1, Item 11.a. requires a Training Program Development Implementation Plan that the applicant has completed and the NRC is reviewing as part of the ESBWR design certification. Therefore, Item 11.a does not belong in the ITAAC. Item 11.b relates to the implementation of the training program itself. Since the training is an operational program, this ITAAC is not needed. The applicant responded stating that, "In Revision 2 to DCD, Tier 1, Item 11.a was deleted and Item 11.b (now Item 8) was modified to contain a results summary report describing the training program." The staff reviewed the revision to the DCD and finds that the applicant's response adequately addressed its concern; RAI 14.3-86 is resolved.

RAI 14.3-87: The ITAAC for "Verification and Validation (V&V)" in DCD Tier 1, Revision 1, Table 3.3-1, Item 12.a required a V&V plan, which the applicant has completed and which the NRC is reviewing as part of the ESBWR design certification. Therefore, Item 12.a does not belong in the ITAAC. Item 12.b relates to the implementation of the V&V itself. The applicant should modify it so that it refers to the implementation of the V&V plan and construct it following the guidance in SRP Section 14.3. The applicant responded stating that, "[i]n Revision 2 to DCD Tier 1, Item 12.a was deleted and Item 12.b (now Item 9) was modified considering the suggested guidance." The staff reviewed the revision to the DCD and, finds that the applicant's response adequately addressed its concern. RAI 14.3-87 is resolved.

RAI 14.3-88: The ITAAC for design implementation in DCD Tier 1, Revision 1, Table 3.3-1, Item 13.a, relates to the development of an implementation plan, which the applicant has completed and which the NRC is reviewing as part of the ESBWR design certification. Therefore, Item 13.a does not belong in the ITAAC. Item 13.b relates to the implementation of the V&V itself. This should be modified to be the implementation of the V&V Plan and should be constructed following the guidance in SRP Section 14.3. The applicant responded and indicated that, "In Revision 2 to DCD, Tier 1, Item 13.a was deleted and Item 13.b (now Item 10) was modified considering the suggested guidance." The staff reviewed the revision to the DCD and finds that the applicant's response adequately addressed its concern. RAI 14.3-88 is resolved.

RAI 14.3-89: The ITAAC for human performance engineering in DCD Tier 1, Revision 1, Table 3.3-1, Item 14.a relates to the development of an implementation plan, which the applicant has completed and which the NRC is reviewing as part of the ESBWR design certification. Therefore, Item 14.a does not belong in the ITAAC. Item 14.b relates to the implementation of the monitoring program itself, which is a COL responsibility subsequent to plant startup. The applicant should modify this ITAAC to refer to the establishment of the human performance monitoring program by the COL licensee and should follow the guidance in

SRP Section 14.3. The applicant responded stating that "In Revision 2 to DCD, Tier 1, Item 14.a was deleted and Item 14.b (now Item 11) was modified to contain a results summary report describing the HPM [human performance monitoring] program." The staff reviewed the revision to the DCD and finds that the applicant's response adequately addressed its concern. RAI 14.3-89 is resolved.

<u>RAI 14.3-355</u>: For ITAAC Item 6 in Table 2.4.2-3, the staff requested that the applicant clarify both its method of deriving the "minimum set of displays" and the correlation between the "minimum set of displays" in the design commitment and its retrievability in the acceptance criteria. In response, the applicant stated that the issues associated with the "minimum set of displays" and the "retrievability of them" are addressed by the HFE DAC ITAAC in Tier 1, Section 3.3. In DCD Tier 1, Section 1.1.1, the term "Inspect for Retrievability" of a display means to visually observe that the specified information appears on a monitor when summoned by the operator. The staff finds this response acceptable, and RAI 14.3-355 is resolved.

<u>RAI 14-210, Supplement 1</u>: Because DAC closure could be performed in several design phases, the NRC requires information on the closure schedule to plan its related activities appropriately. This COL information item will ensure that every COL Applicant referencing the ESBWR DCD provides the NRC with a schedule for DAC closure, even if the initial response to the COL information item is to make a commitment to provide such a schedule at a time when information is mature enough to be able to make reasonable schedule commitments. As such, the staff requested that GEH include a COL information item in the DCD for the COL Applicants/Holders to provide a schedule for DAC closure. The applicant responded that it had updated DCD Tier 2, Revision 5, Section 14.3, Appendix A and had included a COL information item to address the staff's concerns. The staff finds the applicant's response acceptable, and RAI 14.3-210 S01 is closed.

<u>RAI 14.3-211</u>: ITAAC Table 3.3-1 contains 11 items, one for each element of NUREG–0711, "Human Factors Engineering Program Review Model," Revision 2, and the corresponding ESBWR element implementation plan. However, the design commitment column in the ITAAC for each element refers to the overall man-machine interface system and the HFE Implementation Plan, rather than to the specific implementation plans of the pertinent elements. The staff requested that the applicant update the 11 design descriptions provided in Tier 1 to refer to the applicable implementation plans. The applicant responded stating that "GEH will revise the design commitment column in ITAAC Table 3.3-1 in DCD Tier 1, Revision 5, to reference the respective implementation plans." RAI 14-211 was being tracked as an open item in the SER open items. In Revision 6 of the DCD, GEH changed Table 3.3-1 to reference the applicable implementation plans. Therefore, RAI 14-211 is resolved.

<u>RAI 14.3-271</u>: In this RAI, the staff requested the applicant to update the inspections, tests, and analyses and acceptance criteria columns in Table 3.3-1 to ensure that they accurately reflect the methodology described in the final versions of the approved implementation plans. In addition, the staff asked the applicant to review all of the items in the acceptance criteria column to ensure that the text is complete. For example, in DCD Tier 1, Table 3.3-1 Item 1, the acceptance criteria states the following:

> Summary report documents:
>
> a. The OER team members and backgrounds.
> b. The scope of the OER.
> c. The sources of the operating experience reviewed and documented results.
> d. The process for issue analysis, tracking and review.

Because the staff deterimined the above list was incomplete and thus did not provide acceptable acceptance criteria, the staff asked the applicant to update its ITAAC. RAI 14.3-271 was being tracked as an open item in the SER with open items. Revision 6 of the DCD accurately captured the methodology described in final versions of the implementation plans. GEH adjusted the acceptance criteria to reflect complete and meaningful measurements accordingly. Therefore, RAI 14.3-271 and the associated open item are resolved. Subsequent to this change, the ITAAC for procedures and training were deleted because these elements are associated with operational programs. Commission guidance (SECY-02-0067, "Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC) for Operational Programs (Programmatic ITAAC)," April 15, 2002, and the related SRM, dated September 11, 2002; SECY-04-0032, "Programmatic Information Needed for Approval of a Combined License Without Inspections, Tests, Analyses and Acceptance Criteria," February 26, 2004, and the related SRM, dated May 14, 2004; and SECY-05-0197 and the related SRM, dated February 22, 2006.) indicates that operational programs remain subject to inspection and do not require ITAAC.

Based on the staff's review as set forth above, as well as the applicant's implementation of the selection criteria and methodology for the development of the DCD Tier 1, Revision 9, information in DCD Tier 2, Revision 9, Section 14.3, the staff concludes that DCD Tier 1 appropriately describes the top-level design features and performance characteristics of the SSCs, and the DCD Tier 1, Revision 9, information associated with the scope of SRP Section 14.3.9 is acceptable.

Furthermore, the staff concludes that the DCD Tier 1, Revision 9, design descriptions within the scope of SRP Section 14.3.9 can be verified adequately by the ITAAC. Therefore, the staff concludes that the ESBWR ITAAC within the scope of SRP Section 14.3.9 are necessary and sufficient to assure that with respect to these ITAAC, if the inspections, tests, and analyses are performed and the acceptance criteria are met, a facility referencing the certified ESBWR design has been constructed and will be operated in compliance with the design certification and applicable regulations.

14.3.10 Emergency Planning

The applicant did not provide emergency planning ITAAC in DCD Tier 1. As discussed in DCD Tier 2, Revision 9, Section 14.3.8 of the ESBWR DCD, the COL Applicant is responsible for providing the emergency planning ITAAC, and this requirement is consistent with the guidance provided in RG 1.206. In addition, in DCD Tier 2, Section 14.3.10, the applicant provided a COL information item (COL Information Item 14.3.1-A) specifying that the COL Applicant shall provide emergency planning ITAAC based on industry guidance. The staff finds the inclusion of COL Information Item 14.3.1-A in DCD Tier 2, Revision 9, Section 14.3 and the absence of ITAAC for emergency planning in Tier 1, to be acceptable and consistent with the NRC guidance provided in RG 1.206.

14.3.11 Containment Systems

The applicant provided design-basis information, including associated tables and figures, in accordance with the selection criteria and methodology for developing DCD Tier 1 information, as described in DCD Tier 2, Section 14.3 to support the ITAAC for ESBWR SSCs. The applicant organized the Tier 1 information in the systems, structures, and topical areas format shown in the DCD Tier 1, Table of Contents. The staff reviewed the DCD Tier 1 information provided by the applicant, in accordance with the review matrix provided in Appendix 14.3A of this report and in accordance with SRP Section 14.3.11.

The staff's review generated a number of RAIs, several of which the applicant has resolved satisfactorily. The RAIs discussed below are examples of some of the staff's concerns that were resolved.

RAI 14.3-230: The staff considered the following in evaluating the effect of LOCA-generated and latent debris effects on decay heat removal and containment cooling: (a) the GDCS pool consists of a stainless steel liner (DCD Tier 2, Revision 4, Table 6.1-1), (b) the suppression pool consists of a stainless steel liner (DCD Tier 2, Revision 4, Table 6.1-1), (c) "Suppression pool equalization lines have an intake strainer to prevent the entry of debris material into the system that might be carried into the pool during a large break LOCA" (DCD Tier 2, Revision 4, Section 6.2.2.7.2), (d) "The GDCS pool airspace opening to the DW will be covered by a perforated steel plate to prevent debris from entering pool and potentially blocking the coolant flow through the fuel" (DCD Tier 2, Revision 4, Section 6.2.2.7.2), and (e) "The Passive Containment Cooling System (PCCS) heat exchanger inlet pipe is provided with a debris filter with holes no greater than 25 mm (1 inch) to prevent entrance of missiles into the pipe and protection from fluid jets during a LOCA condition" (GEH response to NRC RAI 6.3-42).

However, the staff could not find information in DCD Tier 1, Revision 4, to document and verify these important assumptions. Therefore, the staff asked GEH to add these assumptions to DCD Tier 1, Table 2.15.3-1 and to identify them in DCD Tier 1, Figure 2.15-1. In response, GEH agreed to revise DCD Tier 1 to include this information. The staff considers this response acceptable. The staff confirmed that GEH has made the above changes in DCD Tier 1, Revision 5. The staff considers this response acceptable, and RAI 14.3-230 is resolved.

RAI 14.3-232: DCD Tier 1, Revision 4, did not have an ITAAC to verify that the reactor vessel shield wall is able to withstand the design differential pressure between the reactor vessel annulus and the drywell. Therefore, in RAI 14.3-232, the staff requested that GEH add an ITAAC to DCD Tier 1, Table 2.15.3-2 to verify this design commitment. In response, GEH agreed to update DCD Tier 1, Table 2.15.3-2, ITAAC Item 3, by adding the annulus pressurization loads to verify the structural integrity of the containment internal structures identified in DCD Tier 1, Table 2.15.3-1, which includes the reactor shield wall. The staff confirmed that GEH has made the above changes in DCD Tier 1, Revision 5. RAI 14.3-232 is resolved.

RAI 14.3-233: The acceptance criteria for Item 8 of DCD Tier 1, Revision 4, Table 2.15.3-2, state that "[t]est report(s) demonstrate that each as-built vacuum breaker proximity sensor indicates an open position with the vacuum breaker fully open and indicates a closed position when the vacuum breaker is in the fully closed position." DCD Tier 2, Revision 4, Section 6.2.1.1.2 states that "[t]he vacuum breaker is provided with redundant proximity sensors to detect its closed position." Based on the above, the staff determined that the proximity sensor should identify when the vacuum breaker is open and causing a drywell to wetwell bypass leakage that exceeds the design capacity. That is, the proximity sensor should be able to identify the vacuum breaker open position before it is "fully open." In RAI 14.3-233, the staff asked GEH to revise DCD Tier 1, Table 2.15.3-2, to verify this design feature. In response, GEH agreed that the proximity sensors should be able to identify the vacuum breaker open position before it is fully open. GEH stated that the ITAAC acceptance criteria had been changed from "fully open" to "open". The staff considers the applicant's response to be acceptable, and RAI 14.3-233 is resolved.

RAI 14.3-234: DCD Tier 1, Revision 4, did not provide information needed to verify the following aspects of containment analyses: (a) vacuum breaker area, (b) total number of vertical vents,

and (c) relative elevation of spillover holes. Therefore, in RAI 14.3-234, the staff requested GEH to provide this information in DCD Tier 1. In response, GEH agreed to add the following information to DCD Tier 1 Figure 2.15.1-1: (1) vacuum breaker area: 0.2 square meters (2.2 square feet) each, (2) total number of vertical vents: , 12, and (3) relative elevation of spillover holes, 12,370 mm (40.6 feet). The staff considers the applicant's response to be acceptable and RAI 14.3-234 is resolved.

RAI 14.3-237: The staff found a discrepancy in the PCCS design pressure given in DCD Tier 1 and Tier 2. DCD Tier 1, Revision 4, Table 2.15.4-2, states that "[t]he pressure boundary of the PCCS retains its integrity under the design pressure of 310 kPa gauge (45 psig)." However, in DCD Tier 2, Revision 4, Table 6.2-10 states that "the PCCS design pressure as 758.5 kPa gauge (110 psig)." Based on the above, the staff requested in RAI 14.3-237 that GEH correct this discrepancy. In response, GEH agreed to revise DCD Tier 1, Revision 4, Table 2.15.4-2 to state that "[t]he pressure boundary of the PCCS retains its integrity under the containment design pressure of 310 kPa gauge (45 psig)." The staff considers the applicant's response to be acceptable, and RAI 14.3-237 is resolved.

RAI 14.3-238: DCD Tier 2, Revision 4, Table 6.2-10 provides PCCS design parameters. The staff could not find the necessary information in DCD Tier 1, Revision 4, to verify the following PCCS design parameters: (a) The heat removal capacity for each loop is 11 MWt nominal for pure saturated steam at a pressure of 308 kPa (absolute) (45 psia) and a temperature of 134 degrees C (273.2 degrees F) condensing inside tubes with an outside pool water temperature of 102 degrees C; and (b) The system design temperature is 171 degrees C (340 degrees F). As a result, the staff requested in RAI 14.3-238 that GEH explain how the above design parameters are to be verified.

In response, GEH stated the following: (a) both the inspections, tests, and analyses and the acceptance criteria in DCD Tier 1, Table 2.15.4-2, Item 7 will be revised to include requirements that clearly demonstrate and confirm the capacity of the PCC condensers and design-basis assumptions, and (b) the ITAAC associated with the design and construction of system piping and components (e.g., Table 2.15.4-2, Items 2a and 2b) demonstrate that the system is designed and constructed to meet its design requirements, including system design temperature. ASME code design reports will provide appropriate confirmation of compliance with the design temperature. DCD Tier 2, Revision 9, Section 14.3 describes the process for identification of ITAAC items. The focus of the ITAAC is intended to be on the verification of numeric performance values, in lieu of numeric design values. The staff considers the applicant's response to be acceptable, and RAI 14.3-238 is resolved.

As a result of its review, the staff identified the following issues in RAIs and tracked them as open items in the SER with open items

RAI 14.3-229: Drywell to wetwell bypass leakage capacity is an important assumption used in the containment analyses, but the staff could not find information in DCD Tier 1, Revision 4, to verify the bypass leakage capacity. Therefore, in RAI 14.3-229, the staff requested GEH to add: (1) an item to DCD Tier 1, Section 2.15.3 giving the drywell to wetwell bypass leakage capacity, and (2) an ITAAC to DCD Tier 1, Table 2.15.3-2 to verify this value.

In response, GEH agreed with the NRC request to revise the DCD to add an ITAAC for drywell to wetwell (suppression pool) bypass leakage. GEH proposed to update ESBWR DCD Tier 1, Table 2.15.1-2 to include an acceptance criteria for drywell to wetwell bypass leakage tests that states, "[r]eport(s) document that the results of the drywell to wetwell bypass leakage is less

than or equal to 50 percent of the assumed value in the containment capability design-basis containment response analysis."

In RAI 6.2-145 S02, the staff requested GEH to provide additional justification for this proposed change. In RAI 14.3-229 S01, the staff requested GEH to make the responses to RAIs 14.3-229 and 6.2-145 consistent. RAI 14.3-229 S01 was being tracked as an open item in the SER with open items.

In response to RAI 14.3-229 S01, GEH stated that DCD Tier 1, Section 2.15.1-2 was revised in Revision 5 to be consisted with the bypass leakage acceptance criteria described in DCD Tier 2, Section 6.2.1.1.5.4.3.

GEH's response addresses the staff's concerns and is acceptable to the staff. RAI 14.3-229 and the associated open item are resolved.

Based on the staff's review as set forth above, as well as on the applicant's implementation of the selection criteria and methodology for the development of the DCD Tier 1, Revision 9, information in DCD Tier 2, Revision 9, Section 14.3, the staff concludes that DCD Tier 1, Revision 9, appropriately describes the top-level design features and performance characteristics of the SSCs, and the information associated with the scope of SRP Section 14.3.11 is acceptable.

Furthermore, the staff concludes that the Tier 1 design descriptions associated with the scope of SRP Section 14.3.11 can be verified adequately by the ITAAC. Therefore, the staff concludes that the ESBWR ITAAC with the scope of SRP Section 14.3.11 are necessary and sufficient to assure that with respect to these ITAAC, if the inspections, tests, and analyses are performed and the acceptance criteria are met, a facility referencing the certified ESBWR design has been constructed and will be operated in compliance with the design certification and applicable regulations.

14.3.12 Physical Security

The applicant provided design-basis information including associated tables and figures, in accordance with the selection criteria and methodology for developing DCD Tier 1 information, as described in DCD Tier 2, Revision 9, Section 14.3 to support the ITAAC for ESBWR SSCs. The applicant organized the DCD Tier 1 information in the systems, structures, and topical areas format shown in the DCD Tier 1, Revision 9, Table of Contents. The staff reviewed the DCD Tier 1 information provided by the applicant, in accordance with the review matrix provided in Appendix 14.3A of this report and in accordance with SRP Section 14.3.12, Revision 1, issued January 2010.

The NRC regulation for protecting nuclear power reactors is provided in 10 CFR Part 73, "Physical Protection of Plants and Materials." The regulation includes specific security and performance requirements that when adequately implemented, are designed to protect nuclear power reactors against acts of radiological sabotage, prevent the theft or diversion of special nuclear material, and protect safeguards information against unauthorized release.

The performance requirements for the physical protection of nuclear power reactors are provided in 10 CFR 73.1(a)(1), which bounds the adversarial characteristics of the design-basis threat (DBT), and 10 CFR 73.55. Pursuant to 10 CFR 50.34(c)(2), 50.34(d), 50.54(p)(1) and (p)(2), and 73.55(c)(4), as referenced in 10 CFR Part 52, applicants are required for facility

licenses to prepare and maintain security plans that describe the security-related actions that they will take to protect their facilities against acts of radiological sabotage.

Regulatory requirements and acceptance criteria related to physical protection systems or hardware are, in part, applicable to design certification (i.e., within scope of the design) or may only be applicable to a COL Applicant (outside of a design certification design scope) and are specified in NUREG–0800, SRP Section 14.3.12.

The COL Applicant is required to describe commitments for establishing and maintaining a physical protection system (engineered and administrative controls), organization, programs, and procedures for implementing a site-specific strategy that demonstrates, if adequately implemented, a high assurance of protection of the plant against the DBT. The site-specific physical protection system described must be reliable and available and must implement the concept of defense-in-depth protection in order to provide a high assurance of protection. The security operational programs and the physical protection system are required to meet specific performance requirements of 10 CFR Part 26, “Fitness For Duty Programs”, 10 CFR Part 74, “Materail Control and Accounting of Special Nuclear Material”, and 10 CFR 73.55, 73.56, 73.57, and 73.70 . The COL Applicant’s security program and planning for a safeguard contingency are required to meet 10 CFR 50.34(d) and Appendix C of 10 CFR Part 73, “Physical Protection of Plants and Materials”.Nuclear Power Plant Safeguards Contingency Plans”. The training and qualification program for security personnel and responders is required to meet performance and specific requirements of 10 CFR Part 73, Appendix B, “General Criteria for Security Personnel”. Within this context, the applicant need only address those elements or portions of physical protection systems or features that are considered within the scope of the certified portion of the design. The technical basis for physical protection hardware within the scope of the certified portion of the design must provide the basis for ITAAC acceptability and adequacy.

GEH submitted the following ITAAC for detection and assessment hardware in ESBWR DCD Tier 1, Section 2.19:

2. Physical barriers for the protected area perimeter are not part of the vital area barrier.

3. Isolation zones exist in outdoor areas adjacent to the physical barrier at the perimeter of the protected area that allows 20 feet of observation on either side of the barrier.

4. The intrusion detection system can detect penetration or attempted penetration of the protected area barrier.

6. The external walls, doors, ceiling, and floors in the MCR, the central alarm station (CAS), and the last access control function for access to the protected area are resistant to at least an Underwriter’s Laboratories (UL) level-IV round.

9. An access control system with numbered picture badges is installed for use by individuals who are authorized access to protected areas without escort.

10. Unoccupied vital areas are locked and alarmed with activated intrusion detection systems that annunciate in the Central and Secondary Alarm Stations upon intrusion into a vital area.

11. Security alarm annunciation occurs in the central alarm station and in at least one other continuously manned station not necessarily onsite.

14. Equipment exists to record onsite security alarm annunciation including the location of the alarm, false alarm, alarm check, and tamper indication and the type of alarm, location, alarm circuit, date, and time.

As a result of its review of the ITAAC for Detection and Assessment Hardware, the staff determined that GEH had submitted ITAAC within the DCD that are not within the scope of the design certification and that should be submitted as part of a COL application. Furthermore, the staff needed additional information to complete its review.

In RAI 14.3-440, the staff requested in part that GEH revise the physical security hardware ITAAC in Tier 1 of the DCD in accordance with the approach discussed with the Nuclear Energy Institute (NEI) on October 21, 2008, and consistent with SRP Section 14.3.12. In the RAI, staff indicated that ITAAC Items 2, 3, 4, and 9 for Detection and Assessment Hardware are not within the scope of the certified design.

In response, GEH proposed to revise DCD Tier 1, Section 2.19 and DCD Tier 2, Section 13.6 to delete any items that are outside the scope of the certified design. GEH removed ITAAC Items 3, 4, and 9, which will be submitted by COL Applicants using the same wording. For Item 2, the second required barrier will be addressed by the COL Applicant in an ITAAC that will be provided for the site-specific design elements of Plant Security. In this case, the COL submittal for Item 2, which the applicant deleted in response to this RAI, will have words to the effect of "Physical barriers for the protected area perimeter are not part of vital area barrier and provide one of the two required physical barriers to vital equipment access." GEH revised Item 6 to remove the specificity of bullet resistance of the last access control function for access to the protected area and to apply bullet resistance to at least a UL level-4 round to the external walls, doors, ceiling, and floors of the MCR and the CAS. The applicant revised Item 10 to exclude the secondary alarm station from the locations where the intrusion alarm will annunciate. The COL Applicant will submit new information for Items 6 and 11 to cover the information deleted.

Upon review of the response to RAI 14.3-440, the staff finds the revision to the ITAAC for detection and assessment hardware to be acceptable because it is in conformance with the staff's definition of physical security hardware ITAAC that is within the scope of the design certification, and the ITAAC are sufficient to verify that the hardware, as finally installed and constructed will function as design.

GEH submitted the following ITAAC for Delay or Barrier Design Features in ESBWR DCD Tier 1, Section 2.19:

5. Isolation zones and exterior areas within the protected area are provided with illumination to permit observation of abnormal presence or activity of persons or vehicles.

7. The vehicle barrier system is installed and located at the necessary stand-off distance to protect against the DBT vehicle bombs.

8. Access control points are established to:

    a. Control personnel and vehicle access into the protected area.

b. Detect firearms, explosives, and incendiary devices at the protected area access points.

13. Security - all alarm devices including transmission lines to annunciators are tamper indicating and self-checking (e.g., an automatic indication is provided when the failure of the alarm system or a component occurs, or when on standby power). Alarm annunciation shall indicate the type of alarm (e.g., intrusion alarms, emergency exit alarm, etc.) and location.

As a result of its review of the ITAAC for delay or barrier design features, the staff determined that GEH submitted ITAAC within the DCD that are not within the scope of the design certification and should be submitted as part of a COL application. Furthermore, the staff needed that additional information to complete its review.

In RAI 14.3-440, the staff also indicated that ITAAC Items 7 and 8 for delay or barrier design features are not within the scope of the certified design.

In response to portions of RAI 14.3-440, GEH proposed to revise DCD Tier 1, Section 2.19 by deleting ITAAC Items 7 and 8. GEH indicated that ITAAC Items 7 and 8 will be submitted by the COL Applicants. GEH also deleted ITAAC Item 5 since it was outside the scope of the certified design, even though DCD Tier 2, Section 13.6 provides design criteria for the illumination levels. COL Applicants will submit ITAAC Items 5, 7, and 8 using the same wording. The applicant revised ITAAC Item 13 for clarity.

Upon review of the applicant’s response to RAI 14.3-440, the staff finds the revised ITAAC for delay or barrier design features to be acceptable because they conform to the staff’s definition of physical security hardware ITAAC that is within the scope of the design certification, and the ITAAC are sufficient to verify that the hardware, as finally installed and constructed, will function as designed.

GEH submitted the following ITAAC for systems, hardware, or features facilitating a security response and neutralization in ESBWR DCD Tier 1, Section 2.19.

1. Vital equipment:

   a. Vital equipment shall be located only within a vital area.

   b. Access to vital equipment requires passage through at least two physical barriers.

12. A secondary security power supply system for alarm annunciator equipment and non-portable communications equipment is located within a vital area.

15. Emergency exits through the protected area perimeter and vital area boundaries are alarmed.

16. Central and secondary alarm stations:

   a. Central and secondary alarm stations have conventional (land line) telephone service and other communication capabilities with local law enforcement authorities.

b. Central and secondary alarm stations are capable of continuous communication with security personnel.

As a result of its review of the ITAAC for systems, hardware, or features facilitating security response and neutralization, the staff determined that GEH has submitted ITAAC within the DCD that are not within the scope of the design certification and that should be submitted as part of a COL application. Furthermore, the staff needed additional information to complete its review.

In RAI 14.3-440, the staff requested that GEH revise the physical security hardware ITAAC in Tier 1 of the DCD regarding systems, hardware, or features facilitating security response and neutralization in accordance with the approach discussed with NEI on October 21, 2008, and consistent with SRP Section 14.3.12.

In response, GEH revised ITAAC Item 1 to indicate that access to vital equipment requires passage through a vital area barrier that prevents unauthorized access. GEH provided a revision to DCD Tier 2, Section 13.6.1.1.2 to add the performance standard submitted in response to other areas of RAI 14.3-440.

GEH also revised Item 12 to specify that the secondary security power supply for alarm annunciator equipment and non-portable communications equipment in the CAS is located in a vital area.

GEH revised Item 15 by removing the requirement that protected area perimeter emergency exits are alarmed.

GEH revised Item 16 by removing communication requirements for the secondary alarm station.

GEH further indicated that the COL Applicant will submit new ITAAC Items 12, 15, and 16 to cover the deleted information.

In RAI 14.3-440 S01, the staff requested that GEH revise the physical security hardware ITAAC in Tier 1 of the DCD to address the Part 73 Power Reactor Security Requirements Final Rule and to reflect these changes in Tier 1 and Tier 2 of the ESBWR DCD. In the same RAI, the staff requested that GEH describe the ITAAC that are not within the scope of the ESBWR design.

In response to RAI 14.3-440 S01, GEH revised DCD Tier 1, Section 2.19 with the following ITAAC:

1 a. Vital equipment is located only within a vital area.

1 b-1. Access to vital equipment requires passage through a vital area barrier.

6 a. The external walls, doors, ceiling, and floors in the MCR and CAS are bullet resistant to at least a UL 752 (2006) Level 4.

10 a. Unoccupied vital areas are locked and alarmed with activated intrusion detection systems that annunciate in the central alarm station.

11 b-1. The central alarm station is located inside a protected area and the interior is not visible from the perimeter of the protected area.

12 a. The secondary security power supply system for alarm annunciator equipment contained in the central alarm station and non-portable communications equipment contained in the central alarm station is located within a vital area.

13 a. Security alarm devices including transmission lines to annunciators are tamper indicating and self-checking (e.g., an automatic indication is provided when the failure of the alarm system or of a component occurs or when on standby power) and alarm annunciation shall indicate the type of alarm (e.g., intrusion alarms, emergency exit alarms) and the location.

13 b-1. Intrusion detection and assessment systems provide visual display and audible annunciation of the alarm in the central alarm station.

14 a. Intrusion detection systems recording equipment exists to record onsite security alarm annunciation including the location of the alarm, false alarm, alarm check, and tamper indication and the type of alarm, location, alarm circuit, date, and time.

15 a. Emergency exits through the vital area boundaries are alarmed and secured by locking devices that allow prompt egress during an emergency.

16 a-1. The central alarm station has conventional (land line) telephone service and other communication capabilities with the control room and local law enforcement authorities.

16 b-1. The central alarm station is capable of continuous communication with security personnel.

16 c-1. Non-portable communications equipment in the central alarm station must remain operable from an independent power source in the event of the loss of normal power.

In response to RAI 14.3-440 S01, GEH also added a COL information item to DCD Tier 2, Section 13.6.3. COL Information Item 13.6-20-A indicates that the COL Applicant shall provide the plant and site-specific physical security ITAAC not covered by DCD Tier 1, Section 2.19.

The staff concludes that the GEH has adequately described the Tier 1 ITAAC physical security hardware to be incorporated as part of the standard design. GEH adequately described the plant layout and protection of vital equipment in accordance with the requirements of 10 CFR 73.55 and provided the technical bases for establishing a physical protection system for protection against acts of radiological sabotage. GEH adequately described requirements specific to the design for alarm annunciation records in accordance with 10 CFR 73.70(f). The applicant provided adequate descriptions of objectives, prerequisites, test methods, data required, and acceptance criteria for security-related ITAAC for the certification the ESBWR design. Therefore, the staff concludes that the ESBWR ITAAC within the scope of SRP Section 14.3.12 are necessary and sufficient to assure that with respect to these ITAAC, if the inspections, tests, and analyses are performed and the acceptance criteria are met, a facility referencing the certified ESBWR design has been constructed and will be operated in compliance with the design certification and applicable regulations.

14.3.13 Conclusion

This report documents the staff's review of ESBWR DCD Tier 1, Revision 9, and DCD Tier 2, Section 14.3 which was performed in accordance with the SRP. Based on its review of the ESBWR DCD and the applicant's responses to RAIs issued on Tier 1 and Tier 2 material, the staff finds that the applicant's selection criteria and methodology for the development of Tier 1 information, the implementation of this selection criteria and methodology, and whether the resultant ITAAC are adequate to verify that a facility referencing the ESBWR design has been constructed and will be operated in compliance with the design certification and applicable regulations.

ITAAC SECTION	DCD TIER 1 SECTION TITLE	SRP SECTION	BRANCH(ES)
2.1.1	Reactor Pressure Vessel System	14.3.2 14.3.4	CIB2, EMB, SRSB
2.1.2	Nuclear Boiler System	14.3.3 14.3.4	EMB, CIB2, SRSB
2.2.1	Rod Control and Information System	14.3.5	ICE2
2.2.2	Control Rod Drive System	14.3.3 14.3.4 14.3.5	SRSB CIB2 EMB
2.2.3	Feedwater Control System	14.3.4 14.3.5	ICE2
2.2.4	Standby Liquid Control System	14.3.3 14.3.4 14.3.5	SRSB EMB CIB2
2.2.5	Neutron Monitoring System	14.3.4 14.3.5	ICE2 SRSB
2.2.6	Remote Shutdown System	14.3.5	ICE2
2.2.7	Reactor Protection System	14.3.5	ICE2
2.2.8	Plant Automation System	No Entry	ICE2
2.2.9	Steam Bypass and Pressure Control System	14.3.5	ICE2
2.2.10	Safety-Related Distributed Control and Information System	14.3.5	ICE2
2.2.11	Non-Safety-Related Distributed Control and Information System	14.3.5	ICE2
2.2.12	Leak Detection and Isolation System	14.3.3 14.3.4 14.3.5	ICE2
2.2.13	Engineered Safety Features System Logic and Control System	14.3.5	ICE2
2.2.14	Diverse Instrumentation and Controls	14.3.5	ICE2
2.2.15	Instrumentation and Control Compliance with IEEE Std 603	14.3.5	ICE2
2.3.1	Process Radiation Monitoring System	14.3.5 14.3.8	ICE2 CHPB

ITAAC SECTION	DCD TIER 1 SECTION TITLE	SRP SECTION	BRANCH(ES)
2.3.2	Area Radiation Monitoring System	14.3.5 14.3.8	ICE2 CHPB
2.4.1	Isolation Condenser System	14.3.2 14.3.3 14.3.4	SRSB EMB CIB2
2.4.2	Emergency Core Cooling System – Gravity-Driven Cooling System	14.3.2 14.3.3 14.3.4	SRSB EMB CIB2
2.5.1	Fuel Servicing Equipment	No Entry	
2.5.2	Miscellaneous Servicing Equipment	No Entry	
2.5.3	Reactor Pressure Vessel Servicing Equipment	No Entry	
2.5.4	RPV Internals Servicing Equipment	No Entry	
2.5.5	Refueling Equipment	14.3.2 14.3.7	SBPB SRSB
2.5.6	Fuel Storage Facility	14.3.2 14.3.7	SBPB SRSB
2.5.7	Under-Vessel Servicing Equipment	No Entry	
2.5.8	FMCRD Maintenance Area	No Entry	
2.5.9	Fuel Cask Cleaning	No Entry	
2.5.10	Fuel Transfer System	14.3.7	SBPB
2.5.11	Deleted		
2.5.12	Deleted		
2.6.1	Reactor Water Cleanup/Shutdown Cooling System	14.3.3 14.3.4	SRSB CIB2 EMB
2.6.2	Fuel And Auxiliary Pools Cooling System	14.3.3 14.3.7	SBPB EMB CIB2
2.7.1	Main Control Room Panels	14.3.5 14.3.9	COLP
2.7.2	Radioactive Waste Control Panels	No Entry	
2.7.3	Local Control Panels And Racks	14.3.5 14.3.9	ICE 2

ITAAC SECTION	DCD TIER 1 SECTION TITLE	SRP SECTION	BRANCH(ES)
2.8.1	Fuel Rods and Bundles (DELETED)		
2.8.2	Fuel Channel (DELETED)		
2.9	Control Rods (DELETED)		
2.10.1	Liquid Waste Management System	14.3.7 14.3.8	SBPB CHPB
2.10.2	Solid Waste Management System		SBPB CHPB
2.10.3	Gaseous Waste Management System	14.3.7 14.3.8	SBPB CHPB
2.11.1	Turbine Main Steam System	14.3.3 14.3.7	SBPB CIB2 EMB
2.11.2	Condensate and Feedwater System	14.3.3 14.3.4 14.3.7	SBPB EMB
2.11.3	Condensate Purification System	No Entry	CSGB SBPB
2.11.4	Main Turbine System	14.3.6 14.3.7	SBPB CIB2
2.11.5	Turbine Gland Seal System	14.3.7	SBPB
2.11.6	Turbine Bypass System	14.3.3 14.3.7	SBPB
2.11.7	Main Condenser	14.3.7	SBPB
2.11.8	Circulating Water System	No Entry	SBPB
2.11.9	Power Cycle Auxiliary Water Systems	No Entry	SBPB
2.12.1	Makeup Water System	14.3.3 14.3.7	SBPB
2.12.2	Condensate Storage and Transfer System	No Entry	
2.12.3	Reactor Component Cooling Water System	14.3.3 14.3.7	SBPB
2.12.4	Turbine Component Cooling Water System	No Entry	
2.12.5	Chilled Water System	14.3.3 14.3.7	SBPB

ITAAC SECTION	DCD TIER 1 SECTION TITLE	SRP SECTION	BRANCH(ES)
2.12.6	Oxygen Injection System	No Entry	
2.12.7	Plant Service Water System	14.3.3 14.3.7	SBPB
2.12.8	Service Air System	14.3.3 14.3.7	SBPB
2.12.9	Instrument Air System	No Entry	
2.12.10	High Pressure Nitrogen Supply System	14.3.3 14.3.7	SBPB
2.12.11	Auxiliary Boiler System	No Entry	
2.12.12	Potable Water System	No Entry	
2.12.13	Hydrogen Water Chemistry System (option)	No Entry	
2.12.14	Process Sampling System	No Entry	
2.12.15	Zinc Injection System	No Entry	
2.12.16	Freeze Protection	No Entry	
2.12.17	Station Water System	No Entry	
2.13.1	Onsite AC Power System	14.3.6	EEB
2.13.2	Electrical Wiring Penetrations (see 2.15.1 and 2.16.3.1) (DELETED)	14.3.2 14.3.6 14.3.11	EEB
2.13.3	Direct Current Power Supply	14.3.5 14.3.6	EEB
2.13.4	Onsite Diesel Generator Power Supply Systems	14.3.5 14.3.6	EEB
2.13.5	Uninterruptible AC Power Supply	14.3.5 14.3.6	EEB
2.13.6	Instrument and Control Power Supply	Deleted	
2.13.7	Communication System	No Entry	ICE 2
2.13.8	Lighting Power Supply	14.3.5 & 6	EEB
2.14	Power Transmission	No Entry	EEB

ITAAC SECTION	DCD TIER 1 SECTION TITLE	SRP SECTION	BRANCH(ES)
2.15.1	Containment System	14.3.2 14.3.3 14.3.6 14.3.11	SEB2 EMB SBCV CIB2 EEB
2.15.2	Containment Vessel (see 2.15.1)	-	NA
2.15.3	Containment Internal Structures	14.3.2 14.3.3 14.3.4 14.3.11	SEB2 SBCV
2.15.4	Passive Containment Cooling System	14.3.3 14.3.4 14.3.11	SBCV EMB CIB2
2.15.5	Containment Inerting System	14.3.11	SBCV
2.15.6	Drywell Cooling System	No Entry	SBCV
2.15.7	Containment Monitoring System	14.3.11 14.3.8	ICE2 SBCV
2.16.1	Cranes, Hoists and Elevators	14.3.7	SBPB
2.16.2	Heating, Ventilating and Air-Conditioning Systems	14.3.7	SBCV
2.16.3	Fire Protection System	14.3.7	SFPB
2.16.4	Equipment and Floor Drain System	14.3.7	SBPB
2.16.5	Reactor Building	14.3.2 14.3.5 14.3.6 14.3.7	RSAC SFPB SBPB SEB2 SBCV
2.16.6	Control Building	14.3.2 14.3.5 14.3.6 14.3.7	SBCV, SFPB, SBPB, EGCB
2.16.7	Fuel Building	14.3.2 14.3.7	SFPB, SBPB, SEB2 SBCV
2.16.8	Turbine Building	No Entry	
2.16.9	Radwaste Building	No Entry	
2.16.10	Other Buildings and Structures	No Entry	

ITAAC SECTION	DCD TIER 1 SECTION TITLE	SRP SECTION	BRANCH(ES)
2.17.1	Intake and Discharge Structure	No Entry	
2.18.1	Oil Storage and Transfer Systems	No Entry	
2.18.2	Site Security	No Entry	
2.19	Plant Security System	14.3.12	NSIR
3.1	Design of Piping Systems and Components	14.3.3	EMB
3.2	Software Development	14.3.5 14.3.9	ICE2
3.3	Human Factors Engineering	14.3.9	COLP
3.4	Radiation Protection	14.3.8	CHPB
3.5	Initial Test Program	14.2	CQVB
3.6	Design Reliability Assurance Program	14.3	SPLB
3.7	Post Accident Monitoring Instrumentation	14.3.5	ICE2
3.8	Environmental Qualification of Mechanical and Electrical Equipment	14.3.3 14.3.5 14.3.6 14.3.7	EEB CIB2
4	**Interface Requirements**	1.0	
4.1	Plant Service Water System	1.0 14.3.7	SBPB
5	**Site Parameters**	2.0	RSAC RGS1 RHEB

15. TRANSIENT AND ACCIDENT ANALYSES

15.1 Introduction

In the design control document (DCD) Tier 2, Revision 9, Chapter 15, the applicant discussed the analysis of various anticipated operational occurrences (AOOs) and accidents. The system response analyses are based on the equilibrium core (EC) described in Chapter 4 of DCD and core loading documented in NEDC-33239-P, “Global Nuclear Fuel, GE14 for ESBWR Nuclear Design Report,” Revision 4, March 2009. The staff reviewed the economic simplified boiling-water reactor (ESBWR) transient and accident analyses in accordance with Chapter 15, “Accident Analysis,” of NUREG–0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants (LWR Edition)” (hereafter referred to as the SRP), Draft Revision 3, issued April 1996. The staff used the later version of SRP Section 15.0, Revision 3, issued March 2007, only for defining different event categories.

Because the ESBWR design is based on natural circulation, active components designed to ensure a continuous supply of cooling water (as in the current vintage of boiling-water reactors [BWRs]) are not used. Therefore, a number of transients and accidents do not apply to the ESBWR. In this sense, the ESBWR design is unique. In addition, GE-Hitachi Nuclear Energy (GEH or the applicant) effected a series of instrumentation, mechanical, and electrical design improvements that modified the probability of occurrence of AOOs and accidents. This forced a recategorization of all events. DCD Section 15A is reviewed and approved by the staff in this report.

Because of the uniqueness of the ESBWR design and the recategorization of events in Chapter 15, this review does not strictly follow the SRP for all events. For example, according to Draft Revision 3 of the SRP (1996)[1], all the reactivity transients are AOOs, and the corresponding acceptance criterion of safety limit minimum critical power ratio (SLMCPR) is used. However, in the ESBWR design, most of the reactivity transients (except for control rod withdrawal during startup and power operation) are considered by the applicant and the staff to be in the infrequent category, based on event frequency (a subset of accident category), and hence, 0.025 Sievert (Sv) (2.5 Roentgen equivalent in man [rem]) total effective dose equivalent (TEDE) (10 percent of the dose acceptance criteria specified in Title 10 of the *Code of Federal Regulations* (10 CFR) 52.47(a)(2)(iv)(A)) is used as the acceptance criterion.

15.1.1 Event Categorization

The SRP divides events into AOOs and postulated accidents. The requirements of Appendix A, “General Design Criteria for Nuclear Power Plants,” to 10 CFR Part 50, “Domestic Licensing of Production and Utilization Facilities,” define AOOs as those conditions of normal operation expected to occur one or more times during the life of the nuclear power unit.

SRP Section 15.0, Revision 3, defines postulated accidents as “Unanticipated conditions of operation (i.e., not expected to occur during the life of the nuclear power unit).” DCD Tier 2, Revision 9, Section 15.0, presents the ESBWR transient and accident analysis methodology used by GEH.

GEH proposed a new subcategory of events—infrequent events (IEs)—under the broad category of accidents. GEH proposed this recategorization of events because of the unique

1 The SRP referenced in each Section is the latest revision applicable to that Section.

passive cooling design of the ESBWR, the anticipated lower frequency of event occurrence, and the unique design features, such as the four divisions of safety systems and three channels of process systems (e.g., feedwater control system [FWCS]), which are redundant and fault tolerant. These design features could reduce the frequency of design-basis events (DBEs). In the SRP Section 15.0, Revision 3, design-basis events are defined as follows:

Conditions of normal operation, including AOOs, design-basis accidents (DBAs), external events, and natural phenomena, for which the plant must be designed to ensure functions of safety-related electric equipment that ensures the integrity of the reactor coolant pressure boundary (RCPB); the capability to shutdown the reactor and maintain in a safe shutdown condition; or the capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposures.

As part of the pre-application review, GEH submitted topical report NEDO-33175, "Classification of ESBWR Abnormal Events and Determination of Their Safety Analysis Acceptance Criteria," Revision 1, issued February 2005. In the NEDO report, GEH reviewed the regulatory criteria for event classification for the ESBWR passive plant design to determine the appropriate abnormal event classifications and their associated safety analysis acceptance criteria. GEH provided additional information related to classification of events in its responses to U.S. Nuclear Regulatory Commission (NRC) staff requests for additional information (RAIs).

New initiating events that require consideration within the scope of accidents and transients may result from the new and unique design features of the ESBWR. For example, the original DCD did not include events such as inadvertent actuation of the control rod drive system (CRDS) in the injection mode to the reactor pressure vessel (RPV) or the gravity-driven cooling system (GDCS) inadvertent injection into the reactor vessel. The staff issued RAI 15.0-1 to request the applicant to identify all possible transients and accidents that may result from the unique design features of the ESBWR. In its response GEH stated that it performed such a study, and the results of this systematic review are listed in the RAI response Table 15.0-1. The categorization adheres to the guidance of Regulatory Guide (RG) 1.70, Revision 3, November 1978, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants," i.e., to ensure consideration of systems effects. In addition, GEH stated that the CRDS inadvertent initiation event consequences are bounded by the inadvertent initiation of the isolation condenser system (ICS). GEH also presented the results of a study confirming that all equipment, including the CRDS, in the ESBWR was reviewed to determine whether credible failures in the system or operator error could initiate a new type of DBE. In summary, GEH performed the requested study, which covered all the ESBWR systems and addressed possible new events resulting from the unique design features of the ESBWR; therefore, RAI 15.0-1 is resolved.

The evaluation covered the following event categories:

- Increase in heat removal by the secondary system
- Decrease in heat removal by the secondary system
- Decrease in reactor flow rate
- Reactor reactivity and power distribution anomalies
- Increase in reactor coolant inventory
- Decrease in reactor coolant inventory
- Radioactive release from a subsystem or component

The following sections evaluate the acceptance criteria for four groups of DBEs (AOOs, accidents [Infrequent Events, DBAs], and special events).

15.1.1.1 *Anticipated Operational Occurrences*

AOOs are expected during the life of the plant and require analyses to ensure that they will not cause damage to either the fuel or the RCPB or lead to a worse plant condition.

The designed lifetime of the ESBWR plant is 60 years. In its evaluation, GEH conservatively assumed the plant to operate for 100 years. The conservative definition (as proposed by the applicant and accepted by the staff) of AOOs for the ESBWR includes events with a frequency greater than or equal to 1.0×10^{-2} per reactor year (pry). The acceptance criteria for the AOOs, as given in the SRP, are the following:

- General Design Criterion (GDC) 10, "Reactor design," in Appendix A to 10 CFR Part 50, as it relates to the reactor coolant system (RCS) design having appropriate margin to ensure that the plant does not exceed specified acceptable fuel design limits (SAFDLs) during AOOs

- GDC 13, "Instrumentation and control," which requires the availability of instrumentation to monitor variables and systems over their anticipated ranges for normal operation, for AOOs, and for accident conditions to ensure adequate safety, as well as appropriate controls to maintain these variables and systems within prescribed operating ranges

- GDC 15, "Reactor coolant system design," as it relates to the RCS design having appropriate margin to ensure against breach of the pressure boundary during AOOs

- GDC 17, "Electric power systems," as it relates to providing onsite and offsite electric power systems to ensure that structures, systems, and components (SSCs) important to safety will function during normal operation, including AOOs, and to ensuring sufficient capacity and capability to prevent the reactor from exceeding SAFDLs and design conditions of the RCPB during an AOO

- GDC 20, "Protection system functions," as it relates to the reactor protection system (RPS) being designed to initiate automatic operation of reactivity control systems to ensure that the reactor does not exceed SAFDLs during AOOs

- GDC 25, "Protection system requirements for reactivity control malfunctions," which requires that the RPS design will ensure that the reactor does not exceed SAFDLs in the event of a single malfunction of the reactivity control system

- GDC 26, "Reactivity control system redundancy and capability," as it relates to the system providing reliable control of reactivity changes by accounting for the appropriate margin for malfunctions, such as stuck control rods, to ensure that the reactor does not exceed SAFDLs during AOOs

The specific criteria necessary to meet the requirements of the GDC include the following:

- The plant maintains the reactor water level above the top of active fuel (TAF).

- The plant design should maintain fuel cladding integrity by ensuring that the minimum critical power ratio (MCPR) remains above the applicable staff approved value of the SLMCPR.

- The plant design should maintain pressure in the reactor coolant and main steam systems below 110 percent of the design value 9.58 Megapascals (MPa) (1,375 pounds per square inch gauge [psig]).

- AOOs should not lead to a worse situation without another failure or operator error.

The substantive requirements summarized above apply to every AOO analyzed in Section 15.2 of the DCD. Evaluation of each AOO considers how the requirements are met.

In RAI 15.0-16, staff requested GEH to include the SLMCPR in the technical specifications (TSs). It is the staff's position that the SLMCPR numerical value should be kept as a safety limit in the TS as in the BWR Standard TS. GEH stated that for the ESBWR TRACG methodology, the transient delta-critical power ratio (CPR) uncertainty is inherently combined with the uncertainties included in the evaluation of the conventional BWR SLMCPR. This process allows for the direct calculation of the number of rods subject to boiling transition (NRSBT) for a transient occurring from an initial operating condition corresponding to the operating limit minimum critical power ratio (OLMCPR). Therefore, the NRSTB parameter becomes the cornerstone of the ESBWR TRACG methodology instead of the SLMCPR, which does not inherently exist for the ESBWR methodology. The staff reviewed the applicant's response to the RAI and found it to be unacceptable.

The staff based its position on the following:

- Allowing the removal of the SLMCPR eliminates regulatory control of core analysis issues and takes away a mechanism for the staff to apply conditions that might be needed in some situations to ensure safety. The NRC previously considered and rejected the same request (i.e., removal of the SLMCPR from the TS) from the Boiling-Water Reactor Owners Group and Exelon.

- Use of TRACG for calculating the OLMCPR is not an appropriate basis for removing the SLMCPR from the TS. In its response, GEH referred to the ESBWR TRACG methodology used for the ESBWR OLMCPR calculation. GEH stated that this process allows for the direct calculation of the NRSBT for a transient. GEH maintained that because the SLMCPR is not used to calculate the OLMCPR, it is appropriate not to include the SLMCPR in the TS as assurance that the ESBWR meets the SAFDLs. The staff does not find use of the TRACG methodology to calculate the OLMCPR to be an appropriate basis for excluding the SLMCPR from the TS.

The NRC approved the TRACG methodology for calculating the OLMCPR in the past for BWRs/2–6, and the licensees that currently use the TRACG methodology for calculating the OLMCPR must still have an SLMCPR TS. Specifically, 10 CFR 50.36(d)(1)(i)(A) states, "Safety limits for nuclear reactors are limits upon important process variables that are found to be necessary to reasonably protect the integrity of certain of the physical barriers that guard against the uncontrolled release of radioactivity." The staff has interpreted this section as requiring that the values of the safety limits must remain in a licensee's TS.

The applicant revised their TS Section 2.1.1.2 (Revision 3) which proposed to replace the MCPR safety limit values with a description of what the safety limit protects against (i.e., "greater than 99.9 percent of the fuel rods in the core would be expected to avoid boiling transition"). The proposed description is a fuel condition and is not an acceptance criterion.

The staff disagreed with the proposed change since it was not consistent with the staff's interpretation of 10 CFR 50.36(d)(1)(i)(A).

GEH responded that although using the ESBWR TRACG fuel cladding integrity safety limit reactor core safety limit terminology ensures protection of the fuel cladding for AOOs, it is recognized that a separate lower bound on the steady-state MCPR (i.e., SLMCPR) protects the fuel cladding when the MCPR is not within its limiting condition for operation (LCO) specification. A potential violation of the reactor core safety limit would occur only if the newly defined ESBWR SLMCPR is violated during steady-state operations, or if an AOO occurs when the MCPR is not within its LCO specification. For both of these situations, the process variable MCPR could be used. GEH revised its original response to RAI 15.0-16. The staff reviewed GEH's revised response to RAI 15.0-16 S01, which includes TS Section 2.1.1.1 and the proposed value of the SLMCPR. Based on the applicant's response, RAI 15.0-16 is resolved.

15.1.1.2 *Accidents or Infrequent Events (IEs)*

SRP Section 15.0 defines DBEs as all transients with a frequency that is less than 1.0×10^{-2} pry that may occur during the lifetime of the plant. GEH defines IEs as events with a frequency of less than 1.0×10^{-2} pry; therefore the staff considers IEs as DBEs. The DBE criteria for acceptance include radiological consequence less than that of DBAs. DBAs are defined as postulated accidents that are used to set design criteria and limits for the design and sizing of safety-related systems and components.

GEH submitted DCD Tier 2, Revision 9, Appendix 15A, providing the determination of the event frequency of the IEs. Section 15.A of this report presents the staff's evaluation of the event frequency determination.

The applicant proposed the inclusion of 16 events in this new category. These events include reactivity, power and pressure anomalies such as control rod withdrawal error (RWE), mislocation and misorientation of fuel bundles, and generator load rejection with total bypass failure.

Some of these events are traditionally designated as AOOs for current BWRs, and some of them are new events. Since the acceptance criteria include radiological consequences, IEs are considered as accidents. Acceptance criteria for IEs are bounded by the same criteria that govern accidents. The acceptance criteria for IEs are the following:

- The plant maintains the reactor water level above the TAF.

- The RCPB pressure is less than 10.44 MPa (1,500 psig). This represents the American Society of Mechanical Engineers (ASME) Code Pressure Service Level C, where 10.44 MPa (1,500 psig) is 120 percent of the RCS design pressure 8.72MPa (1,250 psig).

- The radiological consequence is less than 0.025 Sv (2.5 rem) total effective dose equivalent (TEDE), which is, 10 percent of the dose acceptance criteria specified in 10 CFR 52.47(a). The dose acceptance criteria in Table 1 of SRP Section 15.0.3 are fractions of the 10 CFR 52.47(a) dose reference values for accidents other than the loss-of-coolant accident (LOCA), as historically presented. For events having a moderate frequency of occurrence, any release of radioactive material must be such that the calculated offsite doses are a small fraction of the 10 CFR 52.47(a) reference values. The staff has accepted "less than 10 percent" to be a small fraction of the 10 CFR 52.47(a) dose reference values, or 0.025 Sv

(2.5 rem) TEDE (SRP Section 15.0.3, Table 1). The DCD states that 1,000 fuel rods is a bounding number for the fuel damage that meets the 0.025 Sv- (2.5-rem) criterion.

- Staff requested in RAI 15.3-9 that the applicant provide the actual number of rods in transition boiling in DCD Tier 2, Revision 1, Section 15.3.1. In response, GEH stated that the bounding number of rods failed is supported by an engineering evaluation of the number of rods under dryout. This number was estimated based on correlating the number of rods under dryout as a function of the core MCPR. The staff accepts this conclusion. Based on the applicant's response, RAI 15.3-9 is resolved.

- The estimated number of rods in boiling transition is less than half of the assumed 1,000 rods in boiling transition. The calculation of the TEDE assumes 1,000 failed rods. The technical bases for the OLMCPR, SLMCPR, and justification of the 1,000 failed rods are included in NEDE-33083, Supplement 3, "TRACG Application for ESBWR Transient Analysis" dated December 2007.

- The plant maintains containment and suppression pool pressures and temperatures below their design values.

- Control room personnel do not receive an accident dose in excess of 0.05 Sv (5 rem) TEDE for the duration of the event.

The relaxation of the acceptance criteria for less probable events follows the rationale that events assessed as having a high frequency of occurrence must have a small consequence (protection of the SLMCPR), while events assessed as having a lower frequency may have a more severe consequence (i.e., fuel damage may occur, but radiological dose must fall within the limits set forth in 10 CFR 52.47). For current operating BWRs, events with a frequency of less than 1.0×10^{-2} pry may result in cladding failure, fuel failures, or overpressurization. In the ESBWR, there are IEs, such as feedwater controller failure-maximum flow demand, control RWE, and loss of feedwater heating (LOFWH)), that have a frequency of occurrence of less than 1.0×10^{-2} pry, yet the consequences are similar to those of AOOs (i.e., the calculated MCPR is above the OLMCPR, reactor pressure is less than the relief valve set pressure, and the cladding strain is less than the allowed limit). Other IEs may result in fuel damage, overpressurization, or cladding damage.

GEH proposed ASME Code Service Level C (120 percent of the design pressure) as the criteria for RCPB pressure. DCD Tier 2, Revision 9, Section 15.0.1.2(4), defines an accident as a postulated DBE not expected to occur during the lifetime of the plant and with radiological releases not to exceed the calculated exposure in 10 CFR 52.47(a). The DCD also states that an accident equates to ASME Code Service Level C or D acceptance criteria. The staff is not aware of such equivalency, except for anticipated transients without scram (ATWS). ASME Code service levels require justification on a case-by-case basis in a manner similar to ATWS, and GEH did not provide this justification in its response to RAI 15.0-17 and RAI 15.0-17 S01.

In RAI 15.0-17 S02, the staff stated the DCD should include a commitment to perform post-overpressurization event inspection testing to justify continued operation if any event causes an ESBWR RCS to exceed its ASME Code Service Level B (110 percent) pressure limit. In response to RAI 15.0-17 S02, GEH stated that the safety analyses demonstrate that no DBE can cause an ESBWR RCS to exceed its ASME Code Service Level B pressure limit. However, ASME Code Section XI does require adequate inspections and testing to confirm the operability of the safety-related components potentially affected by the hypothetical pressurization event.

Therefore, the ASME Code is used as the basis for the requested post-overpressure event inspections. The applicant updated DCD Tier 2, Section 3.9.3.1.2, in Revision 5 in response to this request. The staff accepts this response based on the requirement for inspections. Based on the applicant's response, RAI 15.0-17 is resolved.

In DCD Tier 2, Section 15.0.1.2(3), the applicant stated, "An infrequent event is defined as a DBE (with or without assuming a single active component failure or single operator error) with probability of occurrence less than 1.0×10^{-2} pry and a radiological consequence less than an accident." In RAI 15.0-26, the staff indicated that the ESBWR IE classification is a subset of the accident category in the SRP and that the radiological consequence of an IE should be less than that of a DBA. The applicant agreed with the staff's comment and revised the text in the DCD to read "radiological consequence less than a design-basis accident." Therefore, based on the applicant's response, RAI 15.0-26 is resolved.

The staff performed independent confirmatory calculations for limiting AOOs with the TRACE/PARCS computer code. Section 21.6 of this report presents the staff's evaluation of the applicant's analyses and the staff's independent calculation results (also discussed in section 6.3.2.3.10 of this report).

The substantive requirements summarized above apply to every IE analyzed in Section 15.3 of this report. Evaluation of each IE considers how the requirements are met.

15.1.1.3 *Design-Basis Accidents*

SRP Section 15.0 defines DBAs as postulated accidents that are used to set design criteria and limits for the design and sizing of safety-related systems and components. The acceptance criteria for accidents include radiological doses less than 0.025 Sv (2.5 rem) TEDE, 0.063 Sv (6.3 rem) TEDE, or 0.25 Sv (25 rem) TEDE (see SRP Section 15.0.3, Table 1), the acceptable radiation dose criteria in 10 CFR 52.47(a)(2), depending on the accident-specific acceptance criterion in Chapter 15 of the SRP.

The DBA category includes the following:

- Fuel-handling accidents (FHAs)
- Main steamline break (MSLB) outside containment
- Control rod drop accident (CRDA)
- Feedwater line break outside containment
- Failure of small line carrying primary coolant outside containment
- Reactor water cleanup/shutdown cooling (RWCU/SDC) system line failure outside containment
- Spent fuel cask drop accident

For LOCAs, the acceptance criteria for the emergency core cooling system (ECCS) specified in 10 CFR 50.46, are as follows:

- The peak cladding temperature must remain below 1,204.4 degrees Celsius (C) (2,200 degrees Fahrenheit [F]).
- For maximum cladding oxidation, the calculated total oxidation of the cladding must nowhere exceed 17 percent of the total cladding thickness before oxidation.
- Total hydrogen generation must not exceed 1.0 percent of the hypothetical amount that would be generated if all the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react.
- The system must maintain the core in a coolable geometry. The system must maintain calculated core temperatures after successful initial operation of the ECCS at acceptably low levels and remove decay heat for the extended period required by the long-lived radioactivity remaining in the core.

Section 6.3 of this report presents the staff evaluation of compliance with 10 CFR 50.46.

15.1.1.4 *Special Events*

The special events category may include a common-mode equipment failure or additional failure beyond the single-failure criteria. ATWS, station blackouts (SBOs), and safe-shutdown fires fall under the special events designation. The acceptance criteria for each of these special events derives from a specific event basis as well as coping, mitigation, and acceptance criteria specified in the NRC regulations and the SRP.

15.1.2 Analytical Methods

TRACG is a multidimensional, two-fluid reactor thermal-hydraulics (T-H) analysis code with a three dimensional neutron kinetics capability. The code is designed to perform transient analyses in a realistic manner with conservatism added, as appropriate, via the input specifications. Section 21.6 of this report provides the staff's evaluation of the assumptions with respect to initial power, scram reactivity, reactivity coefficients, power profiles, and other parameters used in the analyses.

For nuclear analyses, the applicant's suite of codes includes a two-dimensional lattice physics code (TGBLA06) and a three-dimensional core simulator PANAC11. These codes are used in conjunction to perform several analyses to demonstrate ESBWR compliance with GDC. Sections 4.3 and 21 of this safety evaluation report (SER) include additional information on these codes.

GEH transient analyses used the TRACG evaluation model, described in licensing topical report (LTR) NEDE-33083P to analyze only the AOOs in DCD Tier 2, Chapter 15. However, this LTR contained no discussion of IE analyses. In RAI 15.0-27, the staff requested GEH to include discussion of IE analyses.

In response to RAI 15.0-27, GEH submitted Supplement 3 of NEDE-33083, which contains analyses of AOOs, IEs, and special events. Based on the applicant's response, RAI 15.0-27 is resolved.

15.1.3 Nonsafety-Related Systems Assumed in the Analysis

In RAI 15.0-2, the staff requested GEH to provide a list of nonsafety-related systems and equipment credited in the analyses. In response to RAI 15.0-2, GEH submitted a table listing the nonsafety-related systems and equipment used for mitigating transients and accidents described in DCD Tier 2, Chapter 15. In accordance with Criterion 3 specified in 10 CFR 50.36(c)(2)(C), the TS must include limiting conditions of operation (LCOs) for equipment credited in the transient and accident analyses.

In its response to RAI 15.0-2, the applicant described the function of the CRDS as follows:

> Control Rod Drive System (CRDS): The high pressure makeup water function of this system is credited in several event scenarios as backup level control to feedwater. This function of CRDS is nonsafety-related. If credit is not taken for the high pressure makeup water function of the CRDS, then the Isolation Condenser System and Gravity-Driven Cooling System would ensure acceptable inventory control.

In RAI 16.2-33, the staff requested GEH to review the response to RAI 16.0-1 (i.e., bases for the TS) in light of the response to RAI 15.0-2 and identify any changes to TS. In its response to RAI 16.2-33, GEH stated the following:

> Both the RAI 15.0-2 and the RAI 16.0-1 responses indicated that this function is not in the primary success path for mitigating transients and accidents because the safety-related isolation condenser (IC) and GDCS will ensure water inventory is maintained within the acceptance criteria for the applicable event even if the nonsafety-related CRD system makeup water function failed.

The staff requested that GEH revise the DCD to include this information and to include the results of analysis that support this conclusion. The staff also requested the applicant to add a table in Section 15.0 of DCD Tier 2 listing the following nonsafety-related equipment that is credited in the AOO, IE, and/or accident analyses:

- CRDS—makeup water CRDS (not included in the TS)
- Selected control rod run in (SCRRI) (included in the TS)
- Fuel and auxiliary pool cooling system (FAPCS) (not included in the TS)
- FWCS (not included in the TS)
- Rod control and information system (RC&IS) (rod worth minimizer (RWM) and automated thermal limit monitor (ATLM) are included in the TS)
- Steam bypass and control system (included in the TS)

In RAI 15.0-2 S02, the staff requested that GEH confirm that all equipment credited in the analyses be included in the TS. In response to RAI 15.0-2 S02, GEH revised DCD Tier 2, Tables 15.1-5 and 15.1-6, to show the list of nonsafety-grade equipment for which credit was taken in accident analysis.

In RAI 15.0-1 S01 the staff requested GEH to justify why the high-pressure control rod drive (HPCRD) should not be in the TSs. In response to RAI 15.0-1 S01, the applicant stated that "In the case where HPCRD is unavailable for reactor vessel water level control, the system response is similar to the SBO events described in DCD Section 15.5.5 which demonstrates that the level can be maintained above the top of the active fuel with the ICS as the primary success path." DCD Tier 2, Revision 9, Section 15.5.5 presents a description and analysis of SBO where a loss of feedwater flow and control rod drive (CRD) flow is assumed. The SBO analysis bounds the events where operation of the HPCRD is required. Since the acceptance criteria can be met without the HPCRD system, this system is not required to be in the TS. Therefore, RAI 15.0-1 S01 is resolved.

In response to RAI 15.0-2, the applicant stated that the suppression pool cooling mode of FAPCS is credited with long-term cooling of the suppression pool following an inadvertent opening of a safety/relief valve (SRV). With no operation of the FAPCS in the suppression pool cooling mode, the pool would heat up to its scram setpoint and initiate a scram if one has not already occurred. Containment design limits will not be exceeded. Hence, FAPCS is not critical equipment and need not be in the TS.

The applicant, in response to RAI 16.2-33 S01, provided additional information related to the FWCS. The FWCS is credited in the limiting event, Inadvertent Isolation Condenser Initiation (IICI). Failure of the FWCS simultaneously with an IICI event is a detectable and non-consequential random, independent failure, and the automatic function of the FWCS is not in the primary success path for the mitigation of an IICI event.

The applicant stated that nonsafety-related systems or components are assumed to be operational in the following situations:

- When assumption of a nonsafety-related system results in a more limiting transient

- When a detectable and non-consequential random, independent failure must occur in order to disable the system

- When nonsafety-related components are used as backup protection (e.g., the HPCRD system, which is not in the primary success path but is included to illustrate the expected plant response to the event)

In the above circumstances, a nonsafety system failure (1) will not result in a more limiting transient, (2) will occur only when a detectable independent failure disables the system and (3) nonsafety-related systems will be used for backup protection. The staff finds acceptable the assumptions concerning the nonsafety-related systems described above.

Based on the applicant's response, RAIs 15.0-2, 16.0-1, and 16.2-33 are resolved.

15.1.4 Loss of Offsite Power Assumption

In RAI 15.0-4 the staff requested GEH to describe in detail how the ESBWR transient and accident analyses were performed to comply with GDC 17 and in particular, "For new applications, loss of offsite power should not be considered as a single-failure event; rather it should be assumed in the analysis of each event without changing the event category…" GEH addressed compliance with GDC 17 with regard to DCD Tier 2, Chapter 15, analyses in its response to RAI 15.0-4. For 72 hours, no safety-related function requires either offsite

alternating current (ac) power or onsite emergency diesel generator ac power. After 72 hours, the analyses take credit for the nonsafety-related direct current (dc) and ac power. No ESBWR accident analyses assume the availability of offsite power. The ESBWR AOO events do include loss of offsite power. Since the ESBWR has no reactor recirculation pumps that normally receive their power supply from off site, loss of offsite power is not a significant event for the ESBWR. Chapter 8 of this report includes a detailed evaluation of GDC 17. Based on the description of the ESBWR plant features in the applicant's response, RAI 15.0-4 is resolved.

15.1.5 Analysis of Anticipated Operational Occurrences and Infrequent Events for the Initial Core

GEH submitted NEDO-33337, "ESBWR Initial Core Transient and Accident Analyses," Revision 1, April 2009 (which is incorporated into the DCD via reference), which includes the analyses of the initial core and NEDO-33338, Revision 1, "ESBWR Feedwater Temperature Operating Domain Transient and Accident Analysis," issued May 2009 (see DCD Tier 2, Revision 9, Section 15.1.1.9). The staff, with the help of Brookhaven National Laboratory and Oak Ridge National Laboratory (ORNL), respectively, reviewed these LTRs. A summary of the staff's evaluation of the initial core analyses follows.

The evaluation, based on the review of the DCD Tier 2, Chapter 15, incorporates (for each transient) a summary of limiting features from the evaluation of NEDO-33337 and NEDO-33338 The initial core loading (ICL) and the feedwater temperature operating domain (FWTOD) additions (see Figure 15.1-1 of this report) to the DCD were reviewed and evaluated independently of the reference core. It should be noted that the ICL is a simulation of the EC constructed with varying enrichment and poison loadings. These safety evaluations (Sections 15.2 and 15.3 of this report) are based on the reference core DCD Tier 2, Revision 9, with additional segments on the ICL and the FWTOD for each transient. The ICL (NEDO-33337) and the FWTOD core (NEDO-33338) evaluations were reviewed separately.

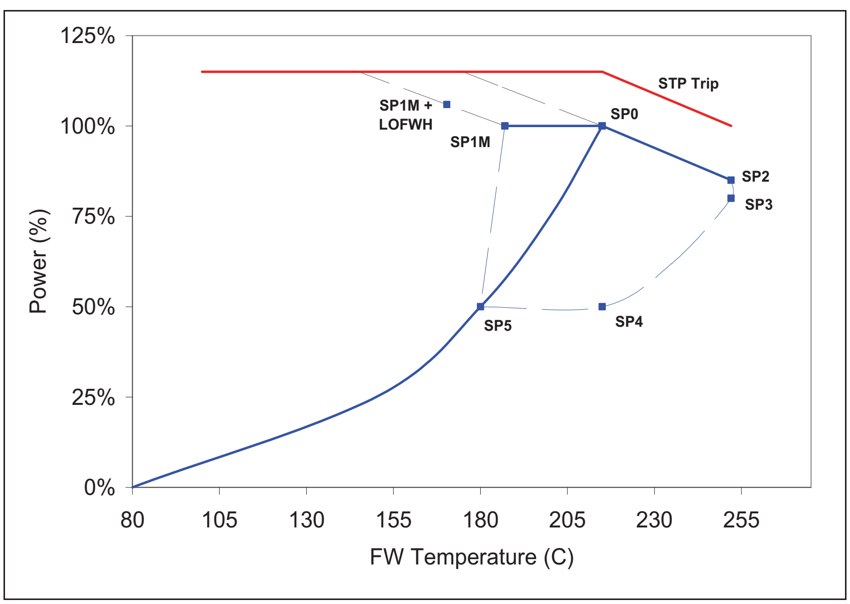

Figure 15.1-1. ESBWR Power-Feedwater Temperature Operating Domain.

The evaluation of NEDO-33337 and NEDO-33338, documented in this report, includes consideration of the limiting characteristics for each transient affected by the ICL and/or the

FWTOD operation so that the most limiting parameters are included. Evaluation of the AOOs and the IEs is based on the acceptance criteria summarized in Sections 15.1.1.1 and 15.1.1.2 of this report.

GEH transient analyses used the TRACG evaluation model, described in LTR NEDE-33083P Supplement 3, to analyze most of the AOOs and IEs. This LTR is based on calculation results for ESBWR AOOs and IEs.

In NEDO-33337, the applicant discussed the analysis of various AOOs. GEH analyzed the following categories of AOOs:

- Decrease in-core coolant temperature
- Increase in reactor pressure
- Reactivity and power distribution anomalies
- Increase in reactor coolant inventory
- Decrease in reactor coolant inventory

In addition, the document discusses IEs and special events. IEs fall under the broad category of accidents, but reflect the unique passive cooling capabilities possible with the ESBWR design. Special events have an extremely low probability of occurrence and, in this case, include ATWS, SBO, etc. The evaluation of the results presented in this report for the ICL is based in large part on the comparison between these results and those obtained for the same transient for the EC.

The primary differences between the ICL and EC are fuel loading and cycle burnup. In an effort to mimic the EC, the ICL has many more fuel types regarding fissile material enrichment. In addition, the ICL has lower gadolinium concentration than the EC. Briefly, the ESBWR fuel assembly has a 10×10 fuel pin structure with two large water rods that correspond to 8 fuel pin positions. Thus, there are a maximum of 92 positions available for fuel pins. Several of these pins contain gadolinium as a burnable poison, with a variety of gadolinia loadings.

These differences result in a softer neutron energy spectrum and an axial power shape that is preferentially more peaked in the bottom half of the core for the initial core compared to the EC. These differences have the following implications for the transients to be considered here:

(1) Scram Worth—The ICL has a more bottom-peaked axial power shape compared to the EC. Thus, as the control rods enter the core from the bottom, they have a more pronounced effect on core reactivity. In addition, the average neutron energy in the ICL core is lower (softer spectrum) than for the EC. This lower neutron energy enhances the neutron absorption in the control rods and their reactivity worth.

(2) Void Reactivity—The ICL will have a lower void reactivity worth because the average neutron energy is lower in the ICL compared to the EC, which implies less under-moderation relative to the EC.

As a result of these two effects, the kinetic response of the initial core is expected to have lower power peaks for transients dominated by void collapse due to primary system pressure increases than the EC.

Finally, the decay heat contribution for the ICL will be lower than that corresponding to the EC. This is due to the lack of actinides in the fuel mix and the lower burnup period. This effect is expected to be small.

The CPR is also an important parameter that must be considered when comparing the initial and ECs. Briefly, the CPR is a measure of the allowable change or variation in the flow and power levels in a given assembly to avoid boiling transition. In the case of transient analyses, ΔCPR/ICPR (initial CPR) measures the change in the CPR as the transient progresses. This combination of parameters is determined for each AOO and IE analyzed.

The analyses of AOO transients are divided into the categories mentioned above. Each of the following is considered:

- Decrease in Inlet Coolant Temperature—one transient was analyzed in this group. This transient involves decrease in feedwater inlet temperature that results from failure in the feedwater heating system. In this case, the SCRRI/select rod insert (SRI) system is not credited in the analysis of the initial core, which results in a 16-percent increase in maximum neutron flux, a 19-percent increase in the maximum of the average heat flux, and a doubling of the ΔCPR/ICPR, compared to the EC. The EC analysis credits the SCRRI system.

- Increase in Reactor Pressure—the nine transients in this group have in common increase in reactor pressure resulting from closure of the main steam line isolation valve(s) (MSIVs). MSIV closure results in a sudden increase in reactor pressure that collapses the core voids. The MSIV closure rate and other mitigating factors characteristic of the transient being analyzed determine the extent of the in-core void collapse. This includes the amount of reactivity added to the core and the rate at which it is added. In all cases, the peak neutron flux is lower for the initial core than for the EC. This is due to the lower void reactivity worth associated with the initial core. The overall change between the ICL and EC analyses for core pressure increase and the maximum of the average heat flux is seen to be minimal. There are significant increases for the ΔCPR/ICPR for selected cases, which can be ascribed to changes in the SCRRI/SRI rod pattern used in the ICL compared to the EC.

- Reactivity and Power Distribution Anomalies—the EC analysis in the DCD Tier 2 applies to the initial core (except for the control RWE during startup with failure of control rod block for which the analysis used the initial core).

- Increase in Coolant Inventory—the two transients analyzed in this group are runout of one feedwater pump and inadvertent initiation of the ICS. The results for both the initial core and the EC are in good agreement.

- Decrease in Coolant Inventory—three transients are considered in this group. These transients are not characterized by a single theme, but are the result of increased flow out of the core or decreased flow into the core. The results for both the ICL and the EC are in good agreement.

In all but two cases, the results for the ICL and those for the EC agree. The two exceptions are generator load rejection and turbine trip with total bypass failure. These events are similar in that they both result in a turbine trip; the difference is in the timing of the sequences. In both cases, the maximum neutron flux determined for the ICL is about 20-percent lower than that corresponding to the EC. In addition, the value determined for ΔCPR/ICPR for the ICL is about 26-percent lower than that determined for the EC. Both of these deviations can be attributed to

the lower ICL void reactivity worth. The other parameters, such as the reactor vessel pressures and maximum average heat flux determined for the ICL, are close to the corresponding EC values.

The analyses of IEs considered eight events. These events involved more than one system failure, thus making them less likely than the AOO transients discussed above, but potentially with more severe consequences.

The three events included in the special events category include an SBO and two ATWS events. The SBO assumes that the external power is cut off, and the station has to rely on standby power. After a 72-hour period, the calculated parameters for the ICL and those of the EC are essentially the same. The first ATWS event involves an MSIV closure with standby liquid control system (SLCS) activation, and the second ATWS event involves a loss of condenser vacuum with SLCS activation. In both cases, an initial power pulse is mitigated by feedwater runback and SLCS activation. In both cases, the results for the ICL have a maximum neutron flux that is about 10-percent lower than that determined for the EC.

Despite this difference, the primary system vessel pressure, suppression pool temperature, and containment pressure determined for the ICL case are very close to the results determined for the EC. However, the calculated peak clad temperature for ICL was 835 degrees C (1,535 degrees F) (Table 2.5-4-4a, NEDO-33337, Rev. 1) versus 928.27 degrees C (1,702.9 degrees F) (DCD Tier 2, Revision 9, Table 15.5-4c) for the ICL and EC cases respectively. The lower ICL values relative to the EC are primarily the result of the lower reactivity insertion associated with the lower void reactivity worth of the initial core.

15.1.6 Feedwater Temperature Operating Domain

Appendix 15D to the DCD Tier 2, summarizes the effect of feedwater temperature (FWT) variation on AOOs, IEs, special events, and LOCAs. GEH also submitted NEDO-33338, Revision 1, which provides the analyses of the initial core and the EC of DCD Tier 2, Chapter 15 transients for operation in the FWTOD. The staff, with the help of ORNL, reviewed this LTR. The following summarizes the staff's evaluation of the analyses.

In LTR NEDO-33338, Revision 1, the applicant describes a broadening of the operating domain, which allows for increased flexibility of operation by adjusting the FWT. This increased flexibility is required to accommodate the so-called “soft” operating practices, which reduce the duty to the fuel and minimize the probability of pellet-clad interactions and associated fuel failures.

By adjusting the FWT, the operator can reduce or increase the reactor power without control blade motion and with minimum impact on the fuel duty. Control blade maneuvering can also be performed at lower power levels.

To control the FWT, the ESBWR design includes a seventh feedwater heater with high-pressure steam. FWT is controlled by either manipulating the main steam flow into the No. 7 feedwater heater to increase FWT above the temperature normally provided by the feedwater heaters with turbine extraction steam (normal FWT) or by directing a portion of the feedwater flow around the high-pressure feedwater heaters to decrease FWT below the normal FWT. An increase in FWT decreases reactor power, and a decrease in FWT increases reactor power. At 100 percent of rated thermal power conditions, the addition of the seventh stage feedwater heaters in full service provides an increase of approximately 36.7 degrees C (66 degrees F) in the FWT, which corresponds to a reduction of approximately 15 percent in the core power.

DCD Tier 2, Revision 9, Figure 15.1-3 shows the ESBWR power FWTOD (P-FWTOD). It has two distinct regions: a feedwater temperature increase (FWTI) region and a feedwater temperature reduction (FWTR) region. The FWTI region is used to reduce the power before control blade maneuvering, both during startup and for normal rod-sequence exchanges. The FWTR region allows operating flexibility and could be used to control day-to-day burnup in a manner similar to the power-flow control with operating reactors.

Five major points are defined in the power FWTOD:

(1) SP0 is the nominal operating state point—100 percent power, 100 percent FWT 216 degrees C (420 degrees F).

(2) SP2 is the increased FWT state point—85-percent power and 252 degrees C (486 degrees F). This FWT corresponds to operation with the seventh feedwater heater at full capacity.

(3) SP1 is the reduced FWT state point—100-percent power and 160 degrees C (320 degrees F). This FWT represents a reduction of 55.56 degrees C (100 degrees F), which is the maximum credible FWT transient caused, by a single failure in the FWCS. SP1 is defined only for bounding calculations; the power flow relationship of FWTOD is limited by the SP1M state point.

(4) SP1M is the stability-bounding reduced FWT state point—100-percent power and 187 degrees C (370 degrees F). The FWT of point SP1M is defined on a cycle-specific basis and will be documented in the Core Operating Limits Report (COLR). Point SP1M is defined so that the reactor remains stable following an additional 16.6 degrees C (30 degrees F) FWT reduction caused by an inadvertent LOFWH.

    Note: LOFWH with temperature reductions greater than 16.6 degrees C (30 degrees F) would result in actuation of SCRRI/SRI control blade insertion, which would terminate the transient before instability is a concern.

(5) SP5 is the 50-percent power point at nominal FWT. At power levels below 50 percent, the ESBWR will operate with the nominal feedwater heater configuration (i.e., heater Nos. 1 through 6 on, and no steam supply to heater No. 7).

One of the primary advantages of the FWTOD is realized during reactor startup. Figure 15.1-2 in this report shows a typical startup path for the ESBWR. Up to point SP5, the operator simply pulls control blades with feedwater heaters Nos. 1 through 6 fully open and feedwater heater No. 7 closed, because rod heatup is not a concern when the power level is below 50 percent. As the power increases, the turbine control valve (TCV) opens to provide higher steam flow to the turbine to maintain constant reactor pressure.

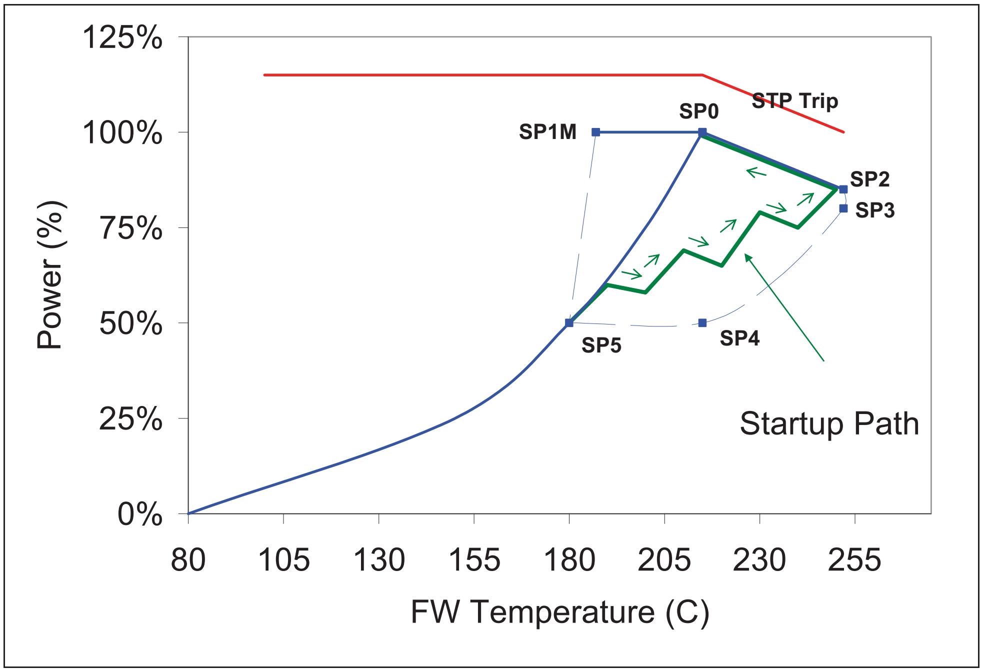

Figure 15.1-2. Typical ESBWR Startup Path.

As the TCV opens, the pressure drop through the valve is reduced, and the turbine inlet pressure is higher, thus providing higher pressure steam to the feedwater heaters, which increases the shell-side steam-condensation temperature. Therefore, the tube-side FWT increases with power automatically.

After point SP5 is reached, the power is high enough that rod heatup limits and possible pellet clad interactions become a concern. To prevent fuel duty issues, the operator will alternate further control blade withdrawals with increases on the steam supply to feedwater heater No. 7. After all the rods are withdrawn following this sequence (rod pull, FWT increase), point SP2 will be reached with the target control blade pattern, while minimizing the duty to the fuel because the blade withdrawals occur at lower power levels. Finally, the full operating power (point SP0) is reached by slowly turning off feedwater heater No. 7 and decreasing the FWT to nominal conditions.

To optimize the core isotopics and burnup management, periodic control rod exchanges are performed with preplanned (preconfigured) control rod patterns. For this periodic rod sequencing, feedwater heater No. 7 is brought slowly back into operation, and the reactor maneuvers from point SP0 to SP2. The rod pattern exchange is performed at the lower power. After the rod exchange, the power is increased by slowly removing feedwater heater No. 7 from operation. Fine reactivity control may be achieved between rod sequence exchanges by partially bypassing feedwater heater No. 7 and operating between the points SP0 and SP1M at reduced FWT. During most of the cycle, gadolinium burnup results in a reactivity increase; thus, burnup control would require slowly increasing the FWT and moving towards point SP0. Towards the end of the cycle, uranium-235 burnup dominates and the reactivity decreases, thus requiring a reduction of FWT (i.e., moving towards state point SP1M).

Operation in the complete FWTOD is possible at some time during the cycle, although most of the operating time should occur at or near point SP0, where the balance of plant has been

optimized for efficiency. The FWTI region in Figure 15.1-3 (i.e., point SP2) is expected to be used during startup and rod pattern exchanges. The FWTR region in Figure 15.1-3 (i.e., the line between SP0 and SP1M) may be used for fine reactivity control or for reactivity stretching towards end of cycle (EOC). In RAI 4.3-25, the staff requested that the applicant discuss whether it is possible to use the region between SP0 and SP1M to provide an-end-of cycle stretch. In response to the staff's RAI 4.3-25, the applicant specified that EOC stretch will not be accomplished by reducing FWT in the ESBWR. If in the future, a licensee intends to implement low-temperature EOC stretch, additional analyses will need to be performed and reviewed.

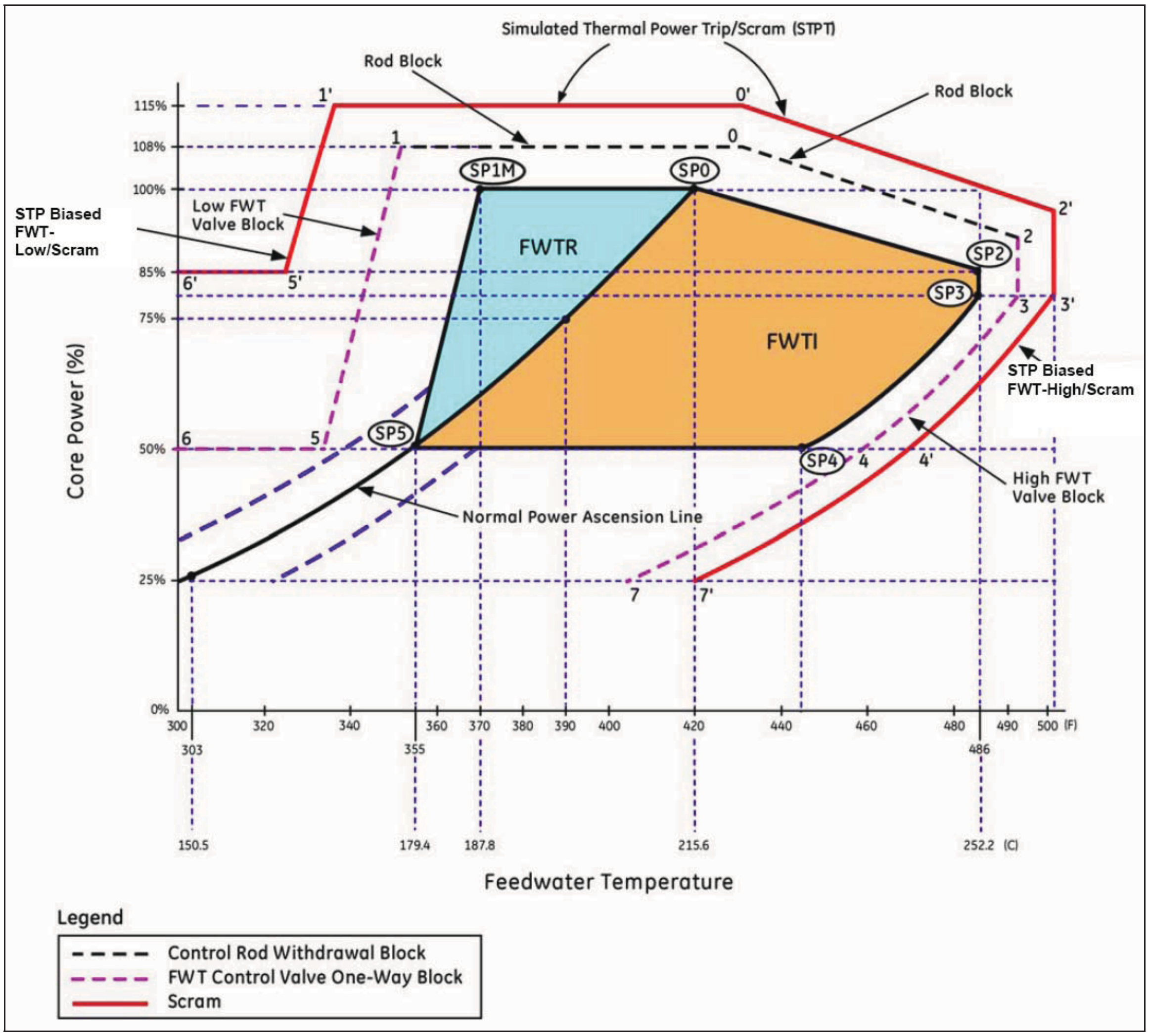

Figure 15.1-3. ESBWR Operating Domain Showing Protection and Control Functions.

Figure 15.1-3 shows a detailed FWTOD map that defines not only the operating domain but also the protection system actuation lines, as well as the control system blocks on control rod and feedwater valve actuation. These blocks indicate the region where operation is restricted by means of the conventional control system. Unintended operation outside this region results in a control room alarm.

The applicant has provided an evaluation of the impact of varying FWT according to this map of the reactor transient response in NEDO-33338, Revision 1, for the initial core design documented in NEDO-33326, Revision 1, "GE14E for ESBWR Initial Core Design Nuclear Report," issued March 2009, and the DCD Tier 2, Revision 9, EC analysis in Chapter 15. The applicant has compared the results against the nominal operating point (SP0) reported in ESBWR DCD Chapter 15 for an EC and in NEDO-33337 for the ICL. All analyses have been performed using the approved version of the TRACG code.

The results of the analyses indicate that the most limiting AOO is the IICI at SP2 state point for both equilibrium and initial cores. The most limiting IE is the generator load rejection with total bypass failure at SP1 for the EC and at SP0 for the initial core. However, the largest change in the CPR when the FWT is changed occurs for the LOFWH event, and therefore, it may have a larger impact on the operating limit MCPR at points SP1M and SP2.

The applicant has evaluated the impact of FWT and cycle-specific conditions for special events, including ATWS, ATWS stability, and LOCA. Ample margins to criteria are demonstrated, and the impact of cycle-specific conditions or FWT is calculated to be insignificant.

15.1.6.1 *FWTOD Summary*

The staff concludes that the applicant has adequately accounted for the effects of the proposed FWTOD extension on the nuclear design and has demonstrated that the fuel design limits will not be exceeded during normal or anticipated operational transients and that the effects of postulated transients and accidents will not impair the capability to cool the core. Based on this evaluation, the staff concludes that the nuclear design of the fuel assemblies, control systems, and reactor core will continue to meet the applicable regulatory requirements. Therefore, the staff finds the proposed FWTOD extension acceptable.

15.1.7 Post-Combined-License Activity

The staff's main conclusion is that the broadening of the ESBWR operating domain by adjusting the FWT is acceptable. However, the following post-combined-license (post-COL) actions are required per TS 5.6.3 "Core Operating Limits Report (COLR)" to satisfy applicable regulatory criteria.

- An operating limit should be established for the OLMCPR that is a function of FWT. Thus, higher operating margin is provided at off-nominal FWTs. The OLMCPR is cycle dependent and will be documented in the COLR.

- The minimum FWT of operating point SP1M should be limited to ensure that stability criteria are satisfied. The FWT of point SP1M will be cycle dependent based on the result of stability analyses and will be documented in the COLR.

- In RAI 4.3-16, the staff requested the applicant to explain how the power-temperature operating domain will be defined. In response to RAI 4.3-16, GEH added a reference to NEDO-33338 in COLR Section 5.6.3 (b)(8) to indicate that the staff had approved the analytical method used for determining the cycle-specific operating thermal and stability limits.

- In RAI 4.3-25, the staff requested the applicant to explain whether it is possible to use the region between SP0 and SP1M to provide an end-of-cycle stretch. In response, the

applicant stated that if a future licensee or applicant intends to use the region between SP0 and SP1M for an end-of-cycle stretch, additional analyses similar to those required for end-of-cycle stretch for operating BWRs will be performed. If in the future, a licensee intends to implement low-temperature EOC stretch, additional analyses as stated above will need to be performed and reviewed. Since the applicant meets the regulatory requirements and based on the applicant's commitment to provide additional analyses for end-of-cycle stretch operation, RAIs 4.3-16 and 4.3-25 are resolved.

15.2 Analyses of Anticipated Operational Occurrences

DCD Tier 2, Revision 9, Section 15.2 provides the analyses of AOOs.

The ESBWR design incorporates several features (in addition to natural circulation cooling), such as the four IC units, instrumentation with triplicate digital electronic circuits, more than 100-percent steam bypass capacity, that forestall the evolution of AOOs into a more serious transient and also reduce reactor scram frequency. Another notable feature is the control rod operation in the SCRRI/SRI configuration. SCRRI is a set of sequential-insertion low-speed control rod gang-assemblies to lower power and prevent a scram. However, SCRRI insertion at high power levels could compress power upwards and possibly threaten thermal limits. SRI is a set of fast, hydraulic full-insertion control rods that lower power to prevent possible violation of thermal limits in anticipation of SCRRI insertion. The combination of lowering the average reactor power level would prevent violation of thermal limits that could arise due to SCRRI malfunction. In addition, core flow will increase with lower power which will decrease the oscillation decay ratios. Independently, the "Oscillation Power Range Monitor" will detect and suppress any T-H instability. Therefore, SCRRI augmented with SRI avoids and prevents violation of the thermal limits as well as instabilities. This conclusion resolves RAI 15.2-5 (see Section 15.2.1.1.2 of this report).

DCD Tier 2, Revision 9, Figure 7.7-1 shows the definitions of RPV water-level ranges. Level designation L0 is about 0.5 meters (m) (1.64 feet [ft]) above TAF. L2 is about 8.5 m (27.89 ft) above TAF and initiates the IC and CRD pump. L3 is about 12.5 m (41.01 ft) above TAF and initiates a reactor low-level scram. L4–6 is the normal operating range. L7 is the high-vessel-level alarm setpoint. L8 is about 14.5 m (47.57 ft) above TAF and initiates a high-reactor-water level scram. These level designations are used in DCD Tier 2, Sections 15.2 and 15.3.

GEH analyzed the following categories of AOOs in the DCD sections indicated:

- 15.2.1 Decrease in Core Coolant Temperature
- 15.2.2 Increase in Reactor Pressure
- 15.2.3 Reactivity and Power Distribution Anomalies
- 15.2.4 Increase in Reactor Coolant Inventory
- 15.2.5 Decrease in Reactor Coolant Inventory

15.2.1 Decrease in Core Coolant Temperature

The staff used the acceptance criteria in Section 15.1.1.1 of this report in evaluating this AOO.

15.2.1.1 *Loss of Feedwater Heating*

15.2.1.1.1 Summary of Technical Information

LOFWH decreases the FWT, which in turn decreases core voids and increases moderation and power. The ESBWR can lose feedwater heating in at least two ways: closing of the heater to the steam extraction line and the feedwater bypassing the heater. The ESBWR design is such that no single failure or operator error will cause LOFWH that would result in a temperature decrease greater than 55.56 degrees C (100 degrees F). The ATLMs will detect a decrease in the FWT, and the diverse protection system (DPS) will initiate SCRRI/SRI insertion to lower power and avert a scram. Although FWT reduction is sensed well before the colder water reaches the core, the analysis conservatively assumes that control rod insertion does not initiate until after core power begins to increase.

When the temperature decrease remains less than or equal to 16.7 degrees C (30 degrees F), the SCRRI/SRI system is not activated, and the power could reach 106 percent of the normal power level, but the ΔCPR/ICPR value is bounded by the inadvertent initiation of an IC, which is analyzed in Section 15.2.4.1 of this report. DCD Tier 2, Revision 9, Table 15.2-5 and Figure 15.2-1, demonstrate the results of the analysis for this transient. The sequential SCRRI/SRI insertion and the calculated power transient clearly indicate that the reactor attains a lower power level and reactor scram is not needed.

15.2.1.1.2 Technical Evaluation

<u>EC</u>: LOFWH decreases core inlet temperature, resulting in increased moderation due to core void collapse and an increase in core power. The ESBWR has design features that (1) limit feedwater maximum inlet ΔT to 55.56 degrees C (100 degrees F), (2) employs the ATLM, which reduces power to avoid exceeding the thermal limits, and (3) includes the DPS. Either the ATLM or the DPS can activate the SRI to lower power in a fast mode to avoid violation of safety limits and reactor scram. The results are summarized in DCD Tier 2, Revision 9, Figure 15.2-1, which indicates that the maximum pressure, water level, and MCPR are all well within normal operating limits. If the FWT decrease is less than 16.67 degrees C (30 degrees F), the SCRRI/SRI is not activated, and the power may increase up to 106 percent of normal power. Either way, the resulting transient is bounded by the IICI, which is discussed in DCD Tier 2, Revision 9, Section 15.2.4.1 and has been found acceptable. This AOO does not induce a more serious condition and does not result in a reactor scram.

In RAI 15.2-5, the staff requested the applicant to explain how the reactor, in the event of a partial failure of the SCRRI, would avoid violating local thermal limits or creating core instability without shutting down the core. This RAI was based on DCD Tier 2, Revision 1, Figure 15.2-1e, which demonstrates the importance of reactivity control. In response to RAI 15.2-5, the applicant provided design description of the SRI system added in Revision 3 of the DCD. SRI is a fast insertion set of control rods that lower power (about 50 percent) followed by the sequential SCRRI control rod insertion that avoids reactor scram. The applicant described how the combined insertions of SCRRI/SRI will prevent the core thermal limits to be exceeded. Based on the applicant’s response, RAI 15.2-5 is resolved because the results satisfy the applicable acceptance criteria listed in Section 15.1.1.1 of this report.

<u>ICL</u>: The LOFWH AOO has been analyzed for the ICL in the same manner using the same assumptions as in the EC. The results of the analysis are close to those for the EC. The feedwater flow increases between 25 and 100 seconds into the transient due to the collapse of

the core voids. In both instances, steady operation is achieved in about 200 seconds at about 50 percent of rated power. Pressure remains normal for about 50 seconds and drifts lower at 200 seconds. The MCPR remains well above the OLMCPR. This AOO, does not result in reactor scram; therefore, the LOFWH AOO for the ICL satisfies the acceptance criteria in Section 15.1.1.1.

FWTOD: GEH analyzed the excessive heat removal events and concluded that the LOFWH AOO is limiting. The decreased FWT results in higher power due to increased moderation and thus a decrease in the MCPR. The most limiting LOFWH AOO is the EC at SP0 state point which results in a limiting ΔCPR/ICPR for SP0. Staff review of the analyses indicates that the assumptions are conservative and the MCPR remains above the SLMCPR and hence the results are acceptable because none of the analyzed cases results in reactor scram.

15.2.1.1.3 Conclusion

The comparison of the initial core and the EC analyses results indicate that the equilibrium analysis at SP0 is limiting for the LOFWH AOOs. As stated above, the results of the analyses for the LOFWH AOOs for EC, ICL, and FWTOD satisfy the acceptance criteria.

15.2.1.1.4 Post-COL Activity

This event is potentially limiting with respect to OLMCPR, because of the effect cycle-to-cycle changes to the SCRRI/SRI rod pattern have on ΔCPR/ICPR. This event will need to be analyzed for each fuel cycle along with the limiting SCRRI/SRI rod pattern which could be changing from cycle to cycle. The SCRRI/SRI requirements are documented in the COLR in accordance with TSs. The OLMCPR is established for the limiting event and documented in the COLR in accordance with TSs.

In RAI 15.2-5 the staff requested the applicant to explain how the reactor, in the event of a partial failure of the SCRRI, would avoid violating the thermal limits. The resolution of RAI 15.2-5 is documented in Section 15.2.1.1.2 of this report and the SCRRI/SRI limitations are in the TS Section 5.6.3, COLR (a)(6), TS 3.7.6.

15.2.2 Increase in Reactor Pressure

The staff used the acceptance criteria in Section 15.1.1.1 of this report in evaluating this AOO.

15.2.2.1 *Closure of One Turbine Control Valve*

15.2.2.1.1 Summary of Technical Information

EC: The SB&PC system includes the TCVs. The DCD states that the SB&PC uses a triplicate digital control system and is not subject to a credible single failure. For the purposes of this analysis, the applicant assumed that one TCV closed inadvertently (i.e., fails) at full power. The SB&PC system will sense the pressure increase and will open the remaining TCVs to maintain pressure. However, this may not be sufficient, and pressure and power will increase, depending on the turbine steam admission design (full or partial arc) and the flow through the remaining three TCVs. With one TCV closed, flow through the remaining three valves is 95 percent for full arc and 85 percent for partial arc in the reference core. Therefore, the partial arc case is conservative with regard to void collapse and resulting power and pressure peak. In addition, fast and slow valve closure was assumed. The full-stroke, rated steam flow closure time could

be either 0.08 seconds (fast) or 2.5 seconds (slow). Both cases were analyzed (i.e., for fast and slow TCV closure times). The analyses presented in DCD Tier 2 are based on partial arc (i.e., the most conservative cases). The analytical results for the fast- and slow-closure cases are presented in DCD Tier 2, Revision 9, Tables 15.2-6 and 15.2-7 and Figures 15.2-2 and 15.2-3 respectively. Both instances are in the pressure increase category.

In the fast-closure case, there is a power peak over 120 percent of rated power at around 1.0 seconds, followed by 110-percent feedwater flow peak at about 6 seconds and steady-state at about 10 seconds. The flow obstruction, increased pressure which caused partial collapse of the core void that increased moderation and power that farther increased pressure. Peak power from void collapse mirrors core reactivity as a function of time from transient initiation. The MCPR remains above the OLMCPR for both the fast-closure and the slow-closure case.

In the slow-closure case, the transients are similar except that the power peak is about 110 percent at 3 seconds and 112-percent feedwater flow peak at about 8 seconds. Steady state occurs at about the same time and to the same levels of power and TCV flow. The pressure shows little change in either the fast- or slow-closure case.

ICL and FWTOD: The same transients have been analyzed for the ICL and the FWTOD and the results for both cases are similar. In the ICL fast closure, the power peak exceeds the scram limit. In the case of FWTOD, because of the higher void fraction, the pressure increase will cause larger reactivity insertion and the power peak also exceeds the reactor scram limit. In both cases, the scram is ignored, and the transients are conservatively analyzed.

15.2.2.1.2 Technical Evaluation

The following comments and conclusions apply to both the fast- and the slow-opening TCV.

EC: Inadvertent closing of one TCV at full power creates a power spike of short duration. However, the associated pressure and MCPR changes are small. The results of this transient meet the acceptance criteria because vessel pressure and core MCPR are well within the acceptance limits and the transient will not cause any other adverse consequence.

ICL: GEH analyzed both instances for fast and slow TCV closing. The results of the fast-closing case bound those of the slow case. The results for the ICL are bounded by the equilibrium case and, therefore, are acceptable.

FWTOD: This transient has been reviewed in the context of the NEDO-33338 review, and the staff finds that it is bounded by IICI events analyzed at the SP2 state point.

15.2.2.1.3 Conclusion

Based on the above analyses, the staff concludes that inadvertent closure of one TCV at full power for the EC, the ICL, or the FWTOD satisfies the acceptance criteria in Section 15.1.1.1 of this report, is bounded by other transients found acceptable in this report, and, therefore, is acceptable. Because the SCRRI/SRI insertion is part of the transient scenario and the OLMCPR depends on fuel core loading, this transient should be analyzed in each cycle as in Section 15.2.1.

15.2.2.1.4 Post-COL Activity

This event (i.e., fast closure) is potentially limiting with respect to OLMCPR which change from cycle to cycle and will be analyzed for each fuel cycle. The OLMCPR is established for the limiting event and documented in the TS Section 5.6.3, COLR (a)(2) in accordance with TS 3.2.2.

15.2.2.2 *Generator Load Rejection with Turbine Bypass System*

The staff used the acceptance criteria in Section 15.1.1.1 of this report in evaluating this AOO.

15.2.2.2.1 Summary of Technical Information

EC: Grid electrical disturbances could cause loss of generator load. To avoid damage to the turbine generator from overspeed, the TCVs are designed and enabled to close very rapidly. TCV closure would increase vessel pressure, but the opening of the steam bypass valves prevents overpower and overpressurization. If there is no other failure, the steam bypass system has the capacity to discharge the entire steam flow at full power. The SCRRI/SRI rod system will then insert control rods to lower reactor power. DCD Tier 2, Revision 9, Table 15.2-8, lists the sequence of events for this transient, and DCD Tier 2, Figure 15.2-4, shows the calculated results. In the interim between the fast closing of the TCVs and opening of the turbine bypass valves (TBVs), there is a sharp pressure pulse accompanied by a power generation pulse that lasts slightly over 1 second. High neutron flux can cause a scram signal. However, the reactor is conservatively assumed not to scram. At the same time, the SCRRI/SRI system activates and the first SRI occurs at 15 seconds, followed by the second at 30 seconds, and so on until the sixth insertion occurs at 75 seconds. SCRRI insertion is complete at 110 seconds. At 200 seconds, the power level reaches 60 percent of rated power. No specific SCRRI group was assigned and the SCRRI results were not used to show acceptable CPR results. However, SCRRI/SRI rod patterns depend on cycle loading, may affect the ratio ΔCPR/ICPR, and are potentially limiting for the OLMCPR. Therefore, their characteristics and requirements are documented in the COLR in accordance with the TS.

ICL: Analysis of the same transient for the ICL yields similar results for the transient characteristics (i.e., peak power, pressure, and feedwater flow). In this case, the transient power stabilizes at about 300 seconds at 40 percent of rated power, with feedwater and steam flows at 30 percent of rated. The peak power exceeds the high thermal flux and the high neutron flux setpoints but the reactor is conservatively assumed not to scram. The MCPR remains above the OLMCPR, and the pressure and core coverage are well above the L2 level

FWTOD: NEDO-33338 presents the transient for the SP2 state point and shows results similar to those described above for the EC and the ICL. The power peak exceeds the high thermal flux scram setpoint (at 100 percent of the ESBWR rated power) but remains below the high neutron flux setpoint. At about 300 seconds into the transient, total power stabilizes at 40 percent (of ESBWR rated power) with feedwater and steam flow at 30 percent of rated steam flow. The reactor power peak (as in the EC and ICL cases) exceeded the high thermal flux scram setpoint.

15.2.2.2.2 Technical Evaluation

EC: The main objective of this AOO is to ensure that the system isolates the turbine generator unit as fast as possible to avoid damage from overspeed. The SB&PC system generates

signals for fast closure of the TCVs, with simultaneous opening of the steam bypass valves and activation of the SCRRI/SRI system to reduce power and limit MCPR values. Overpressurization is prevented by the full-flow turbine bypass capacity in the ESBWR.

Assuming no other equipment failure, total power will peak at about 0.6 seconds, and the feedwater flow will peak at about 35 seconds at 140 percent. SCRRI/SRI rod insertion will stabilize power at about 60 percent with 45-percent feedwater and steam flow.

If the TBVs operate as designed, no vessel overpressurization will occur, and RPV pressure will actually decrease. Rod insertion counterbalances the void reactivity increase from void collapse and the small reactivity increase resulting from fuel temperature. The MCPR stays above the OLMCPR, and reactor operation stabilizes at about 170 seconds.

Within a second after initiation of this transient, during the transition from closing of the TCVs to opening of the TBVs, the core experiences a short pressurization and void collapse, which results in a power spike to about 130 percent of rated power for about 1 second. The power peak is higher than the thermal flux and the thermal flux setpoints, but it is assumed that the reactor does not scram.

The staff questioned the energy deposition and the potential of reaching the cladding strain limits (RAI 15.2-2 S01 and RAI 15.3-11 S01), (Note: the staff requested this information for several transients as the review was progressing. GEH responded collectively in MFN 07-641) GEH responded to questions regarding energy deposition in the fuel during high and fast power peaks not addressed in the DCD. Generator load rejection with bypass failure (GLRBF) is the most limiting such transient. In its response, GEH stated that if the normalized nodal power density (NNPD) of any node in the core during a transient is bounded by the generic NNPD used to define the fast transient limits, then the thermal-mechanical limits are not threatened.

In the applicant’s analyses, the NNPD for the GLRBF was increased by a factor of 1.5 and then compared to the generic GE14 fuel NNPD. The GE14 fuel bounds by a considerable margin the results of the NNPD for the GLRBF including the conservative factor of 1.5. In addition, the GE14E fuel which is used for the ESBWR bounds the GE14 NNPD. Therefore, all AOO transients in DCD Tier 2, Section 15.2 and IEs in Section 15.3 are assured for thermal-mechanical integrity. This includes cladding strain, fuel center melt, and maximum linear heat generation rate (MLHGR).

ICL: The core response is very similar to that of the EC (i.e., the power spike and duration are comparable and the reactor resumes stable operation at 40-percent power at 30-percent feedwater and steam flow). The MCPR remains above the OLMCPR. The acceptance criteria are satisfied in that the MCPR, clad strain, and vessel pressure and water level are within the acceptance criteria; therefore, the transient results are acceptable.

FWTOD: The staff reviewed this transient in the context of the NEDO-33338 and finds that it is bounded by the IICI events analyzed at the SP2 state point.

15.2.2.2.3 Conclusion

The comparison of the initial core and the EC analyses results indicates that the EC analysis at SP0 is limiting in the generator load rejection transients.

As discussed above, the load rejection with turbine bypass transient is well within the acceptance criteria in Section 15.2.1.1.1 in this report. The SCRRI/SRI control rod patterns must be recalculated for each reload because they are fuel-loading dependent. Overpressurization and high-energy deposition are avoided because of the bypass and fast response of the SB&PC system; therefore, the results meet the acceptance criteria.

15.2.2.2.4 Post-COL Activity

This event is potentially limiting with respect to the OLMCPR, because of the effect of cycle-to-cycle changes to the SCRRI/SRI rod pattern on ΔCPR/ICPR.

This event is analyzed for each fuel cycle with the limiting SCRRI/SRI rod pattern, which changes from cycle to cycle. The SCRRI/SRI requirements are documented in the COLR in accordance with the TS. The OLMCPR is established for the limiting event and documented in the TS Section 5.6.3, COLR (a)(2) in accordance with TS 3.2.2.

15.2.2.3 *Generator Load Rejection with a Single Failure in the Turbine Bypass System*

The staff used the acceptance criteria in Section 15.1.1.1 of this report in evaluating this AOO.

15.2.2.3.1 Summary of Technical Information

<u>EC</u>: For this AOO, the system and the plant instrumentation responses are similar to that for the generator load rejection with turbine bypass. When the instrumentation senses generator load rejection, the SB&PC system signals closure of the TCVs and opening of the TBVs. However, in this case, the analysis assumes a single failure in the turbine bypass system. For conservatism, the bypass capacity is assumed to be at 50 percent. DCD Tier 2, Revision 9, Figure 15.2-5, shows core response and core parameter variation as a function of time. Table 15.2-9 lists the sequence of events.

The calculations are based on the assumption that the SB&PC system will signal the bypass valves to initiate opening at 0.02 seconds into the transient and the TCVs will be closed at 0.08 seconds. At 0.15 seconds, the system will sense inadequate bypass, and the plant is scrammed. Control rod insertion initiates at 0.35 seconds. In this AOO, it is assumed that 50 percent of the bypass capacity has failed.

The calculated results are shown in DCD Tier 2, Figure 15.2-5. For a short time (less than 1 second), steam flow decreases because of limited bypass, which increases pressure, neutron moderation, and core power. The power peak lasts less than a second; feedwater flow increases to about 140 percent of normal (due to void collapse) and stabilizes at about 60 percent in about 50 seconds. The vessel dome pressure peaks at about 7.79 MPa (1,130 pounds per square inch absolute [psia]) about 2 seconds into the transient. Peak pressure is below the SRV lift setting. The MCPR remains well above the OLMCPR. This event is potentially limiting with respect to the OLMCPR; therefore, it must be analyzed for each cycle loading and included in the COLR. Because of the void collapse and reactor scram, the water level reaches L3 at about 2.82 seconds. No operator action is required to mitigate this transient.

<u>ICL</u>: The results of the ICL (for the same AOO with the same initial assumptions) are almost identical to the EC. However, the MCPR value remains well above the OLMCPR (Figure 2.3-5g in NEDO-33337, Revision 1) peak pressure is below the SRV setpoint. Because of the void

collapse, the reactor vessel water level reaches L3 at about 2.7 seconds. Scram initiates at .15 seconds.

FWTOD: This AOO is similar to the EC transient. The value of MCPR is well above the OLMCPR, and the peak pressure is significantly below the SRV setpoint. Scram initiates at 45 seconds into the transient, and the L3 level is reached at 3.17 seconds, but the scram level is not exceeded

15.2.2.3.2 Technical Evaluation

EC: As discussed earlier, TBV failure is highly unlikely because the SB&PC system uses a triplicate digital controller. After the system detects inadequate turbine bypass, the reactor scrams and control rod insertion begins at about 0.40 seconds after transient initiation. The resulting pressure and thermal power pulse is less than a second in duration. Should high pressure compress the water to Level 2 for 10 seconds or more, the CRD high-pressure makeup injection will activate. Should the low-level signal remain for 30 seconds, the MSIV and IC will activate.

The vessel pressure remains within acceptable limits, the MCPR remains above the OLMCPR, and no fuel rods are in boiling transition; therefore, the regulatory acceptance criteria are met.

The results provided in DCD Figure 15.2-5(a) show a high and narrow power peak of less than a second's duration. After the initiation of this transient during closing of the TCVs and opening of the TBVs, the core experiences a short pressure pulse resulting in a power spike to about 190 percent of rated power for less than 1.0 second. GEH did not calculate energy deposition to ensure acceptable cladding integrity and fuel cladding interaction. The staff questioned GEH's lack of consideration for energy deposition. In response to RAI 15.2-2 S01, GEH compared the GE14E fuel design limits to the energy deposition in this transient to demonstrate that the energy deposition is almost insignificant. Based on the applicant's response, RAI 15.2-2 S01 is resolved. Staff evaluation of RAI 15.2-2S01 is included in Section 15.2.2.2 of this report.

ICL: The same method (code) and the same assumptions are used as in the EC analysis. The results of the analyses are similar; however, the calculated power peak at the initiation of the transient is about 10-percent higher than in the EC case. The TCVs close at about 0.08 seconds, and at 2 seconds the system senses inadequate bypass and initiates rod insertion and reactor scram. The ensuing power peak lasts less than 0.2 seconds. The resulting MCPR is higher than the OLMCPR.

FWTOD: The staff reviewed this transient in the context of the NEDO-33338 and finds that it is bounded by the IICI events analyzed at the SP2 state point.

15.2.2.3.3 Conclusion

The comparison of the results of the initial core and the EC analyses indicates that the EC analysis at SP0 is the most limiting for generator load rejection with single failure in the turbine bypass system. As discussed above, load rejection with a single failure in the turbine bypass system is well within the acceptance criteria in Section 15.2.1.1.1 in this report. Overpressurization and high-energy deposition are avoided because of the available (50 percent) steam bypass and the fast response of the SB&PC system. The results are within the bounds of the acceptance criteria.

In response to RAI 15.2-2 S01, GEH compared the GE14E fuel design limits to the energy deposition in this transient to demonstrate that the energy deposition is almost insignificant. Based on the applicant's response, RAI 15.2-2 S01 is resolved. Staff evaluation of RAI 15.2-2S01 is included in Section 15.2.2.2 of this report.

15.2.2.3.4 Post-COL Activity

This event is analyzed for each fuel cycle with the limiting SCRRI/SRI rod pattern, which changes from cycle to cycle. The SCRRI/SRI requirements are documented in the COLR in accordance with the TS. The OLMCPR is established for the limiting event and documented in the TS Section 5.6.3, COLR (a)(2) in accordance with TS 3.2.2.

15.2.2.4 *Turbine Trip with Turbine Bypass*

The staff used the acceptance criteria in Section 15.1.1.1 of this report in evaluating this AOO.

15.2.2.4.1 Summary of Technical Information

EC: A variety of causes, such as vibrations, low condenser vacuum, or loss of turbine control fluid pressure, can initiate turbine trip. After turbine trip activation, the SB&PC system will initiate opening of the bypass valves in 0.02 seconds.

At 0.10 seconds, the turbine stop valves (TSVs) are closed, and at 0.20 seconds, the SRI initiates fast rod insertion to limit core power so as to avoid reactor scram and protect the MCPR limits. At 1.5 seconds, SRI initiates insertion and the first group inserts. The second, third, fourth, fifth, and sixth groups insert at 16.5, 31.5, 46.5, 61.5, and 76.5 seconds, respectively. At 121 seconds, the reactor attains a steady-state at about 60 percent of normal power with about 45 percent of normal feedwater flow. Peak feedwater flow occurs at 25–35 seconds at about 135 percent of normal. Vessel pressure shows a very small increase and falls to about 94 percent of the normal operating value. At the new steady state, about 45 percent of steam flow removes about 60 percent of power. The value of the MCPR remains well above the SLMCPR. In DCD Tier 2, Revision 9, Table 15.2-10 and Figure 15.2-6 summarize the results of the calculation. Reactor scram is not activated, and no operator action is required to mitigate this transient. The calculation results are within the range of the acceptance criteria.

ICL: The results of this AOO are similar to the EC transient results. Both cases are based on the same assumptions regarding bypass availability. SCRRI/SRI insertion lowers reactor power and avoids reactor scram. Power stabilizes at about 40 percent of rated with 30 percent of normal flow. Vessel pressure rises slightly, and the MCPR stays above the SLMCPR. The results are within the range of the acceptance criteria.

FWTOD: NEDO-33338 does not explicitly analyze this transient but explained that this transient is bounded by the generator load rejection with a single failure in the turbine bypass system.

15.2.2.4.2 Technical Evaluation

EC: After turbine trip, a fast rise in core pressure causes void collapse, increased moderation, and increased power in the form of a power pulse. Void collapse also causes a brief increase in feedwater flow. The cold-water slug entering the core also contributes to the power pulse. The calculated results show that control rod insertion compensates for the increased reactivity from increased moderation. The reactor reaches a steady-state at a power level of about 60 percent

and a corresponding feedwater flow of about 45 percent. The MCPR remains well above its designated safety limit. This transient is very similar to the load rejection with turbine bypass. The vessel water level remains well above the L3 level, there is no scram, and the reactor attains a stable state. The acceptance criteria are met.

ICL: The transient is similar to the EC transient except that the core power settles at a lower power of about 40 percent of normal with 35 percent of flow. The MCPR value is above the OLMCPR at about 1.0 second. The vessel water level remains well above the L3 level, there is no scram, and the reactor attains a stable state. The acceptance criteria are met.

FWTOD: The staff reviewed this transient in the context of the NEDO-33338 and finds that it is bounded by the IICI events analyzed at the SP2 state point.

15.2.2.4.3 Conclusion

Because of the fast opening of the bypass valves, the calculated results indicate that only a minor power disturbance occurs, no pressure surge takes place, the MCPR remains above the OLMCPR, and the reactor assumes a lower power stable state. Therefore, the results of this transient meet the acceptance criteria, and the transient results are acceptable.

There is no reactor scram. This event is similar to the generator load rejection with turbine bypass and credits the SRI system.

15.2.2.5 *Turbine Trip with a Single Failure in the Turbine Bypass System*

The staff used the acceptance criteria in Section 15.1.1.1 of this report in evaluating this AOO.

15.2.2.5.1 Summary of Technical Information

EC: A variety of causes, such as vibrations, low condenser vacuum, and loss of turbine control fluid pressure, can initiate a turbine trip. Upon activation of a turbine trip signal, the SB&PC system will open the TBVs and initiate TSV closure. In this transient analysis, the applicant assumed that the single failure would result in the loss of 50 percent of the bypass capacity. Since it would require more than a single TBV failure to lose 50 percent of the bypass capacity, this is a conservative assumption. The shortfall in bypass capacity creates a pressure pulse. The pressure increase causes void collapse, increased moderation and rapid increase in neutron power that results in a scram signal. The power pulse has a width of about one third of a second at half-maximum and peaks at over 140 percent of rated power. A small vessel pressure peak occurs at about 2.0 seconds into the transient, which remains well below the SRV setpoint. The TSV closure initiates reactor scram at 0.35 seconds. The opening of the bypass valves ameliorates the pressure pulse. The reactor is essentially shut down in less than 2.0 seconds. The transient calculation terminates at 50 seconds. Because the reactor is shut down with the vessel water level near L3, which is 12.5 m (41 ft) above the TAF, the staff concludes that the reactor is in a stable condition. The MCPR value remains higher than the operating limit OLMCPR. DCD Tier 2, Revision 9, Figure 15.2-7, and DCD Tier 2, Table 15.2-11, show the calculated results.

ICL: For the initial core analysis, this transient is similar to the EC transient. For example, the power peaks at the same value of 150 percent of rated power. In both cases, scram initiates at 37 seconds, and the reactor is shut down in about 2 seconds. The MCPR value is well above

the OLMCPR. The core remains covered at normal pressure and no rods are in boiling transition. This transient meets the AOO acceptance criteria.

FWTOD: The turbine trip with single bypass failure has not been explicitly analyzed in NEDO-33338 because it is shown to be bounded by the IICI event.

15.2.2.5.2 Technical Evaluation

EC: Following turbine trip due to inadequate bypass flow, the RCS pressure peaks, causing void collapse, increased moderation, and the creation of a power peak. The RPS initiates scram, and the reactor total power falls to about 5 percent in less than 3 seconds with simultaneous closure of the TCVs. The vessel pressure peaks at 2.0 seconds but remains well below the SRV setpoint and is decreasing. The MCPR stays well above the OLMCPR and is increasing very quickly. Therefore, the reactor enters a safe-shutdown state, in terms of pressure and MCPR. The results provided in DCD Tier 2, Revision 9, Figure 15.2-7, show a high and narrow power peak about one-third of a second in duration. The applicant has not calculated energy deposition to ensure acceptable fuel cladding interaction. In RAI 15.2-2, the staff requested that GEH explain why it did not consider fuel energy deposition. In the response to RAI 15.2-2, GEH demonstrated that the ESBWR fuel (i.e., GE14E) has the capacity to accommodate a greater amount of thermal energy than that deposited in this transient; therefore, the issue is resolved. (See also related discussion in Section 15.2.1.1.2)

Because the reactor is shut down, and the vessel water is at 12 m (39.37 ft) above TAF (level L3), the staff finds that the reactor is stable. This event is similar to the generator load rejection with a single failure in the turbine bypass system and does not need to be evaluated with each fuel cycle.

ICL: This transient presented in NEDO-33337 is similar to the EC analysis where the feedwater flow is at 150 percent of normal at 30 seconds, the water level is stable, and the vessel pressure is decreasing. The MCPR value is above the OLMCPR.

FWTOD: The staff reviewed this transient in the context of the NEDO-33338 and finds that it is bounded by the IICI events analyzed at the SP2 state point.

15.2.2.5.3 Conclusion

The comparison of the results of the initial core and the EC analyses indicates that the EC at the SP0 state point is the most limiting AOO in the turbine trip with single failure in the turbine bypass system.

The MCPR value for the EC transient is well above the OLMCPR, and pressure and water level are well within normal limits. The staff finds that the FWTOD transient is bounded by other transients. Therefore, the results of this transient meet the acceptance criteria and are acceptable.

15.2.2.6 *Closure of One Main Steam Isolation Valve (MSIV)*

The staff used the acceptance criteria in Section 15.1.1.1 of this report in evaluating this AOO.

15.2.2.6.1 Summary of Technical Information

EC: One MSIV could close under testing conditions (i.e., below certain power levels) without reactor scram. However, at full power, the inadvertent closure of one MSIV will cause closure of all MSIVs, leading to a reactor scram. In this case, the applicant assumed that one MSIV closure at full power does not lead to reactor shutdown. The calculations were performed with this conservative assumption. In DCD Tier 2, Revision 9, Figure 15.2-8 and Table 15.2-12 show the calculated results.

MSIV closure lasts about 3.0 seconds. At the initiation of the closing process, reactor pressure rises, suppressing the core void, which increases moderation and power, but the turbine bypass opens at 2.8 seconds to limit pressure and power increase. Neutronic power peaks at 2.0 seconds but the thermal neutron and the thermal flux do not reach the scram level. At 3.0 seconds the MSIV is closed. Total power assumes a new steady-state at about 101 percent of normal power, with 101 percent of feedwater flow and turbine steam flow at 93 percent of normal. The new steady-state is reached at 40 seconds into the transient. There is a small increase in pressure vessel pressure but the MCPR remains well above the OLMCPR.

ICL: This is similar to the EC transient with an MCPR value of 1.38, a small increase in pressure vessel pressure, power level at 101 percent of the rated level, steam flow at about 93 percent, and a power peak lower than the high thermal flux scram setpoint.

FWTOD: The staff has reviewed this transient in the context of NEDO-33338 and finds that it is bounded by IICI events analyzed at the SP2 state point.

15.2.2.6.2 Technical Evaluation

EC: Under full-power conditions, an MSIV takes 3 seconds to close. During closure, power increases due to void collapse and increased moderation, but fuel temperature reactivity feedback will offset the increase, and total reactivity change returns to zero. Turbine bypass opens at about 2.8 seconds to moderate the pressure and power increase. The calculated results show that the transient has little if any effect on vessel pressure and the MCPR will remain well above the OLMCPR; thus, the acceptance criteria are met. This transient is bounded by the all-MSIV-closure transient, discussed in Section 15.2.2.7 of this report; therefore, it does not need to be reanalyzed for each loading.

ICL: The one MSIV closure transient is almost identical to the EC one MSIV closure transient. This holds true for the power peak and the pressure transient. The MCPR remains above the OLMCPR.

FWTOD: The staff has reviewed this transient in the context of the NEDO-33338 review and finds that it is bounded by IICI events analyzed at the SP2 state point.

15.2.2.6.3 Conclusion

Closure of one MSIV is a minor perturbation in reactor operation without a serious challenge from overpressure, MCPR, the power peak, or the vessel water level. This conclusion assumes that the RPS will work as intended. The pressure stays very close to the operating range, the MCPR is well above the OLMCPR, and the transient assumes a stable-steady state with the core fully covered and does not lead to another transient. Therefore, the EC and ICL results of the analysis of this transient meet the acceptance criteria.

Staff review of this transient for FWTOD finds that it is a mild transient bounded by other events that have been found acceptable.

15.2.2.7 *Closure of All Main Steam Isolation Valves*

The staff used the acceptance criteria in Section 15.1.1.1 of this SER in evaluating this AOO.

15.2.2.7.1 Summary of Technical Information

EC: As stated in Section 15.2.2.6 of this report, inadvertent closure of one MSIV at power levels above the testing power level will cause all of the MSIVs to close. In addition, low steamline pressure, high steamline flow, low water level, or manual action will activate closure of all MSIVs. Total time for completion of MSIV closure is 3.0 seconds. In DCD Tier 2, Revision 9, Figure 15.2-9 and Table 15.2-13 show the calculated results for the evolution of this transient.

MSIV closure initiates a reactor scram on high neutron flux. The same signal also initiates IC operation, which prevents lifting of the SRVs by lowering the RCS temperature and pressure.

The analyses in the DCD conservatively assumed MSIV closure to be completed in 3 seconds, the shortest time in the MSIV closure range, which would thus cause the highest pressure pulse. Vessel pressure reaches a maximum in 4.3 seconds at 7.9 MPa (1,131 psig), while the lowest SRV opening setpoint is at 8.72 MPa (1,250 psig). Control rod insertion is completed within 4.0 seconds, and the MCPR reaches the lowest value at 1.25 seconds, which is well above the OLMCPR. The feedwater flow decreases to about 72 percent of normal at about 4.0 seconds because of increased RCS pressure after void collapse, while core flow increases to about 140 percent of normal. At 20.1 seconds, the L2 vessel water level is reached; at 30.1 seconds, the CRD high-pressure injection is activated; and at 31.82 seconds, the IC valves are fully open and liquid flow from the ICs initiates at about 17 seconds. The reactor water level reaches 11 m (36.1 ft) above TAF in about 20 seconds and keeps rising.

ICL: This transient is similar to the transient in the EC, but the MCPR value is higher. The reactor becomes subcritical in less than a second, and void collapse does not induce a power peak. Pressure will increase to the high-pressure scram setpoint at about 3 seconds but the reactor is already subcritical. IC liquid flow initiates again at about 17 seconds, and the vessel water level starts recovering at about 22 seconds. At that time, the feedwater flow is still above 140 percent of normal.

FWTOD: The staff review of this transient finds that it is bounded by the IICI event that has been found acceptable.

15.2.2.7.2 Technical Evaluation

EC: A variety of circumstances will result in an MSIV closure signal, which also activates turbine bypass, scram, IC initiation, and CRD injection. Assuming that the RPS operates as designed, rod insertion will dominate core reactivity. Within 1.0 second, the reactor becomes subcritical; therefore, void collapse and increased moderation have no effect on power level. Because the ICs and CRD injection initiate simultaneously with the reactor scram signal, operator intervention is not needed. The MCPR remains well above the OLMCPR, and the SRVs are not challenged; therefore, acceptance criteria are met. Because the reactor shuts down within a second and bubble collapse and core pressurization do not create a power spike,

this transient need not be analyzed for different core loadings because no power spike is created that would depend on fuel loading.

ICL: The transient analyses results are almost identical to the EC results, except that the MCPR value remains above the OLMCPR.

FWTOD: The staff has reviewed this transient in the context of the NEDO-33338 review, and the staff finds that it is bounded by the IICI events analyzed at the SP2 state point.

15.2.2.7.3 Conclusion

The comparison of the initial core and the EC analyses indicates that the results of the analyses are similar.

All MSIV closure is a fast-evolving transient where rod insertion, IC initiation, and CRD activation proceed concurrently. Rod insertion dominates the transient reactivity by suppressing the power elevation caused by increased moderation from the pressure pulse.

The pressure increase is within the acceptable range, the MCPR is well above the OLMCPR, and the reactor is shut down in a stable condition with the core covered. Thus, the results of this event meet the acceptance criteria.

15.2.2.8 *Loss of Condenser Vacuum*

The staff used the acceptance criteria in Section 15.1.1.1 of this report in evaluating this AOO.

15.2.2.8.1 Summary of Technical Information

EC: Failure or isolation of the steam jet air injectors and loss of one or more condenser circulating water pumps are common causes of loss of condenser vacuum. Sensing the loss of condenser vacuum, the RPS will initiate turbine trip and reactor scram. Turbine bypass will activate (in 0.02 seconds) to regulate pressure and close the TSVs (in 0.10 seconds). Reactor scram initiation will occur at 0.20 seconds, turbine bypass closure at 6.0 seconds, closure of the MSIVs at 8.0 seconds, and IC activation at 9.8 seconds from the MSIV closure signal. When the vessel water level reaches L2, the high pressure CRD injection initiates (at 24.8 seconds) to control and restore the water level. In DCD Tier 2, Revision 9, Figure 15.2-10 and Tables 15.2-14 and 15.2-15 present the calculated results.

Control rod insertion dominates reactivity response; thus, pressure increase has no effect on power level via void-collapse and increased moderation. For the first 10 seconds, steam flow increases along with feedwater flow. Reactor water level reaches a minimum in 20 seconds at about 1.8 m (5.91 ft) above TAF that is below the L2 level. Because of the fast-acting instrumentation and the TBVs, the vessel pressure trends lower from the operating level. Similarly, as a result of the prompt control rod insertion, the MCPR remains at operating or higher level values. For the initial core, this transient analysis result is almost identical to the EC analysis result.

ICL: Because the reactor shuts down, the results of this AOO are independent of fuel loading, and the results are the same as in the EC analysis.

FWTOD: This event is not analyzed explicitly for FWTOD in NEDO-33338, because it is a mild transient and is not a limiting event. In addition, the reactor is shut down and this transient does not depend on fuel loading.

15.2.2.8.2 Technical Evaluation

EC: Failure or isolation of the steam jet air injectors or loss of one or more condenser circulating water pumps will result in loss of condenser vacuum. When loss of condenser vacuum is sensed simultaneously, the TBVs begin to open to regulate RCS pressure. Reactor scram initiates, and the main TBVs open and initiate MSIV closure, which elevates vessel pressure, collapsing voids, and lowers the reactor water level. The HPCRD injection activates to restore the water level. The MCPR remains well above the OLMCPR, and the high RCS pressure does not challenge the SRVs.

ICL: The ICL transient results are almost identical to those of the EC analysis. The basic feature is the fast response of the RPS and rod insertion that shuts the reactor down. The sequence does not depend on fuel loading.

FWTOD: The staff reviewed this transient in the context of the NEDO-33338 review and finds that it is bounded by the IICI events analyzed at the SP2 state point.

15.2.2.8.3 Conclusion

Loss of condenser vacuum leads to a series of fast actions by the RPS, to scram the reactor, trip the turbine, bypass the existing steam flow, and ensure an adequate RPV water level. Assuming that the instrumentation and the appropriate valves respond according to their design, the vessel pressure remains below operating levels, the MCPR remains well above the OLMCPR, and the reactor is shut down with the core covered and stable. Therefore, the results of this transient meet the acceptance criteria, and the plant response to loss of condenser vacuum is acceptable.

15.2.2.9 *Loss of Shutdown Cooling Function of the Reactor Water Cleanup and Shutdown Cooling System*

There are no specific acceptance criteria for this case because the event is not a transient that involves the reactor.

15.2.2.9.1 Summary of Technical Information

EC, ICL, and FWTOD: This is not a specific AOO; it is a description of a redundant cooling system failure. Therefore, the following information and analyses apply to all three modes of reactor operation. In the ESBWR, the RWCU/SDC system is not a safety system. Nevertheless, it can provide high- and low-pressure water cooling for the core. The system consists of two trains with the necessary piping, heat exchangers, power supply, and instrumentation. In addition to the water cleanup function, the RWCU/SDC provides shutdown cooling where each train takes suction from the RPV and returns cooler water to the feedwater line. Each train has an offsite power supply, but if power is lost, each train has its own independent diesel power supply. In this manner, the system is single-failure proof.

In the event that both trains are lost, the ICS is able to maintain the reactor in stable condition for 72 hours. During refueling, the ICs are unavailable. The GDCS is available to provide

extended decay heat removal for at least 72 hours. After 72 hours, the suppression pool can drain into the vessel via the equalization valves.

Although the RWCU/SDC system is not safety-related, sufficient redundancy exists that the system can be relied upon to provide decay heat removal (closed or open vessel) for extended time periods.

15.2.2.9.2 Technical Evaluation

EC, ICL, and FWTOD: For the shutdown cooling function, each train has its own suction line from the RPV (unlike the current reactors) and returns to the feedwater line. Thus, each of the two RWCU/SDC trains is completely independent of the other. If the single active failure criterion is applied to the RWCU/SDC system, one of the RWCU/SDC trains could be inoperable. However, the operable RWCU/SDC train could achieve cold-shutdown conditions within 36 hours after reactor shutdown.

The RWCU/SDC system, in combination with the ICs, the GDCS, and the water inventory in the suppression pool, is able to provide cooling water for extended periods with a closed vessel or under refueling conditions.

This evaluation does not refer to an AOO; rather, it demonstrates the availability and redundancy of systems able to supply adequate core cooling water for extended periods of time.

15.2.2.9.3 Conclusion

As indicated in the preceding description, the issue in the loss of a train of the RWCU/SDC system is the availability of the redundant train to provide cooling water for removing decay heat after shutdown and, if required, to bring the reactor to a cold-shutdown condition.

There are multiple redundancies for shutdown cooling with the RPV closed or open. The two RWCU/SDC trains have independent supply and discharge lines that are unlike current operating BWRs where, they have a common suction header. The two RWCU/SDC trains also have redundant power supplies. In addition, the ICS could provide core cooling for an additional 72 hours. Under refueling conditions (or with an open vessel) with the RWCU/SDC trains unavailable, the FAPCS could provide cooling. In addition, the GDCS is available to provide core cooling for at least 72 hours. In summary, a fourfold redundancy exists (with either open or closed vessel head) in the ability to supply cooling water for at least 72 hours. Therefore, the possibility of core damage due to RWCU/SDC system malfunction is extremely remote, and the design is acceptable.

15.2.3 Reactivity and Power Distribution Anomalies

15.2.3.1 *Control Rod Withdrawal Error During Startup*

The staff used the acceptance criteria in Section 15.1.1.1 of this report in evaluating this AOO.

15.2.3.1.1 Summary of Technical Information

(Transients in this Section are independent of fuel loading, and thus there is no distinction between EC, ICL, and FWTOD analyses.)

In Tier 2, Section 15.2.3.1, the DCD assumes that during startup, a gang of control rods (or a single rod) is inadvertently withdrawn continuously due to procedural error or a malfunction in the automated rod movement control system. This assumes that the reactor is critical with power less than the low-power setpoint. The RC&IS has a rod worth minimizer (RWM) to prevent any out-of-sequence rod withdrawal. Also RC&IS has restrictions on ganged rod withdrawal sequence such that, if the restrictions are violated, the RC&IS initiates a rod block. The startup range neutron monitor (SRNM) has a period-based scram for periods shorter than 10 seconds.

A typical sequence of events in this AOO begins with the operator withdrawing a rod-gang continuously during startup. No operator action is required to terminate the transient. DCD Tier 2, Revision 9, Section 15.3.8, presents a bounding analysis which does not credit the rod block action. Review of DCD Section 15.3.8 indicates that if the SRNM rod block is not credited, the power spike that follows rod gang withdrawal (either from zero- to 15-percent power) will result in a fuel enthalpy increase that is within the AOO acceptance criteria.

15.2.3.1.2 Technical Evaluation

<u>EC, ICL, and FWTOD</u>: In RAI 15.2-10 and its supplement RAI 15.2-10 S01, the staff requested the applicant's description and the plant's response to a reactivity and power distribution anomaly. In the initial response, the applicant addressed only the electronic part of the system. The staff noted that there were electrical as well as mechanical causes of control rod malfunction and both should be addressed.

In the revised response to RAI 15.2-10 S01, the applicant provided operational information regarding the RC&IS controls FMCRDs employed in the electrical movement of control rods. Mechanical failure of a single relay will not cause an inadvertent RWE. Additionally, failure of the mechanical contact of a switch will not cause RWE because they are single-failure proof with respect to RWE. GEH extended the argument to the electronic equipment also being redundant, failure of which at most will result in the inability to move the associated FMCRD by normal motor movement in the event of a single failure. The response describes several additional improvements in the FMCRDs to support the argument that RWEs are unlikely and can result only from multiple failures. Should an RWE take place a period-based rod block for SRNM occurs for periods shorter than 20 seconds and a scram for periods shorter than 10 seconds. The response references DCD Tier 2, Sections 15A and 7.7.2. The staff accepts this response because both the mechanical and the electronic elements are single failure proof, and there exist additional means to block rod withdrawal or scram the reactor. Based on the applicant's response, RAI 15.2-10 is resolved.

In addition, DCD Tier 2, Revision 9, Section 15.3.8 provides analyses for the case in which the SRNM rod block is not credited with stopping rod or rod-gang withdrawal. The analyses indicate that the energy deposition in the fuel (with conservative adiabatic assumptions) will meet the AOO acceptance criteria. Therefore, the results of this transient analysis are acceptable. Neither reactor scram nor operator action is required to mitigate this AOO.

<u>ICL and FWTOD</u>: The startup RWE transient and the reactor response are independent of fuel loading; as indicated above, for this AOO, there is no differentiation of the EC analyses.

15.2.3.1.3 Conclusion

The comparison of the initial core and the EC analyses results indicates that the analyses are similar, which is to be expected because the actions involved are independent of fuel loading.

The staff agrees that DCD Tier 2, Revision 9, Section 15.2.3, supports the conclusion that transients that may result from an inadvertent rod or rod-gang withdrawal from a critical reactor (or reactor power up to 15 percent of nominal power) meet the AOO transient acceptance criteria. Reactor scram could be invoked. Operator action is not required to mitigate this AOO.

15.2.3.2 *Control Rod Withdrawal Error during Power Operation*

15.2.3.2.1 Summary of Technical Information

(This section is independent of fuel loading; thus, there is no distinction between EC, ICL, and FWTOD.)

The staff used the acceptance criteria in Section 15.1.1.1 of this report in evaluating this AOO.

During power operation, the ATLM performs the rod block monitoring function as a dual channel subsystem of the RC&IS. One channel monitors the MCPR, and the other monitors the MLHGR. The rod block algorithms for both channels are based on actual online core thermal data to protect the MCPR and MLHGR setpoints. Regardless of the origin of the rod withdrawal malfunction, the activation of rod blocks protects the thermal limits. Therefore, an inadvertent rod (or rod gang) withdrawal will be terminated without operator intervention or reactor scram.

The power operation RWE transient and the reactor response are independent of fuel loading; therefore, for this transient, there is no differentiation of the EC from the ICL and FWTOD cores.

15.2.3.2.2 Technical Evaluation

<u>IC, ICL, and FWTOD</u>: The ATLM continuously monitors the MCPR and MLHGR limits and intervenes to prevent violation of either limit due to a rod (or rod-gang) withdrawal error. Because there are two channels, the signal is single-failure proof, no reactor operator action is required, and no scram signal will be generated.

DCD Tier 2, Revision 9, Section 15.3.8 presents a bounding analysis in which the rod block action is not credited. Staff review of Section 15.3.8 indicates that if the SRNM rod block is not credited, the power spike that follows rod-gang withdrawal (either from zero- or 15-percent power) will increase the fuel enthalpy and the increase will be within the AOO acceptance criteria.

15.2.3.2.3 Conclusion

The ATLM system is single-failure proof and protects the fuel. Regardless of the cause of rod withdrawal, the ATLM will intervene to stop rod withdrawal and protect the thermal limits. This action does not require operator intervention or reactor scram therefore, the result is acceptable. In addition, the probability of failure of the ATLM system that would result in an inadvertent rod (or rod-gang) withdrawal is extremely small.

15.2.4 Increase in Reactor Coolant Inventory

15.2.4.1 *Inadvertent Isolation Condenser Initiation*

15.2.4.1.1 Summary of Technical Information

The staff used the acceptance criteria in Section 15.1.1.1 of this report in evaluating this AOO.

<u>EC and ICL</u>: In its analysis of IICI, the applicant assumed that all IC units are activated. Therefore, this is a bounding case. While IC flow initiates at about 10 seconds from activation, the system establishes full IC flow in about 32 seconds. Due to cold-water injection, core power increases due to increased moderation, reaches a maximum at about 50 seconds, and returns to normal level at about 200 seconds. Feedwater flow decreases accordingly to keep the vessel water approximately at the same level. Accounting for the IC liquid flow, the total water injection is about equal to full feedwater flow. An MCPR value of 1.25 is reached at about 150 seconds into the transient. The analysis assumes that the system operates without additional failures. DCD Tier 2, Revision 9, Figure 15.2-11, shows the calculated results, and DCD Tier 2, Revision 9, Table 15.2-17, depicts the sequence of events.

With a power peak at about 110 percent and total water injection (feedwater plus IC liquid) at about normal feedwater flow, the MCPR is well above the OLMCPR and occurs at 150 seconds into the transient. Vessel pressure stays at the normal operating level. The power increase stabilizes at normal power level at about 300 seconds. There is no power scram, and no operator intervention is needed to mitigate this transient.

<u>FWTOD</u>: NEDO-33338 presents analytical results for this transient. The calculation is for SP2 conditions, and the transient evolves similarly to that in the EC and ICL analyses. The MCPR value is above the OLMCPR and occurs 130 seconds into the transient.

15.2.4.1.2 Technical Evaluation

EC and ICL: Assuming that all four IC units are activated, the transient represents a bounding case. The only reasonable assumption for the simultaneous initiation of all ICs is inadvertent manual operator action. Cold water injection into the vessel increases water density, core moderation, and core power. The transient proceeds relatively slowly with a gradual increase in the thermal power and corresponding variation in the feedwater flow. The calculated results indicate that, in about 300 seconds, the reactor attains equilibrium operation at normal power, with feedwater flow at about 90 percent of normal. The MCPR remains well above the OLMCPR. Vessel pressure stabilizes at a slightly lower level than normal. The core remains fully covered and stable. No scram signal or operator action is required in this transient.

FWTOD: NEDO-33338 presents analyses of the inadvertent IC activation for operation SP2 and shows this case to be limiting. Because of the lower power level and the power increase due to cold injection, the transient peak power is less than 100 percent of rated power. At the initiation of the transient, the IC liquid contribution complements the feedwater flow. Vessel pressure shows no significant variation, and water level stays over 12 m (39.37 ft) above TAF. The MCPR value remains above the OLMCPR. NEDO-33338, Figure 2.3-5 shows the evolution of the transient. The calculated ΔCPR/ICPR of 0.12 for EC at the SP2 state point is indicated as the most limiting of the AOOs.

DCD Section 15.2.4.1.3 states that this transient is potentially limiting with respect to the OLMCPR. Therefore, this transient should be analyzed for each cycle and for FWTOD.

15.2.4.1.3 Conclusion

Inadvertent activation of all four IC units causes a bounding cold-water injection transient. From the above discussion, it is apparent that the pressure remains well within the acceptance limits, the MCPR stays well above the OLMCPR, the core remains fully covered, and the reactor returns to a stable state. Therefore, the results of this AOO meet the acceptance criteria.

15.2.4.1.4 Post-COL Activity

This event is analyzed for each fuel cycle with the limiting SCRRI/SRI rod pattern, which changes from cycle to cycle. The SCRRI/SRI requirements are documented in the COLR in accordance with the TS. The OLMCPR is established for the limiting event and documented in the TS Section 5.6.3, COLR (a)(2) in accordance with TS 3.2.2.

15.2.4.2 *Runout of One Feedwater Pump*

15.2.4.2.1 Summary of Technical Information

The staff used the acceptance criteria in Section 15.1.1.1 in evaluating this AOO.

<u>EC and ICL</u>: The EC and ICL transients are very similar in that the size and timing of the changes after a pump initiates runout are similar.

Three feedwater pumps are running continuously during normal operation. Feedwater pumps are motor-driven with variable speed motors. A runout transient consists of one pump increasing speed (and feedwater flow) to its maximum capacity. The FWCS uses a triplicate digital control system, including a fault-tolerant controller.

The controller contains three parallel processing channels, each with microprocessor-based hardware and associated software necessary to perform the control calculations. The operator interface provides system status and required control functions. The processor is capable of identifying faults and isolating faulty channels. However, two credible single failures could lead to loss of one actuator for one feedwater pump with increasing flow. The analyses presented in DCD Tier 2, Revision 9, Figure 15.2-13, and DCD Tier 2, Tables 15.2-18 and 15.2-19, consider such a case.

When the system senses the increased flow, the feedwater controller will lower feedwater flow to the two operating pumps so that the total flow stays at the predetermined value with a minimal disturbance to the system. This occurs in about 21 seconds. The vessel pressure does not change perceptibly. Fuel temperature and void reactivity change in opposite directions, resulting in small changes in total reactivity compensated by small control rod movement. Feedwater flow changes equalize at about 40 seconds, and reactivity variations stabilize at about 100 seconds into the transient. The MCPR value is above the OLMCPR.

This transient has not been explicitly analyzed in the FWTOD because it is bounded by events that have been analyzed and found to be acceptable.

15.2.4.2.2 Technical Evaluation

<u>EC, ICL, and FWTOD</u>: This transient results in increased feedwater flow caused by runout of a single feedwater pump. Feedwater controller action to reduce feedwater flow promptly compensates for increased feedwater, and the system achieves normal water level at about 40 seconds into the transient, according to the submitted analytical results.

The increase in feedwater causes brief reactivity changes, which are self-compensating as they produce limited variation in power. The transient does not initiate a scram, the MCPR remains well above the OLMCPR, and there is a barely perceptible variation in pressure. No operator action is required to mitigate this transient, and there is no scram. The analytical results meet the acceptance criteria.

15.2.4.2.3 Conclusion

Single feedwater pump runout creates a minor disturbance to reactivity, power, feedwater flow, vessel pressure, and reactivity components. The calculated results indicate that vessel pressure remains at normal operating level, the OLMCPR stays well above the SLMCPR, and the core returns to a fully covered and stable position. Therefore the staff concludes that the results of this transient satisfy the acceptance criteria.

15.2.5 Decrease in Reactor Coolant Inventory

15.2.5.1 *Opening of One Turbine Control or Bypass Valve*

15.2.5.1.1 Summary of Technical Information

The staff used the acceptance criteria in Section 15.1.1.1 of this report in evaluating this AOO.

<u>EC</u>: Instrumentation failure, such as actuator or voter failure could cause inadvertent opening of a turbine bypass valve (TBV) or a TCV. Such failure is highly unlikely because the SB&PC system has a triplicate control configuration (see Section 15.2.4.2 of this report) so that no credible single failure can result in TCV or TBV failure.

Inadvertent operator action could cause a TCV or a TBV to open. DCD Tier 2, Revision 9, Figure 15.2-14, shows the evolution of the transient, and DCD Tier 2, Revision 9, Table 15.2-20, lists the sequence of events. The calculated results indicate that, at the initiation of the transient, steam flow increases very briefly, which increases the void fraction, causing a corresponding dip in power and dome pressure. However, the lower pressure increases the feedwater flow, which promptly increases moderation, and power recovers at about 30 seconds into the transient. At this time, the turbine steam flow reduces to about 82 percent of normal, and the TCV flow remains at 15 percent. Regarding reactivity changes, void reactivity dips sharply within the first second of the transient, which is partially compensated for by fuel temperature and the increase in feedwater flow causing void collapse and total reactivity to return to critical at about 30 seconds into the transient. The vessel pressure remains almost unchanged from the normal operating value, the MCPR stays well above the OLMCPR, and the reactor assumes a stable condition while the fuel remains covered. No operator action is required to mitigate this event, and no scram signal is initiated.

ICL: For the initial core, this transient is very similar to the one for EC, but the reactivity oscillations are somewhat smaller. The vessel pressure is reduced somewhat, the MCPR remains well above the OLMCPR, and the reactor assumes a new stable steady state.

FWTOD: The staff reviewed this transient in the context of the NEDO-33338 and finds that it is bounded by IICI events analyzed at the SP2 state point.

15.2.5.1.2 Technical Evaluation

EC, ICL, and FWTOD: In the event of an inadvertent or faulty opening of a TCV or a turbine bypass valve (TBV), assuming the control and instrumentation will operate as designed, the SB&PC system will promptly compensate for the bypass steam, arrest the evolution of the transient, and return the reactor to a stable state without operator intervention or reactor scram. In DCD Tier 2, Revision 9, Table 15.2-20 and Figure 15.2-14 demonstrate the evolution of the event. There is a small increase in steam flow after transient initiation, with a corresponding oscillation in feedwater flow and steady steam flow from the TCV or bypass flow. At about 30 seconds into the transient, the system establishes a new steady-state. The vessel pressure is reduced by a small amount from the normal operating level, the OLMCPR remains well above the SLMCPR, and the reactor returns to a stable condition.

15.2.5.1.3 Conclusion

Opening of a TCV or TBV creates a minor disturbance, mainly as a result of the automated action of the control system to adjust bypass or turbine control flow. The vessel pressure is subjected to a very small change, the OLMCPR stays well above the operating safety limit, and the reactor achieves a stable and covered core steady-state. Therefore, the results of this transient satisfy the acceptance criteria.

15.2.5.2 *Loss of Non-Emergency ac Power to Station Auxiliaries*

15.2.5.2.1 Summary of Technical Information

Loss of power to the station auxiliaries can result from lightning, storms, wind, load instabilities, loss of load, load rejection, or similar causes that could lead to failure of the unit auxiliary transformer. In this analysis, it is assumed that concurrent with load rejection, simultaneous loss of power occurs on the four power generation buses, which will cause the feedwater and condenser circulation water pumps to be lost. Loss of the circulating water pumps results in loss of the condenser vacuum, which in turn causes turbine trip MSIV closing and reactor scram. The bypass valves will be initially available, but loss of the power buses will produce initiation signals for the ICs and HPCRD injection. With the loss of the station transformer, the station standby diesel generators will activate to provide power to the CRD pumps. The CRD startup signal is generated when the wide range water level indication falls below the L2 level (for longer than 10 seconds). However, CRD injection is delayed by 145 seconds until the diesels are up in power. Water level is regained above the L2 level at about 800 seconds. The MCPR value remains above the OLMCPR. In summary, loss of the station auxiliary power will lead to reactor scram. In DCD Tier 2, Revision 9, Table 15.2-21 shows the sequence of events, and Figure 15.2-15 depicts the time dependent variation of the reactor parameters.

The main condenser loss of vacuum signal has a time delay of 50 seconds. Upon loss of load, feedwater flow decreases briefly, followed by a very short power spike from increased moderation with the closing of the MSIVs. The power spike leads to reactor scram. About

100 seconds after initiation, IC water supply increases sharply, then levels off about 20 seconds later.

FWTOD: From the review of the EC and ICL cores, the staff concludes that this transient is not bounding. The critical success parameter is the water level, which for the SP1 is higher than the SP0 level; for SP2, the final water level may be slightly less than SP0.

15.2.5.2.2 Technical Evaluation

EC and ICL: The reactor pressure remains well below the AOO limit of 110 percent of the design value, the MCPR remains well above the OLMCPR, and the reactor is shut down. CRD high pressure injection controls the water level. CRD and IC injection ensure core cooling and core water level recovery. No operator action is required to mitigate this AOO and bring the reactor to a stable shutdown state. Since the core is covered, in a stable state, and cooled, the staff concludes that the acceptance criteria are met.

FWTOD: The staff reviewed this transient in the context of the NEDO-33338 review and finds that it is bounded by the IICI events analyzed at the SP2 state point.

15.2.5.2.3 Conclusion

Loss of all nonemergency ac power to station auxiliaries leads to turbine trip and reactor scram, with concurrent IC and HPCRD pump activation and injection. After a short pressure-and-power pulse, the vessel depressurizes and power reduces to zero. In the transition to shutdown, the MCPR remains well above the SLMCPR; therefore, the results of this transient meet the acceptance criteria.

15.2.5.3 *Loss of All Feedwater Flow*

The staff used the acceptance criteria in Section 15.1.1.1 of this report in evaluating this AOO.

15.2.5.3.1 Summary of Technical Information

EC: Loss of all feedwater flow could result from operator errors, pump failure, or reactor trip signals. In DCD Tier 2, Revision 9, Table 15.2-22 lists the sequence of events, and Figure 15.2-16, shows the variation of reactor parameters as a function of time. If the feedwater pumps trip, the ensuing reduction of feedwater flow will initiate IC operation. At about 6 seconds into the transient, the feedwater flow decays to zero, the vessel water level drops to RPV Level 2 (it is assumed that initial water height is at normal level), the HPCRD injection initiates at 20 seconds, the ICs reach full flow at 33 seconds and the MSIVs close at 40 seconds. At about 80 seconds, the water level recovers to about 13 m (43 feet) above TAF, and the core is shut down and stable.

ICL and FWTOD: Neither is dependent on fuel loading; therefore, reactor response is the same as in the EC case. The transient is not bounding and has not been explicitly analyzed in the FWTOD.

15.2.5.3.2 Technical Evaluation

EC, ICL, and FWTOD: The RPS will scram the reactor and initiate ICs to ensure water level recovery. During this transient, the vessel pressure quickly drops below normal operating values to about 70 percent of normal in less than 200 seconds.

The OLMCPR remains well above the 1.31 value (the designated OLMCPR value), and the reactor is shut down, with the core covered in a stable, cooled state. Therefore, the results of this transient satisfy the acceptance criteria.

15.2.5.3.3 Conclusion

Loss of all feedwater flow results in a fast reactor shutdown and simultaneous IC and CRD high-pressure injection activation. This transient does not violate any of the AOO acceptance criteria; therefore, the transient analysis is acceptable.

15.2.6 Conclusion of Anticipated Operational Occurrence Review

The staff concludes that the requirements of GDC 10, 13, 15, 17, 20, and 26 have been met. This conclusion is based on the following:

- The applicant meets the requirements of GDC 10 that the SAFDLs are not exceeded.
- The applicant meets GDC 13 requirements by demonstrating that all credited instrumentation is available and that actuations of protection systems automatically occurred at values of monitored parameters that were within the instruments' prescribed operating ranges.
- The applicant meets the requirements of GDC 15 that the design conditions of the RCPB are not exceeded.
- The applicant meets the requirements of GDC 17 and 26 by demonstrating that SAFDLs are not exceeded.
- The applicant meets the requirements of GDC 20 that the reactivity control systems are automatically initiated so that SAFDLs are not exceeded.
- In addition, the review identified IICI as the limiting AOO with respect to the MCPR.

15.3 Analysis of Infrequent Events

This section covers the material in DCD Tier 2, Revision 9, Section 15.3. IEs are defined as events with an expected frequency of less than 1.0×10^{-2} pry of operation. DCD Tier 2, Revision 9, Section 15A.3, presents the expected frequency of these events. Section 15A of this evaluation report provides the staff's evaluation of the event frequency determination.

The staff reviewed DCD Tier 2, Revision 1, Section 15.3, and found that the applicant had not provided a complete source term for the radiological consequence analysis for the IEs identified in the DCD. In RAI 15.3-25, the staff requested that the applicant revise DCD Tier 2, Tables 15.3-13 and 15.3-16, by adding applicable information pertaining to radiological consequence analysis for those IEs listed in the DCD. GEH revised DCD Tables 15.3-13 and

15.3-16 and made the requested changes in the DCD. Based on the applicant's response, RAI 15.3-25 is resolved.

In RAI 15.3-26, the staff noted that design certification requires analyses of all IEs, but only the limiting events will need to undergo analysis during the COL licensing phase. The staff found that the applicant needs to revise DCD Tier 2, Table 15.3-1, to show the results of all IEs. The applicant also needs to analyze the events described in SRP Sections 15.3.7 to 15.3.12 and 15.3.14. GEH incorporated the requested changes in Revision 5 of the DCD. It modified Sections 15.3.9.3 through 15.3.9.5 and added Table 15.3-1b. The changes are responsive to the staff's request, and the issue is resolved. Therefore, based on the applicant's response, RAI 15.3-26 is resolved.

15.3.1 Loss of Feedwater Heating—Infrequent Event

15.3.1.1 *Summary of Technical Information*

The staff used the acceptance criteria in Section 15.1.1.2 of this report in evaluating this IE.

EC: LOFWH with Failure of SCRRI/SRI

LOFWH can occur in at least two ways: (1) the steam extraction line to the heater is closed or (2) the feedwater flow bypasses the heater. LOFWH will decrease FWT and increase core water density, resulting in increased core neutron moderation, and an increase in power. The DCD Tier 2, states that the ESBWR design is such that no single failure or operator error will cause LOFWH greater than 55.56 degrees C (100 degrees F). Normally, LOFWH and the associated temperature decrease will be detected by the ATLM and/or the DPS, either of which will activate the SCRRI/SRI to counter the positive reactivity insertion from cold-water injection and partial void reduction and thus avoid reactor scram. In this case, it is assumed that SCRRI/SRI insertion fails. This event is calculated at the simulated thermal power trip (STPT) scram setpoint. The maximum thermal power rises to 115.4 percent of the normal power level. DCD Tier 2, Revision 6, Figure 15.3-1 and Table 15.3-1a show the calculated results as a function of time. The results indicate that the addition of void reactivity is counterbalanced by fuel temperature reactivity. The power level remains at 115.4 percent of normal power at the end of the calculation at 300 seconds. No operator action is required to mitigate this event. The MCPR value is higher than the OLMCPR.

The MCPR value with failure of SCRRI/SRI indicates that the number of fuel rods to enter boiling transition will be bounded by 1,000; therefore, the expected radiological consequences are within the acceptance limits. The estimated frequency of this event is less than 1.0×10^{-2} pry, which classifies it as an IE, as indicated in DCD Tier 2, Section 15A.3.6.3.

ICL: This event is very similar to that for EC. A notable difference is that the reactivity control fraction is positive (maximum about 2 cents) for the EC case, while the ICL does not require reactivity compensation.

FWTOD: In the FWTOD at higher FWTs, the heating valves of feedwater heater No. 7 are open. Under those conditions, FWT minimum demand failure will result in closure of the No. 7 heater valves. The resulting temperature decrease could be higher than 55.56 degrees C (100 degrees F). The ATLM and/or the DPS will detect the FWT decrease and initiate SCRRI/SRI insertion which is credited for this event. In the event of ATLM, DPS or SCRRI/SRI failure to insert the reactor will scram when the STPT setpoint is reached. No operator action is

required to mitigate this event; however, at the end of the transient the operators must not permit reactor operation at elevated power levels.

Appendix A.3, “Infrequent Events,” to NEDO-33338 presents the calculated results for the SP2 state point. The event is similar to the EC, except that the average fuel temperature reactivity component is higher than the void component and the control component is about 5 cents, which reflects the increased fuel average temperature. This event results in reactor scram.

15.3.1.2 *Technical Evaluation*

EC and ICL: This case involves two separate events. The first concerns the reference core where the maximum temperature decrease in the feedwater is 55.56 degrees C (100 degrees F), and the second is the FWTOD case where it is possible to have FWT differences greater than 55.56 degrees C (100 degrees F).

In the first case, EC maximum ΔT = 55.56 degrees C (100 degrees F), the calculated results indicate that coolant pressure will remain within normal limits, power will rise to 115.4 percent of rated power, and the MCPR will stay above the SLMCPR. Operator action is not required to mitigate the event.

In the second case, FWTOD: maximum ΔT > 55.56 degrees C (100 degrees F),, arises only during the FWTOD operation when heater No. 7 is in service. FWT controller failure (to minimum temperature demand) results in closure of the No. 7 heater steam heating valves and subsequent opening of the high-pressure feedwater heater bypass valves. The resulting decrease in FWT is potentially higher than 55.56 degrees C (100 degrees F). The first credible response is the STPT signal to scram the reactor, but STPT scram is not credited here. This case credits the ATLM and the DPS to scram the reactor when power exceeds 101.0 percent of rated power (85-percent power at SP2 condition). In crediting the scram (initiated by the ATLM or DPS), the analysis of the event has similar results to those obtained for the EC, ICL cases, that are not repeated here. The reactor attains a power level of about 107 percent of rated power. No operator action is required to mitigate this transient.

15.3.1.3 *Conclusion*

This event at the SP2 state point is the limiting event for LOFWH for the equilibrium and the initial core. In this event, it is assumed that all fuel rods entering transition boiling will fail. The number of rods in boiling transition is confirmed to be less than 1,000, based on the ΔCPR/ICPR results.

The maximum pressure remains within the limits of normal operating pressure, the vessel water level remains above 13 m (42.66 ft) (i.e., above the L3 setpoint), the MCPR remains above the SLMCPR, and the reactor stabilizes at 101.00-percent rated power at SP2, about 200 seconds into the event. Therefore, the event resulting from LOFWH with SCRRI/SRI failure to insert satisfies the acceptance criteria.

15.3.1.4 *Post-COL Activity*

This event is potentially limiting with respect to the number of rods in boiling transition. The OLMCPR for each fuel cycle is established for the limiting event and documented in the COLR in accordance with TS.

The thermal-mechanical analysis for each fuel cycle confirms that the calculated results remain within the assumptions of the radiological analysis. Any resulting limits on MLHGR are documented in the TS Section 5.6.3, COLR (a)(1) in accordance with TS 3.2.1.

15.3.2 Feedwater Controller Failure—Maximum Flow Demand

15.3.2.1 *Summary of Technical Information*

The staff used the acceptance criteria in Section 15.1.1.2 of this report in evaluating this IE.

EC: The DCD, referencing the design of the FWCS, states that feedwater controller failure requires several failures of feedwater pumps to result in maximum feedwater demand. The estimated frequency is less than 1.0×10^{-2} pry, which classifies it as an IE (DCD Tier 2, Section 15A.3.5.1). With excess feedwater flow, the water level rises to the high reference point (Level 8, 14.5 m [47.57 ft] above TAF), where feedwater pumps initiate a runback, the main turbine trips, and the reactor scrams. At water Level 8, there is a feedwater isolation signal. However, in this analysis, it is not credited because it does not make a significant difference. In DCD Tier 2, Revision 9, Figure 15.3-2 depicts the sequence of events as a function of time, Table 15.3-1a summarizes the main points of the event, and Table 15.3-3 lists the complete sequence of events.

The calculated results indicate that the feedwater flow is ramped up to 170 percent of normal flow in about 2.5 seconds. At 12.7 seconds, the TBVs open to control vessel pressure. The vessel water reaches L8 at 14.5 seconds. At 15.4 seconds, turbine trip, reactor scram, and feedwater pump runback are activated. The main TBVs complete their opening at 15.5 seconds to relieve vessel pressure. At 15.6 seconds, a scram initiates with rod insertion. The value of the MCPR remains higher than the designated OLMCPR; therefore, no fuel damage or radioactive releases are anticipated. After 20 seconds, the reactor vessel water level falls to L2, which activates the ICs and CRD high-pressure injection operation to recover water level.

ICL: This event is almost the same as the EC event (i.e., the reactor scrams, and the water level falls to L2, which activates the ICs and the CRD high-pressure injection operation to recover water level).

FWTOD: Analyses of the FWTOD events in NEDO-33338 demonstrates that IICIs is the limiting event. Therefore, this event has not been explicitly analyzed in NEDO-33337.

15.3.2.2 *Technical Evaluation*

EC, ICL, and FWTOD: The DCD states that runout of all feedwater pumps requires more than one failure to take place. As such, the anticipated frequency is lower than 1.0×10^{-2} pry, and the event is included in the IE category. The calculated results indicate that the excessive feedwater flow event will cause minimal disturbance to the reactor, in that there is a small and short power peak and a corresponding small pressure peak, but the value of the MCPR will remain well above the designated OLMCPR limit. The reactor scrams, and when the vessel water level falls to L2, the ICs and the CRD high-pressure injection are activated to recover the water level. The reactor is continuously covered above the L2 level, and the vessel pressure remains close to the operating limits. No operator action is required. Therefore, the results of this event meet the Section 15.1.1.2 acceptance criteria of this report.

15.3.2.3 *Conclusion*

This event does not result in any fuel failures or any release of primary coolant, and hence no radiological consequence analyses were performed.

In this event, the excessive feedwater flow causes an insignificant perturbation to vessel pressure but does not violate the fuel SLMCPR. At the end of the event, the reactor is in a stable condition. Therefore, the results of the analyses are acceptable.

15.3.3 Pressure Regulator Failure: Opening of All Turbine Control and Bypass Valves

15.3.3.1 *Summary of Technical Information*

The staff used the acceptance criteria in Section 15.1.1.2 in this report in evaluating this IE.

<u>EC and ICL</u>: The SB&PC system controls vessel pressure and steam turbine bypass. In Section 15.2.5 of this report, the staff examined the accidental opening all TCVs and TBVs. The electronic logic aspects of the SB&PC system are such that it would take multiple failures to accidentally open all of the TCVs and TBVs. Therefore, this event is considered as having a very small probability of occurrence and is categorized as an IE.

DCD Tier 2, Revision 9, Section 15A.3.1.1, estimates that the frequency of this event is less than 1.0×10^{2} pry. In DCD Tier 2, Revision 9, Table 15.3-4 and Figure 15.3-3 illustrate the calculated results of the time-dependent evolution of the event. At 19.30 seconds into the event, turbine inlet low pressure will initiate MSIV closure, which in turn will initiate reactor scram and IC operation at 20.5 seconds. At 24.1 seconds, MSIV closure will be completed, but bypass valves will remain open. Because of increased steam flow, the water level decreases, reaching the RPV L2 level at 31.60 seconds. At 36.50 seconds, the IC begins to return condensate coolant to the vessel, and at 41.80 seconds, the HPCRD injection starts and vessel water level recovery initiates.

As stated in the DCD, the ESBWR has a 105-percent bypass capacity. Opening all of the bypass valves produces rapid depressurization, which results in an increase in the void fraction which reduces power. In the first few seconds, the feedwater system attempts to stabilize operation by increasing feedwater flow (due to lower vessel pressure, as shown in DCD Tier 2, Revision 9, Figure 15.3-3a). The MSIV position-switch scrams the reactor. Simultaneously, the IC steam flow increases to about 20 percent of normal steam flow because the MSIVs are closed. At this time, reactor operation stabilizes, with IC cooling having achieved normal water level. The MCPR value stays well above the safety limit value and increases during the event. No operator action is required to mitigate this event. The reactor scrams.

<u>FWTOD</u>: Analyses of the FWTOD events in NEDO-33338 demonstrates that IICIs is the limiting event. Therefore, this event has not been explicitly analyzed in NEDO-33337.

15.3.3.2 *Technical Evaluation*

<u>EC, ICL, and FWTOD</u>: The important feature is that vessel depressurization leads to decreasing power, reactor scram, and initiation of CRD and IC cooling.

Assuming that the required instrumentation and systems will operate as designed and as expected, the results of this event meet the acceptance criteria, vessel pressure decreases from

the operating pressure, and MCPR increases after event initiation. Therefore, no cladding damage occurs, and the event evolves into a stable state.

DCD Tier 2, Revision 3, Section 15.3.3.1, states that “the event is considered as a limiting fault.” As stated in RAI 15.3-29, the staff did not agree with this characterization of the event. This event is an IE, as noted in other parts of the DCD, and in RAI 15.3-29, the staff requested that the applicant revise this section of the DCD to characterize the event as an IE rather than as a limiting fault. GEH agreed and changed the DCD to show the event as an IE. The staff confirmed that this change was included in DCD Tier 2, Revision 5 therefore, RAI 15.3-29 is considered resolved.

15.3.3.3 *Conclusion*

This event does not result in any fuel failures or any release of primary coolant, and hence, no radiological consequence analysis was performed. Inadvertent opening of the TCVs and TBVs from power results in the following: fast reactor vessel depressurization, decrease in reactor power, vessel isolation, reactor scram, and IC initiation. The calculated results indicate that the vessel pressure remains below normal operating values, the MCPR is well above the designated OLMCPR, and the reactor is cooled by the ICs and assumes a stable state.

The results of this event satisfy the acceptance criteria; therefore, this event analysis is acceptable.

15.3.4 Pressure Regulator Failure: Closure of All Turbine Control and Bypass Valves

15.3.4.1 *Summary of Technical Information*

The staff used the acceptance criteria in Section 15.1.1.2 of this report in evaluating this IE.

EC and ICL: This event assumes failure of the SB&PC system, with closure of all TCVs and TBVs. The DCD states that for this event to occur, more than a single failure is necessary and the probability is exceedingly low. DCD Tier 2, Revision 9, Section 15A.3.2.1, estimates the event frequency as less than 1.0×10^{-2} pry, which classifies it as an IE.

As the TCVs and TBVs begin to close, vessel pressure increases and the core void collapses, which increases moderation and power until reaching the neutron high-flux setpoint at 1.78 seconds, initiating reactor scram. Control rod insertion starts at 2.03 seconds. TCV closure is completed at 2.5 seconds into the event. At about 20 seconds, CRD high-pressure injection activates on RPV L2 to recover vessel water level. In DCD Tier 2, the calculation is carried to 50 seconds. At the end of this time, the reactor has recovered the water level, and dome pressure is about 50 percent of normal operating pressure. Vessel pressure peaks at 6.0 seconds at 114 percent of normal operating pressure, but the MCPR remains above the OLMCPR throughout the event. IC initiation does not take place, even though Level 2 is reached because neither the high dome pressure nor the low-water-level signals are in effect for more than 10 or more seconds and 6 or more seconds, respectively, required for IC initiation. No operator action is required to mitigate this event and the reactor scrams.

FWTOD: Analyses of the FWTOD events in NEDO-33338 demonstrates that IICIs is the limiting event. Therefore, this event has not been explicitly analyzed in NEDO-33337.

15.3.4.2 *Technical Evaluation*

EC and ICL: Amendment 26 to "General Electric Standard Application for Reactor Fuel (GESTAR) II," dated March 29, 2000, approved the change of this event from moderate frequency to IE for BWR/6 plants. The amendment stated that "the classification of the pressure regulator downscale failure as an AOO was also re-evaluated and it was concluded that the expected frequency of the initiating failure was below the moderate frequency event definition, and was reclassified as an infrequent event."

The applicant based the categorization of this event as an IE on the performance of the SB&PC system. The peak pressure reaches 114 percent of operating pressure (i.e., it remains below the 110 percent of design pressure). The MCPR value remains above the OLMCPR; thus, no fuel damage is expected during this event. The reactor recovers water level, and the operator has a number of choices for long-term cooling.

DCD Tier 2, Figure 15.3-4a, indicates a sharp rise in total power. In RAI 15.3-11, the staff questioned the ability of TRACG to represent the sharp energy peak in pressure increase transients. The staff accepted the response that the TRACG is qualified to represent the energy peak. In RAI 15.3-11 S01, the staff requested the applicant to calculate the total energy deposition and the resulting cladding strain.

GEH responded to this request in a way that encompassed all the events that exhibit similar event behavior (i.e., high power, narrow peak events). GEH stated that the GE14E scheduled to fuel the ESBWR reactor is designed to withstand much greater energy deposition than any of the IEs in Section 15.3 of the DCD Tier 2. As stated in Section 15.2.3.1.2 of this evaluation the staff finds that neither fuel melting nor cladding strain is an issue for the ESBWR with GE14E fuel. Therefore, based on the applicant's response, RAI 15.3-11 is resolved.

FWTOD: This event is bounded by other events that have been found acceptable. This IE has not been explicitly analyzed in NEDO-33338.

15.3.4.3 *Conclusion*

This event does not result in any fuel failures or any release of primary coolant, and hence no radiological consequence analysis needs to be performed.

Inadvertent closing of the TCV and TBVs from power, results in fast vessel pressurization, power increase, reactor scram due to high power, and HPCRD activation to recover vessel water level. The calculated results indicate that the vessel pressure exceeds normal operating values but remains well below the SRV setpoint and 110 percent of the design pressure. The MCPR is above the designated OLMCPR, and the reactor assumes a stable state, therefore, the calculated results of this event satisfy the acceptance criteria.

15.3.5 Generator Load Rejection with Total Turbine Bypass Failure

15.3.5.1 *Summary of Technical Information*

The staff used the acceptance criteria in Section 15.1.1.2 of this report in evaluating this IE.

EC and ICL: Significant reduction in generator load initiates a signal for fast closure of the TCVs to avoid turbine overspeed. At the same time, the SB&PC system signals the TBVs to

open (in the fast mode). This analysis examines generator load rejection with total failure of the turbine bypass system. DCD Tier 2, Section 15A.3.4 states that the frequency of this event is estimated to be less than $1.0x10^{-2}$ pry which classifies it as an IE.

Upon receiving a load rejection signal, the TCVs will close and upon detection of insufficient turbine bypass, the RPS will initiate reactor scram. The calculated results indicate a sharp and narrow peak in total power, a dip in feedwater flow (5 seconds) due to increased pressure, and a pressure peak that decays slowly after 5 seconds. The neutron flux exceeds the neutron high-point scram signal, and the RPS initiates a reactor scram at about 0.15 seconds, with rod insertion initiating at 0.35 seconds. Peak dome pressure remains lower than the SRV activation pressure. The calculation ends at about 50 seconds, at which time the feedwater flow is at about 70 percent of rated flow. The MCPR value remains above the SLMCPR (1.18) but at 1.07 for ICL, which is below the SLMCPR. Because of reactor void collapse, the vessel level drops below the L2 level longer than 10 seconds which activates CRD high-pressure injection to recover water level. No operator action is required to mitigate this event. The reactor is shut down. The ICL case of a generator load rejection without bypass is a limiting event.

<u>FWTOD</u>: NEDO-33338, Revision 1, includes the results of generator load rejection with total turbine bypass failure for EC at SP2 (85-percent reactor power). This IE also has a high narrow power peak; the RPS issues a scram signal at 0.20 seconds, and control rod insertion initiates at 0.45 seconds.

The ensuing pressure pulse collapses the core void, and the vessel level falls below the L2 for more than 10 seconds, which activates the high-pressure CRD pumps to recover water level. Loss of reactor water level contributes the initial decrease in feedwater flow due to an increase in vessel pressure. However, at about 10 seconds, feedwater flow is above 100 percent of rated flow. The MCPR remains above the OLMCPR (NEDO-33338, Rev. 1, Figure A.3-2g).

At the state point SP1, for EC, the calculated results show similar events, except that in SP1, the resulting power peak is about 335 percent of the rated power (100 percent), while at SP2 the peak power is about 245 percent above the 85-percent rated full power at SP2. The calculated MCPR for SP1 is 1.14 (NEDO-33338, Rev. 1, Figure A.3-3g). For EC, this is a limiting IE event. Because the SP1 MCPR value is lower than the SLMCPR a radiological evaluation has been performed.

15.3.5.2 *Technical Evaluation*

<u>EC</u>: On sensing loss (or partial loss) of electrical load, the system commands the TCVs to close in the rapid mode, causing a sudden reduction in steam flow, void collapse, and pressure and power spikes. The calculated results (DCD Tier 2, Revision 6, Figure 15.3-5a) show a very narrow high-power peak.

As in the IE described in Section 15.3.4 of this report, there is a very fast energy deposition for this event. The staff requested the applicant (RAI 15.3-11), to calculate the energy deposition along the pellet-clad mechanical interaction and cladding strain.

GEH provided a summary evaluation demonstrating that the GE14 fuel is capable of absorbing the energy deposition from any transient analyzed in this submittal. The staff reviewed the response to RAI 15.3-11 and finds the response acceptable; therefore, RAI 15.3-11 is resolved. (See also Section 15.2.2.2.2 of this evaluation.)

The feedwater flow dips to about 60 percent of normal at about 3 seconds, and the simulated thermal power peaks well above the high-flux neutron scram setpoint at a fraction of a second after TCV closure, which initiates a scram. Dome pressure peaks also at the minimum of the feedwater flow but remains below the SRV setpoint. The increased pressure, decrease in feedwater flow, and void collapse reduce the WR water level below the RPV L2 level for about 20 seconds. This is about the minimum time required to initiate IC operation. In this case, the analysis shows that the ICs do not start, but the CRD high-pressure injection initiates to recover the water level in the vessel. The DCD calculation shows that the MCPR remains above the OLMCPR, which indicates that there is no fuel damage. With the HPCRD injection, the core recovers water level and remains stable and shut down. The results of this analysis indicate that the IE satisfies the acceptance criteria.

ICL: The event is similar to the EC case; however, the MCPR value is lower than the SLMCPR and hence a radiological analysis will be performed.

FWTOD: The phenomenology is the same as in the EC event. However, because of the lower power at SP2, the MCPR is higher than for the EC or ICL events. The results of this analysis indicate that the event satisfies the acceptance criteria.

The SP0 event analysis for the initial core shows that it is the most limiting event. In addition, the results indicate that the pressure increase following the power peak is much lower than the SRV opening setpoint. The reactor vessel water level is above TAF; therefore, the results meet the acceptance criteria.

15.3.5.3 *Conclusion*

Generator load rejection with total turbine bypass failure has a very low probability of occurrence. However, assuming that the SB&PC and the RPS respond as designed, the reactor will scram promptly.

In this event, it is assumed that all rods entering transition boiling fail. The number of rods in boiling transition is confirmed to be less than 1,000, based on the ΔCPR/ICPR results. The results of this event satisfy the acceptance criteria and hence are acceptable.

15.3.6 Turbine Trip with Total Turbine Bypass Failure

15.3.6.1 *Summary of Technical Information*

The staff used the acceptance criteria in Section 15.1.1.2 of this report in evaluating this IE.

EC and ICL: A variety of causes, such as loss of turbine fluid pressure, large vibrations, low condenser vacuum, and vessel high water level, can result in a turbine trip. Turbine trip is followed by fast opening of the bypass valves. Failure of all bypass valves to open would require multiple failures. DCD Tier 2, Section 15A.3.3, estimates the frequency of this event to be less than 1.0×10^{-2} pry, which classifies it as an IE.

The sequence of events (after the scram signal) is nearly identical to that described in Section 15.3.5 of this report for generator load rejection.

In DCD Tier 2, Table 15.3-7 and Figure 15.3-6 show the calculated sequence of events as a function of time. The results indicate a sharp and narrow peak in total power due to void

collapse from the pressure pulse, a dip in feedwater flow (5 seconds) resulting from increased pressure, and a wide pressure peak that decays slowly after 5 seconds. Vessel water level decreases to lower than L2 at 12 seconds. At 0.10 seconds, the TSVs are closed. At 0.15 seconds, the RPS initiates reactor scram, and at .35 seconds, control rod insertion begins. At about 1.2 seconds, the MCPR is at 1.21. Unlike its behavior in the generator load rejection event, the simulated thermal power registers only a small rise above normal at about 1 second into the event, followed by decay to less than 50 percent at 10 seconds. In the long term, CRD injection initiates to recover RPV level. Finally, the dome pressure peaks at about 4 seconds into the event but remains well below the SRV setpoint and well below 110 percent of the design value.

FWTOD: This IE has not been explicitly analyzed because it is bounded by generator load rejection with loss of turbine bypass.

15.3.6.2 *Technical Evaluation*

EC, ICL, and FWTOD: This event is almost identical to the generator load rejection with turbine bypass failure discussed in Section 15.3.5 of this report. The results of the analyses and the conclusions are the same. There is 0.06 CPR difference between the ICL and EC for SP0. The analysis presented in DCD Tier 2, Revision 9, Section 15.3.6, shows that the reactor has stabilized and long-term cooling has been established.

As in the previous two events, a pulse-like power event occurs. Therefore, the staff requested, in RAI 15.3-11, that the applicant calculate the energy deposition along with the associated pellet-cladding mechanical interaction. In their response, GEH demonstrated that the GE14 fuel has significant margin to cladding strain and fuel melt criteria as listed in the DCD Tier 2, Revision 4, Table 15.0-3. The staff reviewed the RAI 15.3-11 response, finds it responsive to the request, and considers this issue resolved. The same issue has also been described in Section 15.2.2.2.2 of this report.

No operator action is required to mitigate the event. The reactor is shut down and stable.

15.3.6.3 *Conclusion*

In this event, it is assumed that all rods entering transition boiling will fail. The number of rods in boiling transition is confirmed to be less than 1,000, based on the ΔCPR/ICPR results.

Turbine trip with total turbine bypass failure has a very low probability of occurrence. However, assuming that the SB&PC and the RPS respond as designed, the reactor will scram promptly without fuel damage or overpressurization. Therefore, the calculated results satisfy the acceptance criteria.

15.3.7 Control Rod Withdrawal Error (RWE) during Refueling

15.3.7.1 *Summary of Technical Information*

The staff used the acceptance criteria in Section 15.1.1.2 of this report in evaluating this IE.

EC, ICL, and FWTOD: The DCD states that there is no postulated set of circumstances that results in an inadvertent control RWE while in the refueling mode. The applicant based this conclusion on system interlocks ensuring against inadvertent criticality. In addition, removal of

the highest worth rod or two rods (one of which is the highest worth) in the same hydraulic control unit will not make the reactor critical; this is an ESBWR design feature.

To minimize the possibility of inserting fuel into any cell without control rods inserted, the design requires that all control rods be fully inserted before fuel is loaded into the core. The design achieves this protection through the use of interlocks. For example when the mode switch is in the “refuel” position, the interlock prevents the reload platform from moving over the core if a control rod is withdrawn and fuel is in the hoist, and it also prevents rod withdrawal.

Control rod withdrawal in the refueling mode can occur if the refueling platform is not over the core and the hoist is not loaded with fuel. The possible selections (in the RC&IS) are “single” and “gang.” In this case, the interlock prevents a second rod from moving in the “single” setting or a pair of rods in the “gang” mode. Also, the physical design of the fuel that needs to be removed (four assemblies) before the control rod is removed prevents upward control rod removal from a cell.

The estimated frequency of this event is less than 1.0×10^{-2} pry, which classifies it as an IE, as indicated in DCD Tier 2, Revision 9, Section 15A.3.11.3.

15.3.7.2 *Technical Evaluation*

Analysis of this IE is independent of fuel loading; therefore, the following discussion applies to EC, ICL, and FWTOD.

The design precludes inadvertent criticality because multiple failures must take place to cause criticality under refueling conditions. However, the applicant did not explain the statement that multiple failures are needed to cause criticality. For example, the ESBWR control rod system consists of mechanical, electrical, and pneumatic/hydraulic components and is therefore subject to mechanical, electrical, pneumatic/hydraulic, and operator error failures. DCD Tier 2, Section 15A.3.11, does not indicate how operator error or a combination of equipment failure and operator error could result in control rod withdrawal causing criticality during refueling.

The description of the interlocks effective during refueling in current BWR-4 plants is similar to the description of those in the ESBWR.

A recent announcement disclosed that in 1999 a BWR-4 plant in Japan experienced inadvertent withdrawal of three control rods during refueling, which caused criticality. According to the description, this event resulted from a combination of mechanical failures and operator error.

In DCD Tier 2, Section 15.3.7, the applicant stated that there is no postulated set of circumstances that results in an inadvertent RWE during refueling because of interlocks and design improvements. In RAI 15.3-19, the staff requested the applicant to provide the basis, using applicable information, for reaching this conclusion and the analyses demonstrating the magnitude of the consequences of an RWE under refueling conditions. GEH’s response details the results for an RWE under refueling conditions and states that the probability is extremely small, as it would require multiple component failures and/or operator errors. In addition, GEH provided quantitative evaluation of the results of such an event (based on the 1999 event in Japan) to demonstrate that the potential consequences are insignificant.

The GEH response to RAI 15.3-19 S01 (MFN 08-564) presented results of refueling criticality transients and compared to measurements to the Japanese incident (Shika Electric 1999). The

calculated values of radiation dose at the top of the vessel water are 1.3×10^{-8} millisieverts (mSv) ($1.3x10^{-9}$ rem) and the measurements at the charcoal filters designed to detect this radiation is below detection limits. Likewise pocket dosimeters designed to detect worker exposure to gamma radiation were also below the detection level. The calculated peak power achieved in the Shika reactor was about 15 percent of rated power. The calculated maximum energy deposition was 41 to 49 calories per gram (cal/g) (73.8 to 88.2 British thermal units per pound mass [BTU/lbm]) of UO_2 . These values are much lower than the fuel limit of 150 cal/g (270 BTU/lbm) of UO_2. The consequences of a RWE event for ESBWR are expected to be less severe than the BWR-4. Neither radiation exposure nor material limits were approached; therefore, the staff agrees with GEH's evaluation and finds the results acceptable. Based on the applicant's response, RAI 15.3-19 is resolved.

15.3.7.3 *Conclusion*

The RWE during refueling is an extremely unlikely occurrence for the ESBWR because of interlocks, procedures, and reloading practices. The calculated probability for an RWE during refueling is extremely small, but also the estimated consequences are minimal to insignificant, as evidenced by the criticality event that occurred in a Japanese reactor in 1999. Based on the ESBWR design, with the higher amount of water above the core (the high water level shields the refueling crew), the expected results of a RWE during refueling would be even less significant. Regarding potential fuel damage, the estimated amount of thermal energy released during such an event challenges neither fuel integrity nor cladding strain. The IE would evolve slowly, thus offering the opportunity to the operator to respond by inserting control rods to absorb the excess reactivity. The IE analyses indicate that the results meet the acceptance criteria and therefore are acceptable.

15.3.8 Control Rod Withdrawal Error During Startup with Failure of Control Rod Block

15.3.8.1 *Summary of Technical Information*

Analysis of this IE is independent of fuel loading; therefore, the following discussion applies to EC, ICL, and FWTOD.

The staff used the acceptance criteria in Section 15.1.1.2 of this report in evaluating this IE.

In this analysis, the applicant assumed that during startup, a control rod assembly or a single control rod is either inadvertently withdrawn or the automated control rod system malfunctions. The RC&IS prevents the withdrawal of any out-of-sequence rod. Also, if a rod assembly withdrawal sequence is violated, the RC&IS will initiate a rod block. In addition, the SRNM has a period-based reactor trip function that will either initiate a rod block if the period is less than 20 seconds or a reactor scram if the period is shorter than 10 seconds.

Two sets of calculations were performed: (1) rods are continually pulled, and the short-period (20 seconds) rod block fails and the short period trip (10 seconds) is credited and (2) the short-period trip and the rod block logic of both channels of the RWM fail. In either case, the short-period scram (10 seconds) will terminate the event. Should the SRNM based short-period scram fail, the average power range monitor (APRM) high-flux will scram the reactor and terminate the event.

The assumptions for the analysis are that the reactor is critical at near zero power at 271 degrees C (520 degrees F), the rod worth of the withdrawn rods is 3-percent Δk, and the control

rod worth speed is 28 mm (1.1 in.) per second (i.e., the nominal FMCRD withdrawal speed and, for rod-gang withdrawal, the reactor period monitored by any SRNM are the same).

The calculated enthalpy change for the 10-second period scram initiated from 1.1 to 4.6 seconds into the event, depending on core exposure and with a conservative addition of 2.23 seconds, is 66.2 joules per gram (J/g) (28.46 British thermal units [BTU] per pound) compared to the SRP fuel cladding failure criteria (i.e., 712 J/g [306.1 BTU per pound]). (See Figure 15.3-7a in the SRP, Appendix B, Section 4.2.) If the source range neutron monitoring (SRNM) fails and the average power range monitor (APRM) scram is credited, the results are also within the acceptance criteria. In the second case (APRM at 15-percent power at a high-flux scram initiated between 7.8 to 9.2 seconds by the APRM), the results are 523 J/g (224.8 BTU per pound), which are again within the acceptance value of 712 J/g (306.1 BTU per pound). The analysis was performed using the staff-approved PANAC11 code evaluation.

The estimated frequency of this event is less than 1.0×10^{-2} pry, which classifies it as an IE, as indicated in DCD Tier 2, Revision 9, Section 15A.3.12.3.

15.3.8.2 *Technical Evaluation*

The DCD states that multiple failures (or an inadvertent operator action) are necessary to cause an uncontrolled rod (or rod assembly) withdrawal. The analyses included calculations to assess the impact from about zero power (i.e., less than or equal to the low power setpoint). The energy deposition model assumes adiabatic heating, which is a conservative assumption, and nominal rod (or rod gang) withdrawal rates for the FMCRD mechanism. After event initiation, the reactor promptly scrams on the period trip function. The average at-peak axial location enthalpy increase is well within the acceptance limits. The corresponding pressure and MCPR changes remain negligible. Should the period trip failed, the reactor would scram because of the APRM 15-percent (or higher) power scram setting. The adiabatic heatup assumption adds a degree of conservatism. Therefore, the acceptance criteria are satisfied. No operator action is required to mitigate this IE.

15.3.8.3 *Conclusion*

Because the conservatively calculated fuel enthalpy is within the acceptance criteria of 712 J/g (306.1 BTU per pound), the transient analyses results are acceptable. In the control rod (or rod assembly) withdrawal error at startup, the reactor should scram on the period meter at a very early stage and generate a small amount of energy deposition. In either case, the acceptance criteria are satisfied, and the analyses results are acceptable.

15.3.9 Control Rod Withdrawal Error during Power Operation with Automated Thermal Limit Monitor (ATLM) Failure

15.3.9.1 *Summary of Technical Information*

The staff used the acceptance criteria summarized in Section 15.1.1.2 of this report in evaluating this IE.

<u>EC, ICL, and FWTOD</u>: Analysis of this IE is independent of fuel loading; therefore, the following discussion applies to EC, ICL, and FWTOD.

The ESBWR is equipped with an ATLM, which is a subsystem of the RC&IS. The ATLM has two channels and monitors the MCPR and the MLHGR. Should the reactor reach either of these limits due to control rod withdrawal, the system will remove the rod withdrawal permissive. Potential causes of rod (or rod assembly) withdrawal include procedural operator error and/or malfunction of the automated rod withdrawal sequence control logic. DCD Tier 2, Section 15A.3.13, estimates the probability of an RWE during power operation to be less than 1.0×10^{-2} pry. The calculation distinguishes between an error during automatic rod movement and manual rod movement. In the first case, the calculation takes credit for the ATLM. The ATLM is based on actual core thermal limit information. In the case of manual control rod withdrawal the ATLM will remove the rod motion permissive when the core reaches any thermal limits.

The DCD states that in either case, the ATLM system will halt the progression of the event before any limits are violated. In the case of operator error or malfunction in the automated rod withdrawal sequence logic, the dual-channel multichannel rod block monitor (MRBM) will stop further rod withdrawal to protect the fuel. The DCD estimates that the potential damage will be limited to fuel failure of fewer than 1,000 rods and no fuel melt; therefore, the offsite dose will be within the acceptance criteria. No operator action is required to mitigate this IE.

15.3.9.2 *Technical Evaluation*

EC, ICL, and FWTOD: The DCD categorizes this event as highly unlikely. The ESBWR is equipped with the ATLM system, which would be able to arrest rod withdrawal based on actual core data, such as inlet and coolant temperatures, core power, core power distribution, and other parameters. The ATLM is a dual-channel subsystem, not subject to single failure. However, the DCD provides no reference regarding its classification as a safety-grade system, nor does it refer to TSs on this system. The guidance in SRP Section 15.0 and 10 CFR 50.36, requires that SSCs related to protection of SAFDLs should be safety grade.

In RAI 15.0-15, the staff requested the applicant to describe the bases for the reclassification of the RWE, including the initiating actions/events and mitigating strategies from all modes of operation to address the potential for a gang withdrawal error and to identify the proposed acceptance criteria for the new event classification. In RAI 15.3-33, the staff requested that the applicant provide additional information regarding this event and in particular, an evaluation of barrier performance. The ATLM is included in the TS. GEH responded to RAI 15.0-15 and stated that the ESBWR design is such that the probability of RWE during power operation is very low, and hence it is categorized as IE. The basic design feature at issue is the provision of two systems (ATLM and MRBM) to prevent RWE during power operation. Section 15A of this report presents the staff’s evaluation of the event frequency. Based on the applicant’s response, RAI 15.0-15 is resolved.

GEH responded to RAI 15.3-33 by revising the DCD to include two sets of RWE during power operations: one for the AOO and the other for an IE that includes ATLM failure. In the first case it is assumed that the ATLM responds as designed, which resulted in no thermal limits being violated. Should the ATLM fail the multiple rod block monitoring subsystem will be activated to control rod bocks in the RC&IS to prevent core thermal limit violation. For the case of ATLM failure GEH performed radiological assessments and concluded that the 1,000 rods failure analyses results are within the acceptance criterion of 0.025 Sv (2.5 rem). Based on the applicant’s response, RAI 15.3-33 is resolved.

15.3.9.3 *Conclusion*

The control RWE during power operation with ATLM failure has been reclassified in the DCD as an IE. The reclassification is based on the design differences between the ESBWR and conventional BWRs that result in the SRP classification of the event as an AOO for conventional BWRs. The reclassification is based on the redundancy of systems in the ESBWR that prevent rod withdrawal that could result in violation of thermal limits. Control rod withdrawal could occur because of instrumentation failure or operator error. In the first case, the redundant instruments (with self-diagnosis) have an extremely small probability of failure. In the second case, another system (the MRBM) will intervene to stop control rod withdrawal and prevent (or limit) fuel damage. It is worth noting that the instrumentation will act in anticipation of the reactor exceeding thermal safety limits. Because the staff has accepted the instrumentation and its capabilities, this review finds the reclassification and the fuel damage estimate acceptable.

15.3.10 Fuel Assembly Loading Error, Mislocated Bundle

15.3.10.1 *Summary of Technical Information*

The staff used the acceptance criteria in Section 15.1.1.2 of this report in evaluating this IE.

<u>EC, ICL, and FWTOD</u>: The evaluation of mislocated assemblies involves consideration of at least two fuel assemblies being in interchanged positions. If one is assumed to operate at a lower power level, the other will operate at a higher power. The plant is instrumented so that the core monitor can recognize the mislocated assemblies, allowing the operator to intervene and minimize the consequences of the mislocated fuel. That would be the case if the higher power assembly is next to an automatic fixed in-core probe or a local power range monitor. However, should another fuel assembly be located between the instrument and the higher power assembly, the core monitor will probably not recognize the mislocation. In this case, the possibility exists that the assembly may operate above the thermal limits. Should a mislocated assembly suffer thermal and/or mechanical damage resulting in leaking fuel rods; the application of existing leak detection techniques and local power suppression methods will minimize the radioactive leakage.

The maximum power at which the mislocated assembly will operate is limited by the detection capability of the core monitoring system. The DCD presented a conservative case, bounding for any mislocated assembly. First, it assumed that all fuel rods in the affected assembly will be damaged and become leakers. Then it assumes that all four assemblies surrounding the affected assembly experience damage to all their rods.

In addition, it added a factor of 1.4 to account for fission product inventory differences over an operational cycle, and it added a factor of 2.5 to account for the variation of cycle-dependent bundle power as a ratio of the end-of-cycle average bundle power. This amounts to a factor of 3.5 bounding the end-of-cycle fission product inventory.

The estimated frequency of this event is less than 1.0×10^{-2} pry, which classifies it as an IE, as indicated in DCD Tier 2, Revision 9, Section 15A.3.14.3.

15.3.10.2 *Technical Evaluation*

EC, ICL, and FWTOD: The staff approved Amendment 28 to GESTAR II which allowed the event category change from AOO to IE, which has an acceptance criteria of 10 percent of the radiation dose limit specified in 10 CFR 52.47(a)(2)(iv)(A).

As stated above, the primary safeguards against fuel loading errors are design features and loading procedures to minimize the probability of a misloading event. The applicant has implemented these safeguards in the fuel and plant design. In addition, GDC 13 requires the provision of instrumentation to monitor local operating power versus anticipated power levels. Both design features and loading procedures have been implemented.

In addition, the applicant noted that a mislocated fuel bundle in the immediate vicinity of an automatic fixed in-core probe or a local power range monitor will be readily detected (after startup) and power could be suppressed to minimize fuel leakage. A once-removed mislocated assembly from the detector may not be identified, but the power mismatch is limited. On such occasions, it is possible that the mislocated bundle will operate outside its thermal and/or mechanical limits and damage the cladding. Monitoring will detect fuel leakage (from any cause), and the leakage can be minimized by suppressing power to the segment with the leakers.

The staff established that the GEH estimated potential site boundary dose rate, is conservative as described in GESTAR II, Amendment 28, Revision 1, "Misloaded Fuel Bundle Event Licensing Basis Change to Comply with SRP Section 15.4.7," dated August 23, 2004. DCD Section 15.3.11.3 describes core verification requirements and confirmation of assumptions, as summarized in the following:

> The NRC requires licensees to certify that core verification procedures have the following characteristics:
>
> During fuel movement, each move (location, orientation, and seating) is observed and checked at the time of completion by the operator and spotter.
>
> After completion of the core load, the core is verified by a video recording of the core using an underwater camera.
>
> Two independent reviewers perform the verification of the bundle serial number location, orientation, and seating. Each independently records the bundle serial numbers on a core map, which is verified with the core design loading pattern. The licensees are expected to follow the above procedures during refueling.

15.3.10.3 *Conclusion*

In this section, the applicant analyzed the fuel misloading error event. The analysis assumed that one of two interchanged assemblies is operating in a location of higher power and is one location removed from a detection device and, therefore, is subject to potential thermal mechanical damage. The DCD makes a bounding calculation and estimates the site boundary exposure for plants with main steamline high reactivity trip and for plants without this feature. In the first case, the DCD demonstrates that the exposure criteria are satisfied. In the second case (which depends on the site dispersion factor), the DCD back-calculated the minimum dispersion factor necessary to meet the 10 CFR 52.47(a)(2)(iv)(A) and (B) criteria.

In addition, the review established that the design has the required instrumentation and controls to monitor and control local power as required by GDC 13. The staff finds that the proposed design satisfies the GDC required control instrumentation and conservative estimate of the site boundary exposure to meet the criteria of 10 CFR 52.47(a)(2)(iv)(A) and (B). Therefore, the fuel misloading analysis is acceptable.

15.3.10.4 *Post-COL Activity*

In accordance with TS 5.6.3, Item C and consistent with the staff's SER related to the Global Nuclear Fuel's (GNF) request for proposed Amendment 28 to GESTAR II, March 8, 2007, "Final Safety Evaluation for GNF Topical Report, Amendment 28, the following conditions are to be met by individual licensees:

1. Plants seeking to apply the infrequent incident must confirm that their core refueling verification procedures meet the requirements defined in Section 5.3, Fuel Loading Error Analysis requirements, of the GESTAR US Supplement. This confirmation will be documented every refueling outage through the reload design documentation and the analysis basis stated in the Supplemental Reload Licensing Report (SRLR).

2. Should a bundle mislocation and seating occur and go undetected, the plant-specific acceptance of this categorization for the plant will be revoked, and the classification of this event will revert from "infrequent incident" to an "anticipated operational occurrence classification" immediately.

15.3.11 Fuel Assembly Loading Error, Misoriented Assembly

15.3.11.1 *Summary of Technical Information*

The staff used the requirements summarized in Section 15.1.1.2 of this report in evaluating this IE.

<u>EC, ICL, and FWTOD</u>: The probability that a misoriented assembly is placed in a core position and not detected is very small. Proper fuel orientation has five different visual indications: (1) the fastener springs and spacers to maintain channel clearance are located in the corner toward the center of the control rod, (2) identification on the assembly handle points toward the adjacent control rod, (3) the channel spacing buttons are adjacent to the control rod passage area, (4) the assembly identification numbers located on the fuel assembly handles are readable from the center of the assembly, and (5) there is cell-to-cell replication. Based on the above, the staff considers the probability of a misoriented assembly not being detected to be very small.

The letter from the staff, to Lingenfelter, GEH, March 8, 2007, "Final Safety Evaluation for GNF Topical Report, Amendment 28," is related to the GNF request for proposed Amendment 28 to GESTAR II. It suggests that the analysis is the same as that for the mislocated assembly, described in Section 15.3.10.2 of this report.

15.3.11.2 *Technical Evaluation*

The staff's review of DCD Tier 2, Section 15.3.10.2, indicates that the analysis is a conservative bounding calculation for assembly mislocation and fuel burnup; therefore, it is acceptable. For the same reasons, the analysis of the misoriented assembly is acceptable.

The estimated frequency of this event is less than 1.0×10^{-2} pry, which classifies it as an IE, as indicated in DCD Tier 2, Revision 9, Section 15A.3.15.3.

15.3.11.3 *Conclusion*

In this section, the applicant analyzed the fuel assembly misorientation. The DCD makes a bounding calculation and estimates the site boundary exposure for plants with main steamline high-radiation trip and for plants without this feature.

In the first case, the applicant demonstrated that the exposure criteria are satisfied. In the second case, which depends on the site dispersion factor, the applicant back-calculated the minimum dispersion factor necessary to meet the criteria.

In addition, the review established that the design has the required instrumentation and controls to monitor and control local power as required by GDC 13. The staff therefore, finds that the fuel misloading analysis is acceptable.

15.3.11.4 *Post-COL Activity*

In accordance with TS 5.6.3, Item C and consistent with the staff's SER related to the Global Nuclear Fuel's (GNF) request for proposed Amendment 28 to GESTAR II, March 8, 2007, "Final Safety Evaluation for GNF Topical Report, Amendment 28, the following conditions are to be met by individual licensees:

(6) Plants seeking to apply the infrequent incident must confirm that their core refueling verification procedures meet the requirements defined in Section 5.3, Fuel Loading Error Analysis requirements, of the GESTAR US Supplement. This confirmation will be documented every refueling outage through the reload design documentation and the analysis basis stated in the Supplemental Reload Licensing Report (SRLR).

(7) Should a bundle be misoriented and go undetected, the plant-specific acceptance of this categorization for the plant will be revoked, and the classification of this event will revert from "infrequent incident" to an "anticipated operational occurrence classification" immediately.

15.3.12 Inadvertent Shutdown Cooling Function Operation

15.3.12.1 *Summary of Technical Information*

The staff used the acceptance criteria in Section 15.1.1.2 of this SER in evaluating this IE.

<u>EC, ICL, and FWTOD</u>: This event concerns the power increase resulting from misoperation of the RWCU/SDC system either during power operation or during startup. Malfunction of the SDC leads to lower temperature cooling water entering the core, which results in reactivity insertion and power increase. The DCD does not quantify the resulting temperature differences

or the reactivity insertion. However, the DCD states that, if there is no operator action, the system will assume a new power level, but with or without operator action, the system will not violate the thermal limits. During startup, RWCU/SDC malfunction will increase the reactivity insertion rate and may result in a scram. Either way, the system will not violate the thermal limits. Any potential event ensuing from this event is bounded by the LOFWH event analyzed previously.

15.3.12.2 *Technical Evaluation*

ICL, EC, and FWTOD: In RAI 15.3-34, the staff requested the applicant to quantify the range of expected temperature limits and the resulting reactivity and reactivity-rate to justify the statement that with or without operator action the plant thermal limits will not be violated. The GEH response provided a conservative quantification of the expected sequence of events. The response was in terms of results of TRACG analyses from start-up and power operation conditions. Conservative results were assured because of the conservative input assumptions in the analyses. During start-up a reactor scram on high flux or short period may occur or the reactor may reach a higher stable power level. Likewise during power operation no thermal limits are reached nor violated. Appropriate text modification to DCD Tier 2, Section 15.3.12.2 reflects the response to RAI 15.3-34. The results indicate that the peak power and the stable power to be achieved by the event are a very small portion of the rated power, thus the design has an excess heat transfer capacity to cool the core in this instance. Therefore, this event does not represent a threat to core thermal limits, nor pressure vessel integrity, and it is acceptable. Based on the applicant's response, RAI 15.3-34 is resolved.

15.3.12.3 *Conclusion*

GEH submitted a conservative quantification of the event resulting from an inadvertent activation of the RWCU/SDU during startup or low-power operation. The results indicate a benign event that does not pose a threat to either the core or the pressure vessel and does not require operator intervention. The results satisfy the acceptance criteria and the event analysis is acceptable.

15.3.13 Inadvertent Opening of a Safety/Relief Valve

15.3.13.1 *Summary of Technical Information*

The staff used the acceptance criteria in Section 15.1.1.2 of this report in evaluating this IE.

EC, ICL, and FWTOD: Inadvertent SRV opening could result from a valve malfunction or an operator error. SRV discharge is directed to the suppression pool, which could overheat (if the operator does not close the SRV), triggering reactor scram on high suppression pool temperature. In this case, the analysis assumes no operator action. The calculated results show that, in about 30 seconds, the reactor will assume a new power level and in 412.5 seconds, the reactor will scram (DCD Tier 2, Revision 9, Figure 15.3-8a). The vessel pressure settles at a slightly lower operating pressure, and the MCPR remains well above the OLMCPR. The operator will monitor suppression pool temperature and water level and isolate makeup from external sources as necessary. DCD Tier 2, Revision 9, Section 15A.3.8, estimates the frequency of this event as less than 1.0×10^{-2} pry, which classifies it as an IE.

In estimating the frequency of inadvertent SRV opening the applicant refers to a factor attributed to the triplicate electronic control system (also in Sections 15.3.1, 15.3.3, of this report) i.e., the

applicant based the valve failure probability on the electronic portion of the control system and ignored the mechanical aspects of valve failure. In RAI 15.3-16 the staff requested GEH to justify their choice or revise the probability values in Section 15A.3. The GEH response also documents the mechanical aspects of TBVs and SRVs in accident analyses and calculated revised failure probability values. Based on the applicant's response that accounted for the electrical as well for the mechanical causes of valve failure, RAI 15.3-16 is resolved.

15.3.13.2 *Technical Evaluation*

ICL, EC, and FWTOD: The analytical results indicate that this is an inconsequential event, with or without operator intervention to close a discharging SRV. Neither fuel damage nor overpressurization occurs, and the MCPR remains well above the designated operating limit (DCD Tier 2, Revision 9, Figure 15.3.8). These results satisfy the acceptance criteria for fuel damage and overpressurization; therefore, the event analysis is acceptable.

15.3.13.3 *Conclusion*

Assuming no operator action is taken to close the SRV the reactor will shut down on high suppression pool temperature. This event meets the pressure and fuel damage acceptance criteria. Therefore, the staff considers it acceptable and RAI 15.3-16 resolved.

15.3.14 Inadvertent Opening of a Depressurization Valve

15.3.14.1 *Summary of Technical Information*

The staff used the requirements summarized in Section 15.1.1.2 of this report in evaluating this IE.

A depressurization valve (DPV) could open as the result of a valve malfunction or operator error. The difference between the DPV and the SRV (Section 15.3.13 of this report discusses the SRV) is that the DPV is bigger and also discharges into the drywell, where it could raise the drywell pressure (within a few seconds) to the reactor scram setpoint. The opening of a DPV amounts to a depressurization event in that the SB&PC system will close the TCVs to stabilize the reactor vessel pressure at a slightly lower pressure and the reactor will resume operation at a slightly lower than normal power. However, the DPVs discharge into the drywell, and the reactor will scram on high drywell pressure. DCD Tier 2, Revision 9, Section 15A.3.9, estimates the frequency of this event as less than 1.0×10^{-2} pry, which classifies it as an IE.

15.3.14.2 *Technical Evaluation*

This is an inconsequential event in the sense that the plant does not get close to fuel damage or overpressurization and does not need operator action to mitigate the event. The acceptance criteria are satisfied.

15.3.14.3 *Conclusion*

This event meets the pressure and fuel damage acceptance criteria. Therefore, the results of the event analyses are acceptable.

15.3.15 Stuck-Open Safety/Relief Valve (SRV)

15.3.15.1 *Summary of Technical Information*

The staff used the acceptance criteria in Section 15.1.1.2 of this report in evaluating this IE.

<u>ICL, EC, and FWTOD</u>: A stuck-open SRV is attributed to valve malfunction (either electronic or mechanical), regardless of whether the opening resulted from inadvertent operator action or a high-pressure signal. More specifically, in this event, the applicant assumed that the SRV remains open after issuance of a scram signal and reactor shutdown. The SRV discharges into the suppression pool. The calculated sequence of events (after scram) indicates that depressurization begins at 10 seconds; vessel water level reaches RPV level L2 at 19.3 seconds, activating HPCRD injection; and low steamline pressure activates MSIV closure which, in turn, activates the ICs. At 154 seconds, the ICs are in full operation. As expected, vessel pressure keeps falling at a steady rate, and the MCPR value continues to increase well above the normal operating value. The estimated frequency of a stuck-open relief valve is less than 1.0×10^{-2} pry, classifying this event as an IE. (See DCD Tier 2, Revision 9, Section 15A.3.10.1.) Operator actions consist of monitoring the suppression pool temperature and water level and isolating external sources to the containment as necessary.

15.3.15.2 *Technical Evaluation*

Mitigation of this event depends on successful removal of decay heat. With successful operation of the HPCRD and the ICs, the system ensures decay heat removal and water level recovery. In this case, neither the pressure nor the MCPR value comes close to limiting values; therefore, the acceptance criteria are satisfied, and the results of the analysis are acceptable.

In RAI 15.3-23, the staff requested the applicant to justify the event category and to address the mechanical performance history related to this event. In its response to this RAI, the applicant noted that the SRVs and safety valves addressed in DCD Tier 2, Section 15.3.15, event evaluations are designed in accordance with the ASME Code, Section III, Subsection NB (Class 1), as described in DCD Tier 2, Sections 3.9 and 5.2.2. Per the qualification testing described in DCD Tier 2, Sections 3.9 and 3.10, for these valves, the design deformation limit criteria for disk-to-seat geometry and disk movement clearances are applied to ensure that the valve performance requirements for pressure response during dynamic load conditions and reclosure leak tightness are met, up to and including exposure to Service Level D loads. The applicant also indicated that it would revise the DCD to describe in more detail its evaluation of the potential for inadvertent opening of a relief valve. As a result, Revision 5 to DCD Tier 2, Section 15A.3.8 describes the controlled operation of these valves using procedures and the design of the human-system interface in the control room to support the determination that operator error resulting in inadvertent SRV actuation is negligible. The staff considers the applicant’s clarification of the SRV and safety valve (SV) design as consistent with the ASME Code, and the evaluation of a potential inadvertent SRV opening described in the DCD to be acceptable.

In RAI 15.3-23 S01, the staff requested the following: clarification of the statement that allowable limits are beyond the elastic region yet do not exhibit deformation, the assigned zero value for operator error regarding SRVs and/or SVs, and the need to update the DCD regarding those responses. In their response GEH stated that deformation does not take place until exposure to service level D. In such cases the exposed valve does not re-enter service until after rigorous qualification testing takes place. Regarding operator error, GEH modified the

DCD Section 15A.3.8 to demonstrate that operator error is negligible. In addition to the response to RAI 15.3-23 S01 GEH added a DCD markup for Sections 15A.3.8.1, to 15A.3.8.3. The staff reviewed the revised Sections 15A.3.8.1/2/3 and finds they are responsive to the staff's request and GEH justified the frequency values used for the SRV and SV response. Therefore, the staff considers RAI 15.3-23 to be resolved.

15.3.15.3 *Conclusion*

As stated above the staff concludes that the design of the ESBWR with respect to potential malfunction of SRVs satisfies the NRC regulations and, therefore, is acceptable.

Summary of Staff Review Findings for Sections 15.3.1 to 15.3.15

There are five criteria for this review: reactor vessel water level, RCPB pressure, radiological consequences, containment and suppression pool pressure and temperature, and control room radiation exposure. Control room radiation exposure is discussed in this report; therefore, the four criteria are applicable in the review of Section 15.3.

(8) <u>Vessel Water Level</u>

There is no core uncovery during any of the IEs evaluated above, the reactor water level is always above the TAF and therefore, meets the acceptance criteria.

(9) <u>RCPB Pressure</u>

An increase in vessel dome pressure occurred in several events, but none reached the SRV setpoint. The ESBWR vessel is large and difficult to over pressurize in the context of the events considered in Section 15.3. Therefore, RCPB meets the acceptance criteria.

(10) <u>Radiological Consequences</u>

Radiological consequences of IEs are relatively less severe than the DBAs evaluated in Section 15.4 of this report and are bounded by the radiological consequences of the DBAs.

Evaluation of the actual radiological values is not part of this review. In this context, the limiting IE is the ICL generator load rejection with total turbine bypass failure. Several events that depend on fuel loading and are characterized as post-COL activity items could become as limiting (or more limiting) with different future fuel loadings.

(11) <u>Containment and Suppression Pool Pressure and Temperature</u>

Inadvertent opening of an SRV (or a stuck-open SRV) will discharge steam into the suppression pool, and if not corrected, the reactor will scram on high suppression pool temperature. Similarly, with inadvertent opening of a DPV that discharges into the drywell, the reactor will scram on high drywell pressure. Both types of scram are credited with terminating the events.

No required operator actions are identified to mitigate the events. However, if as a result of an event, the reactor is not at normal operating parameters, the operator is expected to intervene to bring the reactor within normal operating conditions.

15.3.16 Liquid-Containing Tank Failure

15.3.16.1 *Regulatory Criteria*

The staff used the requirements summarized in Section 15.1.1.2 of this report in evaluating this IE. In addition, the staff reviewed DCD Tier 2, Revision 5, Section 15.3.16, in accordance with the guidance and acceptance criteria described in SRP Section 11.2 and Branch Technical Position (BTP) 11-6, "Postulated Radioactive Releases Due to Liquid-Containing Tank Failures," issued March 2007. The requirements for this analysis were initially located in SRP Section 15.7.3 with the same title. The requirements have not changed as the approach, content, and format of BTP 11-6 are consistent with those of SRP Section 15.7.3. The following acceptance criteria are applicable:

- 10 CFR 20.1301, "Dose limits for individual members of the public," and 10 CFR 20.1302, "Compliance with dose limits for individual members of the public," as they relate to limits for liquid effluent concentrations in unrestricted areas; these criteria apply to releases resulting from the liquid waste management system (LWMS) during normal plant operations and AOOs.

- GDC 60, "Control of releases of radioactive materials to the environment," as it relates to the design of LWMS components and structures housing the LWMS to control releases of liquid radioactive effluents.

- GDC 61, "Fuel storage and handling and radioactivity control," as it relates to the ability of structures housing the LWMS to control releases of liquid radioactive wastes.

The relevant requirements of the regulations identified above are met by using the regulatory positions and guidance contained in the following:

- SRP Section 11.2

- SRP Section 11.2, BTP 11-6

- SRP Section 15.7.3, issued July 1981 (and with the SRP updated in March 2007) (The requirements of SRP Section 15.7.3 have been relocated to BTP 11-6.)

- RG 1.143, Revision 2, November 2001, "Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants," as it relates to the design of the LWMS and structures housing this system, as well as to the provisions used to control leakages.

15.3.16.2 *Summary of Technical Information*

DCD Tier 2, Revision 9, Section 11.2, describes the design of the LWMS and its functions in controlling, collecting, processing, storing, and disposing of liquid radioactive waste generated as a result of normal operation, including AOOs. The LWMS includes the equipment drain subsystem, the floor drain subsystem, the chemical drain subsystem, and the detergent drain subsystem. DCD Tier 2, Revision 9, Figure 11.2-1, provides an overview of the LWMS process diagram depicting all subsystems, while Figures 11.2-1a, 11.2-1b, 11.2-3, and 11.2-4 present specific design details for each subsystem. DCD Tier 2, Revision 9, Figure 11.2-2, provides an LWMS process stream information directory and simplified flow diagram. DCD Tier 2, Revision

9, Sections 9.3, 9.2, and 10.4 describe the equipment and floor drain drainage systems and origins and discharges of nonradioactive effluents. DCD Tier 2, Revision 9, Figures 1.2-21 to 1.2-25, present the general arrangements of the LWMS within the radwaste building.

The LWMS and its components are housed in the radwaste building and located in radiologically controlled access areas. DCD Tier 2, Revision 9, Figures 11.2-1a and b, show the tanks, processing equipment, pumps, valves, ion exchangers, filters, and other components. All LWMS tank overflows are routed to building sumps and drains, which are pumped to their respective drain tanks. Subsystem tanks and components are vented to the radwaste building ventilation system, as described in DCD Tier 2, Revision 9, Section 9.4. The cubicles where tanks are located are lined with steel liners to avoid releases of radioactive materials in the environment. Concrete walls are coated with sealants for additional protection and minimization of radioactive waste (e.g., in the form of contaminated concrete). The LWMS treatment system components are arranged in shielded enclosures and compartments to minimize exposure of plant personnel during operation, inspection, and maintenance. The COL holder will subject the LWMS to pre-operational tests, and there are provisions for periodic inspections of major components to ensure the integrity of the LWMS subsystems and components. The ITAAC are described in ESBWR DCD Tier 1, Revision 9, Section 2.10.1 and Tables 2.10.1-1 and 2.10.1-2.

Each subsystem of the LWMS incorporates one or more tanks to hold liquid wastes. The equipment drain subsystem includes three collection tanks, each with a capacity of about 140,000 liters (37,000 gallons), and two sample tanks, each with the same capacity. The floor drain subsystem consists of two collection tanks, each with a capacity of about 130,000 liters (34,000 gallons), and two sample tanks, each with the same capacity. The chemical drain subsystem consists of one collection tank with a capacity of about 4,000 liters (1,060 gallons). The detergent drain subsystem includes two collection tanks, each with a capacity of about 15,000 liters (4,000 gallons) and two sample tanks with the same capacities. The LWMS comprises several subsystems such that any of the systems can segregate liquid wastes from various sources and process them separately. The subsystems maintain the segregation of process streams to support the most appropriate treatment of wastes by the LWMS.

Cross-connections between subsystems provide additional flexibility in processing wastes by alternate methods and provide redundancy if one subsystem were to become inoperative. The LWMS normally operates on a batch basis. There are provisions for sampling at important process points. The detection and alarm of abnormal conditions and administrative controls provide protection against accidental discharge.

15.3.16.3 *Technical Evaluation*

The staff evaluated a potential release of radioactive liquid waste following the postulated failure of a tank and its components, located outside of containment, as part of its review of DCD Tier 2, Revision 5, Section 15.3.16, with information drawn from DCD Tier 2, Revision 5, Sections 11.2 and 12.2. Section 12.2 of DCD Tier 2, Revision 9, presents information on the expected inventory of radioactive materials in LWMS tanks. The staff reviewed the LWMS in accordance with the guidance of SRP Section 11.2 and BTP 11-6 (March 2007) or, equivalently, with SRP Section 15.7.3 (July 1981). Staff acceptance of the postulated impact of a failure of a LWMS tank containing radioactive materials is based on the design's meeting the requirements of GDC 60 and 61; the effluent concentration limits of Table 2 (Column 2) of Appendix B, "Annual Limits on Intake (ALIs) and Derived Air Concentrations (DACs) of Radionuclides for Occupational Exposure; Effluent Concentrations; Concentrations for Release to Sewerage," to 10 CFR Part 20, "Standards for Protection Against Radiation"; and RG 1.143, as it relates to the

design of structures housing LWMS components and provisions used to control leakage and minimize spills into the environment.

In reviewing prior DCD Tier 2 revisions, the staff could not confirm that the approach used in assessing the impact of tank failure was consistent with the guidance of the SRP and SRP acceptance criteria. The radiological source term postulated to be released in an unrestricted area is the radioactivity contained in one of several tanks that are part of the LWMS. The evaluation considers the impact of the release of radioactive materials on the nearest potable water supply located in an unrestricted area and whether the impact results in the presence of radioactivity in potable water above the concentration limits of Appendix B (Table 2, Column 2) to 10 CFR Part 20.

A review of DCD Tier 2, Revision 3, Section 15.3.16, by the staff indicated that the technical approach was not consistent with that described in SRP Section 11.2 (BTP 11-6). The analysis assumed that such tanks are located in compartments with sealed concrete walls designed to hold the expected amounts of liquid wastes in the event of a tank failure. DCD Tier 2, Revision 3, Section 15.3.16.1, states that because of these design features, it is unlikely that a major event would result in the release of liquid radioactive wastes into the environment. The approach takes credit for the presence of coated concrete surfaces that contain the volume of the tank in the compartment where the tank is located. The proposed approach is inconsistent with the SRP, which states that "Credit for liquid retention by unlined building foundations will not be given regardless of the building seismic category because of the potential for cracks." The SRP does not allow credit for retention by coatings or leakage barriers outside of the building foundations. Also, DCD Tier 2, Revision 3, states that this design feature applies only to tanks containing "high-level liquid radwaste." This implies that tanks containing low-level liquid radwaste would not be located in compartments that afford the same level of protection. As a result, the applicant's analysis considers only a single pathway involving the airborne volatilization of radioactivity via the heating, ventilation, and air conditioning (HVAC) system and releases into the environment via the plant stack. Finally, DCD Tier 2, Revision 3, Sections 11.2 and 11.4, emphasize the use of liquid waste processing systems located in treatment bays so as to facilitate truck access and loading and unloading. These design features are in contrast to those designed to minimize spills and leaks into the environment, and it is not clear if the placement of skid-mounted radwaste processing systems in treatment bays would provide the same level of protection as that provided for tanks located in cubicles.

In RAIs 15.3-4 and 15.3-5, and in a related supplemental RAI 2.4-29 S01, the staff requested the applicant to address these inconsistencies with the NRC's guidance and acceptance criteria in SRP Section 11.2 and BTP 11-6, as the SRP precludes the assumption of sealed concrete walls in containing releases of liquid radioactive waste. In addition, the staff asked the applicant to provide additional details on "special design features" to support the approach, to update the radiological assessment, and to discuss why the release of the postulated inventory of radioactive materials to surface or ground water is not limiting as compared to the current case where only the volatile airborne fraction of radioactivity (as radioiodines) is assumed to be released in the environment. The staff also asked the applicant to describe the method, basis, assumptions, and parameters used in the analysis; to update the text and tables in DCD Tier 2, Revision 3, and Section 15.3.16; and to update the text and tables as they apply to DCD Tier 2, Section 2.4.13, and Table 2.0-2 of DCD Tier 2, Revision 3.

In response, the applicant agreed that BTP 11-6 does not allow credit for sealing concrete walls to contain releases of liquid wastes from tanks, and it committed to the use of steel liners in cubicles where liquid radwaste tanks are located. The commitment also includes provisions,

where sumps are located in tank cubicles, to pump liquids from such sumps to the appropriate radwaste subsystem for processing. The applicant has updated Section 11.2.2.3 of DCD Tier 2, to indicate that rooms where tanks are located will be lined with steel to prevent accidental releases of radioactivity in the environment. Similar revisions were made in DCD Tier 2, Revision 5, Sections 15.3.16.1 and 12.2.1.4.

The staff finds that the inclusion of a steel liner in tank cubicles and the use of sumps to collect and pump liquids to the radwaste system are acceptable mitigating features, consistent with BTP 11-6 and RG 1.143, and in compliance with GDC 60 and 61 for the control of releases of radioactivity in the environment during normal operations and AOOs. The staff evaluated the corresponding revision of DCD Tier 2, Revision 5, and finds the changes acceptable as the inclusion of steel liners in cubicles housing radwaste system tanks would mitigate the release of liquid radwaste to the environment. This approach is also consistent with the guidance of SRP Section 11.2 and BTP 11-6 in including mitigating engineering design features. Based on the applicant's response, RAIs 15.3-4, 15.3-5, and 2.4-29 S01 are resolved.

Given that the proposed design precludes the likelihood of a release of radioactivity in ground or surface water, the staff evaluated the applicant's analysis that considers the release of the volatile fraction of radioactivity contained in water and comprised of radioiodines and the impact on members of the public in unrestricted areas, based on the assumptions given in DCD Tier 2, Revision 5, Tables 15.3-17 and 15.3-18. The amount of radioiodines assumed for this analysis consist of the cumulative radioactivity inventory contained in seven tanks, ranging in capacity from 4 to 140 m^3 (about 1,100 to 37,000 gallons). The analysis assumes that the entire inventory of radioiodines is released in the radwaste building and vented outdoors, with no credit taken for treatment. The analysis assumes an atmospheric dispersion factor of 2.0×10^{-3} seconds (s)/m^3 for a receptor located at the exclusion area boundary (EAB). The applicant's results, presented in DCD Tier 2, Revision 5, Table 15.3-19, indicates an inhalation dose TEDE of 0.72 mSv (0.072 rem), for an offsite receptor. The staff confirmed the result and concludes that the dose complies with the 10 CFR 20.1301 dose limit of 1 mSv (0.1 rem) for members of the public.

The staff finds that the inclusion of such design features to mitigate the consequences of the failure of a tank and its associated components is acceptable, consistent with BTP 11-6 and RG 1.143, and in compliance with GDC 60 and 61 for the control of releases of radioactivity into the environment. The basis for the staff's acceptance is the capability of these design provisions to prevent radioactivity from entering a potable water supply system and to prevent the plant from exceeding the limits of 10 CFR Part 20, Appendix B, Table 2 (Column 2), in the nearest source of potable water located in an unrestricted area. The applicant's alternate analysis of a postulated failure of a tank indicates that doses to members of the public from the release and inhalation of volatile radioiodines comply with the dose limits of 10 CFR 20.1301 and 20.1302. Therefore, the staff concludes that the design provisions incorporated by the applicant are acceptable in mitigating the effects of the failure of a tank and its associated components involving radioactive liquids.

Under the provisions of 10 CFR 52.47(b)(1), a DCD application is required to propose ITAAC for the LWMS. The ITAAC are described in ESBWR DCD Tier 1, Revision 9, Section 2.10.1 and Tables 2.10.1-1 and 2.10.1-2. In summary, the relevant ITAAC include the following:

- Confirming the description and functional arrangement of the LWMS

- Assessing the pressure and leakage integrity of the LWMS when subjected to hydrostatic testing pressures expected during operation

- Confirming the installation of steel liners in cubicles housing LWMS tanks and vessels for the purpose of ensuring that, in the event of a tank rupture, the effluent concentration limits of Table 2 (Column 2) of Appendix B to 10 CFR Part 20 will not be exceeded at offsite locations.

15.3.16.4 *Conclusion*

The staff finds that the analyses and impact of the postulated failure of a tank and its components, located outside of containment are consistent with NRC's regulatory requirements and guidance. The applicant has met the requirements of GDC 60 and 61 with respect to the control of releases of radioactive materials to the environment by providing design features to reduce the potential impact of the failure of a radioactive liquid-containing tank and its associated components. Such a release will not result in concentrations of radioactive materials exceeding the limits of 10 CFR Part 20, Appendix B, and Table 2 (Column 2), in the nearest source of potable water located in an unrestricted area.

The staff concludes that the applicant has evaluated the postulated failure of a tank and its associated components and that the design is acceptable, meets the requirements of GDC 60 and 61 for the control of releases of radioactive materials to the environment, and provides an adequate level of safety during normal reactor operation, including AOOs. Based on the above review, the staff determines that the ESBWR LWMS design meets the guidelines of SRP Section 11.2 and BTP 11-6 and, therefore, is acceptable.

15.4 <u>Analysis of Accidents</u>

15.4.1 Design-Basis Accidents

In DCD Tier 2, Revision 9, Section 15.4, the applicant performed radiological consequence assessments of the following five reactor DBAs using the hypothetical set of atmospheric dispersion factors (χ/Q values) provided in DCD Tier 1, Revision 9, Table 5.1-1, and DCD Tier 2, Table 2.0-1. Given that all other aspects of the design are fixed, these χ/Q values determine the required minimum distances to the EAB and the low-population zone (LPZ) for a given site to provide reasonable assurance that the radiological consequences of a DBA will be within the radiological dose limits specified in 10 CFR 52.47(a)(2) and 10 CFR 100.21. No specific reactor site is associated with the ESBWR design. The DBAs analyzed in DCD Tier 2 include the following:

- FHA (DCD Section 15.4.1)
- LOCA (DCD Section 15.4.4)
- MSLB outside containment (DCD Section 15.4.5)
- Failure of small lines carrying primary coolant outside containment (DCD Section 15.4.8)
- Failure of reactor water cleanup system line outside containment (DCD Section 15.4.9)

In addition, in DCD Tier 2, Revision 9, Section 15.4.7, the applicant performed a radiological consequence assessment for the feedwater line break outside containment. This event is neither listed as a DBA nor required to be analyzed for its radiological consequences in SRP Section 15.0.3 and RG 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors."

Both SRP Section 15.0.3 and RG 1.183 list the BWR CRDA as a DBA and require its radiological consequences to be analyzed. In DCD Tier 2, Section 15.4.6, the applicant stated that the radiological consequence of a CRDA need not be considered because such an accident is extremely unlikely with the improved design of the ESBWR, and furthermore, there is no credible basis for the control rod drop to occur.

The ESBWR design employs the FMCRD, which has several new features that are unique and not found in current BWR locking piston control rod drives. DCD Tier 2, Revision 9, Section 4.6.1 describes the FMCRD system, and the staff evaluated and accepted the system in Section 4.6 of this report. For the CRDA to occur in the ESBWR design, it is necessary for both Class 1E separation and detection devices to fail or have the simultaneous failure of the rod block interlock and of the latch mechanism in conjunction with the occurrence of a stuck rod on the same FMCRD.

In 1996, the NRC certified the advanced boiling-water reactor (ABWR) design in Appendix A, "Design Certification Rule for U.S. Advanced Boiling Water Reactor," to 10 CFR Part 52, then titled "Early Site Permits; Standard Design Certifications; and Combined Licenses for Nuclear Power Plants," with the same fine motion control rod drive (FMCRD) design as that provided in the ESBWR. The staff accepted the FMCRD design in the ABWR without the applicant analyzing its potential radiological consequences of a CRDA because such an accident was considered to be extremely unlikely. However, the staff did evaluate the radiological consequences for this event in the ABWR SER (NUREG–1503, "Final Safety Evaluation Report Related to the Certification of the Advanced Boiling Water Reactor Design, Main Report," issued July 1994), Section 15.4.1, and found that the radiological consequences of a postulated CRDA using the FMCRD design at the EAB, LPZ, and control room were well within the dose acceptance criteria.

In DCD Tier 2, Section 15.4.6.5 the applicant stated that conservative analyses were performed for the ESBWR using adiabatic heat retention in the fuel and maximum expected control blade worth. In addition, the ESBWR design proposed a higher hypothetical set of X/Q values than those certified for the ABWR design. The analyses included the initial and EC loadings. The applicant reported that during a postulated CRDA, the fuel enthalpy rise remains well below the lower bound clad failure limits in Appendix B of Revision 3 to SRP Section 4.2. Based on the acceptance of the ABWR FMCRD and the results of the conservative calculations reported by the applicant the staff accepted the FMCRD design in the ESBWR. Because the clad failure limits are not violated there is no need for radiological analyses. However, because CRDA analysis is required (See SRP Section 15.4.9 and SRP Section 4.2, Appendix B, Revision 3, to provide the interim acceptance criteria and guidance for the reactivity-initiated accident [RIA]), the applicant performed such an analysis that is reported in Section 15.4.6 of this report.

In DCD Tier 2, Section 15.4.10, the applicant stated that the radiological consequences of a spent fuel cask drop accident need not be considered because the fuel building design is such that a spent fuel cask drop height of 9.2 m (30 ft) cannot be exceeded. In RAI 15.4-5 the staff requested information on the fuel building design and configuration to preclude a postulated spent fuel cask drop. SRP Section 15.7.5 requires a design-basis radiological consequence analysis only if a cask drop can occur exceeding 9.2 m (30 ft). The applicant's response to RAI 15.4-5, provided fuel building figures showing spent fuel cask movements and lifting heights as security-related sensitive information in accordance with 10 CFR 2.390. The staff finds that the cask drop distance is within the 9.2-m (30.2-ft) height limit specified in SRP Section 15.7.5. Therefore, neither the staff nor the applicant analyzed the radiological consequences for a spent fuel cask drop accident. Based on the applicant's response, RAI 15.4-5 is resolved.

In DCD Tier 2, Section 15.4.7, the applicant provided its radiological consequence analysis. The staff considers the radiological consequence of this event to be bounded by that resulting from the MSLB accident outside containment for all light-water BWRs; therefore, this event is neither listed as a DBA nor required to be analyzed for radiological consequences by SRP Section 15.0.3 or RG 1.183. Nevertheless, the staff reviewed the applicant's analysis and finds that the calculational methods used for the radiological consequence assessment are reasonable and acceptable. Furthermore, the staff confirmed that the radiological consequences calculated by the applicant are indeed bounded by those resulting from the MSLB accident outside containment as analyzed by the applicant for the ESBWR.

In DCD Tier 2, Section 15.4.9, the applicant provided a radiological consequence analysis. Neither SRP Section 15.0.3 nor RG 1.183 lists this event as a DBA, nor is it required to be analyzed for its radiological consequences. During promulgation of Appendix A to 10 CFR Part 52, the Advisory Committee on Reactor Safeguards (ACRS) Subcommittee on ABWR specifically recommended that the applicant analyze this event as a DBA to determine the radiological consequences. Accordingly, the applicant analyzed this event for the radiological consequences for the ESBWR, and the staff reviewed this event as analyzed and documented by the applicant.

Therefore, the staff concludes that the five selected DBAs identified above and analyzed by the applicant are consistent with those identified in SRP Section 15.0.3 and RG 1.183, and therefore, finds the selection to be acceptable.

In DCD Tier 2, Revision 9, Section 15.4, the applicant concluded that the ESBWR design will provide reasonable assurance that the radiological consequences resulting from any of the above five DBAs will be within the offsite dose criteria (i.e., as specified in 10 CFR 52.47(a)(2)) of 0.25 Sv (25 rem) TEDE and the control room operator dose criterion, specified in GDC 19, "Control room," of Appendix A to 10 CFR Part 50, of 0.05 Sv (5 rem) TEDE. The applicant reached this conclusion by using reactor accident source terms provided in NUREG–1465, February 1995, "Accident Source Terms for Light-Water Nuclear Power Plants," and in RG 1.183, and a set of hypothetical χ/Q values (discussed in Section 2.3.4 of this report). No specific reactor site is associated with the ESBWR design.

The χ/Q values are the relative atmospheric concentrations of radiological releases at the receptor point in terms of the rate of radioactivity release. In lieu of site-specific meteorological data, the applicant provided a reference set of hypothetical χ/Q values for the ESBWR. DCD Tier 1, Revision 9, Table 5.1-1, and DCD Tier 2, Revision 9, Table 2.0-1, list the ESBWR hypothetical χ/Q values. The χ/Q values indicate the atmospheric dilution capability. Smaller χ/Q values are associated with greater dilution capability, resulting in lower radiological doses. The radiological consequence doses are directly proportional to the χ/Q values. The hypothetical χ/Q values in the DCD are back-calculated from the dose acceptance criteria to minimize the fission product removal credit assumed for the engineered safety feature (ESF) systems in the ESBWR design.

Therefore, any COL applicant that references the ESBWR design should show that its proposed site-specific χ/Q values fall within the reference set of hypothetical χ/Q values used by the applicant in DCD Tier 1 and Tier 2 in order to demonstrate that the COL application meets the offsite dose criteria specified in 10 CFR 52.47(a)(2) and the control room operator dose criterion specified in GDC 19 of Appendix A to 10 CFR Part 50. This is identified as COL Information Item 2.0-1-A.

15.4.2 Fuel-Handling Accident

15.4.2.1 *Regulatory Criteria*

The staff reviewed DCD Tier 2, Revision 9, Section 15.4.1, in accordance with the guidance provided in SRP Section 15.0.3 and RG 1.183. The staff evaluated the radiological consequences of an FHA against the dose acceptance criteria specified in SRP Section 15.0.3 and RG 1.183, of 0.063 Sv (6.3 rem) TEDE at the EAB for any 2-hour period following the onset of the postulated fission product release and 0.063 Sv (6.3 rem) TEDE at the outer boundary of the LPZ for the duration of exposure to the release cloud. The staff also used a criterion of 0.05 Sv (5 rem) TEDE for evaluating the radiological consequences in the control room from a postulated FHA for the ESBWR design, pursuant to GDC 19 of Appendix A to 10 CFR Part 50.

RG 1.183 provides guidance on radiological consequence analyses to licensees of operating power reactors that choose to implement an alternative source term (AST) pursuant to 10 CFR 50.67, which has the same regulatory dose criteria specified in 10 CFR 52.47(a)(2) (i.e., 0.25 Sv [25 rem] TEDE) and GDC 19 (0.05 Sv [5 rem)] TEDE). Although RG 1.183 was written to apply to currently operating power reactors, the staff finds that its guidance on radiological acceptance criteria, formulation of the source term and DBA radiological consequence analysis modeling also applies in the review of the ESBWR design.

15.4.2.2 *Summary of Technical Information*

In DCD Tier 2, Section 15.4.1, the applicant presented its analyses of the radiological consequences of a postulated FHA. An FHA is postulated to result from a failure of the fuel assembly lifting mechanism, leading to a raised fuel assembly being dropped onto the reactor core or into the spent fuel storage pool. Any fission products released as a result of a fuel assembly drop in the refueling pool will be released into the reactor building atmosphere and then to the environment. Fission products released as a result of a fuel assembly drop onto the reactor core are assumed to be released directly to the environment by means of the cask doors on the west side of the fuel building. The staff requested that the applicant provide the source term assumptions used in its FHA radiological consequence analysis (RAI 15.4-1). The applicant provided the information requested in DCD Tier 2, Revision 2.

In RAI 15.4-1 S01, the staff requested that the applicant provide the FHA radiological consequence analyses for both a fuel assembly drop onto the reactor core and into the spent fuel storage pool. In its response to RAI 15.4-1 S01, the applicant provided the requested analyses in DCD Tier 2, Revision 4. The results indicate that the fuel building release is bounding, due to the higher control room X/Q values. In its radiological analyses, the applicant assumed, in accordance with the guidance in RG 1.183, that fission products are directly released to the environment within a 2-hour period without credit for any fission product removal processes. Therefore, the staff finds this portion of applicant’s response to be acceptable.

In RAI 15.4-1 S02, the staff requested that the applicant provide administrative controls to mitigate radiological consequences of an FHA in accordance with the guideline provided in Section 5.3 of Appendix B to RG 1.183. This section states that if the containment (e.g., the ESBWR reactor building or fuel building) is open during fuel-handling operations, the TS allowing such operations should include administrative controls to close the open penetrations within 30 minutes. Such administrative controls will generally require that a dedicated individual be present, with necessary equipment available, to restore containment closure, should an FHA occur.

In response to RAI 15.4-1 S02, the applicant stated that the reactor building or fuel building doors that potentially open following the implementation of COL Information Item 2A.2-2-A, "Confirmation of the Reactor Building X/Q Values," will have X/Q values that are less than the X/Q values used in the ESBWR DCD, Revision 5, to meet the dose limit. COL Information Item 2A.2-2-A, in DCD Tier 2, Revision 9, Section 2A2.5 states the following:

> If the X/Q values (for a release from any door or personnel air lock on the east sides of the Reactor Building or Fuel Building have X/Q values that would result in doses greater than the bounding dose consequence reported for the FHA in the ESBWR DCD, Revision 5) are not bounded by the X/Q values in the ESBWR DCD, Revision 5, for a release in the Reactor Building, the affected doors or personnel air locks must be administratively controlled prior to and during movement of irradiated fuel bundles.

Based on the applicant's response, RAI 15.4-1 is resolved.

15.4.2.3 *Technical Evaluation*

The staff has reviewed the applicant's analyses and finds that the calculational methods used for the radiological consequence assessment are acceptable and the radiological consequences calculated by the applicant meet the relevant dose acceptance criteria specified in SRP Section 15.0.3 and RG 1.183. The applicant conservatively postulated that a total of two spent fuel assemblies experience damage to the cladding on all fuel rods. One fuel assembly is dropped either into the spent fuel storage pool or onto the reactor core, which impacts fuel assemblies (equivalent to one fuel assembly) in the pool or in the reactor core. In its evaluation the staff considered the wet weight of a dropped fuel assembly, a drop height, and a factor of two reductions to obtain the kinetic energy in a fuel assembly drop through water. The staff finds the total number of failed fuel rods is less than the total fuel rods in two fuel assemblies. The applicant assumed that these two damaged fuel assemblies had undergone 24 hours of decay time and that all fission products in the gap of every rod in the two damaged fuel assemblies were instantaneously released.

The ESBWR TS 3.9.7, "Decay Time," requires the reactor to be subcritical for at least 24 hours before refueling operation. Therefore, the FHA could occur no earlier than 24 hours following reactor shutdown. The applicant assumed a radial peaking factor of 1.7 for the damaged rods in accordance with the guideline provided in RG 1.183. The kinetic energy developed in this drop is conservatively assumed to be dissipated in damage to the cladding on all fuel rods in two fuel assemblies. All fission product inventories in the fuel rod gap of the damaged fuel rods are assumed to be instantaneously released because of the accident.

Instantaneous release of noble gases and radioiodine vapor from the gaps of the broken rods (8 percent of iodine-131, 10 percent of krypton-85, and 5 percent of other iodine and noble gas in the reactor core are assumed to be in all fuel rod gaps) is assumed to occur, with the released gases bubbling up through the fuel pool water (with an effective decontamination factor of 200 for total iodine). These gap fractions and the effective decontamination factor are consistent with the guidance provided in RG 1.183. The applicant assumed that iodine in the particulate form is not volatile; therefore, it is not released. In accordance with the RG 1.183 guidance, the applicant assumed that the particulate cesium iodide (CsI) is instantaneously converted to the elemental form of iodine when it is released from the fuel into the pool water.

For the control room, the applicant assumed that the room will not be isolated during the postulated FHA, and the control room emergency filtration unit (CREFU) will not be operational. The applicant further assumed that the normal control room ventilation system will be operational during this event with no credit for fission product removal. The applicant used a normal control room ventilation system flow rate of 270 liters per second (l/s) (572 cubic feet per minute [cfm]) as an unfiltered air in-leakage rate into the control room envelope for conservatism.

The applicant evaluated the maximum 2-hour TEDE to an individual located at the EAB, the 30-day TEDE to an individual at the outer boundary of the LPZ, and the 30-day TEDE to an individual in the control room. The resulting doses are less than the dose acceptance criteria specified in RG 1.183 and SRP Section 15.03. The staff performed an independent confirmatory dose calculation and found that the staff's results agree with the applicant's values. Both the applicant's and the staff's results meet the relevant dose acceptance criteria at the EAB, LPZ, and control room.

The staff performed independent radiological consequence calculations for the FHA occurring 24 hours after reactor shutdown, coincident with a loss of the spent fuel pool cooling capacity. The ESBWR design does not rely on safety-related equipment to cool the pools that contain spent fuel. The reactor building is provided with passively acting relief devices that allow the building to vent to the environment if the spent fuel pool cooling lost during refueling operation. The staff finds that the radiological consequence resulting from the FHA coincident with a loss of the spent fuel pool cooling capacity still meets the relevant dose acceptance criteria as stated above at the EAB, LPZ, and control room.

15.4.2.4 *Conclusion*

The staff concludes that the ESBWR design, as bounded by the hypothetical X/Q values proposed by the applicant, will provide reasonable assurance that the radiological consequences of a postulated FHA at the EAB and LPZ will be well within the dose criteria in 10 CFR 52.47(a)(2) (i.e., 25 percent of 25 Sv or 0.063 Sv [6.3 rem] TEDE) and that the radiological consequences to an individual in the control room as a result of a postulated fuel FHA will be within the dose criterion in GDC 19 (0.05 Sv [5 rem]) TEDE. Therefore, the staff finds the radiological consequence analysis provided by the applicant to be acceptable.

15.4.3 Loss-of Coolant Accident Containment Analysis

Staff evaluation of this Section is included in Section 6.2 of this report.

15.4.4 Loss-of-Coolant Accident ECCS Performance Analysis

Staff evaluation of this Section is included in Section 6.3 of this report.

15.4.5 Loss-of-Coolant Accident Inside Containment Radiological Analysis

15.4.5.1 *Regulatory Criteria*

The staff reviewed DCD Tier 2, Section 15.4.4, in accordance with SRP Section 15.0.3 and RG 1.183. The staff evaluated the radiological consequences of a LOCA against the dose criteria specified in 10 CFR 52.47(a)(2), of 0.25 Sv (25 rem) TEDE at the EAB for any 2-hour period following the onset of the postulated fission product release and 0.25 Sv (25 rem) TEDE

at the outer boundary of the LPZ for the duration of exposure to the release cloud. The staff used a criterion of 0.05 Sv (5 rem) TEDE to evaluate the radiological consequences from a LOCA in the control room of the ESBWR design, pursuant to GDC 19 of Appendix A to 10 CFR Part 50. The staff used the applicable guidance in RG 1.183 in its review of the radiological consequence analyses.

15.4.5.2 *Summary of Technical Information*

In DCD Tier 2, Section 15.4.4, the applicant analyzed a hypothetical design-basis LOCA. The applicant concluded that certain bounding sets of assumed χ/Q values specified in DCD Tier 1, Table 5.1-1, and DCD Tier 2, Table 2.0-1, in conjunction with the use of (1) the passive containment cooling system (PCCS) in the containment, (2) the natural deposition of fission product aerosol in the containment, (3) an essentially leak tight containment barrier, and (4) the control of the pH of the water in the containment to prevent iodine evolution, are sufficient to provide reasonable assurance that the calculated radiological consequences of a postulated design-basis LOCA will be within the relevant dose criteria established in 10 CFR 52.47 and GDC 19.

To support its conclusion, the applicant submitted the following LTR and three research reports:

- Licensing Topical Report, NEDE-33279P, "ESBWR Containment Fission Product Removal Evaluation Model" (GE Licensing Topical Report), Revision 2, dated July 9, 2008. This report provides the methodology used by the applicant to evaluate the radiological consequences of a postulated design-basis LOCA.

- Research Report, VTT-R-04413-06, "Estimation and Modeling of Effective Fission Product Decontamination Factor for ESBWR Containment—Part 1," issued October 2006 (VTT Report No. 1).

- Research Report, VTT-R-04413-06, "Estimation and Modeling of Effective Fission Product Decontamination Factor for ESBWR Containment—Part 2," issued December 2006 (VTT Report No. 2).

- Research Report, VTT-R-06771-07, "Estimation and Modeling of Effective Fission Product Decontamination Factor for ESBWR Containment—Part 3," Revision 2, issued March 2008 (VTT Report No. 3).

The staff found that the most relevant aspects concerning fission product distribution, transport, and removal following the postulated LOCA involve (1) the PCCS operation, (2) natural deposition of fission product aerosol within the containment, and (3) control of the pH of the water in the containment to prevent iodine evolution. The four reports listed above provide information on these aspects. In RAI 15.4-6, the staff requested that the applicant incorporate the radiological consequence analyses provided in these reports into DCD Section 15.4 or incorporate the reports into DCD Chapter 15 as appendices. In response, the applicant revised DCD Tier 2, Section 15.4, Revision 5, and incorporated the radiological consequence analyses from the LTR and the VTT reports in Revision 6 as the staff requested. The staff finds that the applicant's response is acceptable and therefore, this open item is resolved.

The applicant postulated the following three LOCA scenarios:

(1) RPV bottom drain line break with automatic depressurization system (ADS) operating and with degraded low-pressure makeup system

(2) RPV bottom drain line break with ADS failure and with degraded high-pressure makeup system

(3) Loss of preferred power with ADS operating and with degraded low-pressure makeup system

The applicant originally proposed accident scenarios (1) and (2) above, stating that the reactor core uncovers and fission product release timing is shortest for these scenarios. For accident scenarios (1) and (2), the use of a fully depressurized, low-pressure accident sequence in conjunction with the source term described in NUREG–1465 is appropriate because the release fractions for the source terms presented in NUREG–1465 are intended to be representative or typical of those associated with a low-pressure core melt accident. Both accident scenarios (1) and (2) have the same initiating event with different accident sequences.

The staff accepted the accident scenarios proposed by the applicant but requested that the applicant add one additional accident sequence, "loss of preferred power with ADS operating and with degraded low-pressure makeup system," since it is the most dominant contributor to the core damage frequency for the ESBWR. The applicant accepted the staff's request and agreed to evaluate the above three accident scenarios as representative of the spectrum of ESBWR LOCAs.

In RAI 15.4-17, the staff requested that the applicant describe each of the above three LOCA accident scenarios in more detail, complete with the sequence of events; operation and availability of the ESF systems, including the suppression pool; fission product transport pathways; and fission product release timing. In its response, the applicant revised DCD Tier 2, Section 15.4, Revision 5 and incorporated the above three LOCA accident scenarios in more detail, complete with the sequence of events; operation and availability of the ESF systems, including the suppression pool; fission product transport pathways; and fission product release timing. The staff finds that the applicant's response is acceptable and therefore, RAI 15.4-17 is resolved.

The applicant performed and provided the radiological consequence analysis only for accident scenario (1) above in DCD, Revision 3. In RAI 15.4-7, the staff requested that the applicant provide the same radiological consequence analyses for accident scenarios (2) and (3) above as for accident scenario (1). The staff requested the applicant to incorporate these two remaining radiological consequence analyses into the LTR NEDE-33279P, "ESBWR Containment Fission Product Removal Evaluation Model" and DCD Section 15.4. In addition, the staff requested that the applicant compare and discuss the results of the radiological consequences and fission product removal rates in the containment for all three accident scenarios.

In response, the applicant stated that it will revise the LTR and DCD Tier 2, Section 15.4, accordingly. Subsequently, in Revision 1 of the LTR and Revision 5 of the DCD, the applicant provided the same radiological consequence analyses for accident scenarios (2) and (3) in addition to the accident scenario (1) and discussed the results of the radiological consequences

and fission product removal rates in the containment for all three accident scenarios. The staff finds the applicant's response is acceptable and therefore, this open item is resolved.

Proposed DCD Tier 2, Revision 3, Section 15.4, states that the applicant's radiological consequence analyses are based on the NUREG–1465 ASTs and the methodology in RG 1.183. On-the-other-hand, the applicant also stated in DCD Tier 2, Section 15.4.4.2.1, that the core remains covered throughout the accident, and there is no fuel damage. The statement in DCD Tier 2, Section 15.4.4.2.1, was inconsistent with NUREG–1465 and RG 1.183. In RAI 15.4-8, the staff requested that the applicant rectify the inconsistencies in these statements. Specifically, the staff requested the applicant to review the entire LTR and Section 15.4.4 to ensure that no further discrepancies exist. In response, the applicant stated that it would revise the LTR and DCD Tier 2, Section 15.4.4, accordingly. In LTR Revision 2 and DCD Revision 5, the applicant revised the LTR and DCD Tier 2, Section 15.4.4, to be consistent with NUREG–1465 and RG 1.183. The staff finds that the applicant's response is acceptable and therefore, RAI 15.4-8 is resolved.

In RAI 15.3-25, the staff requested that the applicant provide complete source term information for the radiological consequence analysis for IEs. In response to RAI 15.3-25, the applicant provided the requested source term information in tabular form, including a complete fission product inventory of the core at 4.59×10^9 joules per second (J/s) (4,590 megawatts) thermal, along with its technical bases in DCD Tier 2, Section 15.3. In RAI 15.4-9, the staff requested that the applicant include this source term information (i.e., fission product inventory) in DCD Tier 2, Section 15.4.4. The applicant incorporated this information in Revision 5 of the DCD. Therefore, based on the applicant's responses, RAI 15.3-25 and RAI 15.4-9 are resolved.

All of the fission product releases caused by a postulated LOCA are the result of either a containment atmosphere leak through the reactor building (i.e., reactor building leakage), a containment atmosphere leak bypassing the reactor building (i.e., PCCS leakage), or a main steamline isolation valve leakage bypassing the turbine building (i.e., MSIV leakage). The ESBWR design does not have ESF systems outside of the containment; therefore, the applicant did not consider leakage from the ESF systems as part of its radiological consequence analysis (the SRP and RG 1.183 require a radiological consequence analysis for ESF system leakage).

The ESBWR containment consists of a drywell, a wetwell, a PCCS, and supporting systems to remove and control fission product leakage to the environment following a postulated LOCA, with rapid isolation of all pipes and ducts that penetrate the containment boundary. It is designed to prevent the uncontrolled release of fission products to the environment. The applicant stated that the containment will be built and tested periodically to ensure a leak rate at design pressure of less than 0.35 percent by weight per day (wt%/d) at the calculated peak containment pressure associated with a LOCA for the entire duration of the accident recovery (i.e., 30 days). Both the applicant and the staff used this leak rate in their respective radiological consequence analyses. The ESBWR design provides neither an ESF filtration (e.g., charcoal adsorbers) nor a safety-related containment spray system in the containment.

All containment leaks are released into the reactor building except for two potential leak paths that bypass the reactor building. The applicant assumed that a small fraction of 0.35 wt%/d containment leak rate through the PCCS (i.e., less than 0.01 wt%/d) would leak into the air space directly above the PCCS, and subsequently leak directly to the environment without mixing with the reactor building atmosphere (i.e., reactor building bypass). The applicant further assumed that the MSIV leak rate is less than 1.57 l/s (200 ft^3 per hour), and that leakage is released directly into the environment without mixing with the turbine building atmosphere.

These assumed leak rates are used by the applicant and by the staff for the radiological consequence analyses. The feedwater isolation valve lines are located in the main steam tunnel that is open to the turbine building.

In RAI 15.4-11, the staff requested that the applicant include the PCCS leak rate test in a pre-operational test program as an ITAAC item and in the TS as surveillance requirements. In response to RAI 15.4-11, the applicant included the PCCS leak rate test in ESBWR Chapter 16, TS Section 5.5.9, and in DCD Tier 1, Table 2.15.4-1. Therefore, RAI 15.4-11 is resolved. The ESBWR TS specify the maximum allowable containment and MSIV leak rates and the surveillance requirements.

The reactor building, a reinforced concrete structure that forms an envelope completely surrounding the containment, is designed to seismic Category 1 criteria. The reactor building isolation is designed to be tested under accident conditions. During normal plant operation, the potentially contaminated areas of the reactor building are maintained at a slightly negative pressure relative to the adjoining areas by exhausting the reactor building air through the nonsafety-related normal reactor building HVAC system. Following a postulated DBA, the reactor building is automatically isolated to provide a holdup for the decay of airborne fission products. The normal reactor building HVAC system will continue to operate following the postulated LOCA, only if power is available. Neither the applicant nor the staff claimed fission product mitigation by the normal reactor building HVAC system.

The applicant originally assumed in the ESBWR DCD Tier 2, Revision 4, that the effective mixing volume of $1.6\text{x}10^4$ m^3 (5.65×10^5 ft^3) will be available for mixing, for holdup, and for decay of fission products before leaking from the reactor building to the environment, and that an overall reactor building leakage rate will be less than 50 percent per day. The applicant stated that the reactor building envelope is not intended to provide a leak-tight barrier against radiological fission product release; however, the reactor building is capable of periodic testing to ensure that the leakage rates assumed in the radiological consequence analyses is met. The staff requested in RAIs 15.4-26 and 6.2-165 that the applicant (1) identify the flow paths to be isolated and the method to be used to verify the leak rate, (2) state whether the leakage rate test to meet the 50 percent-per-day limit is specified in the ESBWR TS, and (3) include this leak rate verification in Tier 1 as an ITAAC item to be confirmed at the COL stage. In its response to RAIs 15.4-26 and 6.2-165, the applicant identified the flow paths to be isolated and the method to be used to verify the leak rate. The applicant stated that the leakage rate test to meet the 50-percent-per-day limit is specified in ESBWR TS 3.6.3.1.4. The applicant included this leak rate verification in DCD Tier 1, Table 2.16.5-2, Revision 4, as an ITAAC item.

Subsequently, in DCD Revision 5, the applicant revised the reactor building mixing volume and its leakage rate. The revised effective mixing volume is now $1.2\text{x}10^4$ m^3 ($4.11\text{x}10^5$ ft^3), and its leakage rate is an exfiltration rate of 141.6 l/s (300 cfm). To justify these changes, the applicant provided an analysis of the reactor building mixing and leakage using the GOTHIC computer code. The staff reviewed the applicant's analysis and accepted the revised effective mixing volume and leakage rate. Section 6.2.3 of this report presents the basis for the staff's acceptance. Based on the applicant's responses, RAIs 15.4-26 and 6.2-165 are resolved.

The PCCS is designed to remove decay heat and fission products from the containment atmosphere following a postulated LOCA. The PCCS heat exchangers receive a steam-gas mixture with airborne fission products from the drywell atmosphere, condense the steam, and return the condensate with condensed fission products to the RPV though the GDCS pools. The non-condensables, including noble gases and volatile fission products, are drawn to the

suppression pool through a submerged vent line driven by the differential pressure between the drywell and wetwell. The non-condensables will again become airborne into the wetwell air space and flow back into the drywell during vacuum breaker openings.

The ESBWR design provides a suppression pool to condense steam and remove fission products following a postulated LOCA. The sequence of a postulated LOCA include, among other things, the operation and availability of the suppression pool as a passive fission product control and removal system. The accident scenarios evaluated involve the reactor bottom drain line breaks that result in a blowdown of the RPV liquid and steam to the drywell by means of the severed pipe. The resulting pressure buildup drives the mixture of steam, water, and other gases down through vents to the downcomers and into the suppression pool water, thereby condensing the steam and reducing the containment pressure.

The staff assumed, as specified and stipulated in 10 CFR 52.79(a)(1), the postulated LOCA to result in substantial meltdown of the core with subsequent release into the containment of appreciable quantities of fission products. The fission product release occurs in phases over a 2-hour period. The initial blowdown to the suppression pool will not include significant quantities of fission products. Subsequent fission product releases from reactor safety valves to the suppression pool will remove some fission products by the suppression pool water. The applicant assumed a decontamination factor of 10 for any particulate fission product and for iodine in its elemental form. RAI 15.4-7 requested that the applicant provide fission product removal rates in the containment for the entire period of the accident. In response to RAI 15.4-7, the applicant did not provide the information on fission product removal rates by the suppression pool as a function of time (i.e., for a period of 30 days) for accident scenarios 2 and 3 in Revision 4 of the DCD. Subsequently, in DCD Tier 2, Revision 5, the applicant provided this information in LTR NEDE-33279P. The information provided by the applicant is consistent with the guidance provided in RG 1.183 and therefore, RAI 15.4-7 is resolved.

The applicant assumed leakage of the MSIVs at the TS limit of 0.0623 m^3 per minute total (200 ft^3 per hour).

In RAI 15.4-10, the staff requested the applicant to explain whether the MSIV leakage in the turbine building is included in the total containment leakage rate of 0.5 wt%/d. The applicant's response stated that it is not included in the total containment leakage rate of 0.5 wt%/d. Based on the applicant's response, RAI 15.4-10 is resolved. Subsequently, the total containment leakage rate of 0.5 wt%/d was revised to 0.4 wt%/d in DCD Revision 5, and to 0.35 wt%/d in DCD Revision 6.

In RAI 15.4-19 the staff requested that the applicant include the main steam drain lines along with the main steam lines in the analyses of the loading conditions of the safe-shutdown earthquake (SSE) in the DCD. The main steamlines are classified as seismic Category 1 from the RPV interface to the outboard seismic restraint of the downstream MSIV. The steamlines and their associated branch lines outboard of the last reactor building seismic restraint, including the main steam drain lines, are dynamically analyzed to SSE conditions that determine the flexibility and structural capabilities of the lines under SSE conditions.

The main condensers are also dynamically analyzed to SSE conditions to ensure that fission products leaked through the MSIVs are enclosed.

In its response to RAI 15.4-19, the applicant stated that (1) the main steamlines and drain lines are designed to meet seismic Category I criteria and analyzed to dynamic loading criteria, (2)

the MSIV fission product leakage path to the main condenser is analyzed to demonstrate structural integrity under SSE loading conditions, and (3) the ITAAC in DCD Tier 1, Table 2.11.1-1, now requires the turbine main steam system piping and MSIV fission product leakage path to be able to withstand an SSE without loss of structural integrity. Based on the applicant's response, RAI 15.4-19 is resolved.

15.4.5.3 *Staff Evaluation*

15.4.5.3.1 Accident Source Terms

In SECY-94-302, "Source Term-Related Technical and Licensing Issues Relating to Evolutionary and Passive Light-Water-Reactor Designs," dated December 19, 1994, the staff proposed to use only the "coolant," "gap," and "early in-vessel" releases from NUREG–1465 for the radiological consequence assessments of DBAs for the passive advanced light-water reactor (ALWR) designs and exclude "ex-vessel" and "late in-vessel" releases. These source terms encompass a broad range of accident scenarios, including significant levels of core damage with the core remaining in the vessel. The former set of scenarios represents the most severe reactor accidents from which the plant could be expected to return to a safe-shutdown condition. As stipulated in 10 CFR 52.47(a)(2), an applicant performing a radiological consequence of accident analysis shall assume a fission product release from the core into the containment. Note 6 to this regulation states the following:

> The fission product release assumed for this evaluation should be based upon a major accident, hypothesized for purposes of site analysis or postulated from considerations of possible accidental events. Such accidents have generally been assumed to result in substantial meltdown of the core with subsequent release into the containment of appreciable quantities of fission products.

The staff considered the inclusion of the "ex-vessel" and the "late in-vessel" source terms in NUREG–1465 to be unduly conservative for DBA purposes. Such releases will result only from core damage accidents with vessel failure and core-concrete interactions. For passive ALWRs, the estimated frequencies of such scenarios are low enough that they need not be considered credible for the purpose of meeting the requirements of 10 CFR 52.47(a)(2). The Commission approved the staff-recommended technical position to use only the coolant, gap, and early in-vessel releases from NUREG–1465 for the radiological consequence assessments of DBAs for passive ALWR designs.

The objective of NUREG–1465 is to define an accident source term for regulatory application for future light-water reactors (LWRs). The intent was to capture the major relevant insights available from severe accident research to provide a more realistic portrayal of the amount of the postulated accident source term. These source terms were derived from examining a set of severe accident sequences for LWRs of current design. Because of general similarities in plant and core design parameters, these results are considered to be applicable to passive LWR designs. The NRC has used this source term in evaluating the Westinghouse AP600 and AP1000 standard reactor design certification applications.

RG 1.183 provides guidance to licensees of operating power reactors on acceptable applications of ASTs pursuant to 10 CFR 50.67. This RG establishes an acceptable AST based on insights from NUREG–1465 and establishes the significant attributes of other ASTs that may be found acceptable by the staff for operating LWRs. RG 1.183 also identifies acceptable radiological analysis assumptions for use in conjunction with the accepted AST for operating

power reactors. The applicant followed the applicable guidance in RG 1.183 for the ESBWR design.

15.4.5.3.2 Radiological Consequence Analysis

In DCD Tier 2, Section 15.4.4, the applicant analyzed a hypothetical design-basis LOCA. The applicant concluded that certain bounding sets of hypothetical χ/Q values specified in DCD Tier 1, Table 5.1-1, and DCD Tier 2, Table 2.0-1, in conjunction with the use of the PCCS in the containment, the natural deposition of fission product aerosol in the containment, an essentially leak tight containment barrier, and the control of the pH of the water in the containment to prevent iodine evolution, are sufficient to provide reasonable assurance that the calculated radiological consequences of a postulated design-basis LOCA will be within the relevant dose criteria established in 10 CFR 52.47(a)(2) and GDC 19.

15.4.5.3.2.1 *Primary Containment Atmosphere Leakage*

The ESBWR design does not provide an active containment atmosphere cleanup system. Instead, the design relies on natural aerosol removal processes for deposition in the containment structural surfaces and the PCCS condensers, such as gravitational settling and plateout through diffusiophoresis and thermophoresis. The GEH LTR, NEDE-33279P, and VTT Reports 1, 2, and 3 discuss the removal of airborne activity from the containment atmosphere. The applicant provided a nonsafety-related containment spray system for accident management following a severe accident as part of the ESBWR fire protection system design. The containment spray system design is not safety-related and is not intended to be used during or following the postulated LOCA. Therefore, radiological consequence assessments give no credit for removing fission products by the containment spray system.

(4) Iodine Removal

The ESBWR passive containment design utilizes a unique PCCS to transport decay heat from a damaged reactor core to a water-pool heat sink and thereby to reduce the containment pressure. Following any initial pressure transients associated with reactor vessel blowdown, long-term heat rejection in the ESBWR is accomplished by heat rejection to the PCCS water pools by the flow of steam drawn into the cool condenser tubes of the PCCS. Steam produced by boiling in the reactor vessel enters the containment by way of the open direct pressure vent lines and flows into the PCCS. The condensate from the PCCS returns to the GDCS pool and subsequently returns to the reactor vessel. Thus, water is maintained in the reactor vessel by a supply from the GDCS pool.

While the design of the PCCS should prevent reactor core damage, the applicant and staff assumed substantial meltdown of the core as a result of the postulated LOCA, with subsequent release into the containment of appreciable quantities of fission products as stipulated in 10 CFR 52.47(a)(2) and in 10 CFR 100.21.

Condensation occurring in the PCCS tubes driven by the boiling of water in the reactor vessel provides a very effective means of scrubbing radioactive iodine in the drywell, and in time, most of the drywell iodine will be captured in the PCCS condensate. Since the PCCS condensate drains back into the reactor vessel, most of the iodine will reside in the water of the reactor vessel. NUREG–1465 specifies that, after an accident, iodine entering the containment from the reactor core is composed of at least 95-percent CsI, with the remaining 5 percent comprising elemental iodine and a small amount of hydriodic acid. However, about 3 percent of

elemental iodine in contact with some organic compounds will produce organic iodides. Therefore, the iodine in the containment will consist of 95-percent particulate iodine as CsI, 4.85-percent elemental iodine (I_2), and 0.15-percent organic iodine. The composition of the iodine in the ESBWR is consistent with the composition specified in NUREG–1465 and RG 1.183.

Both gaseous and particulate iodine can be scrubbed from the drywell in the PCCS condenser tubes and delivered back to the reactor vessel by the draining condensate. Within the boiling vessel, the CsI in particulate/aerosol form will subsequently dissociate to form Cs^+ and I^-. Here, the aqueous I_2 and methyl iodine (CH_3I) together with the dissociated I^- may undergo complex chemical reactions in the high-radiation environment of the boiling reactor vessel, producing a wide range of chemical and ionic forms of iodine, including volatile I_2. Dissolved I_2, much of which was originally in the form of CsI when initially released to the containment atmosphere, may subsequently return to the containment atmosphere as gaseous iodine at the surface of the water pool in the reactor vessel and subsequently be carried to the containment atmosphere by the steam leaving the reactor vessel.

This ionized iodine again flows into the PCCS where it can be dissolved into the condensate and reintroduced to the reactor vessel. Therefore, it may be postulated that there is a continuous refluxing of iodine from the PCCS to the reactor vessel, and from the reactor vessel back into the containment atmosphere, and back into the PCCS tubes. Meanwhile, airborne volatile iodine in the containment atmosphere will be adsorbed on the walls and wetted surfaces of the containment and removed by gravitational settling and plateout through diffusiophoresis and thermophoresis. The staff believes that the combination of production (sources) and removal (sinks) will lead to a steady-state concentration of gaseous iodine in the containment atmosphere that will leak to the environment at a design-basis leak rate.

In RAI 15.4-29, the staff requested that the applicant explain the iodine transport phenomena in the ESBWR containment and perform a rate analysis of steady-state iodine transport within the containment including iodine revolatilization (source) from the reactor vessel and iodine removal by the PCCS condenser and by natural deposition (sink). In response, the applicant provided an analysis report titled "Iodine Re-Volatilization from the Reactor Pressure Vessel During Late-Stage ESBWR LOCA." The applicant's response addressed the evolution of iodine in the volatile elemental iodine form in the event of a change in pH of the water pool in the RPV from alkaline to acidic conditions in the course of a LOCA. Detailed evaluation of the applicant's response to this RAI by the staff follows in Section (3) below of this report.

(5) Aerosol Removal

Applying credit for aerosol removal through the PCCS requires input from T-H analyses in the containment. The basis document defining the revised accident source term, NUREG–1465, does not specify an associated T-H scenario, methodology, or acceptance criteria for fission product removal. The AST regulatory guidance, RG 1.183, also does not specify these items. NUREG–1465 describes a source term derived from an examination of a set of severe accident sequences for LWRs and is intended to be representative or typical and does not imply a specific scenario, much less the worst case.

In the past, the staff and industry have evaluated aerosol removal through well-established models of spray removal or condensation. The ESBWR design relies on natural deposition processes in the PCCS that depend strongly on local T-H conditions. While gravitational settling is relatively easy to understand, aerosol removal through diffusiophoresis and

thermophoresis is much more complex. Diffusiophoresis is associated with steam condensation on the heat sinks and depends on the condensation steam mass flux. Thermophoresis relies only on the temperature gradient close to the surface on which the particles will be deposited.

Thermophoresis is more subtle than the other two natural deposition processes. Because the temperature gradient cannot be measured or easily calculated, its model uses the heat flux at the surface divided by the thermal conductivity of the gas adjacent to the surface as an equivalent measure of the driving force. Simultaneous occurrence of the two phoretic processes introduces an additional level of complexity.

The applicant used the MELCOR code to establish T-H boundary conditions and to estimate fission product removal rates in the containment by the PCCS. The MELCOR code is an NRC severe accident code and is a fully integrated, engineering-level computer code with the primary purpose of modeling the progression of a severe reactor accident and estimating fission product source terms. In DCD Tier 2, Table 15.4-5, the applicant provided aerosol removal coefficient values starting at the onset of a gap release through the first 12.5 hours into a DBA. The values ranged from 0 to 6.5 per hour.

In its independent evaluation of aerosol removal coefficients, the staff considered the same natural processes for removing aerosols from the containment atmosphere as described above. These processes include the sedimentation mechanism of gravitational settling, such as aerosol agglomeration, and the phoretic mechanisms of diffusiophoresis and thermophoresis in the PCCS.

For the staff's independent evaluation of aerosol removal coefficients, the staff contracted with Sandia National Laboratories (SNL) to evaluate aerosol removal coefficients and to perform quantitative analyses of uncertainties in predicting the aerosol removal rates, both in the containment and the main steamlines. Sandia used a MELCOR ESBWR containment-only model, incorporating the three accident scenarios described in Section 15.4.5.2 above. The NUREG–1465 radiological source term for the gap release and in-vessel release phases were used in place of the source term predicted in the fully integrated MELCOR analysis. The uncertainty analysis considered those MELCOR parameters known to affect aerosol settling and depletion to be uncertain within a range of values, represented by an assumed distribution function.

The staff's contractor used a Monte Carlo method which randomly samples the uncertain parameters. The uncertain parameter distributions were randomly sampled 150 times. The sampled values were then incorporated into 150 realizations of the containment-only ESBWR MELCOR model. The model results were used to calculate the distribution of aerosol removal rates in the ESBWR containment and the main steamlines.

In its evaluation of aerosol removal rates, SNL used the containment geometry (e.g., volume, upward-facing surface area) provided by the applicant and the fission product release timing, fractions, and release rates described in NUREG–1465. The staff's analyses considered the following principal uncertainties in aerosol properties and aerosol behavior:

- Aerosol size and distribution
- Aerosol void fraction and particle shape factors
- Aerosol material density
- Nonradioactive aerosol mass
- Particle slip coefficient

- Sticking probability for agglomeration
- Boundary layer thickness for diffusion deposition
- Thermal accommodation coefficient for thermophoresis
- Ratio of thermal conductivity of particle to gas
- Turbulent energy dissipation
- Multipliers on heat and mass transfer to containment shell

After several discussions between the staff and the contractor, engineering judgment was used in choosing the parameters, as well as identifying the range and distribution of their values.

(6) Containment Pool Water Chemistry

Iodine in the form of CsI is soluble in the containment pool water. Some of it may be converted into the elemental form, (I_2), which can be released into the containment atmosphere. The released radioactive elemental iodine may leak out of the containment atmosphere to the reactor building and, subsequently, to the environment. To minimize formation of elemental iodine, the pH of the containment pool water should be kept basic.

The ESBWR design includes three pertinent water pools: the PCCS pool in the reactor building and the GDCS pool and suppression pool in the containment. During normal plant operation, the pH of these pools will be between 6 and 7. In RAI 15.4-28, the staff requested that the applicant provide pH values for water in each pool (i.e., the PCCS pool, GDCS pool, and suppression pool), the RPV, and the lower drywell following the postulated LOCA for the duration of the entire accident period (i.e., 30 days). In response to this RAI, the applicant provided pH values for water in each pool following the postulated LOCA for the duration of the entire accident period in VTT Report No. 3. The applicant determined the pH for the various pools inside containment for 30 days after the postulated LOCA for the three accident scenarios described in Section 15.4.3.2 above using input from VTT Technical Research Centre of Finland. VTT used the commercially available Chemsheet code to calculate, among other things, pH values in the containment water pools following a DBA. The staff reviewed the report and finds that the applicant addressed and provided the pH values for water in each pool following the postulated LOCA for the duration of the entire accident period in VTT Report No. 3. The applicant concluded from this report that the containment pool water pH remains at a value above 7, and therefore, iodine trapped in the pools does not re-evolve into the containment atmosphere. For this reason the staff finds the pH and iodine transport analyses in the report to be acceptable. Based on the applicant's responses, RAIs 15.4-28 and 15.4-29 are resolved. The bases for the staff's acceptance are described below.

The pH of the containment pool water after a LOCA is determined by acidic and basic chemical species released to the containment from different sources in the plant. The most significant effect on reducing containment water pH results from the hydrochloric acid produced by radiolytic decomposition of electric cable jackets. The applicant estimated the generation of hydrochloric acid by radiolytic decomposition of cable jacketing using the methodology described in NUREG/CR–5950, "Iodine Evolution and pH Control," issued December 1992. The applicant assumed that 92 percent of the cables reside in the lower drywell and the remaining 8 percent of the cables are in the upper drywell. The applicant made a conservative assumption by scaling the hydrochloric acid formation rates upward by 125 percent.

In RAI 15.4-14, the staff requested that the applicant identify the amount of cable insulation material used in the ESBWR containment and include it in DCD Tier 1 as an ITAAC item. In response to this RAI, the applicant revised DCD Tier 1, Revision 4, Section 2.15.1 and

Table 2.15.1-1, to include exposed cable mass. Therefore, based on the applicant's response, RAI 15.4-14 is resolved.

Nitric acid is produced by the irradiation of air and water. The applicant used the methodology described in NUREG/CR–5950 to determine the amount of nitric acid in the containment pools. This methodology considers the production of nitric acid to be proportional to the time-integrated radiation dose rate for gamma and beta radiation. The applicant made a conservative assumption by scaling the nitric acid (HNO_3) rates upward by 125 percent. The applicant made an additional conservative assumption by including the formation of HNO_3 in the water vapor in the containment atmosphere in addition to its formation in the water pools. HNO_3 is a strong acid and will lower the pH.

NUREG–1465 specifies that 5 percent of the total core cesium inventory is discharged to the suppression pool during the gap release phase, and an additional 20 percent is discharged during the early in-vessel phase. In both cases, cesium is released as cesium hydroxide (CsOH) and CsI. The cesium that is not in the form of CsI is assumed to exit the RCS in the form of CsOH. The applicant performed a sensitivity analysis of pH as a function of the amount of CsOH formation (at 100, 50, 25, 10, and 0 percent) to study the effect of uncertainty in cesium formation. The applicant assumed 50-percent cesium formation. CsOH is a strong base and will increase the pH.

Sodium pentaborate is a buffering solution primarily used as a backup means for criticality control within a postaccident RPV. Sodium pentaborate is injected directly into the RPV by the SLCS.

$$Na_2B_{10}O_{16} + 16\ H_2O \leftrightarrow 2\ Na^+ + 2\ H_2BO_3^- + 8\ H_3BO_3$$

Since boric acid is a relatively weak acid and sodium hydroxide (formed by the union of a sodium ion and hydroxyl ions) is a strong base, their solution has a buffering effect and will control pH in the containment pools at values higher than 7.0. The staff considers the buffering action of sodium pentaborate an important factor in enhancing the pH control of containment pools.

To minimize formation of elemental iodine, and thus to prevent its release into the containment atmosphere (and subsequent leakage to the reactor building at a design-basis containment leak rate and then to the environment from reactor building), the pH of the containment pool water must be kept near 7.0 (neutral) or preferably to basic. VTT Report No. 3 shows that the pH in the RPV becomes acidic at 704 hours, in the low drywell at 603 hours, and in the GDCS at 12 hours. The pH in the wetwell remains permanently at basic. The applicant used these pH values in its determination of aerosol removal rates in the containment in performing the radiological consequence analysis.

For the staff's independent evaluation of the containment pool water pH, the staff contracted with SNL. Sandia used the iodine pool model developed by their laboratory for the NRC in the MELCOR code to evaluate pH in the containment pools. Based on the findings and conclusions, the staff concluded that the containment pool water pH remains above 7.0, and therefore, iodine trapped in the water pools does not re-evolve into the containment atmosphere, which confirms the applicant's analysis.

15.4.5.3.2.2 *Main Steamline Isolation Valve Leakage*

The MSIVs automatically isolate the four main steamlines that penetrate the drywell in the postulated LOCA. Two MSIVs are on each steamline, one inside the drywell (i.e., inboard) and one outside the drywell (i.e., outboard). The MSIVs are functionally part of the primary containment boundary, and design leakage through these valves provides a leakage path for fission products to bypass the reactor building and enter the environment as a ground-level release.

The applicant assumed that the inboard MSIV failed to close in one of four main steamlines and its outboard MSIV leaks at a maximum allowable MSIV leakage of 1.58 l/s (200 standard cubic feet per hour [scfh]) specified in the ESBWR DCD TS. The applicant modeled one main steamline with the leak as a single main steamline and combined the three remaining nonleaking main steamlines into one equivalent main steamline. This leak rate is based on a design-basis LOCA maximum peak containment pressure of 0.33 MPa (48 psig). The applicant did not credit any reduction in drywell pressure or the MSIV leakage rate of 1.58 l/s (200 scfh) after 24 hours following the postulated LOCA. Leakage rates were held constant for the entire duration of the accident (i.e., 30 days) for conservatism. The DCD TS specifies the maximum allowable MSIV leak rate.

The applicant's analysis did not take credit for aerosol and iodine removal in the main steamlines or in the main steam drain lines. The applicant's analysis did take credit for aerosol and iodine removal in the main condensers, referencing BWR Owner's Group Topical Report NEDC-31858P, "BWROG Report for Increasing MSIV Leakage and Elimination of Leakage Control System," September 1993. In 1996, the staff accepted this topical report in reactor licensing for reactor plants that use the accident source terms specified in the Atomic Energy Commission's Technical Information Document (TID)-14844, "Calculation of Distance Factors for Power and Test Reactor Sites," issued 1962.

However, the ESBWR design uses the ASTs to meet the radiological consequence evaluation factors as expressed in TEDE as required by 10 CFR 52.49(a)(1) and 10 CFR 100.21. Therefore, the use of TID-14844 accident source terms is no longer acceptable to the staff. In RAI 15.4-22, the staff requested that the applicant provide the model, method, and assumptions used for fission product removal in the main condensers and justify the use of a TID-14844 accident source term for this pathway in estimating its radiological consequences. In response to this RAI, the applicant referenced VTT Report No. 3, which demonstrates that the aerosol removal rates using the MELCOR analysis were higher than those rates using the BWROG methodology. In addition, the applicant stated that it did not claim any credit for aerosol deposition in main steamlines and drain lines and therefore its analysis was more conservative. The staff finds the BWROG methodology used by the applicant for determining aerosol removal rates is more conservative than the MELCOR analysis used by the staff, and therefore, the applicant's response to be acceptable. Based on the applicant's response, RAI 15.4-22 is resolved.

15.4.5.3.2.3 *Reactor Building Leakage*

Section 6.2.3 of the ESBWR DCD Tier 2 describes the reactor building functional design, including reactor building leakage.

GDC 16, "Containment design," states that reactor containment and associated systems shall be provided to establish an essentially leak tight barrier against the uncontrolled release of

radioactivity to the environment and to ensure that the containment design conditions important to safety are not exceeded for as long as postulated accident conditions require. The applicant stated that GDC 16 does not apply because the reactor building is not considered to be a leak tight barrier.

The staff considered the applicant's statement with respect to the applicability of GDC 16. The applicant assumed that the reactor building leakage to the environment is no greater than 141.6 l/s (300 cfm). This assumption directly affects the results of the design-basis radiological consequence analyses required by 10 CFR 52.47 and the control room operator dose stated in GDC 19. The staff requested in RAI 15.4-26 that the applicant provide the method to be used to verify the reactor building leak rate and include the leakage rate as a TS and ITAAC.

In response to RAI 15.4-26, the applicant provided (1) the maximum leak rate that could occur from the reactor building under design-basis conditions (i.e., 141.6 l/s [300 cfm]) and (2) the method to be used to test reactor building leakage. The leakage rate test is specified in ESBWR TS 3.6.3.1.4 and identified in ESBWR DCD Tier 1, Section 2.16.5, as an ITAAC item.

To justify the 141.6 l/s (300 cfm) leak rate, the applicant provided an analysis of the reactor building mixing and leakage using the GOTHIC computer code. The staff reviewed the applicant's analysis and accepted the revised effective mixing volume and leakage rate. Section 6.2.3 of this report presents the bases for the staff's acceptance. Based on the applicant's response, RAI 15.4-26 is resolved.

15.4.5.3.2.4 *Control Room Radiological Consequence Analysis*

In DCD Tier 2, Section 15.4, the applicant reported the results of its radiological consequence analysis for personnel in the main control room (MCR), relying on the CREFU to limit the radioactivity to which personnel may be exposed. Section 6.5 of this report describes the staff's review and evaluation of the CREFU in more detail.

The original ESBWR design in DCD Tier 2, Revisions 0 through 2, included a passive control room emergency bottled air breathing system (EBAS) and did not provide an ESF atmosphere cleanup filtration system for the control room. Subsequently, in DCD Tier 2, Revision 3, the applicant changed its ESBWR control room design to provide the CREFU as an active containment ESF atmosphere cleanup filtration unit, designed to remove fission products from the control room habitability area and to pressurize the control room with nonradioactive air from outside following postulated DBAs. The CREFU is a safety-related system and a subsystem of the control building HVAC system located in the control building; it is designed to seismic Category 1 criteria. The CREFU, in ESBWR DCD Tier 2, Revision 3, replaces the passive control room EBAS provided in the original ESBWR design in DCD Tier 2, Revisions 0 through 2. In RAI 15.4-27, the staff indicated that it is aware of possible design changes that include the EBAS and requested that the applicant state whether the design changes are complete. The applicant responded that the EBAS is no longer applicable to the ESBWR design. Based on the applicant's response, RAI 15.4-27 is resolved.

DCD Tier 2, Revision 9, Section 6.4, describes the CREFU design, and Section 6.5 of this report provides the staff's evaluation. The applicant assumed an unfiltered air in-leakage rate of 5.66 l/s (12 cfm) in DCD Tier 2, Table 15.4-5, in its control room radiological consequence analysis. In RAI 15.4-30, the staff requested that the applicant include the pre-operational testing of assumed control room unfiltered air in-leakage rate in DCD Tier 1, Table 2.16.2-1 as an ITAAC item and in DCD Tier 2, Chapter 16 as a TS surveillance requirement in generic TS

(GTS) 3.7.2 and bases, and in GTS 5.5.12, "CRHA Boundary Program," in accordance with guidance provided in the STS generic change traveler, which the industry owners groups' Technical Specifications Task Force (TSTF) had submitted for staff review as TSTF-448, "Control Room Habitability," Revision 3, on August 8, 2006, as supplemented on December 29, 2006. The staff made TSTF-448-A, Revision 3, available for incorporation into operating plant TS, design certification GTS, and COL plant-specific TS on January 17, 2007

In ESBWR DCD Tier 1, Revision 4, the applicant specified the testing of assumed control room unfiltered air in-leakage rate in Table 2.16.2-6 as an ITAAC item and included its surveillance requirement in DCD Tier 2, Revision 4, Chapter 16, Section 5.5.12. The staff finds the response to RAI 15.4-30 to be acceptable and therefore RAI 15.4-30 is resolved.

In Revision 3 to the DCD, the applicant did not provide complete figures and tables showing the design features that will be needed by the COL applicant to generate site-specific control room χ/Q values at the COL stage. In RAI 2.3-9, the staff asked the applicant to provide figures showing control room intake, unfiltered in-leakage, and postulated DBA release locations to the environment. These figures are intended to provide a basis for determining the distances and directions between potential accident release pathways and intake and in-leakage pathways to the control room necessary to evaluate the radiological consequences. In response to this RAI, the applicant included the requested information in DCD Revision 5. Based on the applicant's response, RAI 2.3-9 is resolved.

In Revision 3 to the DCD, the applicant revised the control room χ/Q values in DCD Tier 1, Table 5.1-1, and Tier 2, Table 2.0-1, by listing them as standard plant site design parameters. Two sets of control room χ/Q values are provided for the reactor building, PCCS/reactor building roof, and turbine building release pathways; the first set represents unfiltered in-leakage and the second set represents the filtered air intake. In RAI 15.4-31, the staff requested that the applicant state which set of control room χ/Q values it used for the control room radiological consequence analysis and why. In response to this RAI, the applicant provided the requested information in DCD Tier 2, Revision 4. Based on the applicant's response, RAI 15.4-31 is resolved.

In DCD Tier 2, Section 15.4, the applicant reported the results of its radiological consequence analysis for personnel in the MCR during a design-basis LOCA, relying on the CREFU to limit the radioactivity to which the personnel may be exposed. After performing an independent radiological consequence dose calculation, the staff finds that the ESBWR control room design meets the 0.05 Sv (5 rem) TEDE criterion in GDC 19 for the postulated LOCA.

15.4.5.3.2.5 *Technical Support Center Radiological Consequence Analysis*

The technical support center (TSC) provides an area and resources for use by the applicant to provide plant management and technical support to the reactor operating personnel located in the control room in the event of an emergency. The TSC relieves the reactor operators of peripheral duties and communications not directly related to reactor operations and prevents congestion in the MCR.

The TSC is a required facility specified by the NRC regulation, 10 CFR Part 50, Appendix E, Section IV.E.8, as it relates to providing emergency facilities and equipment for use in an emergency. 10 CFR Part 50, Appendix A, GDC 19 requires the applicant to provide equipment at appropriate locations outside the control room with a design capability for prompt hot shutdown of the reactor and with a potential capability for subsequent cold shutdown of the

reactor. Its functional criteria are specified in NUREG–0696, February 1981, "Functional Criteria for Emergency Response Facilities," and the radiological acceptance criterion is specified in NUREG–0737, Supplement No. 1, January 1983, "Clarification of TMI Action Plan Requirements."

NUREG–0737 requires, among other things, radiological protection to assure that radiation exposure to any person working in the TSC would not exceed 0.05 Sv (5 rem) whole-body, or its equivalent to any part of the body, for the duration of an accident. The SRP Section 15.0.3 states that the radiation protection design of the TSC is acceptable if the total calculated radiological consequences for the postulated fission product release fall within the exposure acceptance criterion specified for the control room of 0.05 Sv (5 rem) TEDE for the duration of an accident.

In Section 13.3 of the ESBWR DCD Revision 6, the applicant describes the TSC design requirements and the staff evaluated it in Section 13.3 of this report. The applicant stated among other things, that the TSC is provided with radiological protection and monitoring equipment necessary to ensure that the radiation exposure to any person working in the TSC would not exceed 0.05 Sv (5 rem) TEDE for the duration of the accident. The staff audited the applicant's dose calculations and performed an independent TSC dose calculation generating the same results. Therefore, the staff finds that the TSC radiological consequence analysis provided in the ESBWR DCD is acceptable

15.4.5.3.2.6 *Hypothetical Atmospheric Dispersion Factors*

Because no specific site is associated with the ESBWR design, the applicant defined the offsite boundaries (EAB and LPZ) only in terms of various hypothetical χ/Q values. DCD Tier 1, Revision 9, Table 5.1-1, and DCD Tier 2, Revision 9, Table 2.0-1, list the hypothetical reference χ/Q values used in the radiological consequence analyses for the ESBWR design. Section 2.3.4 of this report provides the staff's evaluation of the hypothetical reference χ/Q values used for the control room radiological consequence evaluation. The staff will review site-specific χ/Q values for a COL application that references the ESBWR design. If site-specific χ/Q values exceed the referenced χ/Q values used in this evaluation (e.g., poorer dispersion characteristics), a COL applicant may need to consider compensatory measures, such as increasing the size of the site or providing additional ESF systems to meet the relevant dose limits given in 10 CFR 52.47 and GDC 19.

15.4.5.4 *Conclusion*

The staff performed an independent confirmatory dose calculation and found that the staff's results agree with the applicant's values. Both the applicant's and the staff's results meet the relevant dose acceptance criteria for the EAB, LPZ, and control room. Therefore, the staff concludes that the ESBWR design, is bounded by the hypothetical χ/Q values proposed by the applicant, will provide reasonable assurance that the radiological consequences of a LOCA at the EAB and LPZ will be within the dose criteria set forth in 10 CFR 52.47(a)(2) (i.e., 0.25 Sv [25 rem] TEDE) and that the radiological consequences to an individual in the control room as a result of a postulated LOCA will be within the dose criterion established in GDC 19 (0.05 Sv [5 rem] TEDE). Therefore, the staff finds the radiological consequence analysis provided by the applicant to be acceptable.

15.4.6 Main Steamline Break Outside Containment

15.4.6.1 *Regulatory Criteria*

The staff reviewed DCD Tier 2, Section 15.4.5, in accordance with SRP Section 15.0.3 and applicable guidance provided in Appendix D to RG 1.183. The staff evaluated the radiological consequences of this DBA against the dose acceptance criteria specified in SRP Section 15.0.3 and RG 1.183 of 0.025 Sv (2.5 rem) TEDE for an accident-initiated iodine spike and 0.25 Sv (25 rem) TEDE for a preaccident iodine spike at the EAB for any 2-hour period following the onset of the postulated fission product release. The staff used a criterion of 0.05 Sv (5 rem) TEDE to evaluate the radiological consequences from a postulated MSLB accident in the control room of the ESBWR design, in accordance with GDC 19 of Appendix A to 10 CFR Part 50.

15.4.6.2 *Summary of Technical Information*

The applicant postulated that one of the four main steamlines will rupture between the containment outer isolation valve and the TCV. The radiological consequences of a break outside containment will bound those from a break inside containment. The accident evaluated is the complete severance of a main steamline outside the containment at a location downstream of the outermost MSIV. The applicant presented its analyses of the radiological consequences of a postulated MSLB accident outside containment in DCD Tier 2, Section 15.4.5 and Tables 15.4-10 through 15.4-13. The main MSIVs are assumed to isolate the break within 5 seconds, as specified in the ESBWR DCD Tier 2, Revision 9, TS 3.6.1.3. The staff assumed the duration of this event to be 5.5 seconds, which includes an additional 0.5 seconds for MSIV response time. No other release mitigation (i.e., plateout, holdup, dilution) is assumed, and no fuel damage is projected to occur. The only radioactivity available for release from this event is the activity that was in the reactor coolant and steamlines during the normal plant operation before the break.

Following isolation of the main steam supply system (i.e., MSIV closure) ADS initiates depressurization. Once the reactor system has been depressurized, the GDCS automatically begins reflooding the reactor vessel, and therefore, no fuel damage is projected to occur. The radioactivity in the released coolant is assumed to be released to the environment instantaneously from the turbine building as a ground-level release.

The applicant concluded in DCD Tier 2, Revision 3, that no more than 8.2328×10^4 kilograms (kg) (181,339 pounds mass [lbm]) of reactor coolant will be lost through the break before automatic isolation and that less than 4.705×10^3 kg (103,634 lbm) of that will be lost as steam. In RAI 15.4-2, the staff requested the applicant to provide the source term information used in the MSLB accident analysis. In its response to RAI 15.4-2, the applicant stated that it is revising the MSLB event to determine exact mass release values. In RAI 15.4-2 S01, the staff requested that the applicant provide, among other things, revised steam and water mass releases for the MSLB accident. In its response, the applicant provided the revised steam and water mass releases stating that it will include the revised radiological consequence analysis of this event in its forthcoming Revision 5 to the ESBWR DCD Tier 2.

In DCD Tier 2, Revision 5, the applicant concluded that no more than 45,593 kg (101,513 lbm) of reactor coolant will be lost through the break before automatic isolation and that less than 21,084 kg (46,482 lbm) of that will be lost as steam. In DCD Tier 2, Revision 5, the applicant evaluated the dose to operators in the control room. The resulting 30-day TEDE to an individual in the control room is less than the GDC 19 dose criteria. The applicant assumed that the

control room will be isolated during this event and the CREFU is credited for removing fission products. The applicant assumed an in-leakage rate of 5.66 l/s (12 cfm) of unfiltered air into the control room envelope. Based on the applicant's response, RAI 15.4-2 is resolved.

15.4.6.3 *Staff Evaluation*

The staff performed an independent radiological consequence dose calculation for the two scenario cases for the MSLB accident described below.

For Case 1, the staff assumed that a temporary increase in the primary coolant iodine concentration (iodine spike) occurred as a result of the power/pressure transient caused by the MSLB accident. Before the accident, the staff assumed that the ESBWR reactor was operating at the equilibrium limit of 7.4 kilobecquerels per gram (kBq/g) (0.2 microcuries per gram [μCi/g]) for dose equivalent iodine-131 (DEI-131) in the primary coolant, as specified in the ESBWR TS. The iodine spike generated during the accident is assumed to increase the release rate of iodine from the fuel by a factor of 500. This increase in the release rate will result in an increased concentration in the primary coolant during the course of the accident.

For Case 2, the staff assumed that previous reactor operation had resulted in a primary coolant iodine concentration equal to the maximum instantaneous ESBWR TS limit of 0.148 megabecquerel per gram (MBq/g) (4 μCi/g) for DEI-131.

For both cases, the staff's independent radiological consequence dose calculation confirmed the applicant's assertion that a postulated MSLB accident meets the dose criterion provided in SRP Section 15.0.3 and in RG 1.183 at the EAB and LPZ, as well as the GDC 19 criterion of 0.05 Sv (5 rem) TEDE for the control room.

15.4.6.4 *Conclusion*

The staff performed an independent confirmatory dose calculation and found that its results agree with the applicant's values. Both the applicant's and the staff's results meet the relevant dose acceptance criteria at the EAB, LPZ, and control room.

Therefore, the staff concludes that the ESBWR design, as bounded by the applicant's proposed hypothetical χ/Q values, will provide reasonable assurance that the radiological consequences of an MSLB accident at the EAB and LPZ will be within the dose criteria specified in SRP Section 15.0.3 and RG 1.183 of 0.025 Sv (2.5 rem) TEDE for an accident-initiated iodine spike and 0.25 Sv (25 rem) TEDE for a preaccident iodine spike. Furthermore, the radiological consequences to an individual in the control room as a result of a postulated MSLB accident will be within the dose criterion set forth in GDC 19 of 0.05 Sv (5 rem) TEDE. Therefore, the staff finds the applicant's radiological consequence analysis to be acceptable.

15.4.7 Control Rod Drop Accident

As stated in Section 4.6 of this report the staff accepted the FMCRD as a system for which a CRDA is a very unlikely event; therefore, radiological analysis is not required.

15.4.7.1 *Regulatory Criteria*

As indicated in Section 15.4.7 of this evaluation the staff used SRP Section 15.4.9, and Section 4.2, Appendix B, Revision 3, to provide the interim acceptance criteria and guidance for

the RIA. RIAs consist of postulated accidents that involve a sudden and rapid insertion of positive reactivity. This accident scenario includes a CRDA for BWRs. The uncontrolled movement of a single control rod out of the core results in a positive reactivity insertion which promptly increases local core power. Fuel temperatures increase rapidly, prompting fuel pellet thermal expansion. The reactivity excursion is initially mitigated by Doppler feedback and delayed neutron effects followed by reactor trip.

Fuel Cladding Failure Criteria

(12) The high cladding temperature failure criteria for zero power conditions is a peak radial average fuel enthalpy greater than 170 cal/g (306 BTU/lbm) for fuel rods with an internal rod pressure at or below system pressure and 150 cal/g (270 BTU/lbm) per fuel rod with an internal rod pressure exceeding system pressure. For intermediate (greater than 5-percent rated thermal power) and full-power conditions, fuel cladding failure is presumed if local heat flux exceeds thermal design limits (e.g., CPR).

(13) The pellet/cladding mechanical interaction failure criteria are a change in radial average fuel enthalpy greater than the corrosion-dependent limit depicted in SRP Section 4.2, Appendix B, Figure B-2 (BWR).

Core Coolability Criteria

Fuel rod thermal-mechanical calculations, employed to demonstrate compliance with Criteria 1 and 2 below, must be based on design-specific information accounting for manufacturing ranges and modeling uncertainties using NRC-approved methods including burnup enhanced effects on pellet power distribution, fuel thermal conductivity, and fuel melting temperature.

(14) Peak radial average fuel enthalpy remains below 230 cal/g (414 BTU/lb).

(15) Peak fuel temperature must remain below incipient fuel melting conditions.

(16) Mechanical energy generated as a result of (a) nonmolten fuel-to-coolant interaction and (b) fuel rod burst must be addressed with respect to reactor pressure boundary, reactor internals, and fuel assembly structural integrity.

(17) There must be no loss of coolable geometry as a result of (a) fuel pellet and cladding fragmentation and dispersal and (b) fuel rod ballooning.

Fission Product Inventory

The total fission-product gap fraction available for release following any RIA would include the steady-state gap inventory (present before the event) plus any fission gas released during the event. The steady-state gap inventory would be consistent with the non-LOCA gap fractions cited in RG 1.183 (Table 3) and RG 1.195, “Methods and Assumptions for Evaluating Radiological Consequences of Design Basis Accidents at Light-Water Nuclear Power Reactors” (Table 2) and would depend on operating power history. Whereas diffusion governs the fission gas release into the rod plenum during normal operation, pellet fracturing and grain boundary separation are the primary mechanisms for fission gas release during the transient.

15.4.7.2 *Technical Information*

Section 4.6 of this report discusses the FMCRD system design features provided to reduce the occurrence of CRDAs. In DCD Section 15.4.6.2, the applicant listed the following highly unlikely events for postulating a CRDA:

- The reactor is at less than 5-percent power.
- Failures of both safety-related separation detection devices or failure of the rod block interlock occurs.
- The latch mechanism fails.
- A simultaneous additional failure causes the occurrence of a stuck rod on the same FMCRD.
- The control rod is withdrawn without the operators noticing that the control rod withdrawal did not result in a neutron flux increase.
- The stuck rod has to become unstuck.

15.4.7.3 *Staff Evaluation*

Based on the design features, the applicant believes that the ESBWR design incorporates sufficient safeguards to negate its susceptibility to excess reactivity events. Initially, the ESBWR DCD did not include design requirements for a CRDA analysis. The staff was concerned that several scenarios might lead to an excess reactivity event and that each scenario would require exploration to ensure that it was not beyond design basis. If any scenario were to be credible, acceptance criteria (e.g., coolability, radiological consequences) would need to be developed and an acceptable accident analysis performed to demonstrate that these criteria were satisfied. The inclusion of this family of accidents may involve changes to the proposed ESBWR TS (e.g., LCOs, ESF actuation system setpoints) and the ESBWR DCD (e.g., Sections 4.2, 4.6, and 15).

In RAI 4.6-23 and RAI 4.6-23 S01, the staff requested the applicant to describe any enhanced features (with respect the ABWR design) or design requirements developed for the ESBWR to minimize the probability of an excess reactivity addition event. The staff also requested the applicant perform a failure modes and effects analysis to discuss the probability and potential consequences for each scenario leading to an excess reactivity event. The staff reviewed the control rod drop event frequency estimates provided by GEH in response to RAI 4.6-23 and RAI 4.6-23 S01. The design and testing of the control rod and CRD mechanism include a number of diverse and redundant features for preventing a rod drop event, which is an indicator of high reliability in the design. Based on its review of key design and operational features and the applicant's fault-tree analysis, the staff concludes that GEH has provided a reasonable estimate of the rod drop frequency.

In RAI 4.6-23 S02, the staff requested the applicant demonstrate compliance with GDC 28, "Reactivity limits," and guidance provided in SRP Section 4.2, Appendix 4B. The staff also considered the applicant's control rod drop event frequency evaluation and regulatory requirements provided in response to RAI 4.6-23 S02. Based on the potential consequences of an unrestricted reactivity excursion and to ensure compliance with GDC 28, the staff concludes

that the ESBWR design must demonstrate RCPB integrity and acceptable radiological consequences for the CRDA, irrespective of the probability of a CRDA.

GEH provided the CRDA analyses in the response to RAI 4.6-23 S02 and RAI 4.6-38. GEH utilized a combination of the nuclear core simulator, PANACEA, and the T-H code, TRACG, for the analyses.

Compliance with GDC 28 is demonstrated by analysis of the consequences of a postulated CRDA. The staff notes significant conservatism in the analysis. In particular, the adiabatic assumption precludes any void formation (which would insert negative reactivity during the accident). Also, the calculations assumed that the worth of the dropped rod, regardless of its position during the startup withdrawal sequence, is added to a critical reactor.

The analysis appropriately assumed that the control rod is dropped from its fully inserted position to the position of the drive and explicitly accounted for the effects of exposure.

The staff notes that the calculation did not include either operator error or calculational biases and uncertainties. The staff, however, has reviewed the applicability of PANACEA version 11 (PANAC11) to evaluating nuclear characteristics for the ESBWR. The staff found that PANAC11 is suitable for calculations of blade worth for the ESBWR. The staff has approved previous versions of PANACEA to provide control blade worth and control rod drop shape information to downstream transient evaluations. Therefore, the staff is reasonably assured that the calculations are indicative of the expected ESBWR behavior.

The staff found that the low enthalpy rises are a result of low blade worth (less than 80 cents in all cases). Therefore, the staff finds that the calculational results indicating large margin are expected. The staff is reasonably assured that consideration of modeling biases, uncertainty, and operator error would not result in changes to the analytic result on the order of magnitude of the available margin. The large margins to cladding failure for the ESBWR initial core provide the staff with reasonable assurance that, for the core design described in the DCD, the radiological consequences are bounded by the DCD analyses and that barrier integrity is demonstrated.

NUREG–0471, "Generic Task Problem Descriptions (Categories B, C, and D)," issued June 1978 (ADAMS Legacy Library Accession No. 7811270090, Microfiche Address 94439:153, 94439:289), describes Unresolved Safety Issue (USI) D-3, "Control Rod Drop Accident." This issue is an ACRS generic concern which involves assessing the uncertainties in calculations of the CRDA, including the choice of a negative reactivity insertion rate due to a scram and the potential difference between results of a two-dimensional calculation and a three-dimensional calculation. The response by the applicant to RAI 4.6-38 refers to the analysis performed in response to RAI 4.6-23 S02. The response briefly describes a reload licensing screening approach, analysis procedures, and analytical results. The applicant performed the analyses using the PANAC11 (PANACEA version 11) three-dimensional simulator in a transient mode with six delayed neutron groups. PANAC11 calculates the fuel enthalpy rise according to an adiabatic model (by integrating transient power) and explicitly accounts for blade worth, nominal blade pull during startup, and radial power shapes. Section 4.3 of this report provides a detailed evaluation of the reactivity aspects.

The staff finds that GEH followed the SRP Section 4.2, Appendix B, interim acceptance criteria and analyzed the CRDA. Based on the applicant's response, RAIs 4.6-23 and 4.6-38 are resolved.

Since this accident does not result in any fuel failures or release of any primary coolant to the environment, core coolability and fission product criteria do not apply.

15.4.7.4 *Conclusion*

The staff concludes that the rod drop accident analysis is acceptable and meets the requirements of GDC 13 and 28. This conclusion is based on the following findings:

- The applicant met GDC 13 requirements by demonstrating that all credited instrumentation was available and that actuations of protection systems, automatic and manual, occurred at values of monitored parameters that were within the instruments' prescribed operating ranges.

- The applicant met GDC 28 requirements by providing reactivity control systems features that mitigate postulated reactivity accidents that could result in damage to the RCPB greater than limited local yielding or damage that impairs core cooling capability significantly.

The staff has evaluated the applicant's analysis of the assumed CRDA and finds the assumptions, calculation techniques, and consequences acceptable. Because the calculations predict peak fuel temperatures below melting conditions, prompt fuel rupture with consequent rapid heat transfer to the coolant from finely dispersed molten UO_2 presumably did not occur. The pressure surge results in a pressure increase below Service Limit C as defined in Section III of the ASME Boiler and Pressure Vessel Code (SRP Section 15.4.9-6, Revision 3, March 2007) for the maximum control rod worths assumed. The staff believes that the calculations are sufficiently conservative in both the initial assumptions and analytical models to maintain primary system integrity. Section 21.6 of this report provides additional information for the use of PANAC11.

15.4.7.5 *Post-COL Activity*

For use in assessing ESBWR reload cores, GEH has developed a conservative criterion for the maximum static control blade worth below which the enthalpy rise curve in Appendix B of Revision 3 to SRP Section 4.2 would not be exceeded. This criterion will be applied to future ESBWR reload cores to determine whether additional calculations are needed. Only if necessary, will the enthalpy rises be calculated using a conservative adiabatic methodology or a best-estimate methodology that has been approved by the NRC.

In accordance with TS 5.6.3, Item C and as discussed in response to RAI 4.6-23 S02, licensees will perform cycle-specific confirmatory evaluations based on an NRC-approved or NRC accepted method for reload cores to ensure that all requirements pertaining to a postulated CRDA are met.

15.4.8 Feedwater Line Break Outside Containment

Staff evaluation of this Section is included in Section 15.4.1 of this report.

15.4.9 Failure of Small Lines Carrying Primary Coolant outside Containment

15.4.9.1 *Regulatory Criteria*

The staff reviewed DCD Tier 2, Section 15.4.8, in accordance with guidance provided in SRP Section 15.6.2, Revision 2, and SRP Section 15.0.3. RG 1.183 neither provides guidance nor lists this event as a DBA. The staff considers the radiological consequence resulting from this event to be bounded by that resulting from the MSLB accident outside containment for all light-water BWRs.

The staff evaluated the radiological consequences of this DBA against the dose acceptance criterion specified in SRP Section 15.0.3 of 0.025 Sv (2.5 rem) TEDE at the EAB for any 2-hour period following the onset of the postulated fission product release. The staff used a criterion of 0.05 Sv (5 rem) TEDE for evaluating the radiological consequences in the control room of the ESBWR design, in accordance with GDC 19.

GDC 55, “Reactor coolant pressure boundary penetrating containment,” contains a provision to ensure isolation of all pipes that are part of the RCPB and which penetrate the containment building. Exempted from these specifications are small-diameter pipes (instrument lines) that must be continuously connected to the primary coolant system to perform their necessary functions. For these lines, methods of mitigating the consequences of a rupture are necessary because the lines cannot be automatically isolated.

15.4.9.2 *Summary of Technical Information*

For the ESBWR design, the applicant postulated an instantaneous and circumferential rupture of an instrument line that is connected to the primary coolant system outside of the containment, but inside of the reactor building at a location where it may not be isolated automatically for 30 minutes at normal reactor operating temperature and pressure. The applicant assumed that, 30 minutes after initiation of this event, the operator will detect the pipe break, scram the reactor, and initiates reactor depressurization. The applicant assumed the duration of this event to be 5.9 hours (0.5 hours to detect and 5.4 hours to depressurize the reactor). The applicant presented its analyses of the radiological consequences of a postulated small line break accident outside containment but inside the reactor building in DCD Tier 2, Section 15.4.8 and Tables 15.4-17 through 15.4-19.

The applicant estimated that 1.48×10^4 kg (32,595 lbm) of primary coolant will be released through the break until the reactor is depressurized before it is isolated and that 4.0×10^3 kg (8,834 lbm) of the primary coolant will flash to steam and be available for release. All of the iodine available in the flashed steam is assumed to be released via the reactor building to the environment without any mitigation. Furthermore, the applicant assumed that the iodine in the primary coolant was at the maximum equilibrium limit of 0.148 MBq/g (4 μCi/g) for DEI-131, as specified in the ESBWR TS.

The applicant evaluated the radiological consequence doses at the EAB and LPZ, and to reactor operators in the control room. The applicant assumed that the control room will be isolated during this event and the CREFU will be operational to remove fission products. The applicant used an in-leakage rate of 5.66 l/s (12 cfm) of unfiltered air into the control room envelope. The applicant analyzed the control room dose over a 30-day period. The resulting 30-day TEDE to an individual in the control room is less than the GDC 19 dose criterion.

In RAI 15.4-3, the staff requested that the applicant (1) state if applicant has taken any exceptions to the guidance provided in SRP Section 15.6.2, (2) provide steam and water break flow rates and reactor building leak rate used in dose calculation, and (3) provide a copy of dose calculation performed. In response to RAI 15.4-3, the applicant stated that it did not take any exceptions to the guidance provided in SRP Section 15.6.2 and provided information requested in items (2) and (3) above. The staff performed an independent dose calculation using the information provided by the applicant and confirmed the applicant's results meeting the dose acceptance criteria specified in SRP Section 15.0.3.

In RAI 15.4-3 S01, the staff requested that the applicant add the duration of the event, fission product release point, and site boundary and control room atmospheric dispersion values used in DCD Tier 2, Table 15.4-17. In DCD Tier 2, Revision 5, Table 15.4-17, the applicant provided the information requested in RAI 15.4-3 S01. The staff finds that the estimated duration of this event is consistent with the guidance provided in SRP Section 15.6.2 and the control room atmospheric dispersion values used are the same as those provided in DCD Tier 1, Table 5.1-1, and DCD Tier 2, Table 2.0-1. The applicant added the fission product release point in DCD Tier 2, Revision 5, Table 15.4-17. Therefore, based on the applicant's response, RAI 15.4-3 is resolved.

15.4.9.3 *Staff Evaluation*

While performing past licensing reviews, such as those for the AP600, AP1000, and ABWR, the staff determined that a small line break accident is expected to result in radiological consequences less than a small fraction of the dose reference values specified in 10 CFR 50.34(a)(1) and 10 CFR 52.47(a)(1). Furthermore, the staff believes that the radiological consequences resulting from a small line break accident are bounded by those resulting from the MSLB and the RWCU line failure outside containment. However, the staff performed an independent radiological consequence dose calculation for this event and confirmed the applicant's assertion that a postulated small line break accident indeed meets the dose criteria in SRP Section 15.0.3 and RG 1.183 at the EAB and LPZ, as well as the GDC 19 criterion of 0.05 Sv (5 rem) TEDE for the control room.

15.4.9.4 *Conclusion*

The staff performed an independent confirmatory dose calculation and found that its results agree with the applicant's values. Both the applicant's and the staff's results meet the relevant dose acceptance criteria for the EAB, LPZ, and control room.

Therefore, the staff concludes that the ESBWR design, as bounded by the applicant's proposed hypothetical χ/Q values, will provide reasonable assurance that the radiological consequences of a small line break accident at the EAB and LPZ will be within the dose criterion specified in SRP Section 15.0.3 and RG 1.183 of 0.025 Sv (2.5 rem) TEDE. Furthermore, the radiological consequences to an individual in the control room as a result of a postulated MSLB accident will be within the dose criterion set forth in GDC 19 of 0.05 Sv (5 rem) TEDE. Therefore, the staff finds the applicant's radiological consequence analysis to be acceptable.

15.4.10 RWCU/SDC Line Failure Outside Containment

15.4.10.1 *Regulatory Criteria*

Neither SRP Section 15.0.3 nor RG 1.183 lists this event as a DBA; therefore, the NRC does not require that it be analyzed for its radiological consequences. However, during promulgation of Appendix A to 10 CFR Part 52, the ACRS Subcommittee on the ABWR specifically recommended that the applicant analyze the radiological consequences from failure of the RWCU system line outside of containment Since SRP Section 15.0.3 and RG 1.183 provides guidance for the MSLB accident , the applicant analyzed, and the staff reviewed, the radiological consequences of this event for the ESBWR as a substitute for the failure of the RWCU system line outside of containment event.

The staff evaluated the radiological consequences of this event against the dose acceptance criteria specified in SRP Section 15.0.3 and RG 1.183 of 0.025 Sv (2.5 rem) TEDE for an accident-initiated iodine spike and 0.25 Sv (25 rem) TEDE for a preaccident iodine spike at the EAB for any 2-hour period following the onset of the postulated fission product release. The staff also used a criterion of 0.05 Sv (5 rem) TEDE to evaluate the radiological consequences in the control room of the ESBWR design, in accordance with GDC 19.

15.4.10.2 *Summary of Technical Information*

In DCD Tier 2, Revision 9, Section 15.4.9 and Tables 15.4-20 through 15.4-23, the applicant presented its analyses of the radiological consequences of a postulated RWCU system line failure outside containment.

The applicant assumed that the break will be instantaneous and circumferential and will occur on the downstream side of the outermost containment isolation valve, but on the upstream side of the RWCU demineralizer. The applicant assumed 66 seconds of break flow time (a 46-second built-in delay time for flow differential pressure instrumentation to activate an isolation signal and 20 seconds for the motor-operated isolation valve to close). The applicant further assumed that no fuel damage would result as a consequence of this event. The only radioactivity available for release from this event is the activity that was in the reactor coolant and the RWCU system during the normal plant operation before the break. The applicant limited the initial break flow rate to 2.218×10^3 kg/s (1.76×10^7 pounds mass per hour [lbm/hr]), assuming two-phase critical flow for limiting diameter piping inside containment. The applicant further assumed that no more than 1.33×10^5 kg (2.93×10^5 lbm) of reactor coolant would be lost through the break before automatic isolation and that less than 5.0×10^4 kg (1.10×10^5 lbm) would be lost as steam.

In RAI 15.4-4, the staff requested that the applicant (1) state if any operator actions are credited in the event of a RWCU/SDC system line failure, (2) provide the break flow rate and break flow duration used in the radiological consequence dose calculation, and (3) provide a copy of dose calculation performed. In response to RAI 15.4-4, the applicant stated that it did not credit operator actions and provided the break flow rate and break flow duration used in the radiological consequence dose calculation, and a copy of dose calculation performed. The staff performed an independent dose calculation using the information provided by the applicant and confirmed the applicant's results meeting the dose acceptance criteria specified in SRP Section 15.0.3.

In RAI 15.4-4 S01, the staff requested that the applicant add (1) the duration of this event, (2) fission product release point, and (3) site boundary and control room atmospheric dispersion values used by the applicant to Table 15.4-21. In response to RAI 15.4-4 S01, the applicant revised Table 15.4-4 and added the requested information in Table 15.4-21 of DCD Revision 5. Therefore, based on the applicant's response, RAI 15.4-4 is resolved.

The applicant evaluated the radiological consequence doses at the EAB, LPZ, and to operators in the control room. The applicant assumed that the control room will be isolated during this event and the CREFU will be operational. The applicant used an in-leakage rate of 5.66 l/s (12 cfm) of unfiltered air into the control room envelope. The applicant analyzed the control room dose over a 30-day period. The resulting 30-day TEDE to an individual in the control room is less than the GDC 19.

15.4.10.3 *Staff Evaluation*

The staff provided no specific regulatory guidance for evaluating the radiological consequences for this event in RG 1.183 or in SRP Section 15.0.3. Therefore, the staff reviewed this event using the guidance provided for the MSLB accident in SRP Section 15.0.3 and RG 1.183.

The staff performed an independent radiological consequence dose calculation for the following two scenario cases for this event. In Case 1, the staff assumed that a temporary increase in the primary coolant iodine concentration (iodine spike) occurred as a result of the power/pressure transient caused by this event. Before the accident, the staff assumed that the ESBWR reactor was operating at the equilibrium limit of 7.4 kBq/g (0.2 µCi/g) for DEI-131 in the primary coolant, as specified in the ESBWR TS. The iodine spike generated during the accident was assumed to increase the release rate of iodine from the fuel by a factor of 500. This increase in the release rate will lead to an increasing concentration of DEI-131 in the primary coolant during the course of the accident. In Case 2, the staff assumed that previous reactor operation had resulted in a primary coolant iodine concentration equal to the maximum instantaneous ESBWR TS limit of 0.148 MBq/g (4 µCi/g) for DEI-131. The staff's independent radiological consequence dose calculation for this event confirmed the applicant's assertion that a postulated small line break accident meets the dose criteria provided in SRP Section 15.0.3 and RG 1.183 at the EAB and LPZ, as well as the GDC 19 limit of 0.05 Sv (5 rem) TEDE for the control room.

15.4.10.4 *Conclusion*

The staff performed an independent confirmatory dose calculation and found that its results agree with the applicant's values. Both the applicant's and the staff's results meet the relevant dose acceptance criteria for the EAB, LPZ, and control room.

Therefore, the staff concludes that the ESBWR design, as bounded by the applicant's assumed χ/Q values, will provide reasonable assurance that the radiological consequences of this event at the EAB and LPZ will be within the dose criterion specified in SRP Section 15.0.3 and RG 1.183 of 0.025 Sv (2.5 rem) TEDE for an accident-initiated iodine spike and 0.25 Sv (25 rem) TEDE for a preaccident iodine spike. Furthermore, the radiological consequences to an individual in the control room as a result of this event will be within the dose criterion set forth in GDC 19 of 0.05 Sv (5 rem) TEDE. Therefore, the staff finds the applicant's radiological consequence analysis to be acceptable.

15.4.11 Spent Fuel Cask Drop Accident

Staff evaluation of this Section is included in Section 15.4.1 of this report.

15.5 Special Events

Historically, non-DBEs that are evaluated in BWR safety analysis reports or DCDs have been termed “special events.” The applicant retained this classification for the ESBWR. The applicant established the following criteria for special events:

- Events postulated in 10 CFR Part 50 to demonstrate some specified prevention, coping, or mitigation capabilities, without specifically requiring a radiological evaluation

- Events that include a common-mode equipment failure or additional failures beyond the single-failure criterion

The applicant analyzed special events in DCD Tier 2, Revision 9, Chapter 15. In some cases, these events form the technical bases for conclusions drawn in other sections of this report. In such instances, the applicant presented results in the corresponding section of the DCD. The staff will correspondingly reference the appropriate section in the report.

15.5.1 Overpressure Evaluation

Section 5.2.2 of this report presents the results of the staff’s evaluation of RPV overpressure protection.

15.5.2 Shutdown without Control Rods

The SLCS, which is evaluated in Section 9.3.5 of this report, provides for reactor shutdown without control rods.

15.5.3 Shutdown from outside the Main Control Room

Section 7.4.2 of this report evaluates shutdown from outside the MCR.

15.5.4 Anticipated Transient without Scram

An ATWS event results from an AOO, as defined in Appendix A to 10 CFR Part 50, followed by the failure of the reactor trip portion of the protection system specified in GDC 20. Since protection systems (e.g., the reactor trip system) must satisfy the single-failure criterion, the assumed failure of the reactor trip must be caused by multiple failures or a common-mode failure. Therefore, an ATWS event cannot be classified as either an AOO or a DBA.

The failure of the reactor to shut down during certain transients can lead to unacceptable RCS pressure, fuel conditions, or containment conditions. For a conventional BWR, AOOs with failure to scram could lead to unacceptable conditions, such as closure of the MSIVs or turbine trip with bypass available, if unmitigated unstable power oscillations are allowed to grow.

Safety issues associated with ATWS have been evaluated since the early 1970s. During NRC evaluations of vendor models and analyses addressing such events, ATWS was formally identified as USI A-9, “Anticipated Transients Without Scram.” NUREG–0460, “Anticipated

Transients Without Scram for Light Water Reactors," issued in 1980, presents the staff studies and findings regarding USI A-9. In 1986, the agency resolved USI A-9 through promulgation of 10 CFR 50.62, also known as the ATWS rule. The ATWS rule does not require ATWS analyses. SECY-83-293, "Amendments to 10 CFR Part 50 Related to Anticipated Transients Without Scram Events," dated July 19, 1983, and the *Federal Register* notice of the final rule (Volume 49, page 26036) present the bases for current regulatory requirements related to ATWS, including the associated regulatory evaluation.

15.5.4.1 *Acceptance Criteria*

The provisions of 10 CFR 50.62 specify the prescriptive requirements for ATWS. This regulation requires BWRs to have the following mitigating features for an ATWS event:

- An SLCS capable of injecting a borated water solution with reactivity control equivalent to the control obtained by injecting 5.43 l/s (86 gallons per minute [gpm]) of a 13 percent by weight sodium pentaborate decahydrate solution at the natural boron-10 isotope abundance into a reactor vessel with a 637.5 centimeter (cm) (251-in.) inside diameter

- An alternate rod insertion (ARI) system that is designed to perform its function in a reliable manner and that is independent from sensor output to the final actuation device

- An SLCS initiation that is automatic and designed to perform its function in a reliable manner for plants granted a CP after July 26, 1984

- Equipment to trip the reactor coolant recirculation pumps automatically under conditions indicative of an ATWS

The staff determined that this latter requirement does not apply to the ESBWR because the ESBWR does not contain recirculation pumps. The staff reviewed the ESBWR DCD to determine if the applicant had provided comparable actions.

The staff also compared BWR performance during an ATWS to the criteria used in the development of the ATWS safety analyses described in NEDO-24222, "Assessment of BWR Mitigation of Anticipated Transients Without Scram," issued December 1979. The criteria include the following:

- Limiting the peak vessel bottom pressure to less than the ASME Service Level C limit of 10.34 MPa (1,500 psig)

- Ensuring that the peak cladding temperature, maximum cladding oxidation, and coolable geometry remain within the limits specified in 10 CFR 50.46

- Limiting peak suppression pool temperature to less than the containment design temperature

- Limiting the peak containment pressure to a maximum of the containment design pressure

Finally, SRP Section 15.8 provided guidance for the staff's review of BWR ATWS. SRP Section 15.8 provides the applicable GDC that form the regulatory basis of the ATWS rule, as listed below:

- GDC 12, “Suppression of power oscillations,” which requires that oscillations are either not possible or can be reliably and readily detected and suppressed

- GDC 13, “Instrumentation and control,” which requires a control and monitoring system to monitor variables and systems over their anticipated ranges for normal operation, AOOs, and accident conditions

- GDC 14, “Reactor coolant pressure boundary,” which requires an extremely low probability of failure of the coolant pressure boundary

- GDC 16, “Containment design,” which requires that containment design conditions important to safety are not exceeded as a result of postulated accidents

- GDC 35, “Emergency core cooling,” which specifies that fuel and clad damage, should it occur, must not interfere with continued effective core cooling and that clad metal-water reactions must be limited to negligible amounts

- GDC 38, “Containment heat removal,” which requires that the containment pressure and temperature be maintained at acceptable low levels following any accident that deposits reactor coolant in the containment

- GDC 50, “Containment design basis,” which requires that the containment not exceed the design leakage rate when subjected to the calculated pressure and temperature conditions resulting from any accident that deposits reactor coolant in the containment

The ATWS rule specifies two requirements: (1) light-water cooled plants must have prescribed systems and equipment that have been determined to acceptably reduce risks attributable to ATWS events and (2) licensees must demonstrate the adequacy of the features specified in the rule. In addition, all required equipment and systems must be designed to perform their functions in a reliable manner. Design and quality assurance criteria for the required systems and equipment should meet or exceed the criteria established in conjunction with the ATWS rulemaking, as described in Appendix A to SRP Section 7.1A, dated December 4, 1997, to ensure adequate independence, diversity, and reliability as required by the ATWS rule.

15.5.4.2 *Summary of Technical Information*

For ATWS prevention and mitigation, the ESBWR provides the following:

- An ARI system that utilizes sensors and logic that are diverse and independent of the RPS

- Electrical insertion of FMCRDs that also utilize sensors and logic that are diverse and independent of the RPS

- Automatic feedwater runback that operates under conditions indicative of an ATWS

- An SLCS that automatically initiates under conditions indicative of an ATWS

The mitigation of ATWS events is accomplished by a multitude of equipment and procedures. These include ARI, FMCRD run-in, feedwater runback, ADS inhibits, and the SLCS. The following are the initiation signals and setpoints for the above responses:

- For ARI and FMCRD run-in, the following apply:
  - High pressure, or
  - Level 2, or
  - Either RPS scram command or SCRRI/SRI command and elevated power levels exist after time delay, or
  - Manual
- For SLCS initiation, the following apply:
  - High pressure and SRNM ATWS permissive for 3 minutes, or
  - Level 2 and SRNM ATWS permissive for 3 minutes, or
  - Manual ARI/FMCRD run-in signals and SRNM ATWS permissive for 3 minutes
- For feedwater runback, the following apply:
  - High pressure and SRNM ATWS permissive, or
  - Either RPS scram command or SCRRI/SRI command and elevated power levels persist after time delay, or
  - Manual ARI/FMCRD run-in
- ADS inhibit, the following apply:
  - High pressure and APRM not downscale for 1 minute, or
  - Level 2 and APRM not downscale, or
  - MCR controls manually inhibit the ADS under ATWS condition
- For HPCRD, the following apply:
  - Level 2 with maximum 10-second delay, or
  - Level 2 with maximum 145-second delay during loss of offsite power
- For IC, the following apply:
  - Closure of MSIV, or
  - High pressure for 10 seconds, or
  - Level 2 with 30-second delay or Level 1

15.5.4.3 *Staff Evaluation*

The ESBWR has an SLCS capable of automatically injecting 18.4 l/s (291 gpm) of sodium pentaborate solution into the RPV with the simultaneous operation of both accumulators. The 5.4 l/s (86-gpm) equivalency specified in the ATWS rule for the 637.5-cm (251-in.) RPV (i.e., sodium pentaborate decahydrate solution of 13 percent by weight at 5.4-l/s [86-gpm] for a 637.5-cm [251-in.] vessel) is satisfied by the 18.4 l/s (291 gpm) provided for the 706.1-cm (278-in.) EBSWR vessel. The staff evaluated compliance with this portion of the ATWS rule, as described in Section 9.3.5 of this report, and concluded that the applicant had satisfied these requirements.

The ATWS rule requires that the SLCS initiation must be automatic and must be designed to perform its function in a reliable manner for plants granted a CP after July 26, 1984. Section 9.3.5 of this report provides a detailed evaluation of the SLCS.

GDC 26, 27, and 28 require the SLCS, which is described in Section 9.3.5 of this report. Because the new CRD design eliminates the previous common-mode failure potential and there is a very low probability of simultaneous common-mode failures of a large number of FMCRDs, the staff considers a failure to achieve shutdown to be unlikely. The staff believes that the provisions of the ATWS rule continue to require the SLCS. In addition, the ESBWR incorporates automatic initiation of the SLCS under conditions indicative of an ATWS to meet the rule specified at 10 CFR 50.62.

The ESBWR incorporates electric-hydraulic FMCRDs, which provide motor-driven scram and hydraulic scram. In response to a scram signal, the control rods are inserted hydraulically by means of the stored energy in the scram accumulator, similar to the currently operating BWR CRDs. In the ESBWR, a scram signal is also given simultaneously to insert the FMCRD electrically by means of the FMCRD motor drive. This diversity (i.e., hydraulic and electric methods of scramming) provides a high degree of assurance for rod insertion on demand.

The ESBWR has an ARI system that is independent of the RPS from sensor output to the final actuation device. The ARI system has redundant scram air header exhaust valves. The ARI system is designed to perform its function in a reliable manner and is independent of the existing RPS system from sensor output to the final actuation device. Chapter 7 of this report provides a detailed evaluation of the ARI and RPS.

As stated in the evaluation criteria, the ATWS rule incorporates prescriptive requirements because it clearly reflects the BWR use of forced core flow circulation. Because the ESBWR uses natural circulation, there are no recirculation pumps to be tripped. Hence, the ESBWR cannot implement recirculation pump trip (RPT) logic.

The ESBWR does implement an ATWS automatic feedwater runback feature, which provides a reduction in water level, core flow, and reactor power, similar to the RPT in a forced circulation plant. This feature prevents reactor vessel overpressure and possible short-term fuel damage for the most limiting ATWS events. The staff finds that the feedwater runback feature is comparable to the RPT feature provided in BWRs with forced recirculation with respect to the requirements of 10 CFR 50.62(c)(5).

The ATWS rule is also specific as to the use of locking-piston CRDs. The ESBWR, however, uses the FMCRD design with both hydraulic and electrical means to achieve shutdown. Section 4.6 of this report describes this CRDS. The use of the FMRCD design reduces the common-mode failure potential of the existing locking-piston CRD by eliminating the scram discharge volume (potential mechanical common-mode failure) and by having an electric motor run-in diverse from the hydraulic scram feature. This latter feature allows rod run-in, if scram air header pressure is not exhausted because of a postulated common-mode electrical failure and simultaneous failure of the ARI system, thus satisfying the intent of the ATWS rule.

The staff issued RAI 15.5-5 regarding the manner in which the applicant credited operator actions. The staff sought clarification because the TRACG analysis of ATWS does not appear to include operator actions; discussion in the DCD suggests otherwise. In response, the applicant noted that the TRACG analysis of ATWS MSIV closure transient response evaluation assumes operator action to achieve the following:

(18) Maintenance of level at TAF + 1.524 m (5 ft) after the initial automatic feedwater runback

(19) Depressurization of the reactor, if the heat capacity temperature limit curve is reached

Since the above operator actions are consistent with the operator actions specified in Emergency Procedure Guidelines (EPG), the staff agrees with this response. Based on the applicant's response, RAI 15.5-5 is resolved.

The applicant analyzed several classes of transients to provide assurance that, based on a low estimated frequency of occurrence, unacceptable plant conditions will not occur in the event of an ATWS. The applicant demonstrated that RCS pressures will not exceed the ASME Code Service Level C limits of 120 percent of the RPV design peak pressure of 10.44 MPa (1,500 psig). The applicant performed this analysis using the TRACG systems code. The NRC staff reviewed the applicability of TRACG for the ESBWR ATWS analysis as presented in NEDE-33083-P, Supplement 2, "TRACG Application for ESBWR Anticipated Transients Without Scram Analyses." Section 21.6 of this report, as well as the staff's SER on NEDE-33083P (which is contained in the NEDE document), provides the staff's evaluation of NEDE-33083P, Supplement 2.

The applicant also used TRACG to analyze ESBWR stability during ATWS scenarios. Section 4.A.2 of this report addresses ESBWR stability during ATWS scenarios.

15.5.4.4 *Analysis*

To establish compliance with the criteria identified in Section 15.5.4.1 of this report, the applicant analyzed ATWS scenarios initiated by the following AOOs:

- MSIV Closure—The maximum values from this event are, in most cases, bounding of all events considered.

- Loss of Condenser Vacuum—Pressurization rate and energy addition to the pool may be as severe as those in the MSIV closure scenario.

- Loss of Feedwater Heating—This scenario may be limiting in terms of peak cladding temperature.

- Loss of Normal Ac Power to Station Auxiliaries—This scenario could challenge the capability of the plant to mitigate an ATWS event because of reduced available equipment.

- Loss of Feedwater Flow—This event is analyzed to demonstrate the plant's capability to mitigate ATWS events initiated by low level trips.

- Generator Load Rejection with Single Failure in the Turbine Bypass System–This event is not limiting, but was analyzed for completeness

- MSIV Closure without Scram in Combination with the ARI, FMCRD Run-in-Failure, and Automatic SLCS—This is the most limiting event. The results indicate that the peak calculated reactor pressure of 9.68 MPa (1,390 psig), containment pressure of 0.31 MPa (30 psig), suppression pool temperature of 72.8 degrees C (163 degrees F), and peak cladding temperature of 927.8 degrees C (1,702 degrees F) are all within the acceptance criteria and hence are acceptable.

The staff completed confirmatory neutronics analysis to determine the effects of localized boron concentration on the effective multiplication factor of the ESBWR core. For this analysis, the

staff used the Monte Carlo N-Particle Transport Code and a flux-squared, adjoined weighting factor to determine core criticality in situations with limited boron distribution.

Feedwater Water Temperature Operating Domain

The applicant has analyzed the impact of the proposed P-FWTOD extension on the limiting ATWS events. NEDO-33338, Revision 1, specifically calculates the following ATWS events at off-nominal FWT:

(20) MSIV closure
(21) Loss of condenser vacuum

The applicant analyzed these two ATWS events at the reduced FWT point (SP1) and increased FWT point (SP2) for the initial core design. In addition, GEH analyzed MSIV closure ATWS at both SP1 and SP2 conditions for the equilibrium cycle. These analyses used standard ATWS assumptions, with a combination of nominal and bounding inputs. The results show sufficient margin to ATWS criteria, including peak reactor pressure, peak clad temperature, suppression pool temperature, and containment pressure.

15.5.4.5 *USI A-9—Anticipated Transient without Scram*

The staff published its technical findings regarding this USI in Volume 4 of NUREG–0460, and the publication of the ATWS rule resolved the issue. The ESBWR design meets the ATWS rule, and hence this issue is resolved.

15.5.4.6 *Conclusion*

The staff concludes that the plant design adequately addresses ATWS events and meets the requirements of 10 CFR 50.62. This conclusion is based on the following:

- The applicant's plant design includes ATWS risk reduction features prescribed by the ATWS rule.

- These features are independent and diverse from the reactor trip system and are designed to be reliable, as required under the ATWS rule.

- The applicant has also proven or referenced information, analyses, and risk assessments that demonstrate that it has considered limiting ATWS transient and event sequences. Based on this information, the applicant has determined that features included in the design, in accordance with the ATWS rule, result in reasonable assurance, based on a low estimated frequency of occurrence, and that unacceptable plant conditions, as defined during the ATWS rulemaking, will not occur as a result of ATWS events.

- The applicant has provided an acceptable diverse scram system.

15.5.4.7 *Post-COL Activity*

SRP Section 15.8, Section III.4.C states that on a cycle-specific basis, the licensee must confirm that the ATWS analysis of record, based on new fuel design or power-density change, bounds the plant-specific core configuration. This is covered by TS 5.6.3, Item C.

15.5.5 Station Blackout

As required by 10 CFR 50.63, each light-water-cooled nuclear power plant must be able to withstand and recover from an SBO (i.e., loss of the offsite electric power system concurrent with reactor trip and unavailability of the onsite emergency ac electric power system) of a specified duration. In particular, 10 CFR 50.63 requires that, for the SBO duration, the plant must be capable of maintaining core cooling and appropriate containment integrity. The rule also identifies the factors that must be considered in specifying the SBO duration.

15.5.5.1 *Acceptance Criteria*

The provisions of 10 CFR 50.63(a) (2) requires the following:

> [T]he reactor core and associated coolant, control, and protection systems, including station batteries and any other necessary support systems, must provide sufficient capacity and capability to ensure that the core is cooled and appropriate containment integrity is maintained in the event of an SBO for the specified duration. The capability for coping with an SBO of specified duration shall be determined by an appropriate coping analysis.

As noted in RAI 15.5-8, Regulatory Position 3.2.7 of RG 1.155, "Station Blackout," states that the ability to maintain appropriate containment integrity during a loss of all ac power should be addressed. In DCD Tier 2, Section 15.5, the applicant addressed containment integrity in terms of design limits on pressures and temperatures. In RAI 15.5.-8, the staff requested that the applicant add a discussion to this section explaining what provisions are present to ensure valve position indication and closure for containment isolation valves that may be in the open position at the onset of an SBO.

In response to RAI 15.5-8, the applicant revised DCD Tier 2, Section 15.5.5.1, in Revision 5 to state that, "SBO requirements related to the required power for valve position indication and containment isolation closure verification are met." All containment isolation valves are safety-related. All containment isolation valve position indications are supplied from the safety-related dc system through the uninterruptible power supply, and hence the position indications are available to the operator at the onset of an SBO. Based on the applicant's response, RAI 15.5-8 is resolved.

The staff performed its systems review of SBO using Regulatory Position 3 of RG 1.155. SRP Section 8.4, issued March 2007, incorporates the guidance of Regulatory Position 3, which states the following:

- Section 3.2.1: Assume a 100-percent rated thermal power for 100 days.
- Section 3.2.2: Determine core cooling and decay heat removal capability.
- Section 3.2.3: Ensure adequate inventory.
- Section 3.2.4: Evaluate design adequacy and capability, including potential failures of equipment necessary to cope.
- Section 3.2.5: Consider use of nonsafety-related equipment.

- Section 3.2.6: Consider timely operator actions.

- Section 3.2.7: Address the ability to maintain appropriate containment integrity.

15.5.5.2 *Summary of Technical Information*

For its SBO analysis, the applicant used the following assumptions and inputs:

- The reactor is operating at 100-percent rated power and 100-percent rated nominal core flow, with nominal dome pressure and normal water level.

- The nominal American National Standards Institute/American Nuclear Society Standard 5.1-1994, “Decay Heat Power In Light Water Reactors,” decay heat model is assumed.

- SBO starts with loss of all ac power, which occurs at time zero. Auto bus transfer is assumed to fail.

- The loss of ac power trips the reactor, feedwater, condensate, and circulating water pumps. A turbine load rejection is also initiated.

- The reactor scram occurs at 2 seconds from the loss of power supply to the feedwater pumps because loss of feedwater flow results in a scram signal with a delay time of 2 seconds.

- Bypass valves open on load rejection signal and close 6 seconds later because of a loss of condenser vacuum or to control the reactor pressure when it begins to drop because of the reactor scram or both.

- The MSIVs close automatically 30 seconds after the water level reaches Level 2 or because of a loss of condenser vacuum; the valves are fully closed at 5 seconds.

- The CRD pumps are unavailable because of a loss of ac power. No safety systems are credited, with the exception of three ICs.

- ICs are automatically initiated upon loss of feedwater pump power buses at 3 seconds to remove decay heat following the scram and isolation. IC drain flow provides initial reactor coolant inventory makeup to the RPV.

- The analysis credits no automatic or manual action when the vessel reaches Level 2 or Level 3.

- Vessel depressurization occurs, and the inventory of vessel and other components remains constant. Changes in level are observed as a result of changes in liquid temperature and pressure.

Using these assumptions and initial conditions, the applicant analyzed the SBO scenario employing the TRACG computational code to conclude that, during a 72-hour coping period that credits no operator actions, the ESBWR is placed and maintained in a hot-shutdown condition. The coolant inventory is such that it remains above Level 1 in the vessel. As a result of ICS operation, coolant is not released into the drywell or wetwell. Therefore, the applicant asserts that containment integrity is maintained.

15.5.5.3 *Staff Evaluation*

The applicant used TRACG to analyze the SBO scenario. The staff had not previously determined that TRACG is qualified for this analysis. To establish qualification, the staff reviewed the existing TRACG qualification documentation that applies specifically to the ESBWR. The staff determined that the conditions predicted during the SBO scenario are within the limits of a LOCA and the ATWS scenarios for which TRACG approval is pending.

The TRACG qualification documentation states that TRACG is qualified to predict ESBWR system responses. The documentation also provides validation of the ability of TRACG to model IC behavior by comparison to test data from the PANTHERS facility. The staff verified TRACG in this respect by conducting a comparison to the staff's confirmatory TRACE calculations. In the SER approving NEDE-33083-P-A, the staff concluded that TRACG adequately modeled IC behavior.

In consideration of the nonlimiting nature of the reactor and system response during the SBO scenario, as well as the stated capability of TRACG to model IC performance, the staff concludes that the TRACG analysis adequately predicts ESBWR performance during an SBO.

The staff issued RAI 15.5-6 to verify that the SBO analysis assumed operation at 100-percent thermal power for 100 days. In response, GEH committed to providing this information in a revision to the DCD. The applicant updated DCD Tier 2, Revision 3, with this change. The staff reviewed and accepted the change. Based on the applicant's response, RAI 15.5-6 is resolved.

The selection of a coping time must be based on site-specific criteria, as required by 10 CFR 50.63. However, because passive plants will not have emergency ac power sources, applicants for such plants need not evaluate SBO coping duration as long as they are able to demonstrate that the design selected is capable of performing safety-related functions for 72 hours. The ESBWR is capable of maintaining the core in a hot-shutdown condition for at least 72 hours using three of the four ICs.

The applicant carried out the TRACG analysis for 20,000 seconds to demonstrate that the ICS is capable of maintaining a collapsed water level above the TAF and that a hot-shutdown condition can be achieved and maintained. The staff reviewed the applicant's analysis and determined that it demonstrated the adequacy of the core cooling, decay heat removal capability, and coolant inventory. The SBO analysis indicates that appropriate containment integrity is maintained throughout the duration of the event.

Because an IC is assumed to be out-of-service, the staff concludes that the applicant considered the potential for failure of equipment necessary to cope with an SBO. The use of nonsafety-related equipment is not assumed, and no operator actions are required.

Section 8.4 of this report provides additional information about the staff's evaluation of SBO.

15.5.5.4 *Conclusion*

The ESBWR reactor core and associated coolant, control, and protection systems, including station batteries and other necessary support systems, provide sufficient capacity and capability to ensure that the core is cooled and appropriate containment integrity in the event of an SBO for 72 hours. The applicant conducted an appropriate coping analysis to demonstrate the

capability for coping with an SBO with a 72-hour duration, and hence, the acceptance criteria are satisfied.

15.5.5.5 *Post-COL Activity*

DCD Tier 2, Revision 9, Section 15.5.5.3 states, “Re-analysis of this event is performed for each fuel cycle.”

15.5.6 Safe-Shutdown Fire

The applicant credits TRACG analysis of SBO to provide conservative results for the safe-shutdown fire scenario because a manual scram is initiated before evacuation of the MCR. The staff reviewed the set of initial conditions for both scenarios and determined that, because all four ICs are assumed to be available in the fire scenario as well as CRD flow, the SBO scenario is bounding for the fire scenario during a control room fire. From a reactor systems standpoint, the SBO review demonstrates system response adequacy as applied to a safe-shutdown fire. Section 9.5.1 of this report provides the staff’s evaluation of safe-shutdown fire from a fire protection perspective.

15.5.7 Waste Gas System Leak or Failure

Section 11.3.3 of this report evaluates waste gas system leak or failure.

15A EVENT FREQUENCY DETERMINATION

15A.1–15A.2 Scope and Methodology

The staff reviewed the methodology used in the determination of the event frequency. The applicant stated that it used the following types of analysis in determining the event frequency:

- For those initiating events explicitly modeled in the ESBWR probability risk assessment (PRA), the frequency of the initiating events is taken directly from the PRA. The staff found only one discrepancy of this type. For the stuck-open relief valve event, Section 15A used a modified number which was lower than that used in the ESBWR PRA.

- The event frequency is determined from actual BWR operating experience, modified to reflect the ESBWR improved design features. For cases in which the analysis depended on specific assumed design features or testing, these features and tests are identified as ESBWR design requirements. The staff verified that the applicant had described these designed features and tests in the appropriate sections.

- For events involving multiple independent hardware failures or human errors, the event frequency is based on conservative estimates of the hardware failures (including common-cause failures) and human errors. The staff verified that the applicant used this approach for the events of turbine trip with total bypass failure, generator load rejection with total bypass failure, LOFWH with failure of SCRRI/SRI, and inadvertent shutdown cooling function operation.

15A.3 Staff Evaluation Results

The staff compared event frequencies and failure probabilities used by the applicant in the analyses with data from operating reactors published in NUREG/CR–6928, “Industry-Average Performance for Components and Initiating Events at U.S. Commercial Nuclear Power Plants,” issued February 2007, and the latest update to NUREG/CR–5750, “Rates of Initiating Events at U.S Nuclear Power Plants 1987-1995,” issued February 1999. The staff found that the parameter values used are consistent with operating experience and in many cases are reflective of the 95th-percentile of the distribution of the operational data. In cases in which a comparison could not be made, the staff examined the impact of increasing the parameter by an order of magnitude to determine whether such increases produced results that exceeded the staff’s acceptance criteria.

The staff also verified that the final event frequencies are at least a factor of three above the criterion for the infrequent event (i.e., less than 3.33×10^{-3}/yr) to account for the modeling uncertainty.

A discussion of the staff’s evaluation of the frequency of each specific event follows.

15A.3.1 Pressure Regulator Failure—Opening of All Turbine Control and Bypass Valves

The SB&PC system controls the reactor pressure during plant operation. The SB&PC system is equipped with a triple-redundant, fault-tolerant digital controller (FTDC). The vendor will confirm the reliability of the FTDC as part of the COL applicant commitment. Upon vendor confirmation, the reliability of the SB&PC controller will meet the requirement that the mean time to failure

(MTTF) be greater than 1,000 years. The controller can either fail high, causing maximum demand, or fail low, causing minimum demand. Assuming that both failure modes are equally possible, the frequency of the controller's failing in a manner to cause maximum demand is estimated to be once in 2,000 years (i.e., 5.0×10^{-4}/yr).

In the initial submittal, the applicant did not address adequately the contribution of mechanical failure to the frequency of the events. In RAI 15.0-25, the staff requested the applicant to discuss the reasons for not considering mechanical failures and clearly state any assumptions made in the analysis regarding mechanical failures. RAI 15.0-25 was divided into multiple questions (i.e., Items (A), (B.1), and (B.2)) to the applicant. Sections 15A.3.2, 15A.3.6, 15A.3.11, 15A.3.12, and 15A.3.13 of this report discuss the applicable responses to RAI 15.0-25.

In response to RAI 15.0-25, Item (A), GEH assessed the mechanical failure of the pressure regulators and concluded that the likelihood of mechanical failure of the pressure regulators is negligible compared to the estimated overall failure frequency of 5.0×10^{-4}/yr. The staff reviewed the applicant's calculations and found them to be within acceptable ranges. Therefore, based on the applicant's response, RAI 15.0-25, Item (A), is resolved.

Based on the design requirement of the SB&PC described in Section 7.7.5 of this report, the staff agrees that this event frequency meets the criterion of less than 1.0×10^{-2}/yr.

15A.3.2 Pressure Regulator Failure—Closure of All Turbine Control and Bypass Valves

As described above in Section 15A.3.1, similar results are found for a closure of all turbine control and bypass valves. The SB&PC system controls the reactor pressure during plant operation. The SB&PC system is equipped with a triple-redundant FTDC. The vendor will confirm the reliability of the FTDC as part of the COL applicant commitment. Upon vendor confirmation, the reliability of the SB&PC controller will meet the requirement that the MTTF be greater than 1,000 years. The controller can either fail high, causing maximum demand, or fail low, causing minimum demand. Assuming that both failure modes are equally possible, the frequency of the controller's failing in a manner to cause minimum demand is estimated to be once in 2,000 years (i.e., 5.0×10^{-4}/yr).

In response to RAI 15.0-25, Item A, GEH assessed the mechanical failure of the pressure regulators and concluded that the likelihood of mechanical failure of the pressure regulators is negligible compared to the estimated overall failure frequency of 5.0×10^{-4}/yr. The staff reviewed the applicant's calculations and found them to be within acceptable ranges. Therefore, based on the applicant's response, RAI 15.0-25, Item (A), is resolved.

Based on the design requirement of the SB&PC described in Section 7.7.5 of this report, the staff agrees that this event frequency meets the criterion of less than 1.0×10^{-2}/yr.

15A.3.3 Turbine Trip with Total Bypass Failure

In RAI 15.0-20, Items (A) through (G), the staff requested the applicant to provide additional information to justify and/or clarify assumptions and statements made in DCD Tier 2, Revision 1, Sections 15A.3.3 and 15 A.3.4. In response to RAI 15.0-20, Items (A) through (F), GEH modified the model of the turbine bypass failure using the linked fault-tree approach. The modeling of the bypass valves failures includes the electric-hydraulic control (EHC) system, related mechanical components, and supporting power supplies. The staff reviewed the

applicant’s calculations and found them to be within acceptable ranges. Therefore, based on the applicant’s response, RAI 15.0-20, Items (A) through (F), are resolved.

Based on the modified turbine bypass failure model and industry data for the frequency of turbine trip, the frequency of turbine trip with total turbine bypass failure is 5.17×10^{-4}/yr. The staff agrees that this event frequency meets the criterion of less than 1.0×10^{-2}/yr.

15A.3.4 Generator Load Rejection with Total Turbine Bypass Failure

In response to RAI 15.0-20, Item (G), GEH proposed three alternatives to estimate the generator load rejection initiating event frequency. The staff agrees with the approach of using traditional generator load rejection frequency data as the initiating event frequency for generator load rejection. The staff reviewed the applicant’s calculations and found them to be within acceptable ranges. Therefore, based on the applicant’s response, RAI 15.0-20, Item (G), is resolved.

Based on the modified turbine bypass failure model and traditional approach for estimating the initiating frequency for generator load rejection, the frequency of generator load rejection with total turbine bypass failure is 1.98×10^{-4}/yr. The staff agrees that this event frequency meets the criterion of less than 1.0×10^{-2}/yr.

Staff has identified and resolved Items (A) through (G) for RAI 15.0-20.

15A.3.5 Feedwater Controller Failure

The FWCS accomplishes both RPV water level control and FWT control. The two functions are performed by two sets of triple-redundant controllers located in separate cabinets, using independent and diverse inputs. Two events of concern may result from failures of the FWCS. One event consists of the FWCS erroneously generating a maximum flow demand, and the other event consists of the FWCS erroneously generating a minimum temperature demand. The simultaneous occurrence of a maximum flow demand and a minimum temperature demand is considered incredible because of the independence of the two control schemes. The random probability of the second controller (e.g., temperature) failing while the first controller (e.g., flow) is failed, and before the effects of the first controller’s failure are mitigated is judged to be insignificant.

15A.3.5.1 *Feedwater Controller Failure—Maximum Flow Demand*

One function of the FWCS is to regulate the flow of feedwater into the RPV to maintain predetermined water level limits during transients and normal plant operating modes. The FWCS is equipped with a dedicated, triple-redundant FTDC, including power supplies, and input and output signals. The FTDC consists of three parallel processing channels, each containing the hardware and software for execution of the control algorithms. The FTDC is designed to a high degree of reliability, and the MTTF of the FTDC is at least 1,000 years. It is assumed that the feedwater flow controller can fail high or fail low with equal probability. Therefore, the frequency of the controller failing in a manner to cause maximum demand is less than once in 2,000 years.

Based on the design requirement of the FWCS described in Section 7.7.3 of this report, the staff agrees that this event frequency meets the criterion of less than 1.0×10^{-2}/yr.

15A.3.5.2 *Feedwater Controller Failure—Minimum Temperature Demand*

One function of the FWCS controls FWT to allow reactor power control without moving control rods. The FWCS is equipped with a dedicated, triple-redundant FTDC, including power supplies, and input and output signals. The FTDC consists of three parallel processing channels, each containing the hardware and software for execution of the control algorithms. The FTDC is designed to a high degree of reliability, and the MTTF of the FTDC is at least 1,000 years. It is assumed that the FWT controller can fail high or fail low with equal probability. Therefore, the frequency of the controller failing in a manner to cause minimum demand is less than once in 2,000 years.

Based on the design requirement of the FWCS described in Section 7.7.3 of this report, the staff agrees that this event frequency meets the criterion of less than 1.0×10^{-2}/yr.

15A.3.6 Loss of Feedwater Heating with Failure of Selected Control Rod Run-In and Selected Rod Insertion

In RAI 15.0-21, the staff requested the applicant to justify assumptions in the frequency estimate for “Loss of Feedwater Heating with Failure of SCRRI”. Based on the responses to RAIs 15.0-21 and 15.0-25, Item (B.1), GEH modified the initiating event frequency estimate by adding the SRI system to back up the SCRRI. The failure frequency calculation for this initiating event reflects the electrical, mechanical, and common-cause failure modes.

Based on the RAI responses and detailed modeling of the failure modes, the estimated failure frequency is 1.51×10^{-3}/yr. The staff agrees that this event frequency meets the criterion of less than 1.0×10^{-2}/yr. Therefore, based on the applicant’s response, RAIs 15.0-21 and 15.0-25, Item (B.1), are resolved.

15A.3.7 Inadvertent Shutdown Cooling Function Operation

In RAI 15.0-22, the staff requested the applicant to justify the assumed interlock frequency and operator error probability for “Inadvertent Shutdown Cooling Function Operation”. In response to RAI 15.0-22, GEH used a linked fault-tree approach to estimate the frequency of inadvertent SDC actuation. The analysis modeled valve functions, testing, and operator errors. Based on this approach, GEH estimated that the frequency of inadvertent SDC mode of operation is about 1.6×10^{-4}/yr.

Based on the RAI responses and improved modeling, the estimated event frequency of this event is 1.6×10^{-4}/yr. The staff agrees that this event frequency meets the criterion of less than 1.0×10^{-2}/yr. Therefore, based on the applicant’s response, RAI 15.0-22 is resolved.

15A.3.8 Inadvertent Opening of a Safety/Relief Valve

In RAI 15.0-23, the staff requested the applicant to justify the assumptions in event frequency estimate for “Inadvertent Opening of a Safety/Relief Valve”. In response, GEH included detailed failure modes of inadvertent opening of an SRV leading to vessel depressurization. Modeled failure modes include incorrect setpoints, vibration-induced failure, excess nitrogen pressure, spurious opening signal, operator error, and common-cause failures.

Based on the RAI responses and detailed modeling, the estimated event frequency of this event is 2.81×10^{-3}/yr. The staff agrees that this event frequency meets the criterion of less than 1.0×10^{-2}/yr. Therefore, based on the applicant's response, RAI 15.0-23 is resolved.

15A.3.9 Inadvertent Opening of a Depressurization Valve

In RAI 15.0-24, the staff requested the applicant to address PRA modeling of the instrumentation and control (I&C) system, including common-cause failures. In response, GEH modified the modeling of this event by using the linked fault-tree approach and including the common-cause failures.

Based on the RAI responses and improved modeling, the estimated event frequency of this event is 5.75×10^{-4}/yr. The staff agrees that this event frequency meets the criterion of less than 1.0×10^{-2}/yr. Therefore, based on the applicant's response, RAI 15.0-24 is resolved.

15A.3.10 Stuck-Open Relief Valve

In RAI 15.0-28, the staff requested the applicant to provide technical basis of unavailability of the ICS. In RAI 15.0-28, the staff noted that in DCD Tier 2, Section 15A.3.10, GEH estimated the initiating event frequency of a stuck-open relief valve by taking credit for the availability of the ICS for the ESBWR. The applicant assumed that the probability of the ICS being unavailable is less than 0.1. However, the applicant provided no justification for this number in this section. The staff asked the applicant to provide the technical basis for this number. The applicant's response to RAI 15.0-28 provided applicable information on the unavailability of the ICS, and explained that the assumed value of 0.1 is conservative because of the simplicity, redundancy, and diversity of the ICS system. The staff agrees with this response. Therefore, based on the applicant's response, RAI 15.0-28 is resolved.

The staff issued RAI 15.0-29 to clarify an inconsistency in the data used for the stuck open relief valve initiating event frequency in the DCD. In RAI 15.0-29, the staff noted that in DCD Tier 2, Section 15A.3.10, GEH provided a best-estimate value for the expected frequency of a stuck-open SRV of 3.28×10^{-4}/yr. However, the traditional number used for existing BWR plants is about 4.6×10^{-2}/yr (see NUREG/CR–5750). In addition, the number used in the ESBWR PRA is 2.23×10^{-2}/yr (see NEDO-33201, Revision 6, Section 2, "ESBWR Design Certification Probabilistic Risk Assessment"). The staff asked the applicant to explain why the ESBWR PRA did not use the best-estimate of ESBWR frequency (i.e., 3.28×10^{-4}/yr).

The applicant addressed RAI 15.0-29 by providing the initiating event frequency of a stuck-open SRV and explaining that the value used in the DCD is lower than that of the existing BWRs because GEH credited the surveillance testing requirement of SRVs during power operation for the ESBWR. The staff agrees with this response. Therefore, based on the applicant's response, RAI 15.0-29 is resolved.

Based on the RAI responses and improved modeling, the estimated event frequency of this event is 2.24×10^{-4}/yr. The staff agrees that this event frequency meets the criterion of less than 1.0×10^{-2}/yr.

15A.3.11 Control Rod Withdrawal Error during Refueling

This event is initiated by one or more operator errors followed by failure of the refueling equipment interlocks. According to the GEH estimate, the frequency of an RWE during

refueling is significantly less than once in 1,000 years, based on the multiple failures required for this event to occur. In response to RAI 15.0-25, Item (B.2), GEH assessed the mechanical failure of the FMCRD and concluded that the likelihood of mechanical failure of the FMCRD is negligible compared to the estimated overall failure frequency. The staff reviewed the applicant's calculations and found them to be within acceptable ranges.

Based on the RAI responses and estimated failure frequency of 1×10^{-3}/yr, the staff agrees that this event frequency meets the criterion of less than 1×10^{-2}/yr. Therefore, based on the applicant's response, RAI 15.0-25, Item (B.2), is resolved.

15A.3.12 Control Rod Withdrawal Error during Startup with Failure of Control Rod Block

The applicant postulated that, during reactor startup, a single control rod is inadvertently withdrawn continuously because of a procedural error by the operator during manual rod withdrawal or a gang of control rods is inadvertently withdrawn because of a malfunction in the automated rod movement control system (ganged rod operation) of the plant automation system when in the automatic startup mode.

GEH estimates that the frequency of an automatic control rod withdrawal is about 1.20×10^{-6}/yr, and the frequency of manual rod withdrawal is about 1.5×10^{-7}/yr. With the consideration of uncertainty, these values are less than 1.0×10^{-2}/yr.

In response to RAI 15.0-25, Item (B.2), GEH assessed the mechanical failure of the FMCRD and concluded that the mechanical failure of the FMCRD is negligible compared to the estimated overall fail frequency. The staff reviewed the applicant's calculations and found them to be within acceptable ranges and agrees with the RAI responses and the failure assessment of this event. Based on the estimated failure frequency of control RWE during startup, the staff agrees that this event frequency meets the criterion of less than 1.0×10^{-2}/yr. Therefore, based on the applicant's response, RAI 15.0-25, Item (B.2), is resolved.

15A.3.13 Control Rod Withdrawal Error during Power Operation

The causes of a potential RWE at power are either a procedural error by the operator, in which a single control rod or a gang of control rods is withdrawn continuously, or a malfunction of the automated rod withdrawal sequence control logic during automated operation, in which a gang of control rods is withdrawn continuously.

GEH estimated that the frequency of an automatic control rod withdrawal is about 1.20×10^{-9}/yr, and the frequency of manual rod withdrawal is about 2.5×10^{-5}/yr. With the consideration of uncertainty, these values are less than 1.0×10^{-2}/yr.

In response to RAI 15.0-25, Item (B.2), GEH assessed the mechanical failure of the FMCRD and concluded that the mechanical failure of the FMCRD is negligible compared to the estimated overall fail frequency. The staff reviewed the applicant's calculations and found them to be within acceptable ranges and agrees with the applicant's RAI responses and the failure assessment of this event. Based on the estimated failure frequency of control RWE during power operation, the staff agrees that this event frequency meets the criterion of less than 1.0×10^{-2}/yr. Based on the applicant's response, Therefore, based on the applicant's response, RAI 15.0-25, Item (B.2), is resolved.

In the preceding paragraphs under RAI 15.0-25, staff has identified and resolved all individual items.

15A.3.14 Fuel Assembly Loading Error, Mislocated Bundle

The loading of a fuel bundle in an improper location with subsequent operation of the core requires three separate and independent errors:

(1) A bundle must be placed in a wrong location in the core.

(2) The bundle that was supposed to be loaded where the mislocation occurred must also put in an incorrect location or discharged.

(3) The misplaced bundles are overlooked during the core verification process performed following core loading.

Based on the industry survey data, GEH estimated that the mislocated bundle frequency is 9.6×10^{-4}/yr.

The staff reviewed the applicant's calculations and found them to be within acceptable ranges. The staff has reviewed the failure assessment of not detecting the mislocated bundle and agrees with the estimated failure frequency. Based on the low probability of not detecting a mislocated bundle and the estimate of the frequency of a mislocated bundle, the staff agrees that this event frequency meets the criterion of less than 1.0×10^{-2}/yr.

15A.3.15 Fuel Assembly Loading Error, Misoriented Bundle

Proper orientation of fuel assemblies in the reactor core is readily verified by visual observation and ensured by verification procedures during core loading. Based on the industry survey data, GEH estimated that the misoriented bundle frequency is 2.4×10^{-3}/yr.

The staff reviewed the applicant's calculations and found them to be within acceptable ranges. The staff has reviewed the failure assessment of not detecting the misoriented bundle and agrees with the estimated failure frequency. Based on the low probability of a misoriented bundle not being detected and the estimate of the frequency of a misoriented bundle, the staff agrees that this event frequency meets the criterion of less than 1.0×10^{-2}/yr.

15A.3.16 Liquid-Containing Tank Failure

Based on the industry survey data, GEH estimated that the frequency of this event is 3.3×10^{-4}/yr. The staff agrees with the assessment that this is a low probability event. Based on the low probability of this event, the staff agrees that this event frequency meets the criterion of less than 1.0×10^{-2}/yr.

15A.4 <u>Conclusion</u>

Based on the above discussions, the staff agrees that each of the events reviewed has a frequency less than 0.01/yr, including consideration of uncertainty.

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10. SUPPLEMENTARY NOTES

11. ABSTRACT (200 words or less)
This final safety evaluation report documents the technical review of General Electric-Hitachi's (GEH's) Economic Simplified Boiling-Water Reactor (ESBWR) design certification. GEH submitted its application for the ESBWR design on August 24, 2005, in accordance with Subpart B, "Standard Design Certifications," of 10 CFR Part 52. The NRC formally docketed the application for design certification (Docket No. 52-010) on December 1, 2005. The ESBWR design is a boiling-water reactor (BWR) rated up to 4,500 megawatts thermal (MWt) and has a rated gross electrical power output of 1,594 megawatts electric (MWe). The ESBWR is a direct-cycle, natural circulation BWR that relies on passive systems to perform safety functions credited in the design basis for 72 hours following an initiating event. After 72 hours, non-safety systems, either passive or active, replenish the passive systems in order to keep them operating or perform post-accident recovery functions directly. The ESBWR design also uses nonsafety-related active systems to provide defense-in-depth capabilities for key safety functions provided by passive systems. The ESBWR standard design includes a reactor building that surrounds the containment, as well as buildings dedicated exclusively or primarily to housing related systems and equipment. On the basis of its evaluation and independent analyses, as set forth in this report, the NRC staff concludes that GEH's application for design certification meets the requirements of 10 CFR Part 52, Subpart B, that are applicable and technically relevant to the ESBWR design. Appendix F includes a copy of the report by the Advisory Committee on Reactor Safeguards, as required by 10 CFR 52.53.

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