9. AUXILIARY SYSTEMS

9.1 Fuel Storage and Handling

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 [SRP] for the Review of Safety Analysis Reports for Nuclear Power Plants," SRP Section 9.1.1, Revision 3, "Criticality Safety of Fresh and Spent Fuel Storage and Handling," issued March 2007. The acceptance criteria for the criticality safety of fresh and spent fuel storage and handling are based on compliance with the following requirements:

- General Design Criterion (GDC) 62, 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 (ANS) 57.1, ANSI/ANS 57.2, and ANSI/ANS 57.3, 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 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, Section 9.1 describes the fuel storage and handling design bases of the economic simplified boiling water reactor (ESBWR). 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, Section 9.1 references topical report NEDC-33374P, Revision 3, to document the analyses of storage rack criticality. NEDC-33374P, Revision 3 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.

New Fuel Storage

DCD Tier 2, 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 reactor building 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 assure that the fully loaded array is subcritical by at least 5 percent Δk. Monte Carlo techniques are employed in the calculations performed to assure that the effective multiplication factor (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 following normal and abnormal new fuel storage conditions were analyzed by the applicant:

- Normal positioning in the new fuel array; and
- Eccentric positioning in the new fuel array.

Spent Fuel Storage

DCD Tier 2, 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 Fuel Building provide for the storage of 3504 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 ten calendar years of plant operation, plus one full core offload. The racks are comprised 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 design criteria are met, the following normal and abnormal spent fuel storage conditions were analyzed:

- Normal positioning in the spent fuel array; and
- Eccentric positioning in the spent fuel array.

The effects of pool moderator temperature on criticality were also evaluated.

To control SFP reactivity, borated stainless steel storage racks are used as part of a strategy to maintain a 5 percent Δk margin to subcriticality 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 whether the design complied with the requirements of GDC 62. The applicant committed to meet the guidance of the RGs 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, Section 9.1, references Topical Report NEDC-33374P, Revision 3, "Safety Analysis Report for Fuel Storage Racks Criticality Analysis for ESBWR Plants," 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 detailed staff review is provided in the safety evaluation (SE) for NEDC-33374P, "Safety Analysis Report for Fuel Storage Racks Criticality Analysis for ESBWR Plants." The staff review included assessment of the criticality analysis methodology proposed by the applicant, 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 appropriate fuel assembly and storage rack data were used in the analyses, 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, D.C. 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. Detailed design calculations and computer program documentation were also reviewed. The staff determined that the detailed design data used in the calculations was appropriately documented and utilized. The staff found in the SE for NEDC-33374P that the methodology employed by the applicant is consistent with that approved for operating boiling water reactor (BWR) new and spent fuel storage criticality evaluations.

During the course of the DCD review, the staff determined that the DCD contained no Tier 1 requirement to maintain subcriticality in the new and SFPs. Additionally, there was no requirement for verification 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 its response, the applicant provided DCD markups which added the Tier 1 new and spent fuel rack subcriticality design requirement and provided Table 2.5.6-1 inspection, test, analysis and acceptance criteria (ITAAC) Items and Acceptance Criteria. As requested by the staff in the RAI, Topical Report 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 determined that the response was acceptable since the ITAAC are based on the essential parameters for criticality safety identified in Appendix A of NEDC-33374P. Accordingly, based on the above and the applicant's response, RAI 14.3-457 is resolved. The staff confirmed that the changes were incorporated in DCD Revision 7.

The scope of the criticality safety analyses presented in topical report NEDC-33374P is limited to analysis of fuel storage racks in the fuel building and in the buffer pool in the reactor building, and no analysis is provided in the topical report for fuel handling of fresh and spent fuel. Section 9.1.6 of DCD Tier 2, Revision 7, includes combined license (COL) item 9.1-4-A, which would require 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 and COL item 9.1-4-A includes all applicable criticality safety issues for new and spent fuel storage and handling. Therefore, the staff finds COL Item 9.1-4-A acceptable.

In the safety evaluation report (SER) for NEDC-33374P, the staff finds that the applicant has 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 comprised of borated stainless steel. Therefore, the staff finds 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 whether the design complied with the requirements of 10 CFR 50.68. The applicant has addressed the requirements of 10 CFR 50.68 by compliance with the additional design and analysis requirements specified in 10 CFR 50.68 (b)(2) or (b)(4). No credit is taken for soluble boron, so the k-effective 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 topical report NEDC-33374P. As discussed above, COL 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 GDC 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 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 NUREG 0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants,” SRP Section 9.1.1, Revision 3, “Criticality Safety of Fresh and Spent Fuel Storage and Handling,” and SRP Section 9.1.2, Revision 3, “Spent Fuel Storage,” issued July 1981. The staff performed a comparison of the SRP version used during the review (i.e., 1981) with the 2007 version of the SRP, SRP 9.1.2 Revision 4, “New and Spent Fuel Storage.” The following are the major review areas included in the 2007 version of the SRP, but not in the 1981 version: (a) the new fuel vault, (b) new fuel storage racks, (c) new fuel criticality monitoring requirements, (d) as low as reasonably achievable (ALARA) considerations, (e) thermal-hydraulic considerations, (f) how the design would preclude load drops on new and spent fuel, (g) radiological shielding of personnel by maintaining adequate water levels in the SFP and buffer pool, (h) the ability to maintain adequate coolant inventory in the SFP and Buffer Pool under accident conditions, (i) avoidance of high density storage racks for hot fuel, (j), methods of preventing pool draining, (k) ability to place a fuel assembly around the periphery of the SFP or the buffer pool, (l) increased minimum amount of fuel that can be stored, and (m) use of appropriate monitoring systems to detect the following: 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. The discussion of new fuel storage was moved to Section 9.1.2 in the 2007 version of the SRP from Section 9.1.1.

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

Reference to GDC 62, Prevention of Criticality in Fuel Storage and Handling, was deleted from Section 9.1.2 of the 2007 SRP because evaluation of criticality with respect to fuel storage was moved entirely 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 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

New Fuel Storage

In ESBWR DCD Tier 2, Section 9.1.1, the applicant provided the design bases, a description, and an 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 the fuel bundles 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 14x2 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, no cooling is needed for the new fuel racks, and no thermal-hydraulic analysis is 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.

Spent fuel Storage

The fuel storage racks provided in the SFP in the FB provide for storage of 3504 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 are comprised 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 bundle decay heat is removed by fuel and auxiliary pools cooling system (FAPCS) recirculation flow to maintain the SFP temperature below 48.9ºC (120ºF) during normal conditions (defined as ten 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 60ºC (140ºF).

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. A dynamic loads analysis was performed by the applicant to determine the capability of the spent fuel storage racks to withstand the combined loads of (1) the dead weight plus buoyancy load, (2) fuel handling loads, (3) the thermal effect, (4) the safe shutdown earthquake (SSE), (5) the safety relief valve discharge (SRVD) load, and (6) the 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 Part 20, "Standards for Protection Against Radiation," of Title 10 of the *Code of Federal Regulations* (10 CFR Part 20), 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 safe-shutdown earthquake). 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 5 percent Δk margin to subcriticality 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.

<u>Thermal-Hydraulic Design</u>

DCD, Tier 2, Section 9.1.2.5, Thermal-Hydraulic Design, provides a discussion of 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°C (250°F). Rack cooling in the spent fuel and buffer pools is provided by FAPCS recirculation flow to maintain the rack exit temperature below 48.9ºC (120ºF) during normal conditions and 60ºC

(140ºF) during a full core offload. The fuel storage racks are designed such that nucleate boiling is prevented in the event of loss of both FAPCS cooling trains.

<u>Storage Rack Cooling Analyses</u>

Section 5 of topical report NEDO-33373 summarizes the 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 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 where 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 FAPCS.

The decay heat generated by the fuel elements accumulated during ten years of operation and the decay heat resulting from a full core offload are determined in a separate referenced calculation, and 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 eight 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 has been applied to the fully-loaded racks within the SFP. To bound potential loading configurations, the applicant has 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 between all 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 input to the CFD model, and the code calculates the velocity distribution in the SFP and through the racks.

The effects of modeling assumptions and methods (turbulence model selection, buoyancy treatment, or mesh density) have been evaluated by sensitivity studies. Variation of input parameters, such as inlet mass flow rate, inlet temperature, loss coefficient, turbulence model (k-ε model vs. SST-k-ω model), reference temperature for buoyancy model, and inlet turbulence intensity was included in the sensitivity studies.

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, which prevents bulk boiling in the racks and nucleate boiling on the fuel assemblies.

Although not required by regulations, the applicant provided additional analyses assuming 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 48.9ºC (120ºF) during normal conditions and 60ºC (140ºF) during a full core offload.

Topical Report 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, "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 American Nuclear Society (ANS) 57.1-1980, "Design Requirements for Light Water Reactor Fuel Handling Systems," and ANS 57.3-1981, "Design Requirements for New LWR Fuel Storage Facilities," as they relate to preventing criticality and to 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, ANS 57.1 or 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 its response, the applicant provided a markup of DCD Tier 2, Section 9.1.1 which addresses the required GDC. The staff reviewed the RAI response and DCD markup and determined it was 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 SER with open items. The staff confirmed that the above changes were incorporated into DCD Tier 2, Revision 6 and the confirmatory item is closed.

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, “Spent Fuel Storage Facility Design Basis”; the applicable portions of RG 1.29, Revision 4, “Seismic Design Classification,” 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, “Design Objectives for Light Water Reactor Spent Fuel Storage Facilities at Nuclear Power Stations.” 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 Revision 1 of RG 1.115, “Protection Against Low-Trajectory Turbine Missiles”; Revision 1 of RG 1.117; and the appropriate paragraphs of ANSI/ANS 57.2. The ESBWR design is a single-unit station, and the requirements of GDC 5 are not applicable 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 Subsection III.1 of SRP Section 9.1.2. 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, “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 ANS 57.2, and appropriate 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 ANS 57.3. Compliance with 10 CFR 20.1101(b) is discussed in Chapter 12 of this report. 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 storage in accordance with the SRP. RAI 9.1-45 was being tracked as an open item in the SER with open items. In its response the applicant identified modifications to DCD Tier 2, Section 9.1.2 to address compliance with the GDCs. The staff reviewed the RAI response and DCD markup and determined it was acceptable since Section 9.1.2.1, “Design Bases,” was revised to address GDC 2, GDC 4, GDC 61, GDC 62 and GDC 63. In addition, each GDC was described in detail with the applicable guidance of RGs and other standards such as 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 and 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 have to be reviewed by the NRC staff 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 is 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 its response, the applicant stated that it would submit two licensing topical reports (LTRs) to provide fuel rack analyses, NEDO-33373, “Dynamic, Load-Drop, and Thermal-Hydraulic Analyses for ESBWR Fuel Racks,” and NEDO-33374, “Criticality Analysis for ESBWR Fuel Racks,” and therefore the COL holder items are no longer required. The staff reviewed the

RAI response and determined it was acceptable since the topical reports include the topics in the COL holder items and including the topical reports 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 it was unable to complete its evaluation of GDC 2, 61, and 62 prior to the submission of the fuel rack analyses. Accordingly, the review of NEDO-33373 and NEDO-33374 was being tracked as an open item in the SER with open item.

The staff SERs on NEDO-33373 and NEDO-33374 documents 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 NEDO-33374, 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 NEDO-33374, this open item is resolved.

GDC 2

The staff verified whether 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 its 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 for the pool concrete structure (except load factors for all cases are equal to 1.0 and the acceptance criteria follow ASME Section III, Division 2, CC-3700.) The staff reviewed the RAI response and DCD markup and determined it was 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 these criteria were included in DCD Tier 2, Revision 6. 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 7, Section 9.1.2.4, "Mechanical and Structural Design," 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, "Rules for Construction of Nuclear Power Plant Components," Subsection NF, "Supports." Additional discussion of the stress analysis and its results are documented in NEDO-33373, which is evaluated by the staff in the SER for NEDO-33373.

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 its 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 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 reviewed the RAI response and DCD markup of Table 9.1-4, "Classification of Equipment," and determined it was 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 its 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 it would revise Table 3.2-1 and Table 9.1-4 to identify that the new fuel inspection stand shall be seismic Category II. The staff finds this clarification acceptable. The modifications were made in DCD Tier 2, Revision 5. Subsequently, Table 9.1-4 was removed from the DCD rev 6 and seismic classification is included in 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.

GDC 4

The staff verified whether 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 7, Section 9.1.5.5, "Fuel Building and Reactor Building Cranes," 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 as well as when the plant is on line. 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 (DW) 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, Table 9.1-7, "Summary of Heavy Load Operations." 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

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.

crane is also used during refueling/servicing as well as when the plant is on line. 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. Light load handling is discussed in Section 9.1.4.3 of this report, and heavy load handling is discussed in Section 9.1.5.3 of this report.

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 that include fuel assemblies and light loads like control rods, burnable poison rods, and flow-limiting orifices that weigh no more than a fuel assembly) would be mitigated. In its response to RAI 9.1-15, the applicant stated that the amount of leakage through the liner in the event of a load-handling accident is limited by designing the pool to withstand dropping of the load 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, "Rules for Construction of Nuclear Power Plant Components," Division 2, "Code for Concrete Reactor Vessels and Containments," 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 its response to RAI 9.1-15 S01, the applicant stated that "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 provide the basis for the equation used to calculate the liner thickness, describe the material properties assumed for the liner, describe the type of impact model assumed, describe how the liner is assumed to fail, and describe how operational experience was considered during the evaluation.

In its response to RAI 9.1-15 S02, 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 (a) provide a description of the alternative analysis, (b) explain how the results of the alternative analysis compare to the original analysis, and (c) describe the structural response of the liner plate due to impact of a dropped fuel assembly. The staff noted that the evaluation in the response to RAI 9.1-15 S03 relied upon reinforcing the liner plate in the region of the leak chase channels in areas that are not covered by spent fuel racks by welding 1.34 inch (34 mm) thick cover plates. Since reinforcement of the liner plate is a special design feature relied upon 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 provide justification why a DCD revision is not considered 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* 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 applicant was asked to include this design requirement in the DCD or to provide justification why a DCD revision is not considered necessary.

In its response to RAI 9.1-15 S04, the applicant included in DCD Tier Section 3.8.4.2.5 a Tier 2* description of the reinforcing liner plate. The staff determined that this description was 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 there will be areas at the bottom of the buffer pool that is 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., 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 its response, the applicant proposed a revision 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 impacts from dropped objects such as a fuel assembly. The staff reviewed 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, "Welding of Pool Liners," and determined they were 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 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 the ESBWR new and spent fuel storage design meets the requirements of GDC 4.

GDC 61

The staff verified whether 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, "Domestic Licensing of Production and Utilization Facilities." In its 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 (BSS), which conforms to ASTM A887, "Standard Specification for Borated Stainless Steel Plate, Sheet, and Strip for Nuclear Application" (UNS Designation S30467, Grade B, 1.75–2.25-percent boron inclusion). There is no welding of the BSS. Fuel rack feet are fabricated from Type 630 (17 to 4 PH) 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 reviewed the RAI response and DCD markup of Section 9.1.2.6, "Material Considerations," and determined they were acceptable since all the materials used in the fabrication process for each type of rack was specified; that is, 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 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 BSS material used in construction of the interlocking panels. This BSS material is UNS S30467, in accordance with ASTM A887, "Standard Specification for Borated Stainless Steel Plate, Sheet, and Strip for Nuclear Application."

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 BSS and/or 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 its response, the applicant explained that there are no plans to use composite materials such as Boral or Metamic as a nuclear absorbing material in the spent fuel. The applicant stated that

BSS is the composite material used as neutron absorbing material in the spent fuel. The applicant also provided the chemical composition of the BSS 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 BSS 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 experiencing the same environment as the spent fuel storage racks. Based on industry experience of operating plants using BSS as neutron absorbing material, recording of surveillance data occurs after completion of the first cycle following installation of the racks and no less 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 with those coupons that have been exposed to the SFP water environment. The staff determined the RAI responses were acceptable since the surveillance test coupons are fabricated from the same BSS material used in the construction of the interlocking panels of the spent fuel storage racks and because 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 and thus ensure 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 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 its response, the applicant addressed the scenario it understood the staff wanted considered where the level of the SFP was reduced by water spilling into two adjacent empty pools. The applicant stated it had determined 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 ten feet. 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 top of active fuel at 72 hours. In its 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 ten feet above top of active fuel. In addition, the applicant stated that assuming 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 2.2 inches of water above top of active fuel at 72 hours. The staff reviewed the RAI responses and determined they are acceptable since even postulating a gross failure of the transfer gate, an adequate water level margin remains above the top of active fuel (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 "Response 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. The staff evaluation of the revised water level and the applicant's "Response 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 in Section 9.1.3.3, Subsection "Audit of the ESBWR Spent Fuel Pool Required Water Inventory," of this report.

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 five years of operation at full power plus one full-core discharge. DCD Tier 1, Section 9.1, Revision 7 states that the SFP (physical structure and cooling capacity) are designed to store fuel from 20 years of operation plus one full-core offload; 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 the ability of the SFP and the buffer pool to accommodate the required storage capacity was verified and to specify how the design accounted for the reduced cooling effectiveness for high-density racks when compared to low-density racks.

In its 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.

Due to a lack of specific design information, the staff determined this 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 in order 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 staff evaluation of the thermal-hydraulic analyses of the spent fuel racks is in the SER for NEDO-33373 and is briefly summarized below. The SER 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 SER 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 appropriate fuel assembly and storage rack data were used in the NEDO-33373 analyses, 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, D.C. 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. Detailed design calculations and computer program documentation were also reviewed. The staff determined that the detailed design data was 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, used 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 thermal-hydraulic analyses performed by the applicant. In RAI 9.1-124, the staff requested the applicant to provide the SFP dimensions and the corresponding fuel pool model components and to clarify the assumptions made in laying out the fuel pool model. The information was requested to clarify the applicant's model and to support the potential NRC staff confirmatory CFD model. In its response, the applicant provided the necessary information to produce a confirmatory CFD model, if needed. The response to RAI 9.1-124 also provided clarification regarding rack assumptions and loss coefficients. The staff determined that the applicant's response was 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 what sensitivity studies it has performed to support its CFD modeling assumptions. In its response, the applicant described a series of related sensitivity studies of the CFD model. A comparable description of sensitivity studies was included in NEDO-33373. The specific mesh density studies cited are for an unspecified model of a BWR SFP and are only considered to be qualitative. The staff determined that the response was 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 its response, the applicant explained that these data are calculated and the mass flow refers to a single bundle. The applicant also explains that the pressure drop was bounding since it was based on fuel for existing reactors rather than the shorter ESBWR fuel. The staff determined that the response was acceptable since the response clarified the information in NEDO-33373, Figure 5.2 and 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 its 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 determined that the responses was 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 SER 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 topical report 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 have been satisfied.

The guidelines in SRP Section 9.1.2, Revision 4 specify that the staff should verify whether 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 its 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 determined that the RAI response and DCD markup of Section 9.1.2.4, "Mechanical and Structural Design," were 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 the above changes were incorporated into DCD Tier 2, Revision 5 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 designing 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 crane uplift forces from a stuck fuel assembly were considered in the rack design for the SFP. In its response, the applicant noted that the load combinations listed in Section 9.1.2.4 of the DCD refer to the dynamic analysis. The uplift force analysis is a separate calculation and 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 3.8.4. In RAI 3.8-69 S01, the staff identified that while the DCD was revised to reference SRP 3.8.4, Appendix D, the loading combinations specified in DCD Section 9.1.2.4 were not in agreement with those in SRP 3.8.4, Appendix D. In its response, the applicant revised DCD Tier 2, Section 9.1.2.4 to include loads and load combinations consistent with SRP 3.8.4 Appendix D. This includes the stuck fuel load-upward force on the racks caused by a postulated stuck fuel assembly. The staff determined that the applicant's response was acceptable since it included loads from SRP 3.8.4, Appendix D in DCD Tier 2 Section 9.1.2.4, and made these loads Tier 2*. The staff also verified that the stuck fuel load is considered in the dynamic analyses in NEDO-33373 Revision 4. Accordingly, based on the above, the applicant's responses, and DCD changes, RAI 9.1-5 and RAI 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 fuel storage racks capability of withstanding all design loads.

The guidelines in SRP Section 9.1.2, Revision 4 specify that the staff should verify whether 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/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, inclined fuel transfer system, and equipment storage pool, as well as any interfaces or pits in the pools. In its 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 its response to RAI 9.1-46 S01. The staff determined the 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 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 10 feet (3.05 meters) above the TAF. In addition, the RAI asked the applicant to provide a COL information item that instructs the 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 pool water level inadvertently to less than 10 feet (3.05 meters) above the active fuel. In its 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 10 feet (3.05 m) 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 there are other kinds of "gates" that could be exempt from this definition. The applicant revised the DCD to refer to them simply as "gates" to avoid confusion. The staff determined that the RAI response and DCD markup were acceptable since the applicant adequately addressed the bottom of the gate with respect to the TAF by adding a Tier 1 ITAAC requiring 3.05 meters (10 feet), which provides adequate shielding and cooling. In addition, a clarification was provided for the 'gates'. Based on the applicant's response and incorporation of the DCD markup in Revision 6 of the DCD, RAI 9.115 S01 is resolved. The staff notes that in the applicant's "Response 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, seismic category of the gates, and the applicant's "Response 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 in Section 9.1.3.3, Subsection "Audit of the ESBWR Spent Fuel Pool Required Water Inventory," of this report.

However, in reviewing DCD Revision 6, the staff identified that the anti-siphon devices should have been included in the applicant's response. Accordingly, in RAI 9.1-130, the staff asked the applicant to (a) revise the DCD, Section 9.1.2.4, to clarify that the anti-siphon holes preserve the water inventory such that it would be at least 10 feet above top of active fuel in the event of a break in the line at a lower elevation, (b) revise the 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 10 feet (3.05 meters) above TAF in the event of a break in the line at a lower elevation, and (c) 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 10 feet above the TAF in the event of a break in the line at a lower elevation. In its response, the applicant agreed to revise the DCD in Revision 7, as requested. The staff determined the RAI response and DCD markup of Tier 1 and Tier 2 are acceptable since all three items were adequately addressed. Item 14 of Tier 1, Section 2.6.2 describes the redundant anti-siphon holes that preserve water inventory 3.05 meters (10 feet) above the TAF for safe shielding. In addition, 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 these DCD changes were incorporated into DCD Revision 7. 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 "Response 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 "Response 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 in Section 9.1.3.3, Subsection "Audit of the ESBWR Spent Fuel Pool Required Water Inventory," of this report.

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

- The SFP and buffer pool are reinforced concrete structures with a stainless steel liner.
- 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 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 its response to RAI 14.3-442, the applicant addressed each of the items:

Regarding SFP and buffer pool materials, the applicant stated that an ITAAC will be added 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 determined that the applicant's RAI response for this item and DCD changes were acceptable since the DCD Tier 1 changes adequately addresses the materials of the SFP.

Regarding SFP and buffer pool liner embedments, the applicant stated that an ITAAC will be added to 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 determined that the applicant's RAI response for this item and DCD changes were 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 gates that connect the SFP to adjacent pools are designed so that the bottom of the gate is at least 10 feet (3.05 m) above the TAF. In addition, the applicant stated that since the buffer pool is a deep pit with 9.5 m of water, an ITAAC is not needed for the buffer pool gates. The staff determined that the applicant's RAI response for this item and DCD changes were acceptable since the Tier 1 changes adequately address the SFP bottom gate location with respect to TAF. The staff also determined 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 "Response 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, seismic category of the gates, and the applicant's "Response 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 in Section 9.1.3.3, Subsection "Audit of the ESBWR Spent Fuel Pool Required Water Inventory," of this report.

Regarding the lines to fill and drain the SFP and buffer pool, the applicant stated that an ITAAC would be added to Tier 1 Table 2.6.2-2 in Revision 6 to the DCD that describes the design feature that lines to fill and drain the SFP and buffer pool enter the pools above the safe shielding water level. However, the revision was not made to Revision 6 of the DCD. In the revised response to RAI 14.3-442, the applicant revised the ITAAC satisfactorily. The staff determined that the applicant's RAI response for this item and DCD changes were acceptable since the 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 an ITAAC will be added to DCD Tier 1 Table 2.6.2-2 in Revision 6 to 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 10 feet (3.05 meters) 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 meters (10 feet) above the TAF for safe shielding. The staff determined that the applicant's response to RAI 14.3-442, as modified by the response to RAI 9.1-130, was acceptable since the ITAAC verify that the anti-siphon vacuum breakers preserve a safe water level. The staff notes that in the applicant's "Response 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 "Response 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 in Section 9.1.3.3, Subsection "Audit of the ESBWR Spent Fuel Pool Required Water Inventory," of this report.

Regarding individual spent fuel racks being spaced less than one fuel assembly apart so that a fuel assembly cannot be inserted between racks, the applicant responded that Topical Report NEDC-33374P, "Safety Analysis Report for Fuel Storage Racks Criticality Analysis for ESBWR Plants" confirms that the gaps between racks are small enough that they cannot accommodate a spent fuel bundle. In addition, in DCD Tier, Revision 7, Table 2.5.6-1, ITAAC were added to confirm that the fuel rack spacing dimensions are within the tolerance used in the fuel storage criticality analyses. The staff determined that the applicant's response to this item was acceptable since the rack spacing assumed in NEDC-33374P is less than one fuel bundle and the DCD Tier 1 adequately 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, Subsection 3.8.4, which describes the design features of the reactor building and fuel building 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 the DCD changes were incorporated into DCD Revisions 6 and 7 as applicable.

Based on the above, the staff finds that the new and spent fuel storage design meets the requirements of GDC 61. Additional discussion related to GDC 61 and the SFP is further discussed in Section 9.1.3.3 is this report.

GDC 63

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

10 CFR 20.1406

The staff evaluated whether FAPCS is designed in accordance with 10 CFR 20.1406 in section 9.1.3 of this report.

DCD Section 3.8.4.2.5, "welding of pool liners," identifies that after construction is finished, each isolated pool is 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. The staff's evaluation of the 10 CFR 20.1406 program as it relates to FAPCS and the pools used for the storage of spent fuel is discussed in Section 12.3 of this report.

10 CFR 20.1101(b)

The staff verified whether 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. 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, "Plant Shielding Description – Fuel Storage", describes the fuel storage pool as being designed to ensure that the dose rate around the pool area is less than 25 μSv/hr (2.5 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 do 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 fuel and auxiliary pools cooling system operates continuously to reduce the radioactive contamination in the pool water for all the major pools in the ESBWR. In order 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. 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 and the staff therefore finds them to be acceptable. The ALARA program is further addressed in Section 9.1.3 related to FAPCS, and Section 12.3 of this report.

10 CFR 50.68

The staff verified whether the design complied with the requirements of 10 CFR 50.68. 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 has addressed the requirements of 10 CFR 50.68 by compliance with the additional design and analysis requirements specified in 10 CFR 50.68(b)(2) or (b)(4). No credit is taken for soluble boron, so the k-effective 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 topical report NEDC-33374P, which is evaluated in the SER 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 concludes that the ESBWR design conforms to GDC 2, 4, and 61. GDC 63 is discussed in SER Section 9.1.3. Because the ESBWR design is a single unit, GDC 5 is not applicable. Based on the discussion above, the staff concludes that the ESBWR design conforms to 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, “Spent Fuel Pool Cooling and Cleanup System,” 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: (a) 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, (b) 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 (c) clarification of the seismic specifications for the SFPCS makeup system and its backup. Although these items were not included in the SRP version used by the staff for its review the staff subsequently did address them in its review.

Discussion of 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 is provided in SER Section 9.1.3.3. Discussion of the minimum operational heat removal capacity of the SFPCS is provided in SER Section 9.1.3.3. Discussion of the clarification of the seismic specifications for the SFPCS makeup system and its backup is provided in SER Section 9.1.3.3.

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), "Radiation Protection Programs," 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 and SECY 95-132, as they relate to non-safety-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 non-safety systems (RTNSS):

  – non-safety 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 non-safety 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 (IC)/passive containment cooling system (PCC) pool for long-term cooling in accordance with SRP Sections 5.4.7, Revision 3, "Residual Heat Removal System," issued

April 1984; 6.2.2, Revision 4, “Containment Heat Removal System,” issued October 1985; and 6.3, Revision 2, “Emergency Core Cooling System,” issued April 1984. 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 IC system 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 PCC 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 FAPCS is to cool and clean pools located in the containment, RB, and FB during normal plant operation. FAPCS provides flow paths for filling and makeup of these pools during normal plant operation and during post-accident conditions, as necessary. 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 C (120 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 C (140 F). All operating modes are manually initiated and controlled from the main control room (MCR), except the SPC mode, which is initiated either automatically on 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 non-safety-related system with the exception of piping and components required for containment isolation, the interface with safety-related reactor water cleanup (RWCU)/SDC piping, and piping and components providing the flow path for post-accident refilling of the IC/PCC 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 and/or accident recovery function have Quality Group B or C[2] and seismic Category I or II classification. Provisions are taken to prevent inadvertent draining of the pools.

There are FAPCS components located outside the RB that support FAPCS makeup to the IC/PCC 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 Fire Protection System (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. This vessel injection function is credited in the ESBWR probabilistic risk assessment (PRA).

9.1.3.3 Staff Evaluation

Compliance with the requirements of GDC 2 may be based in part on adherence to the guidance of Regulatory Positions C.1, "Seismic Design," C.2, "Protection Against Extreme Winds," C.6, "Drainage Prevention," and C.8, "Makeup Water" of RG 1.13, as well as Regulatory Position C.1, "safety-related portions of the system" and Regulatory Position C.2, "non-safety-related portions of the system" of RG 1.29. Compliance with the requirements of GDC 4 may be based on adherence to the guidance of Regulatory Position C.2 of RG 1.13. The ESBWR design is a single-unit station, and 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, "Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations Will Be As Low As Is Reasonably Achievable." Adherence to RG 8.8 has been identified by the applicant as a COL information item in DCD Tier 2, Section 12.1-4-A. This is discussed in Chapter 12 of this SER. The staff finds this acceptable.

GDC 2 and GDC 4

The staff verified whether the design complied with the requirements of GDC 2 and GDC 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 7, Table 3.2-1, "Classification Summary," states that the RB and FB are designed to seismic Category I requirements. This statement is consistent with

2 See Regulatory Guide 1.26, Revision 4, "Quality Group Classifications And Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants"

Regulatory Position C.2 of RG 1.13 and the design criteria specified in SRP Section 3.5.3, Revision 2, “Barrier Design Procedures.” Details of the staff’s review of the seismic design of the RB and FB are found in Section 3.7 of this report. The staff finds the applicant’s declaration that the RB and FB are seismic Category I to be acceptable to meet 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, “Safety Classification, Quality Group, and Seismic Category,” 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 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), “Pool Cooling,” of RG 1.13 to confirm that it is constructed to Quality Group C 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 FAPCS is a non-safety-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 emergency water supplies which are safety related.

Consistent with Regulatory Position C.8 of RG 1.13 and Criterion III.1.f of SRP Section 9.1.3, “Spent Fuel Pool Cooling and Cleanup System,” the staff verified whether 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 that 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, “Fire Protection System,” 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] 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 standard. In its 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. Hence, the applicant assigned the components associated with providing makeup water from the FPS to Quality Group D on this basis. The applicant also stated 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 also stated that the fire pumps are mounted on a seismic Category 1 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

3 See Regulatory Guide 1.26, Revision 4, “Quality Group Classifications And Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants”

classifies the enclosure as seismic Category I, consistent with its response to RAI 9.1-16. In its 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 since the applicant agreed to reclassify the FPS enclosure as seismic Category I and Quality Group D, which is a reliable makeup water source for 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 this classification is identified in DCD Tier 2, Table 1.9-9, "Summary of Differences from SRP Section 9," as a deviation from Criterion II.1.a of SRP Section 9.1.3. The staff finds the 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 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 the SRP acceptance criteria that are expected for pools that rely on active components for providing makeup during this period. The staff confirmed that these modifications were made in DCD Tier 2, Revision 5 and accordingly 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 details of how the safety-related SFP makeup water supplies and water supplies to the IC/PCC pools would be protected from the effects of tornados and other natural phenomena. In its response, the applicant stated that the only safety-related components of the FAPCS that exist outside of the reactor building are the emergency fill-up valves that are attached to the Reactor Building 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 that it conflicted with the response to RAI 9.1-12. In RAI 9.1-16 S01 the staff asked the applicant to clarify its position. In its response, the applicant stated it concurred, and noted that its response to RAI 9.1-12 S01 addressed this inconsistency. The staff noted in RAI 9.1-16 S02 that there were additional apparent inconsistencies in the level of protection afforded FAPCS makeup regarding tornado missiles, and the staff 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 its response, the applicant reiterated that FPS components located outside the RB that 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 the dedicated line from the FPS to the FAPCS is not designed to 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 Reactor Building**.** The staff determined that the response was acceptable, but in RAI 9.1-16 S03 requested the applicant to modify Tier 2 documentation to state this directly. In its response, the applicant proposed a modification to Tier 2 in Revision 6 to the DCD. The staff determined that the RAI response and DCD markup of Tier 2 and are acceptable since the applicant adequately addressed the FPS components (not designed to NFPA standards) and the FPS relationship to FAPCS to support long term makeup to pools in the RB. The staff reviewed and found the Tier 2 modifications in Revision 6 acceptable. 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 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 control building (CB), auxiliary diesel building, RB, and FB. Failure of any of these lines could divert flow from potential refill of the fuel pools or IC/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, in DCD Tier 2, Section 19A.3.1.1 it stated that the 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 fire fighting function and will only be used for refilling of the pools as a RTNSS backup. In RAI 9.1-142, the staff requested the applicant to identify on DCD Tier 1, Figure 2.16.3-1 the dedicated line. In its response, the applicant proposed a modification to DCD Revision 7, Tier 1 and Tier 2 documentation to reflect this level of detail, including modifying DCD Tier 2 Figure 9.5-1 and DCD Tier 1 Figure 2.16.3-1. The staff determined that the RAI response and DCD markups of Tier 1 and Tier 2 are acceptable since the applicant provided sufficient details on Tier 1 Figure 2.16.3-1 and Tier 2 Figure 9.5-1, including showing the dedicated fire protection seismic Category I line to FACPS. This dedicated fire protection (FP) line will not be utilized for water supply to other FP 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 the identified changes were incorporated 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 its response, the applicant stated that RWCU does not support cooling the SFP. Reference to such capabilities was deleted from the DCD. The staff determined the applicant's response was acceptable since the applicant removed references to the RWCU as backup cooling for SFP and related inconsistencies in the DCD on this point. 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 its 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 meters (18.0 feet) 72 hours after loss of pool cooling. NEDO-33373, Section 1.4.1 specifies the height of the spent fuel racks of 3.85 meters (12.63 feet). 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 meters (5.41 feet). Since 1.65 meters is below the safe shielding level of 3.05 meters (10.0 feet), the applicant further specified that plant personnel would not be allowed in close proximity to the SFP during a loss of cooling event, and that pool makeup is achieved from outside the fuel building. The staff determined the response was acceptable since a water level of 1.65 meters (5.41 feet) 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 "Response 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 and the applicant's "Response 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.

The DCD states that dual mode operation of FAPCS is prohibited when only one train of FAPCS is operating. In RAI 9.1-98 the staff asked the applicant to explain how this action was prohibited. In its response, the applicant explained that operation of FAPCS trains will be implemented by operators 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 determined that the RAI response and DCD Tier 2 markup were acceptable since FACPS will be instrumented such that any configuration or alignment can be achieved or precluded as necessary. Also, prohibited modes of operation will be alarmed. The staff confirmed that the DCD markup of Section 9.1.3.2 has been incorporated 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 its response, the applicant indicated that the flow paths are normally isolated and only opened if the FAPCS is in the suppression pool cooling or GDC 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 GDC pools. The staff determined that the RAI response was 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 to the volume of water between the minimum and maximum pool levels due to 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 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 its 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 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 its 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 RWCI/SDC used for LPCI.

The applicant also identified that the RTNSS functions of FAPCS include suppression pool cooling and LPCI which includes the suction line from the suppression pool, all of the piping and components in the cooling and cleaning 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 determined that the RAI response and DCD revision were acceptable since they clarified the portions of FAPCS which are safety-related versus RTNSS. The staff confirmed that the DCD markups were incorporated 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 FAPCS for emergency refill of the IC/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 its 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 Tier 2, Subsection 19A.3.3 in Revision 7 to the DCD to reflect this distinction. The staff determined that the RAI response and DCD markup of Section 19A.3.3 were acceptable since the representative SSCs that meet the RTNSS B criterion are now listed in a consistent manner and ambiguous language concerning safety classification has been removed. Also the piping used for the dedicated FPS makeup water supply to the SFP and IC/PCCS pools was clarified. The staff confirmed that the DCD markups were incorporated 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 the DCD, Revision 6, the staff noted that Tier 2, Section 19A.3.1.2, Containment Integrity, 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 and also RTNSS. In RAI 9.1-136, the staff asked the applicant to clarify this apparent confusion and to revise any other sections of 19A that incorrectly describe RTNSS functions for containment integrity. In its response, the applicant provided markups for DCD Tier 2 Subsection 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 FAPCS to replenish the water boiled off from the SFP and the IC/PCCS pools after 72 hours. The staff determined that the RAI response and DCD revision were 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 DCD markups were incorporated 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, System Description, Detailed System Description, the staff noted a statement in the paragraph referring to, "A reactor makeup water discharge line," which states that "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 its response, the applicant agreed and stated it would correct figures in DCD Tier 1 and Tier 2, Revision 7. The staff determined the RAI response and DCD markup of Tier 1 and Tier 2 figures were acceptable since the relief valves are correctly placed upstream of MOVs F332A/B for Tier 1 Figure 2.6.2-1 and Tier 2 Figure 9.1-1. The staff confirmed that the DCD markups were incorporated 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 Rev. 6, the staff noted that Tier 2, Section 9.1.3.2, Detailed System Description, discusses that there is piping separate from FAPCS pool cooling piping that provides flow paths to refill the IC/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, etc. that will exist or be stored on site such that they can 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 reactor building 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 its response, the applicant stated it will clarify DCD Tier 2, Revision 7, Subsection 9.1.3.2 to indicate the FAPCS makeup water for the SFP and IC/PCCS pools can be supplied from onsite or offsite sources. The staff determined that the RAI response and DCD markup of Section 9.1.3.2 were acceptable since the applicant clarified that the FAPCS makeup water to the SFP and IC/PCCS pools can be supplied from onsite FPS or offsite sources via flanged connection in the yard area. The staff confirmed that the DCD markups were incorporated 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, the staff concludes that the ESBWR design complies with the requirements of GDC 2 and 4.

GDC 61

The staff verified whether the SFP and cooling systems meet the requirements of GDC 61. The staff verified whether 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 motor-operated valves 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 GDC pools and suppression pool and for injecting water to the DW 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 allows 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 under accident conditions (i.e., pool boiling) and how essential equipment is protected against the environmental effects. In its response and in 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 (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 system isolates the FB as described in DCD Tier 2, Revision 3, Section 9.4.2.5, "Instrumentation Requirements." Steam generated by boiling of the SFP water is released to the atmosphere (the ultimate heat sink) from the FB through non-safety related passive relief devices so as to prevent the fuel building 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 non-safety related passive relief devices so as to prevent the RB from exceeding its maximum design pressures. The non-safety 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 determined the applicant's response to RAI 9.1-8 and the design clarifications to DCD Tier 2 Revision 6 were 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 on 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 Chapter 15. The FB devices will open on 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 non-safety 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 DCD markups were incorporated 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 ultimate heat sink.

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 provide a justification for not including an atmosphere cleanup system. RAI 9.1-8 was being tracked as an open item in the SER with open items. In its response to RAI 9.1-8 S01, the applicant stated that design basis accidents (DBAs) associated with the Fuel Building are limited to the Fuel Handling Accident (FHA) and Spent Fuel Cask Drop Accident. Dose consequences for the FHA 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 heating, ventilation and air conditioning (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 23 feet above the top of the active fuel 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, and the time to boil in the SFP following the loss of forced cooling. In its response, the applicant stated that SFP water level during refueling is the same as in normal operation (i.e., approximately 35.6 feet 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 23 feet 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 SPF boiling and SPF water level. The staff notes that in the applicant’s “Response 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 and the applicant’s “Response 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.

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 non-safety related. In the applicant’s supplement to the response a markup of DCD Tier 2, Table 3.2-1, Sections 6.2.3.2, 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 applicant’s response was determined 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 determined that the FB and RB relief devices are properly classified. RAI 9.1.8 is resolved pending confirmatory items to be inserted as part of Revision 8 of the DCD.
(Confirmatory Item 9.1.xxx).

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 its 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 providing a description of controls that will be used to ensure that the required volume of water will be maintained at all times. In its 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 non-safety-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 that demonstrates 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 provide a description of 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 its response to RAI 9.1-9 S02, the applicant provided references to calculations that show that if both trains of 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 meters (10 feet) 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 meters (10 feet) above the TAF. The staff determined that the applicant's response was acceptable since adequate water level will be maintained in the buffer pools 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 "Response 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 and the applicant's "Response 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.

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 °C (140 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 (ASB) 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 C (120°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 C (140 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 its 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 requirements 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 its 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 its 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 cleaning and cooling trains to the DCD in Table 9.1-8, DCD Tier 2, Revision 5. The staff determined that the RAI response was acceptable since the heat load data and balance summary were provided and FAPCS design and performance parameters were added to the DCD. The DCD revision also clarified that one train of FAPCS is capable of removing 9.6 MWt at its design conditions; thus the most limiting heat load conditions in the SFP can be accommodated by FAPCS.

The design capability of FAPCS was later changed 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 determined 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 9.1.3, Revision 2. The staff confirmed that the DCD markups were incorporated into DCD Revision 7. Based on the above, the applicant's responses, and DCD changes, RAI 9.1-10 is resolved.

The staff verified whether design provisions exist to permit appropriate inservice inspection and functional testing of system components. DCD Tier 2, Revision 3, Section 9.1.3.4, "Testing and Inspection Requirements," 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 verified whether design provisions exist to address Regulatory Position C.6 of RG 1.13, such that systems have been designed so 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 meters (10 feet) 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 10 feet (3.05 m) 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 its 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 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 these details in the DCD so 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 its response to RAI 9.1-11 S01, the applicant agreed to make this change. The staff determined that the RAI responses were acceptable since the applicant clarified the emergency makeup header location and that anti-siphon holes are located at the normal water level for all cooling discharge lines and the applicant made 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 above changes were incorporated into DCD Tier 2, Revision 4 and the confirmatory item is closed. The staff notes that in the applicant's "Response 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 "Response 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 its 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 IC/PCC and SFPs. This change ensures that onsite water sources remain available as makeup for the IC/PCC 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 PCC condensers can provide sufficient heat removal capability at and beyond 72 hours to satisfy GDC 34, "Residual heat Removal," and GDC 38, "Containment Heat Removal," requirements 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 being 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 (200 gallons per minute) 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 its 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 determined that the RAI response was acceptable since the minimum makeup water flow rate was determined based on the highest heat load, which occurs at 3-days post-accident, 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 its 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 preoperational testing. The applicant stated that it will update DCD Tier 1, Table 2.16.3-1, "Fire Protection Equipment," to include a requirement for performance of this test, and provided the proposed ITAAC. RAI 9.1-14 was being tracked as an open item in the SER with open items. The staff determined that the RAI response was acceptable since a test will be performed to demonstrate that the diesel-driven fire pump will supply a minimum of 46 m^3/hr ($\geq$200 gpm) flow rate to the IC/PCCS and SFP. This test was added to Table 2.16.3-2, "ITAAC for the Fire Protection System, ITAAC 7a. Accordingly, based on the above, the applicant's responses, and DCD changes, RAI 9.1-14 was resolved.

In RAI 9.1-31, the staff requested that the applicant clarify the discrepancy between its response to RAI 9.1-14 and the statement of DCD Tier 2, Revision 3, Section 9.1.3.2, "System Description," which identifies the maximum heat conditions as resulting from 20 years of operation. In its 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 is unclear since several other RAI responses indicate that there is an approximately 0.7 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 due to margin in the designated FPS flow rate for 10 years of spent fuel, an FPS flow rate of 46 m^3/hr (200 gpm) to the IC/PCCS and SFP is sufficient for 20 years of spent fuel. The staff determined that the RAI 9.1-31 response, when augmented by the RAI 9.1-13 S01 RAI response, was acceptable since the FPS flow is sufficient to cool 20 years of spent fuel. Accordingly, based on the above and the applicant's response, RAI 9.1-31 and RAI 9.1-13 S01, RAI 9.1-31 was 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 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 its 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 C (212°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 set point is exceeded. The F/D units on both trains are flushed to a common backwash receiving tank which is drained to the liquid waste management system. The cleanup system reduces radioactive materials and other contaminants from the SFP, auxiliary pools, suppression pool, and GDC 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) "Advanced Light Water Reactor Utility Requirement Document," Revision 8, Volume III, Section 2.2.3.2. The staff determined that the applicant's response was acceptable because the DCD description of the cleanup system adequately addresses 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 DCD Tier 2, Revision 3, Section 9.1.3, writeup was unclear whether the applicant considered such provisions. The staff requested in RAI 9.1-30 that the applicant provide a description of the provisions. In its 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 the DCD, Tier 2, Section 9.1, "Fuel Storage and Handling," and Chapter 19 ACM "Availability Controls Manual." A summary of the audit, including participants and audit activities may be found in the ADAMS at Accession Number ML101680660. Prior to 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 too) RAIs 9.1-10, 9.1-11, 9.1-18, 9.1-44, 9.1-46, 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 identified that changes to AC 3.7.4 and corresponding sections of the DCD would be made 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, "Emergency Makeup Water." 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 to 9 were identified during the June 3, 2010, audit and open items 10 to 11 were identified during the June 15, 2010, audit.

1. Impact of a seismic event on the SFP to maintain SPF 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 AC

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. AC related to the fire protection system and its bases for water make-up

9. Apparent inconsistency was found between the latest thermal analysis results and the AC B 3.7.1, Emergency Make-up (1921 m^3 vs. 1151 m^3).

10. Clarification of any non-seismic Category 1 and 2 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 "Response 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 feet] above TAF for safe shielding) and these anti-siphon holes are no lower than 9.2 m (30.2 ft) above the TSFA in order to provide safe shielding in the event of a break at a lower elevation. The applicant also states 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 provides a DCD Tier 2 Table 3.2-1, Section 9.1.3.2, and Tier 1 Table 2.6.2-2 markup which will be incorporated into Revision 8 of the DCD.

The staff finds the applicant's response to item 1 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 FAPCS event 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 1 requirements. Accordingly, the staff finds items 1, 3 and 10 are resolved pending inclusion of this information in the DCD through Revision 8 of the DCD. **This is Confirmatory Item 9.1.x.**

For items 2, 4 and 8, the applicant stated that DCD Tier 2, Revision, Chapter 19, ACLCO 3.7.4, "Spent Fuel Pool (SFP) Water Level," requires that the SFP water level shall be no less than 8.5 m (27.9 ft) above 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 1760 m^3 (62,200 cubic feet.) Therefore, a TS surveillance has been added to DCD Tier 2, Chapter 16 (replacing the old ACLCO) that specifies a minimum pool level of $\geq$9.2 m (30.2 ft) above the TSFA in the SFP and reactor building buffer pool. The minimum pool level of 9.2 m (30.2 ft) above the fuel assembly's bounds the volume of 1760 m^3 (62,200 cubic feet) credited for boil-off of the SFP. In addition, the applicant provides a DCD Tier 2 markup which will be incorporated into Revision 8 of the DCD.

The staff finds the applicant's response to item 2 acceptable since the boil off analyses was re-performed with a bounding boil-off volume of 1760 m^3 (62,200 cubic feet) 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 $\geq$9.2 m (30.2 ft) and water temperature $\leq$60 °C (140 °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 <48.9 °C (120 °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°C (140°F). Accordingly, the staff finds items 2, 4 and 8 are resolved pending inclusion of this information in the DCD through Revision 8 of the DCD. **This is Confirmatory Item 9.1.x.**

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 has been performed to determine the minimum makeup water flow rate at 72 hours with consideration of 102 percent core power. This calculation shows that the minimum makeup rate is 159 gpm, which is bounded by the DCD value of 200 gpm. In addition, the applicant provides a DCD Tier 2 AC B 3.7.1 markup which will be incorporated into Revision 8 of the DCD.

The staff finds the applicant's response to item 5 acceptable since the calculations included combined decay heat of the fuel in the reactor and the SFP following a shutdown that occurs 36 days into a cycle in which the reactor is run at 102 percent rated power from 72 hours through 7 days. These conditions are bounding in terms of the combined decay heat of the irradiated fuel in the reactor pressure vessel and SFP and the combined evaporation from the IC/PCCS pools and the SFP. Accordingly, the staff finds items 5 and 7 are resolved pending inclusion of this information in the DCD through Revision 8 of the DCD. **This is Confirmatory Item 9.1.x.**

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 provides a DCD Tier 2 markup related to buffer pool volumes and water levels which will be 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 following a shutdown that occurs 36 days into a cycle in which the reactor is run at 102 percent rated power from 72 hours through 7 days. These conditions are bounding in terms of the combined decay heat of the irradiated fuel in the reactor pressure vessel and SFP and the combined evaporation from the IC/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 (10200 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. Accordingly, the staff finds item 5 is resolved pending inclusion of this information in the DCD through Revision 8 of the DCD. **This is Confirmatory Item 9.1.x.**

For item 9, the applicant stated that the inconsistency between the minimum volumes from 72 hours to 7 days of 1921 m^3 (67840 cubic feet) and 1151 m^3 (40650 cubic feet) has been addressed. The value was determined to be unnecessary detail and was removed from AC B.3.7.1 and the applicant provides a DCD Tier 2 markup related to this information being deleted in 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 minumum water volume for both the IC/PCCS pools and the SFP. This new volume is approximately 3900 m^3 (1.03 x 10^6 gallons) for the 72 hours to 7 days duration which is available in the two firewater storage tanks. Accordingly, the staff finds item 9 is resolved pending inclusion of this information in the DCD through Revision 8 of the DCD. **This is Confirmatory Item 9.1.x.**

For Item 11, the applicant stated the proposed TS for SFP water level requires a minimum water level of 9.20 m (30.2 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 9.20 m (30.2 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 9.20 m (30.2 ft.) above 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 9.20 m (30.2 ft.) under the bounding heat load.

The applicant further explained that the TS limit of 9.20 m (30.2 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 9.20 m (30.2 ft.) above TSFA, there would be significant margin after 72 hours. Therefore, the TS limit of 9.20 m (30.2 ft.) is sufficient to meet the guidelines of SRP 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, (10.3 m (33.8 ft) above the TSFA).

- The SFP and buffer pool have no mechanism by which they can be drained below 9.20 m above 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 9.20 m (30.2 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 9.20 m (30.2 ft.) above 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, Subsection 9.1.3.5), thereby conservatively avoiding false readings due to steam vapors above the actual water level.

In summary, the applicant concluded that the proposed TS limit of 9.20 m (30.2 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 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 with a complete core offload. Under such conditions, the staff concluded 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 that "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. Accordingly, the staff finds item 11 is resolved pending inclusion of this information in the DCD through Revision 8 of the DCD. **This is Confirmatory Item 9.1.x.**

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

<u>GDC 63</u>

GDC 63 requires that appropriate systems shall be provided in fuel storage and radioactive waste systems and associated handling areas (1) to detect conditions that may result in loss of residual heat removal capability and excessive radiation levels and (2) to initiate appropriate safety actions. SRP 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 7, Section 9.1.3.5, "Instrumentation and Control," describes system instrumentation which includes water levels, water temperatures, and flow and pressure for FAPCS.

<u>Surge Tank and Pool Water Level</u>:

The normal FAPCS water source is the skimmer surge tanks, which are filled by overflow from the SFP. The skimmer surge tank level is monitored by a level detector and transmitter mounted on a local panel. 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/low-level alarms.

The IC/PCC 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 non-safety-related display and alarms. Both signals are validated and used to control the valve in the makeup water supply line to the IC/PCC 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 its response the applicant stated that the level instruments on the surge tank provide for automatic makeup water from the condensate storage and transfer system 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 requirements of GDC 63 are satisfied by the safety-related SFP level instruments, which will sound an alarm in the MCR upon a low SFP water level. 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 alarms in the MCR on 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 and how the operators respond to MCR alarms; therefore, the staff generated RAI 9.1-18 S01.

In its response to RAI 9.1-18 S01, the applicant stated that the instrumentation are 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 determined this response was unacceptable since the amount of water between the TAF and the SFP low level alarms was not specified. The applicant stated that no operator actions are needed for 72 hours and the low level set point 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 set point.

In its 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 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 meters (~6.5 feet) above TAF) 72 hours after loss of forced cooling (these calculations are discussed further with RAI 9.1-44 above). Additional alarm setpoints for

TAF and shielding (3.05 m (10.0 ft)) were discussed in the RAI response and the alarm setpoints were included in DCD Tier 2, Revision 5.

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 has been 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 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 applicant was asked 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 its response to RAI 9.1-18 S03, the applicant stated that the level instruments in the buffer pool will be upgraded to safety-related and provide a DCD mark-up for Revision 6.

The staff determined that the applicant's response to RAI 9.1-18 S03 and DCD revisions were acceptable since the water-level instruments in the buffer pool were made safety-related. DCD Revision 6 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 determined that the design clarifications were acceptable since the clarification of the alarm locations is being made to be consistent with the guidelines of SRP 9.1.3, Revision 2, which indicate alarms should initiate both locally and in the MCR. The staff confirmed that these design clarifications were incorporated 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, this change was not implemented 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 it would revise Tier 1 including Table 2.6.2-2 as requested for the buffer pool level instruments in Revision 7 to the DCD.

The staff determined that the applicant's response was acceptable since appropriate information on the buffer pool safety-related water level instrumentation was incorporated in DCD Tier 1, Revision 7. Accordingly, based on the above, the applicant's responses, and DCD changes, RAI 9.1-18 and RAI 9.1-131 are resolved. The staff notes that in the applicant's "Response 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 "Response 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 IC/PCC pools provide accurate level indication during boiling conditions.

In its response to RAI 9.1-41 the applicant indicated that the level instruments in the IC/PCC 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 (due to an increase in water temperature) in the SFP will result in 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, how it will be affected by the increase in temperature and the boiling conditions, and 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 its response to RAI 9.1-41 S01, the applicant stated that no specific instrumentation design has been chosen. 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 and 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, "ITAAC for the Fuel and Auxiliary Pools Cooling Cleanup System," to add an ITAAC description of the SFP level instrumentation.

The staff determined that the applicant's response to RAI 9.1-41 S01 was acceptable since the applicant added an instrumentation ITAAC addressing adequate operating ranges for the SFP and IC/PCC pools. In addition, the staff noted that in the revision to the response to RAI 14.3-449 S02, DCD Tier 1 Revision 6, Table 2.6.2-2, Item 9 was modified 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 "Response 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 "Response 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 potentially radioactive leakage from the fuel storage pools and the FAPCS is collected and processed.

In its 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, "Equipment and Floor Drain System."

The staff determined that the applicant's response to RAI 9.1-17 is acceptable since the leakage from the fuel storage pools has been adequately addressed with leakage channels and a collection/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 DCD Tier 2 was revised 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 IC/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 Liquid Waste Management System. 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 and the staff confirmed that DCD Revision 3 contains the changes as described above.

<u>Water Temperature</u>:

The fuel and auxiliary pools have non-safety-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 cooling and cleanup 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 are acceptable to support RTNSS functions of the FAPCS since they include typical local and MCR controls and indications.

<u>FAPCS System Flow and Pressure</u>:

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 for pump protection with the pressure transmitters that send 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 are acceptable to support RTNSS functions of the FAPCS since they include typical local and MCR controls and indications.

In summary, the requirements of GDC 63 have been met by the ESBWR design. 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 SPF, 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 TAF and after 72 hours fire water or another water source can be provided through safety related connections.

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, “Residual Heat Removal (RHR) System,” and 6.2.2, “Containment Heat Removal Systems.” The staff verified whether 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 IC system 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 SPC, LPCI, DW 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 its 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, DW 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 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, “Probabilistic Risk Assessment and Severe Accidents,” 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 neither low pressure injection nor suppression pool cooling is credited in the safety analyses, they are credited in the ESBWR PRA. The applicant’s response stated that a single train of FAPCS is capable of pumping water from the suppression pool to prevent core damage in the event the GDC is not providing makeup water to the reactor, or by 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 to provide calculations that demonstrate these design parameters are adequate. In its 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 performance requirements as requested in the RAI. In RAI 9.1-20 S03, the staff requested the applicant to provide the performance requirements of the FAPCS. In its response, the applicant committed to add FAPCS heat exchanger performance requirements to the 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 asked the applicant to (a) identify and include in the DCD and NEDO-33201 the FAPCS performance requirements for suppression pool cooling mode during accident conditions considered in the PRA, (b) provide assumptions and results showing the FAPCS and reactor cavity cooling water system (RCCWS) can remove heat as assumed in the PRA, and clarify if the FAPCS can remove heat as assumed for all applicable scenarios evaluated in the PRA. In its 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 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 to the DCD, the staff determined that the design specifications provided in ESBWR DCD, Tier 1 Table 2.6.2 and Tier 2 Table 9.1-8 appear to only pertain to the FAPCS heat exchangers being able to remove 8.3 MW of heat from the suppression pool, while the PRA credits 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 does 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., NPSH issues). In RAI 9.1-20 S05, the staff asked the applicant to provide a write up in DCD Chapter 9.1 and Chapter 19, Tier 2 that gives reasonable assurance that the FAPCS heat exchangers and pumps will be capable of removing the assumed heat load credited in ESBWR PRA, NEDO-33201, Revision 4. In addition, the staff asked that the Tier 1 plant service water system (PSWS) interface requirements be evaluated and modified as appropriate to be consistent with the changes made to DCD Section 9.2 in response to this RAI. In its 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° Kelvin (136.8 °F) based on the maximum FAPCS temperature and the minimum RCCWS temperature. The staff determined that the RAI response was acceptable since the limiting differential temperature is based on the maximum FACPS temperature of 91° C (195.8 °F) and the minimum RCCWS temperature of 15°C (59 °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 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 its response, the applicant stated that while 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 non-use 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 the system can be purged of any gases that are present. Also, plant operation and maintenance procedures assure 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 13.5-4-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 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 utilized to avoid gas accumulation and procedures are used to assure sufficient measures are taken to avoid water hammer and gas binding in pumps with a pointer to Section 13.5.2, "Operating and Maintenance Procedures."

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, 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 FAPCS are relieved by a pressure relief valve. In addition, FAPCS is not immediately placed into service for either LPCI or alternate shutdown cooling 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 net positive suction head (NPSH) is ensured for these functions, consistent with the guidance of SRP Section 6.2.2, "Containment Heat Removal Systems," Revision 4, assuming the respective pool is at saturation temperature for the pressure at its surface.

In its response, the applicant stated that the FAPCS pumps are located approximately 14 meters (~46 feet) below the bottom of the suppression pool, which is a significantly higher available NPSH than exists for pumps performing these same functions in most boiling-water reactors.

In its 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 suppression pool cooling functions. However, the applicant did not provide an actual analysis for 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, "Containment Heat Removal Systems," Revision 4, the staff verified whether 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 or a passive failure of the emergency makeup header pressure boundary.

In its 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 IC/PCC and SFPs. In DCD, Revision 2, the applicant revised Figures 2.6.2-1, "Fuel and Auxiliary Pools Cooling Cleanup System," in Tier 1 and 9.1.1, "New Fuel Storage," in Tier 2, 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 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 for 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 where the water level can be restored and maintained at a level above the top of the reactor core. DCD Tier 2, Subsection 6.3.3, presents the results of the short term (0 to 2000 seconds) emergency core cooling system (ECCS) performance evaluation and Subsection 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 feet) 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 feet) above the TAF, or 8.453 m (27.7 feet) from the RPV bottom). For all credible single failures considered, the long term RPV water level following a LOCA remains higher than 8.453 m (27.7 feet) 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 feet) from the RPV bottom, or 2.5 m (8.2 feet) 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 determined the response was 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 IC/PCC 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 PCC is satisfied assuming a single active or passive failure affecting the makeup line from the FAPCS. In its 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 IC/PCC 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 PCC heat exchangers and two for IC 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 how the sub compartments are shared via valve design assuming a single failure. In addition, the emergency makeup line was revised to include two parallel valves as described regarding 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 IC/PCC pools since the expansion pools are not normally connected. In its 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 IC/PCC 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 determined that the response was 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 7, Section 9.1.3.2, describes the SFP cleanup system. The SFP cleanup system and various auxiliary systems are designated as non-safety-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 has been designed with the capability to maintain acceptable pool water conditions. The staff verified whether that 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 its response, the applicant stated that DCD Tier 1, Revision 1, Figure 2.6.2-1 indicates the capability to discharge water to the liquid waste management system 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 determined that the response was 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 Section 9.1.3.2, "System Description" of the DCD, Revision 5, 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(s) where the design details and justification are provided in the DCD that this line cannot rupture under any circumstances. In its response, the applicant stated it would modify Subsection 9.1.3.2 of the DCD Tier 2 in Revision 6 to state that an analysis would be performed consistent with DCD Tier 2 Subsection 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 determined that the RAI response was 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.

<u>ITAAC</u>

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

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

2. FAPCS is designed to provide drywell spray and alternate shutdown cooling.

3. In Section 9.1.3.1, “System Description,” for FAPCS in DCD, Revision 5, it describes the portions of the FAPCS that are not specifically defined as safety related as being seismic Category II. This quality is not mentioned in Table 2.6.2-2, “ITAAC for the Fuel and Auxiliary Pools Cooling Cleanup System.”

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, 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 its response, the applicant stated the following:

Item 1: Regarding the design feature that the FAPCS consists of two physically separated cooling and cleanup trains, the applicant stated and the staff confirmed that the design commitment is covered by existing ITAAC. This is acceptable to the staff and the staff agrees that existing ITAAC are adequate.

Item 2: Regarding the design feature that the FAPCS is to provide drywell spray and alternate shutdown cooling, the applicant stated these functions are not safety-related and are not credited in the licensing basis, and therefore should not be included in 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.

Item 3: Regarding the fact that the system description in DCD, 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 FAPCS is designed to seismic Category II and therefore it does not need to be mentioned in the Tier 1 Table 2.6.2-2, “ITAAC For The Fuel and Auxiliary Pools Cooling Cleanup System.” 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 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 are used on all submerged FAPCS piping to prevent unintended drainage of the pools, the applicant noted that an ITAAC added to 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. The response to RAI 14.3-442 is further discussed in section 9.1.2.3 of this report.

Item 6: Regarding the design feature addressed in DCD Tier 2 Revision 5 that the suppression pool suction line is conservatively designed to preclude a rupture between the pool and the containment isolation valves, the applicant stated that this design commitment is covered by the ITAAC in Tier 1 Table 3.1-1, Item 3. 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 Tier 2 DCD Section 9.1.3.2 describing why the piping would not crack. The applicant clarified that an analysis will be performed on the suppression pool suction line, in accordance with DCD Tier 2, Subsection 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, part 6 of this RAI is resolved.

Item 7: Regarding the design feature that the electrical power supplies, control and instrumentation of the two FAPCS trains and their supporting systems are electrically and physically separated, the applicant referred the staff to the applicant's response to RAI 14.3-394, S01 and the corresponding Tier 1 markup. The staff determined that this response was unacceptable. The RAI 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 Tier 1 that assure the control cables, instrument cables, and power cables for equipment in the two FAPCS trains are physically and electrically separated. In its response, the applicant stated it would add a new ITAAC in DCD Tier 1, Revision 6 (Table 2.6.2-2, item 16) to test and inspect that "(t)he 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 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 DCD Tier 1, Rev 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 Tier 1 and new ITAAC (Table 2.6.2-2, item 18a, item 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 those identified for 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 determined that cumulatively the responses were 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 assure that heavy loads drops would not impact the electrical equipment of both trains of 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 FAPCS pool cooling piping that provide flow paths for post-accident makeup water transfer, the applicant noted the separation of the piping presently is 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 8 parts of RAI 14.3-443, RAI 14.3-443 is resolved.

DCD, Revision 5, Section 9.1.3.1, "System Description," describes FAPCS as being a nonsafety-related system with the exception of the piping and components relied upon for containment isolation, refilling the IC/PCC pools and SFP, and interface with the Reactor Water Cleanup/Shutdown Cooling system. Seismic Category I piping was shown on DCD Tier 1, Figure 2.6.2-1 but 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 1 piping identified in DCD Tier 1, Table 2.6.2-1; however, no piping was so identified. In RAI 14.3-444**,** the applicant was asked to revise DCD Tier 1 Section 2.6.2 ITAAC for Seismic Category 1 piping to reference Figure 2.6.2-1 or modify Table 2.6.2-1. In its response, the applicant committed to adding the FAPCS piping described as safety-related to Tier 1 Table 2.6.2-1 and Tier 2 Table 9.1-3. The applicant also stated it would add the GDCS interconnecting pipes to 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 FAPCS was added to Tier 1 including the GDCS interconnecting pipes. However, the RAI response was not fully implemented in DCD Revision 6. In RAI 9.1-134, the staff requested the applicant to clarify the list of FAPCS safety-related items. In its 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 determined the RAI response was acceptable since the applicant clarified that FAPCS has four safety-related items as was described in the response to RAI 14.3-444. The staff confirmed that the DCD changes were incorporated in the DCD Revision 7. Accordingly, based on the above and the applicant's response, RAI 9.1-134 is resolved.

<u>10 CFR 20.1101(b)</u>.

The staff verified whether 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. 10 CFR 20.1101 was previously addressed in Section 9.1.2 of this report related to fuel storage pools.

DCD Tier 2, Section 12.3.1.4.2, "Fuel and Auxiliary Pool Cooling System", described 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 filter demineralizer units that 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 filter demineralizer 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/hr (2.5 mrem/hr) and normal personnel access is permitted.

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

<u>10 CFR 20.1406</u>

The staff evaluated the fuel storage pools liner welds in accordance with 10 CFR 20.1406 in section 9.1.2.3 of this report.

DCD Tier 2, Table 12.3-18, "Regulatory Guide 4.21 Design Objective and Applicable DCD Subsection Information," describes the provisions 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. 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 IC/PCCS pools to the Liquid Waste Management System. 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 IC/PCCS pools are also equipped with stainless steel liners, and equipped with leak detection drains. All leak detection drains are designed to permit free gravity drainage to the Liquid Waste Management System. 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 Leak Detection and Isolation System (LD&IS).

The staff finds that these design provisions for FAPCS conform to the guidelines of RG 4.21 and meets the requirement of 10 CFR 20.1406. Section 12.3 of this report further addresses the ESBWR design in accordance with 10 CFR 20.1406.

Operating Experience Considerations

Inspection and Enforcement (IE) Bulletin 84-03, “Refueling Cavity Water Seal,” was issued to address the potential failure of refueling cavity seals to assure 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. Additional information concerning refueling cavity seal failures was provided by Information Notice (IN) 84-93, “Potential for Loss of Water from the Refueling Cavity.” IN 84-93 noted that refueling cavities can also be drained due to 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, RAI 9.1-128 and RAI 9.1-128, Supplement 1, were issued by the staff 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. Tier 2 Table 1C-2, “Operating Experience Review Results Summary – IE Bulletins,” was revised to show that information pertaining to IE Bulletin 84-03 is provided in Tier 2 Subsections 6.2.1.1.2, 9.1.4.21, and 12.4.4, and these subsections of the DCD were also revised to reflect the applicant’s response. Tier 2 Section 9.1.4.8 was also revised to provide additional information concerning the seal plugs discussed in (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

The refueling cavity bellows seal (RCBS) for ESBWR is described in Tier 2 Section 6.2.1.1.2 and 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 Tier 2 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.

Tier 2 Section 9.1.4.8, “Reactor Servicing Equipment,” indicates that prior to refueling, the main steam and the depressurization valve (DPV)/IC 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 factor of safety of five or more, and include redundant seals (one pneumatic and one mechanical). Each seal is individually leak tested prior to 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 Tier 2 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 (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 due to 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 has been 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 the fire protection system. 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.

The IFTS is described in Tier 2 Section 9.1.4.12 and the staff’s evaluation of the IFTS design and potential for draining the refueling cavity is provided in Section 9.1.4 of this report. Consequently, no further evaluation of IFTS is provided in this section.

FMCRD maintenance is discussed in Tier 2 Section 4.6.2.1.4. 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 that are specified in 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 has been effective in preventing catastrophic drainage from BWR control rod drive 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 Tier 2 Section 13.5.2 and are subject to NRC inspection.

Based on operating experience, this approach has been 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 Tier 2 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 and separator storage pool, upper fuel transfer pool, and reactor well all have local, non-safety-related, panel-mounted level transmitters that annunciate high/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

The impact and mitigation of refueling cavity leakage is discussed in Tier 2 Section 12.4.4, “Refueling Operations.” As indicated in 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 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 meters (19.7 feet) 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. Dose considerations associated with refueling operations are evaluated in Section 12.4 of this report.

E. Procedural Controls for Maintaining Refueling Cavity Integrity

Tier 2 Section 13.5.2, “Operating and Maintenance Procedures,” 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 and that plant operating procedures, procedures for performing maintenance, and procedures related to refueling cavity integrity will be included in this plan (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 for performing periodic maintenance and inspection of the refueling cavity seal and the main steam and isolation condenser system plugs. The procedures specified in Tier 2 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 Tier 2 Section 13.5.2 are necessary and appropriate, and are considered to be acceptable by the NRC staff.

The considerations referred to in Tier 2 Sections 9.1.4.8 and 9.1.4.21 and discussed above assure 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 determined that the RAI response and DCD changes were 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 determined 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 concludes 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, equipment designed to handle heavier loads that also are used to maneuver light loads are discussed in the LLHS section of the DCD, and their application for light loads is evaluated in this section.

9.1.4.1 Regulatory Criteria

The staff reviewed the LLHS in accordance with SRP Section 9.1.4, Revision 3, “Light Load Handling System (Related to Refueling),” issued March 2007. The staff’s acceptance of the ESBWR design is based on meeting the relevant requirements of the following GDC and 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 requirement of GDC 5 is not applicable to the single unit. Compliance with the requirements of GDC 61 and GDC 62 depends on adherence to the guidance of ANSI/ANS 57.1-1992, "Design Requirements for Light Water Reactor Fuel Handling Systems."

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 fuel building (FB) crane, reactor building (RB) crane, refueling machine, fuel-handling machine, inclined fuel transfer system (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.

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.

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.

Refueling Machine

The refueling machine located in the RB is used to transport fuel and reactor components to and from 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 designed to meet the requirements of NUREG-0554, Single Failure Proof Cranes and ASME NOG-1, Rules for Construction of Overhead and Gantry Cranes.

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, and 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, "Single Failure Proof Cranes for Nuclear Power Plants" and ASME NOG-1, "Rules for Construction of Overhead and Gantry Cranes (Top Running Bridge, Multiple Girder)."

<u>Fuel Transfer System</u>

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 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.

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.

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.

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.

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 has been designed such that one hook pin and one main beam of the cruciform is capable of carrying the total load of 176 tons (160 metric 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 Nuclear Regulatory Commission (NUREG)-0612, "Control of Heavy Loads at Nuclear Power Plants" issued July 1980, and ANSI-14.6, "Standard for Special Lifting Devices."

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 is 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 is 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 whether 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 its 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 determined that this approach was acceptable, but in RAI 9.1-33 S01 requested the applicant to revise the DCD to include a reference to RG 1.29 for meeting GDC 2 as 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, “ITAAC for the Cranes, Hoists and Elevators,” will demonstrate conformance to RG 1.29. In its response, the applicant committed to revise DCD Tier 2 to include a reference to RG 1.29 for meeting GDC 2 as related to fuel handling components. It also noted that reactor and fuel building cranes have been re-classified to seismic Category I in DCD Tier 1, Revision 5 and clarified that the seismic Category I reactor and fuel building cranes described in Table 2.16.1-1 ITAACs 5 and 6 are designed to withstand the effects of a safe shutdown earthquake condition thus demonstrating conformance to RG 1.29. The staff determined that the response to RAI 9.1-33 was acceptable, in combination with the additional DCD changes discussed since with these changes the design meets 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 will be in accordance with Sections 3.7.2, “Seismic System Analysis,” and 3.7.3, “Seismic Subsystem Analysis,” of the SRP and is addressed in Sections 3.7.2 and 3.7.3 of this report.

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, “Fuel Transfer System,” 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, “Fuel Transfer System.” 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 its 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 assure that the failure of one sensor does not result in lack of knowledge of the carriage position. Control interlocks are provided to assure 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 determined that the RAI response was 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 resolved.

In DCD Tier 2, Revision 3, Section 9.1.4.12, the applicant 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 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 the IFTS components. RAI 9.1-35 was being tracked as an open item in the SER with open items. In its response, the applicant indicated it would revise DCD Tier 2, Figure 9.1-2, "Inclined Fuel Transfer System," Table 3.2-1, "Classification Summary," and Table 9.1-4, "Classification of Equipment" to make the boundaries of seismic design classifications clear. The modifications were made in DCD Tier 2, Revision 5 to clearly define the seismic classification of the IFTS components. The staff determined that the RAI response was 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 its 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 it would revise Table 3.2-1 and Table 9.1-4 to identify that the new fuel inspection stand shall be seismic Category II. The staff determined that the RAI response was acceptable since the applicant clarified the seismic design classification of the new fuel inspection stand. The staff confirmed that the modifications were made in DCD Tier 2, Revision 5. Subsequently, Table 9.1-4 was removed from the DCD rev 6 and seismic classification is included 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 its response, the applicant clarified how the guidelines of NUREG-0612 and ANSI 14.6 will be met. The staff determined 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 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, "Seismic Classification," and 3.2.2, "Quality Group Classification," 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 concludes 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 whether the design complied with the requirements of GDC 61 and GDC 62 and the guidelines of ANSI/ANS 57.1. DCD Tier 2, Rev. 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 its response, the applicant agreed to modify the DCD to provide this information. In Section 9.1.4.1, "Design Bases" and 9.1.4.5, "Refueling Equipment" of DCD Tier 2, Rev. 4, the applicant states that both the refueling machine and the fuel handling machine always maintain a safe water shielding depth of at least 2591 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 meters (10 feet). In RAI 9.1-50 S01, the staff asked the applicant to justify this discrepancy with SRP 9.1.2 and RG 1.13. In its response to RAI 9.1-50 S01, the applicant stated that the interlock height of 2591 mm (8.5 ft) is the actual height of water above the top of active fuel that is provided with the normal full up interlock installed on either the refueling and/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 2591 mm interlock height. In its 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 2591 mm (8.5 ft) of water above the top of a single fuel assembly provides adequate shielding during transit. In addition, the applicant stated it would include reference to the dose rates from the shielding calculation in DCD Tier 2, Rev. 6. The proposed mark-up of the DCD, Rev. 5, provided in the RAI response, stated 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/hr (27 mrem/hr) 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 its 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 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 the staff's RAI 12.2-27. The staff also requested that the applicant justify how the estimated operator dose provided in their response to this RAI supplement meets the criteria stated in ANSI/ANS 57.1-1992, which states that the maximum dose rate to an operator for fuel handling equipment should not exceed 2.5 mrem/hr. In the applicant's 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 meters (10 feet), 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 they would increase the water coverage in the RB refueling floor pools from 2.59 m (8.5 feet) to 2.74 m (9 feet) over a raised assembly and would provide additional shielding (equivalent to one foot of water) to the refueling machine design. The applicant stated that these changes would be made in Rev. 7 of the DCD. The staff determined that the RAI response was 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 these changes were incorporated into DCD Revision 7.

In Section 9.1.4, "Design Bases," Section 9.1.5.2, "General," and Table 9.1-5, "Reference Codes and Standards," of DCD, Revision 5, the applicant referenced only NUREG-0554, "Single-Failure Proof Cranes for Nuclear Power Plants," 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 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 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 its response, the applicant agreed to add ASME NOG-1 as a reference in DCD Tier 2, Revision 6, in Subsections 9.1.4.5 and 9.1.5.2 and Table 9.1-5. The staff determined that the RAI response was acceptable since the applicant referred to the ASME 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), "Review Procedures," 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. Figures showing overall system arrangement, including reactor well, the buffer pool, the upper fuel transfer pool, the inclined fuel transfer pool, the fuel building storage pool, the spent fuel storage pool, the lower fuel transfer pool, cask pool, and the inclined fuel transfer system were not provided by the applicant in DCD Tier 2, Revision 5. 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 its response, the applicant stated that in response to RAI 14.3-441 (discussed below), it would list the fuel and reactor building overhead cranes, as well as the refueling machine and fuel building machine hoists, as single failure proof with an ITAAC in DCD Tier 1. The staff confirmed that this was so modified in Revision 6 to the DCD. The applicant also clarified that the nuclear island plan figures for the different reactor and fuel building elevations, Figures 1.2-1 to 1.2-11, are included in DCD Tier 2, Section 1.2. These figures show the overall light load handling system arrangement related to refueling. The staff determined that the RAI response was 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 "Reference Codes and Standards," states that NUREG-0554, "Single Failure Proof Cranes for Nuclear Power Plants," is applicable 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. Tier 1 Section 2.16.1, "Cranes, Hoists, and Elevators," and Table 2.16.1-1, "ITAAC for Cranes, Hoists and Elevators" 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. 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 its response, the applicant stated it would so revise the DCD in Revision 6. Subsequently, the staff in RAI 14.3-441 S01 requested the applicant to enhance its response by providing a greater level of detail in the ITAAC for single failure proof cranes. In its response, the applicant stated it would so revise DCD Tier 1 in Revision 6. The staff determined that the RAI response was acceptable since DCD Tier 1, Table 2.5.5-1, "ITAAC For The Refueling Equipment," will be revised to specify greater level of details for the refueling machine and fuel building machine hoists to provide reasonable assurance that they are single failure proof. The staff confirmed that the changes were incorporated into DCD Revision 6. Accordingly, based on the above, the applicant's responses, and DCD changes, RAI 14.3-441 is resolved.

Sections 12.3 and 12.4 of this report discuss the staff's evaluation of whether the designs of the fuel-handling system and the spent fuel transfer process will result in occupational radiation exposures during spent fuel handling being ALARA.

Section 15.4.1, "Fuel Handling Accident," of this report discusses the staff's evaluation of radiological consequences of fuel-handling accidents. Section 15.4.10 "Design Bases Accidents," 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 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 its 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 there are no components that 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 determined that the response was acceptable since the applicant clarified the redundant nature of the grappling devices. In addition, ITAAC are provided in DCD Tier 1 Revision 7, Table 2.16.1-1 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 its response dated September 8, 2006, the applicant stated that administrative controls are applied to the tabulated listing of the cranes and refueling equipment provided in Table 9.1-6, "Heavy Load Equipment Used to Handle Light Loads and related Refueling Handling Tasks." The applicant stated that the development of the site-specific procedures to govern these administrative controls is a COL Holder Action 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 Action Items be changed to COL Items, which can provide information to allow the staff to conclude whether safe load paths, routing plans, and administrative controls satisfy the regulatory requirements prior to issuance of a COL. In DCD Revision 4, Tier 2, the applicant modified the text to state that these are to be provided by the COL applicant, and modified the COL 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 determined that the RAI response was acceptable since the proposed action items are consistent with guidance of RG 1.206 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 was resolved.

DCD Tier 2, Revision 1, Section 9.1.4.1, states that, where applicable, DCD Tier 2, Table 9.1-5, "Reference Codes and Standards," 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 its 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 determined that the RAI response was 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 GDC 62. In RAI 9.1-43, the staff requested that the applicant revise the DCD to include such a statement. In its response, the applicant indicated it would add references to the ANSI/ANS standard, which it did in DCD Tier 2, Revision 5. The staff determined that the applicant's response was acceptable since conformance to ANSI/ANS 57.1 was addressed. Accordingly, based on the above, the applicant's responses, and DCD changes, RAI 9.1-43 is resolved.

In SRP Section 9.1.4, "Light Load Handling System (Related to Refueling)," acceptance criteria for meeting the relevant requirements of GDC 61 and GDC 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 fuel building crane, reactor building crane, fuel prep machine, control component change mechanism, inclined fuel transfer system, 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 its response, the applicant stated it would revise Subsection 9.1.4 in the DCD Tier 2 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 DCD Tier 2 Subsection 9.1.4.1 is being revised to clearly state 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 determined that the RAI response was acceptable since the applicant addressed the ANSI/ANS 57.1-1992 guidelines for interlocks. The staff confirmed that Revision 6 incorporated the DCD changes. 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 Section 9.1.4.5, "Refueling Equipment," of DCD Tier 2, Revision 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 2591 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 its response to RAI 9.1-108, the applicant stated that the interlock referred to in the 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 determined that the RAI response was 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.

Section 9.1.4.9, "In-Vessel Servicing Equipment," of DCD, Revision 5, discusses moving the instrument strongback with the Reactor Building 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 Table 9.1-5, "Reference Codes and Standards," in the next revision to the DCD Tier 2 to identify the standards and codes to which these hoists are to be constructed and operated. In its 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 reactor building overhead crane auxiliary hoist is constructed and operated in the same manner as the main hoist of the Reactor Building overhead crane, thus meeting the same standards listed in Table 9.1-5 of the DCD. The staff determined that the RAI response was acceptable since the use of CMAA-70 conforms to the guidance of SRP 9.1.5 Revision 1. Accordingly, based on the above and the applicant's response, RAI 9.1-109 is resolved.

In Section 9.1.4.12, "Fuel Transfer System," DCD, Revision 5, it 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 its response, the applicant stated 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 its 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 that can lead to flooding or loss of core cooling is not credible due to 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 determined that the RAI response was acceptable since the design uses redundant valves to prevent draindowns and the ITAAC was clarified 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.

In Section 9.1.4.12, "Fuel Transfer System," DCD, Revision 5, it 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 whether this protection is single failure proof. In its 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 determined that the response was 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 these diverse and redundant sensors and interlocks be specifically discussed in the DCD. In its response, the applicant stated it would include the sensors and interlocks for opening the bottom and drain valves listed in its response to RAI 9.1-111 in Revision 6 of the DCD Tier 2. The staff determined that the response was acceptable since the sensors and interlocks would be added to the DCD. The staff confirmed that the DCD changes were incorporated in DCD Tier 2, Revision 6. Accordingly, based on the above, the applicant's responses, and DCD changes, RAI 9.1-111 is resolved.

In Section 9.1.4.12, “Fuel Transfer System,” DCD, Revision 5, it states that the inclined fuel transfer system tubes and supporting structure can withstand an SSE without failure of the basic structure or compromising the integrity of adjacent equipment and structures. In Revision 5, Section 9.1.4.12 was changed to state that cooling is provided for two instead of one freshly removed fuel assemblies in the incline fuel transfer system. 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 at least two fuel assemblies are contained in the transport device (i.e., carriage) during the seismic event. In its response, the applicant stated that the seismic event assumes two fuel assemblies are contained in the fuel transfer tube and would modify the DCD to make that clear. The staff determined that the RAI response was acceptable since the applicant added fuel assemblies to the list of items discussed in conjunction with an SSE. The staff confirmed that DCD Revision 6 was so revised. Accordingly, based on the above and the applicant’s response, RAI 9.1-112 is resolved.

In Section 9.1.4.12, “Fuel Transfer System,” DCD, Revision 5, it states 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, (2) that 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 its response, the applicant referenced its response to RAI 12.4-19 S03, questions 1, 2, and 3. Rooms of interest for RAI 9.1-113 are identified and are said to 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 determined that the RAI response was acceptable since the response to RAI 12.4-19 S03 includes the description of methods used to control personnel access during fuel transfer. In addition, DCD Tier 2, Revision 7 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 GDC 62.

Section 9.1.4, “Light Load Handling System (Related to Refueling)” in the DCD only indirectly addresses transfer of spent fuel to a cask. Section 9.1.4.3, “Spent Fuel Cask,” of the DCD, Revision 5, 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 its response, the applicant pointed out that the DCD states that the fuel building overhead crane has the capacity to lift a 165-ton load, which bounds anticipated SFP pool cask weights. The staff determined that the applicant’s response was acceptable since the rated load capacity allows the single-failure proof FB overhead crane to safely lift a spent fuel cask and 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 its response, the applicant clarified in its response that the equipment pool gate is removed and installed to support the drain down and reflooding of the reactor well, while the buffer pool gates are installed and removed to support fuel movement. The applicant indicated that these actions are described in DCD Tier 2, Revision 6 Section 9.1.4.15. 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 make source to the IC/PCCS pools while the water in the buffer pool is not. The applicant indicated that this clarification would be added to DCD Tier 2, Section 9.1.4.15. The staff determined that the RAI response was 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 DCD changes were incorporated into DCD revision 7.

In RAI 14.3-445, the applicant was asked to explain why the reactor pressure vessel head strongback was not added to ITAAC or specified as Tier 1 material. In its response, the applicant stated that the reactor pressure vessel head strongback is non-safety related and thus the design details of the strongback do not meet the criteria for inclusion in Tier 1. The staff determined that the RAI response was acceptable since the strongback serves no safety function and accordingly, the strongback 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 its 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 determined that the RAI response was 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 6 includes COL 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, "Overhead Heavy Load Handling Systems," issued March 2007. The staff's acceptance of the ESBWR design is based on meeting the relevant requirements of the following GDC and regulations:

- GDC 1, 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, "Single Failure-Proof Cranes for Nuclear Power Plants," 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, "Seismic Design Classification," 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, "Spent Fuel Storage Facility Design Basis". The ESBWR design is a single-unit station, and the requirement of GDC 5 is 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 DW servicing equipment, the lower DW servicing equipment, the main steam tunnel servicing equipment, and other servicing equipment.

In DCD Tier 2, Revision 3, Section 9.1.5.3, "Applicable Design Criteria for All OHLH Equipment," it is stated 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/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 fuel shipping cask handling involves a long duration lift, low speed, and spotting control, the design incorporates thermal protection features.

DCD Tier 2, Revision 3, Section 9.1.5.3 also states that transportation routing drawings are made covering 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 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 SER are also applicable to this system.

FB 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/servicing as well as when the plant is on line. 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 NOG-1, “Rules for Construction of Overhead and Gantry Cranes.”

RB 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 DW head, RPV head insulation, RPV head, dryer, chimney partitions, separator strongback, and RPV head stud-tensioning equipment.

The RB crane is used during refueling/servicing as well as when the plant is on line. 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 NOG-1.

Upper DW Servicing Equipment

The upper DW arrangement provides servicing access for the main steam isolation valves (MSIVs), feedwater isolation valves, safety/relief valves (SRVs), DPVs, IC system valves, GDCS valves, and drywell cooling coils, fans, and motors. Access to the space is from the RB through either the upper DW personnel lock or the equipment hatch. Equipment is removed through the upper DW equipment hatch. Platforms are provided for servicing the feedwater isolation valves and MSIVs, the SRVs, and the drywell cooling equipment with the objective of reducing 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 (a) a release of radioactivity, (b) a criticality accident, or (c) the inability to cool fuel within the reactor vessel or SFP.

Lower DW Servicing Equipment

The lower DW arrangement provides for servicing, handling, and transportation operations for fine motion control rod drives (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° 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.

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/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 are required to operate. Inadvertent load drops by the main steam tunnel servicing equipment cannot cause (a) a release of radioactivity, (b) a criticality accident, or (c) the inability to cool fuel within reactor vessel or SFP.

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 (a) a release of radioactivity, (b) a criticality accident, (c) the inability to cool fuel within reactor vessel or SFP, or (d) 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 Section 9.1.5.2, "General," of the DCD, Revision 6:

> 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 its response, the applicant agreed to do so and the staff verified that Revision 7 has incorporated this reference into the DCD. The staff finds the RAI response acceptable since Section 5.1.1(6) of NUREG-0612 was added 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 light and overhead heavy-load handling systems 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 its 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 determined that the response to RAI 9.1-38 was acceptable 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 DCD changes were incorporated 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, "Fuel Building and Reactor Building Cranes," stated 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 its 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. This information must be reviewed by the staff 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 its response, the applicant proposed to modify the COL Holder items in DCD Tier 2, Section 9.1.6, to COL Items 9.1-4-A (fuel-handling operations) and 9.1-5-A (handling of heavy loads.) The staff determined the response was 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 above changes were incorporated into DCD Tier 2, Revision 4 and the confirmatory item is closed.

The Acceptance Criteria in SRP 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 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 its 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 determined that the response was acceptable, since the applicant clarified conformance to GDC 1 and ANSI N14.6, ASME B30.9, and NUREG-0554 in accordance with SRP 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 9.1.5 for GDC 2 states 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 in the context of GDC 2. In RAI 9.1-101, the staff asked the applicant to address compliance with GDC 2. In its response, the applicant stated it would revise Subsection 9.1.5.2, DCD Tier 2 in Revision 6 to commit the OHLHS to complying with NUREG-0554, thus meeting the criteria of GDC 2. The staff determined that the response was acceptable, since the applicant clarified conformance to GDC 2, NUREG-0554, and RG 1.29 on accordance with SRP 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 9.1-101 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 heavy load drop analyses performed by the applicant. 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 its response, the applicant stated that the reactor building 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, ANSI N14.6, ASME/ANSI B30.9, ASME/ANSI B30.10, and NUREG-0612. NUREG-0612 allows the use of 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 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 determined that the RAI response was 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 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, "System Description," did not provide a description of 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 its response, the applicant stated it would list the fuel and reactor building 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 Application Item 9.1-5-A that will call for the COL applicant to develop heavy load safe paths and routing plans. The staff determined that the RAI response was acceptable, since COL Item 9.1-5-A addresses physical arrangements and is in accordance with the guidance of RG 1.206 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, "Operation Responsibilities," 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 9.1.5, Section III.4.C.i states, "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 its response, the applicant stated it would revise DCD Tier 2, Subsection 9.1.5.2 in Revision 6 to address the guidance of the SRP 9.1.5 Section III.4.C.i. In addition, the applicant stated it would revise Subsections 9.1.5.8 and 9.1-5-A to commit the OHLHS to meeting the QA program recommendations of NUREG-0554 and the program elements added to the DCD Tier 2 Section 9.1.5.2. The staff determined the RAI response was acceptable since the applicant revised the DCD to conform to the guidelines of SRP 9.1.5, Section III.4.C.i. The staff confirmed the DCD changes were incorporated into DCD Revision 6. Accordingly, based on the above and the applicant's response, RAI 9.1-104 is resolved.

Subsection III.4.C.ii.(1) of Section 9.1.5 SRP states, "[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." Section 9.1.5.5 of the DCD, Revision 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 its response, the applicant stated it would revise DCD Tier 2, Subsection 9.1.5.5 in Revision 6 to 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 determined the RAI response was acceptable since the applicant revised the DCD to be consistent with the guidelines of SRP 9.1.5, Section III.4.C.ii.(1). The staff confirmed the DCD changes were incorporated into DCD Revision 6. Accordingly, ased on the above and the applicant's response, RAI 9.1-105 is resolved.

In Section 9.1.5.2, "General," of the DCD, Revision 5, the applicant commits to having hoists, cranes, or other lifting devices comply with, among other standards, ASME/ANSI B30.9. Subsection III.4.C.ii.(2) of Section 9.1.5 "Overhead Heavy Loads Handling Systems," SRP 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." 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 choice to not specify metallic material (chain or rope) for construction of slings. In its response, the applicant stated it will revise the existing COL Item related to the handling of heavy loads to ensure 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 Subsections 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 determined the RAI response was acceptable since the applicant included RIS 2005-25, supplement 1 in the heavy load handling program, which addresses SRP 9.1.5, Section III.4.C.ii.(2). The staff confirmed the DCD changes were incorporated into DCD Revision 6. Accordingly, based on the above and the applicant's response, RAI 9.1-105 is resolved.

Table 9.1-5 "Reference Codes and Standards," DCD, Revision 5 states that NUREG-0554, "Single Failure Proof Cranes for Nuclear Power Plants," 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." ESBWR DCD, Tier 1 Section 2.16.1, "Cranes, Hoists, and Elevators," and Table 2.16.1-1, "ITAAC For The Cranes, Hoists and Elevators" 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 “single failure proof” design criteria for the above listed cranes and hoists should be listed in Tier 1. 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 its 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 a minimum set of tests should be included in the ITAAC for single failure proof cranes. 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 its response, the applicant added the requested tests to the ITAAC, referencing ASME NOG-1 in the acceptance criteria. The staff determined the response was acceptable since the included tests and the use of ASME NOG-1 conforms with the guidance of SRP 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 important-to-safety design features that were omitted from 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 ITAAC:

- Cranes and hoists, or monorail hoists pass over the centers of gravity of heavy equipment that is to be lifted.

- The PCC and GDC 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 its 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 determined the RAI response was acceptable since ITAAC conforms to the safety commitment in Tier 2.

- The PCC system is not required to be operable during refueling and Tier 2, Section 9.1.5.6 will be revised to delete references to the PCC system regarding load drops. For the GDCS, an ITAAC will be added to DCD Tier 1 that the GDC is not susceptible to a load drop that could result in the GDC not being able to meet TS for modes 5 and 6. The applicant also clarified that the protection of the GDC components could be provided by restricting the movements of heavy loads through interlocks or the spatial separation of the GDCD components. The staff determined the RAI response was acceptable since ITAAC verifies that the GDC 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 DCD will be revised in Revision 6 to indicate that crane interlocks, and not floor arrangement, preclude transporting heavy loads over fuel storage pools. The staff determined the RAI response was acceptable since the applicant clarified it would use interlocks to prevent transporting heavy loads over fuel storage pools and added an ITAAC to verify the interlocks.

The staff confirmed the DCD changes were incorporated 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 concludes 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 7, 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 NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants" (hereafter referred to as the SRP) Section 9.2.1, Revision 6, "Station Service Water System," issued March 2007. 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 upon conformance with:

- GDC 2, "Design Basis for Protection Against Natural Phenomena," as it relates to the capability of the design to maintain and perform its safety function following an earthquake.

- GDC 4, "Environmental and Dynamic Effects Design Bases," as it relates to the dynamic effects associated with water hammer.

- GDC 5, "Sharing of Structures, Systems, and Components," as it relates to the capability of shared systems and components important to safety to perform required safety functions.

- GDC 44, "Cooling Water," as it relates to transferring heat from structures, systems, and components 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.

The PSWS is a non-safety-related system; however, the system provides defense-in-depth (DID) for the ESBWR passive plant design. In addition to the SRP guidance, the NRC staff's evaluation of DID 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, "Regulatory Treatment of Non-Safety Systems," of this report; (2) confirming that failure of DID systems and components will not adversely impact safety-related SSCs;(3) confirming that availability controls 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

In DCD Tier 2, Revision 7, Section 9.2.1, "Plant Service Water System," the applicant described 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 PSWS that are not a 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 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 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. Remote 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 in 36 hours assuming the most limiting single active failure. A simplified diagram for PSWS is shown in DCD Tier 2 Figure 9.2-1, “Plant Service Water System Simplified Diagram,” Tables 9.2-1, “PSWS Heat Loads,” and 9.2-2, “PSWS Component Design Characteristics,” in DCD Tier 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 and applicable regulations such as GDC 2, 4, 5, 44, 45 and 46. 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, “Reactor Component Cooling Water System,” and 9.2.8, “Turbine Component Cooling Water System,” 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 non-safety-related (NSR) active system, must be relied upon in order to achieve cold shutdown conditions in accordance with TS requirements, and these systems should be highly reliable and capable of achieving and maintaining cold shutdown conditions. In addition, there should be no single failure of these systems which would result in inability to terminate use of the passive safety related systems and achieve cold shutdown. The PSWS should be capable of cooling the plant to Mode 5 conditions within 36 or 37 hours in order to satisfy ESBWR TS requirements. Numerous TS sections require Mode 5 entry. The PSWS, which is designated as RTNSS (including their support systems), is subject to enhanced design, quality, reliability, and availability provisions and are relied upon for performing functions as discussed in Tier 2 of DCD, Appendix 19A, “Regulatory Treatment of Non-Safety Systems.” Sufficient information needs to be included in Tier 1 and Tier 2 of the DCD in order to demonstrate that the PSWS is adequate for achieving and maintaining cold shutdown conditions (i.e., cooldown from Mode 4 to Mode 5) and for performing RTNSS functions and that applicable design consideration have been satisfied.

On March 19-20, 2009, the staff conducted a regulatory audit of the supporting information for ESBWR DCD, Section 9.2, “Water Systems,” including the plant service water system (Section 9.2.1), reactor component cooling water system (Section 9.2.2) and nuclear island chilled water subsystem (Section 9.2.7). The audit was primarily focused on the review of these systems in 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. This audit is referred to several times throughout the remainder of this section.

A. PSWS Classification and Quality Assurance Provisions

DCD Tier 2 Section 3.2, “Classification of Structures, Systems and Components,” specifies the classification of SSCs based on safety importance and other considerations. The staff’s evaluation of the classification designations that are specified is provided in Section 3.2 of this report. This section of the staff’s evaluation confirms that the appropriate classification designations are specified for the PSWS 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 PSWS (RTNSS Criterion C) as discussed in Appendix 19A, Section 19A.8, “Proposed Regulatory Oversight.” The staff reviewed DCD Tier 2 Figure 9.2-1 and confirmed that the classification designations on the simplified diagrams are consistent with those that are listed for PSWS in DCD Tier 2 Table 3.2-1, “Classification Summary.” In particular, the following classification designations are specified in Table 3.2-1 for PSWS:

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

- QA Requirement S is specified for the PSWS in Revision 6 of the DCD as stated in the applicant’s response to RAI 3.2-6 S02. Based on the RAI response, RTNSS components/systems that were identified under Revision 5 of the DCD as QA Requirement E are to be changed to QA Requirement S under 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 and the staff determined this change is 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 non-safety-related SSCs that are classified as RTNSS Criterion C because augmented seismic design standards do not apply. As stated in DCD Section 19A.8.3, “Augmented Design Standards,” 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, “Seismic Design Classification,” March 2007 - Revision 4, regarding nonsafety-related systems. In RAI 9.2-12 and RAI 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 its response, 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 and TCCWS. It will be designed to seismic requirements to be specified in Appendix 19A to DCD Tier 2 and Section 3.2, “Classification of Structures, Systems and Components” of the DCD. Chapter 22 of this report provides the staff 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, “Environmental and Dynamic Effects Design Bases,” 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(s). 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/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 prior to system operation and after maintenance operations, and the use of a check valve at each pump discharge to prevent backflow into the pump.

Water hammer considerations were a topic for discussion at the March 19-20, 2009, audit and were addressed in the applicant’s response 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 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 the applicant’s 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 being addressed in 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 the water hammer issues; therefore, the staff included the issue in RAI 9.2-24 to further ask the following:

- The amount of back leakage through the pump check valves that is considered to be excessive needed to be specified and explained, and how excessive check valve back leakage or system voiding will be prevented from occurring over time needs to be described.

- A description was needed for how proper operation of the automatic air release/vacuum valves will be assured over time.

- Valve actuation/stroke times that are considered to be appropriate (especially with respect to the air operated valves) needed to be specified and explained, and how these times will be maintained as the plant ages needed 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. A detailed hydrodynamic analysis will be performed during the detailed design phase by the applicant 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 through 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 9.75m (32 ft), column separation is prevented by taking care to ensure 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 preoperational startup testing described in DCD 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, “Operating and Maintenance Procedures,” to include provisions to ensure that procedures developed for RTNSS systems will address water hammer.

Based on the staff’s review of the applicant’s response to RAI 9.2-11 and RAI 9.2-24, the staff concludes that water hammer has been adequately addressed 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 Item 13.5-4-A. Accordingly, based on the above and the applicant’s response, RAIs 9.2-11 and 9.2-24 as it relates to water hammer are resolved. Based on the above, the staff concludes that the PSWS meets GDC 4 in accordance with the guidance of SRP Section 9.2.1.

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 SRP 9.2.1 guidance, the staff review of the PSWS against GDC 44, 45, and 46 is based on that the PSWS 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, “Summary of Differences from SRP Section 9,” and Section 9.2.1, the applicant determined that GDC 44, 45, and 46 were not applicable for the PSWS among other systems. 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, to address conformance to GDC 44, 45, and 46. The applicant also clarified that the PSWS satisfied 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 loss of offsite power,
- 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 portions of the GDC 44 requirements that apply to the heat removal function under normal operating conditions apply to the PSWS. The staff reviewed the PSWS 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 Section 9.2.1, and determined that PSWS satisfies GDC 44, 45, and 46 with respect to its normal operation function. However, in a design-basis accident, decay heat is transferred to the isolation condenser/passive containment cooling (IC/PCC) 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 concludes 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 determined 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 provided to detect and control leakage of radioactive contamination into and out of the PSWS. The design is considered to be acceptable by the staff if the PSWS simplified diagrams 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 RAI 9.2-8, RAI 9.2-8 S01, and RAI 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 its 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 could be isolated, identified, and repaired without immediately impacting plant operation; the PSWS design includes provisions for grab sampling; and the COL item (COL 11.5-2-A in DCD Section 11.5.7) will make provisions for sampling cooling tower blowdown as referenced in DCD Tier 2, Table 11.5-5, "Provisions for Sampling Liquid Streams." The DCD requires continuous effluent monitoring either directly on the effluent of PSWS or another downstream process effluent (i.e., one detector could monitor the combined effluent of PSWS and circulating water) to ensure monitoring prior to release to the environment. The staff determined that the RAI responses were acceptance 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 radiation monitors (including alarm functions) were not described in Section 9.2.1 and were not shown on the PSWS simplified diagrams and 10 CFR 20.1406 was not adequately addressed. Therefore, the staff requested in RAI 9.2-26 that the applicant revise Section 9.2.1 and the simplified diagrams as appropriate to address the requirements of 10 CFR 20.1406.

In its response, the applicant stated that radioactive leakage into 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, post accident 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, Rev. 5, Subsection 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 Rev. 5, Section 9.2.2.2). Consequently, there is essentially no potential for plate failure and cross contamination.

DCD Tier 2 Section 9.2.1.2 describes 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 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 Section 9.2.2.2 describes 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 Subsection 9.2.2.5 describes that RCCWS surge tank levels are used to monitor losses of cooling water, and detect intersystem leakage intrusions into 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 plant service water system and minimize contamination of the facility and the environment.

DCD Tier 2, Table 12.3-18, "Regulatory Guide 4.21 Design Objective and Applicable DCD Subsection Information," describes similar provisions related to PSWS 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)
- Decreasing the spread of contaminant from the source (design objective 4)

The staff finds that these design provisions for the PSWS meets the requirement of 10 CFR 20.1406 and conforms to the guidelines of RG 4.21. Section 12.3 of this report further addresses the ESBWR design in accordance with 10 CFR 20.1406. The staff determined the RAI response was acceptable since the applicant clarified why radiation monitors do not need to be described for the PSWS and 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 PSWS is classified as RTNSS Criterion C. Appendix 19A, Section 19A.8.3, “Augmented Design Standards,” indicates that RTNSS Criterion C systems incorporate the DID principles of redundancy and physical separation to ensure adequate reliability and availability. Section 19A.8.3, “Augmented Design Standards,” 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, Section 19A.8.3 indicates that RTNSS Criterion C equipment is qualified to The Institute of Electrical and Electronics Engineers, Inc. (IEEE) Standard 344-1987, “Recommended Practice for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations-Description,” to only demonstrate structural integrity.

RTNSS Criterion C systems in the ESBWR design, such as the PSWS, do not require augmented design standards to assure 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 Section 9.2.1.2, “System Description,” describes that in the event of a LOPP, the PSWS supports the RCCWS in bringing the plant to the cold shutdown condition in 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 these systems are available and reliable. Therefore, design goals for PIP 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 FAPCS in providing a backup source of lower pressure injection and suppression pool cooling. 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 concludes this graded design approach is acceptable considering the design function of the PSWS under the regulatory criteria for this nonsafety system.

G. PSWS Capability and Reliability

In RAI 9.2-24, the staff requested the applicant to specifically address information concerning the PSWS functions that are subject to regulatory treatment of non-safety systems (RTNSS), focusing on PSWS capability and reliability. The key points that were included in this RAI included:

- 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 support of resolution of 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 results of the audit and the RAI response are discussed throughout the remainder of this section.

<u>(1) 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 in this regard 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
- Motor and air operated valve 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 meters per second (15 feet per second) or less. In the staff's experience and in accordance with general engineering practice, piping velocities between 1.2 and 4.6 meters per second (4-15 feet second) are reasonable, thus long term internal pipe wear is expected to be minimal. The remaining items noted above were reviewed by the staff 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 Item. The staff confirmed that Revision 6 of the DCD has incorporated this RAI proposed change. The staff determined that the response to RAI 9.2-24 regarding PSWS descriptive information and flow considerations was acceptable since the applicant clarified the basis for the design parameters included in the DCD and the applicant clarified 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 provides a listing of 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 in this regard and requested in RAI 9.2-24 that the applicant revise DCD Section 9.2.1 to address heat transfer and address 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 in the RAI responses. Single train failure during cooldown results in the greatest heat load per PSWS train at 80.8 (MW) (2.75 x 10^8 BTU/hr). This transient mode occurs with one train of PSW in operation (two (2) PSW pumps) and all heat loads are dissipated through two (2) RCCWS heat exchangers and two (2) TCCW 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. 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 RCCWS heat exchangers (3) and PSW pumps (2) on the active PSWS train. Two PSWS pumps provide sufficient cooling capacity to the RCCWS heat exchangers in order to bring the plant to the cold shutdown condition within 36 hours. This mode of operation removes 74.8 MW (2.55 x 10^8 BTU/hr) from RCCWS using one train of PSWS. The RAI response provided for DCD Table 9.2-2 states that each of the PSWS cooling tower is capable of removing a minimum of 83.5 MW. (2.85 x 10^8 BTU/hr) Based on the staff review, for the two bounding condition noted above, there is at least an 83.5 MW/80.8 MW (2.85 x 10^8 BTU/hr / 2.75 x 10^8 BTU/hr) or 3.3% 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 x 10^7 BTU/hr / 2.75 x 10^8 BTU/hr) or 368% 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 mark-up related to the clarification to the PSWS heat loads and PSWS component design characteristics in Table 9.2-1 and Table 9.2-2. The staff confirmed that Revision 6 of the DCD has incorporated these RAI proposed changes. The staff determined that the response to RAI 9.2-24 regarding heat transfer was 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 Section 9.2.1, the PSWS consists of two fully redundant (train A and train B), 100% capacity trains with each train consisting of a total of 2-50% pumps powered by separate standby diesel generators (EDG). Although the two trains are normally cross-connected via air operated valves, they can be split out if necessary from the control room. The staff determined that clarification was needed for when off-site power is not available and requested in RAI 9.2-24 that the applicant revise Section 9.2.1 to address single failure of the cooling tower basin, backup power for the self cleaning strainer functions and air-operated valves.

The response to RAI 9.2-24 addressed in detail each of the above noted items. All MOVs fail as is during a loss of preferred power and can change position once power is restored via the EDG. 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 TBCCW heat exchangers fail open thus maintaining PSWS system flows.

Based on the staff review of the RAI response, the staff concludes that single failure consideration have been properly addressed due to the redundancy of the design, components emergency power supply availability, and components failure position on a LOPP. In addition, train redundancy ensures that single failure of any air-operated valve will not impact the other train. The staff determined that the response to RAI 9.2-24 regarding single failure and backup power was acceptable since the applicant clarified how the single-failure and backup power attributes of the PSWS are included in the DCD.

(4) PSWS Pump Net Positive Suction Head

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

In order to provide minimum system flow, the PSWS design should assure 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 that the water levels in the service water basins are monitored to ensure sufficient NPSH. However, the specific minimum NPSH for the PSWS pumps; the minimum service water basin water level that is 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); 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 how COL applicants will know to periodically confirm that adequate levels exist in the service water basins were not described. Therefore, the staff requested in RAI 9.2-23, RAI 9.2-23 S01, and RAI 9.2-24 that the applicant address NPSH and additional questions including addressing design alarms features in the MCR available to the operators. In addition, the staff asked the applicant to revise Section 9.2.1 to include this information and to establish COL information items and interface requirements as appropriate.

In the applicant’s response to these RAIs, DCD Revision 5 changes and Revision 6 mark-ups were provided. It states in Revision 5, Section 9.2.1.2 that 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 sufficient NPSH at design flow is provided to the PSWS pumps. In addition, the change to Tier 1, Section 4.1, “Plant Service Water System – Interface Requirement,” stated that the PSWS pumps must have sufficient available net positive suction head at the pump suction location for the lowest probable water level of the heat sink. In its response to RAI 9.2-23 S01, the applicant revised the description of the interface between the standard plant design for the ESBWR and conceptual design to be addressed by COL applicants to include consideration of NPSH under worst case conditions. In addition, DCD

Tier 2, subsection 14.2.8.1.51, "Plant Service Water System Preoperational Test," describes a series of individual component and integrated system tests to demonstrate acceptable pump suction under the most limiting design flow conditions.

The staff determined that the responses to RAIs 9.2-23 and 9.2-24 regarding PSWS pump NPSH were 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 concern of NPSH is resolved. The staff confirmed that Revision 6 of the DCD incorporated these RAI proposed changes.

(5) Operating Experience

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

- Generic Letter (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 its 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, "System Description," 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, "COL Information," identifies a corresponding COL Item, 9.2.1-1-A, Material Selection." The staff determined that the RAI response, with the addition of COL Item 9.2.1-1-A, was 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 ESBWR because it was issued for safety-related systems. The staff believes that while the PSWS is not safety-related, it performs DID functions, and there is no basis to conclude that the provisions of the GL should not apply to those systems that perform these functions. DID systems are different from typical non-safety-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 assure 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 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 DID functions over the life of the plant. Likewise, the applicant's responses to the other operating experience items that are referred to in Chapter 1 that pertain to DID systems and components need to be revised accordingly to address the operating experience considerations as they pertain to these important systems rather than inappropriately dismissing the items based on system safety classifications.

In the applicant's response to RAI 9.2-24, it was stated that the ESBWR PSWS is not committed to meeting the recommendations of GL 89-13. DCD Table 1C-1, "Operating Experience Review Results Summary – Generic Letters," states that 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 in the cooling water system design. The RAI responses added;

- 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 determined that the RAI 9.2-24 regarding operating experience was acceptable since applicable operating experiences have been properly addressed for the RTNSS, non safety 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.

<u>(6) 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, Section 9.2.1.4, "Testing and Inspection Requirements," 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 standards of the 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. Motor-operated valves 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 DID functions over the life of the plant. The PSWS design bases indicates that provisions are included to permit inspection of components and equipment. Also, the system description indicates that valves are arranged for ease of in-service inspection. Section 9.2.1.4, "Testing and Inspection Requirements," indicates that provision is made for periodic inspection of components to ensure the capability and integrity of the system, and that motor-operated valves are inspected to ensure plant availability. The periodic inspection and testing was determined to be incomplete; therefore, the staff requested in RAI 9.2-24 that the applicant revise Section 9.2.1.

In the applicant's response to RAI 9.2-24, it was 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 Appendix 19A.8, "Proposed Regulatory Oversight," and 19A.8.4.9, "Component Cooling – HVAC, Cooling Water, Chilled Water, and Plant Service Water," 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 system. Such SSCs may include RTNSS components.

The staff determined that the periodic inspections and testing aspect of the RAI 9.2-24 response is acceptable since the PSWS will be monitored under the Maintenance Rule Program to 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

Section 9.2.1.5, "Instrumentation Requirements," indicates that the PSWS is operated and monitored from the main control room, and that it can also be operated from the remote shutdown panels. Other instrumentation that was briefly described includes 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 its response, the applicant explained all the instruments, controls, and alarms in the MCR for the PSWS and revised the DCD accordingly. To address to this RAI, the applicant made changes to DCD Tier 2 Section 9.2.1.5 in Revision 5 providing the instruments and controls and alarms in the MCR. The staff determined the RAI response and DCD changes were acceptable since DCD Tier 2, Revision 5, Section 9.2.1.5 identifies the instrumentation controls and alarms necessary for PSWS operation and indicates 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 RAI 9.2-6 and RAI 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 its 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 staff to resolve the RAI. Subsequently, the applicant added more details in the existing simplified diagrams of Figures 9.2-1 and 9.2-2.

As a follow-up to RAI 9.2-6 S01, RAI 9.2-24 was generated and asked the following:

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

- A more detailed description of how the PSWS detects gross leakage is needed, and the instrumentation that is credited needs to be specified.

- DCD Tier 2 Section 9.2.1.5, "Instrumentation Requirements," 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 Section 9.2.1.2, "System Description," 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 applicant's response to RAI 9.2-24, the staff performed the following:

- Reviewed the available Phase 1 design drawings and PSWS proprietary drawings updated for the PSWS Value Added Board for header temperature and pressure detectors.

- Reviewed the provided drawings that provide monitoring of system flow in the Main MCR and can be used to assist in leak detection.

In the response to RAI 9.2-24, the applicant noted that 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 standby diesel generators. This ensures 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 the staff's review, it was determined that the drawings were adequate in the placement of PSWS instrumentation which includes 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 PSWS pump trip based on the pump discharge valve failing to open was reviewed and determined to be adequate since it provides pump protection against a no-flow condition.

The response to RAI 9.2-24 also stated that DCD Tier 2 Section 9.2.1.5 will be revised under Revision 6 to specify that a PSWS pump will trip if the pump discharge valve fails to open ensuring minimum flow conditions are maintained. The staff determine 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, RAI 9.2-6, and RAI 9.2-24 as relating to simplified diagrams are resolved. The staff confirmed that Revision 6 of the DCD has incorporated the RAI proposed changes.

9.2.1.3.2 COL Information

The staff reviewed DCD Section 13.5.3 “COL Information,” COL 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 American National Standards Institute (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 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 above generic COL information item might not cover the non-safety-related systems such as PSWS and RCCWS in the ESBWR. In response to RAI 9.2-11 S04, the applicant revised Section 13.5.2 of the DCD to clarify that the water hammer procedures for the RTNSS systems will be included as a part of COL Item 13.5-4-A. Therefore, the staff finds COL Item 13.5-2-A acceptable regarding procedure development for the PSWS.

The applicant identified one COL Item, COL 9.2.1-1-A, “Material Selection,” specifically for the PSWS. This item is discussed above in Section 9.2.1.3.1 of this report, under Item D.5, “Operating Experience,” with respect to GL 89-13. The staff determined this particular item is considered to be acceptable, which addresses aspects of GL 89-13 related to material determinations.

9.2.1.3.3 Availability Controls

As discussed in DCD Tier 2, Revision 7, Appendix 19A, Section 19A.8.1, “Regulatory Oversight – Availability Controls,” regulatory oversight is applied to each system that is designated as RTNSS to ensure adequate reliability and availability to perform RTNSS functions. 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 Availability Controls Manual (ACM). Appendix 19A, Table 19A-2, “RTNSS Functions,” identifies that the PSWS is a support system and that the PSWS ‘Availability Controls’ are the ‘Maintenance Rule,’ which means that the availability of the PSWS is addressed by the Maintenance Rule performance monitoring rather than by a specific ACM entry.

The PSWS is subject to the ACM through the systems it supports. Table 19A-2 classifies PSWS as a support system for RCCWS, which is classified as a support system for the standby diesel generators (SDGs) and for the nuclear island chilled water system (NICWS). NICWS supports building heating, ventilation, and air conditioning (HVAC), which supports the fuel and auxiliary pools cooling system (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 (ACs) for the SDGs in AC 3.8.1, “Standby Diesel Generators – Operating,” and AC 3.8.2, “Standby Diesel Generators – Shutdown;” and for 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 ACs that are specified for the SDGs and FAPCS.

ACM 1.1, “Definitions”, 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(s) 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(s) are also capable of performing their related support function(s). Since PSWS is a support system for RCCWS, NICWS, FAPCS, and SDGs, if PSWS becomes unavailable, then the system in which is support becomes unavailable and the applicable ACM action statement would then apply.

Based on the above, the staff finds the availability controls for the PSWS acceptable since the PSWS is subject to the maintenance rule and indirectly subject to the ACM via the PSWS being a RTNSS support system and the ACM definitions.

9.2.1.3.4 Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC)

DCD Revision 5 Tier 1, Section 2.12.7, "Plant Service Water System," provides ESBWR design certification information and ITAAC for the PSWS. Tier 1 information for balance-of-plant SSCs is evaluated in Section 14.3.7 of this report, and 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 Tier 1 information for PSWS.

In RAI 14.3-69, the staff requested that the applicant to revise DCD Tier 1, Section 2.12.7, to include a system description and system drawing, design commitment, and ITAAC scope for PSWS. In its 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, "Plant Service Water System," the applicant provided a design description, ITAAC Table 2.12.7-1, 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 Section 9.2.1. ITAAC details were addressed as part of the March 19-20, 2009 audit and RAI 9.2-24. The applicant response to the staff's questions regarding the lack of specific details for the RTNSS Criterion C acceptance criteria was stated as:

> 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 determined that the RAI response was acceptable since the PSWS Tier 1 information is adequate and reasonable based on the ESBWR graded approach for this RTNSS Criterion C, non-safety 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 for supporting the post-72-hour cooling function of the PSWS. The PSWS is relied upon to remove $2.02x10^7$ MJ ($1.92x10^{10}$ BTU) over a period of 7 days without active makeup. Consequently, verification of compliance with the interface requirements shall be achieved by inspections, tests, and analyses that are similar to those provided for the certified design. The COL applicant referencing the certified design shall 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, in order to provide minimum system flow, the PSWS design should assure 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 Section 9.2.1 to include this information and to establish COL information items and interface requirements as appropriate. In the applicant's response to this RAI a DCD Revision 6 mark-up was provided. The change to Tier 1, Section 4.1 stated that the PSWS pumps must have sufficient available net positive suction head 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, subsection 14.2.8.1.51, "Plant Service Water System Preoperational Test," describes a series of individual component and integrated system tests to demonstrate acceptable pump suction under the most limiting design flow conditions.

The staff determined that the responses to RAIs 9.2-23 and 9.2-24 regarding PSWS pump NPSH were 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 are resolved. The staff confirmed that Revision 6 of the DCD has 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

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

DCD Tier 2, Revision 7, Section 14.2.8.1.51, “Plant Service Water System Preoperational Test,” describes the preoperational test program for the PSWS. The staff finds the objective of the PSWS preoperational test program to be appropriate since it 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 very general terms to address the considerations that apply to PSWS, this approach for this non safety 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, “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 Section 14.2.8.1.51 to address the testing of automatic air release/vacuum valves. In its response to RAI 9.2-24, the applicant provided a DCD mark-up of Section 14.2.8.1.51 with the addition of testing of the automatic air release/vacuum valves. The staff determined the response was acceptable since the addition of testing of the automatic air release/vacuum valves assures 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 has incorporated this RAI proposed change.

9.2.1.4 Conclusion

The staff concludes 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 NRC policies that have been established with respect to its RTNSS C function.

9.2.2 Reactor Component Cooling Water System

9.2.2.1 Regulatory Criteria

The staff reviewed the reactor component cooling water system (RCCWS) based on guidance provided in NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants” (hereafter referred to as the SRP) Section 9.2.2, Revision 4, “Reactor Auxiliary Cooling Water System,” issued March 2007. 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:

- General Design Criteria (GDC) 2, “Design Basis for Protection Against Natural Phenomena,” as it relates to the capability of the design to maintain and perform its safety function following an earthquake.

- GDC 4, “Environmental and Dynamic Effects Design Bases,” as it relates to the dynamic effects associated with water hammer.

- GDC 5, “Sharing of Structures, Systems and Components," as it relates to the capability of shared systems and components important to safety to perform required safety functions.

- GDC 44, “Cooling Water,” as it relates to transferring heat from structures, systems, and components (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.

The RCCWS is a non-safety-related system; however, the system provides defense-in-depth (DID) for the ESBWR passive plant design. In addition to the SRP guidance, the NRC staff’s evaluation of DID 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, “Regulatory Treatment of Non-Safety Systems,” of this report; confirming that failure of DID systems and components will not adversely impact safety-related SSCs; confirming that availability controls are established as appropriate; and confirming that proposed inspections, tests, analyses, and acceptance criteria (ITAAC) and initial test program specifications are adequate.

9.2.2.2 Summary of Technical Information

DCD Tier 2, Revision 7, Section 9.2.2, “Reactor Component Cooling Water System,” 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, “Classification Summary,” indicates that part of the RCCWS (P21) is a non-safety-related system located in the reactor building (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, fuel building (FB) and electrical building (EB) and is designated as quality group “D” and non seismic. The RCCWS has regulatory treatment of nonsafety related systems (RTNSS) functions.

In the event of a loss of preferred power (LOPP), the RCCWS supports the fuel and auxiliary pools cooling system (FAPCS) and the reactor water cleanup/shutdown cooling system (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 standby onsite alternating current power supply diesel generators. Tables 9.2-3, “RCCWS Normal Heat Loads,” and 9.2-4, “RCCWS Component Design Characteristics,” of DCD Tier 2 tabulate the RCCWS design heat loads and component design characteristics.

While the RCCWS is a non-safety-related system, it performs DID functions and is also subject to RTNSS as described in DCD Tier 2 Appendix 19A, "Regulatory Treatment of Non-Safety Systems." As stated in DCD Tier 2 Section 19A.4.2, "Assessment of Uncertainties," 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 fuel and auxiliary pools cooling system (FAPCS) that provide low pressure injection and suppression pool cooling are added in 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 boiling-water reactor (BWR) designs in that the ESBWR RCCWS is non-safety-related system because the RCCWS removes heat only from the chilled water system, RWCU/SDC, FAPCS, and standby onsite AC power diesel generator, 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

Nonsafety-related (NSR) active systems must be relied upon in order to achieve cold shutdown conditions in accordance with Technical Specification (TS) requirements, and these systems should be highly reliable and capable of achieving and maintaining cold shutdown conditions. In addition no single failure of these systems should result in inability to terminate use of the passive safety related systems and achieve cold shutdown in accordance with GDC 44. These NSR systems should be capable of cooling the plant to Mode 5 conditions within 36 or 37 hours in order to satisfy ESBWR Technical Specification (TS) requirements. Numerous Technical Specification Sections require Mode 5 entry. NSR 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 Tier 2 of the Design Control Document (DCD), Appendix 19A, "Regulatory Treatment of Non-Safety Systems." Sufficient information needs to be included in Tier 1 and Tier 2 of the DCD in order to demonstrate that these systems are adequate for achieving and maintaining cold shutdown conditions (i.e., cooldown from Mode 4 to Mode 5) and for performing RTNSS functions and that applicable design consideration have been satisfied.

On March 19-20, 2009, the staff conducted a regulatory audit of the supporting information for the ESBWR Design Control Document (DCD), Section 9.2, "Water Systems," including the plant service water system (Section 9.2.1), reactor component cooling water system (Section 9.2.2) and nuclear island chilled water subsystem (Section 9.2.7). The audit was primarily focused on the review of these systems in 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 Agency-wide Documents Access and Management System (ADAMS) at Accession Number ML101250439. This audit is referred to several times throughout the remainder of this section.

A. RCCWS Classification and Quality Assurance Provisions

DCD Tier 2 Section 3.2, "Classification of Structures, Systems and Components," specifies the classification of SSCs based on safety importance and other considerations. The staff's evaluation of the classification designations that are specified is provided in Section 3.2 of this report. 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 that is described in DCD Tier 2 Section 3.2 and that the designations properly reflect the regulatory oversight provisions that pertain to RCCWS (RTNSS Criterion C) as discussed in DCD Tier 2 Appendix 19A, Section 19A.8, "Proposed Regulatory Oversight." The staff reviewed simplified drawings, 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 Table 3.2-1, "Classification Summary." In particular, the following classification designations are specified in Table 3.2-1 for RCCWS:

- The RCCWS is designated Safety Class N which is used for non-safety-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, "Quality Group D," this quality group generally applies to non-safety-related SSCs that satisfy specified industry codes and design standards and are subject to one or more significant licensing requirement or commitment. The staff concludes that this is appropriate quality group 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 main control room, are designated Seismic Category II. These items are designed to structurally withstand the effects of a safe shutdown earthquake (SSE). Other portions of the RCCWS are located in the TB, RB, FB and EB and are designated as non seismic. 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.

- Quality Assurance Requirement S is specified for the RCCWS in Revision 6 of the DCD as stated in the applicant's response to request for information (RAI) 3.2-6 S02. Based on the RAI response, RTNSS components/systems that were identified under Revision 5 of the DCD as QA Requirement E are to be changed to QA Requirement S under 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 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 concluded that Revision 6 of the DCD has incorporated this RAI proposed change and the staff determined this change is acceptable.

B. GDC 2

RCCWS is a non-safety-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 (non seismic building). SRP Section 9.2.2 indicates that the requirements of GDC 2 can be met for a non-safety-related system based on meeting Regulatory Position C.2 of Regulatory Guide (RG) 1.29, "Seismic Design Classification," regarding non-safety-related systems. In RAI 9.2-12, and RAI 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 its 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 a safe-shutdown earthquake. It will be designed to seismic requirements to be specified in Appendix 19A to DCD Tier 2 and Section 3.2 of the DCD Tier 2. Chapter 22 of this report provides the staff 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 the portions of Regulatory Position C.2 of RG 1.29 regarding non-safety-related systems because the failure of the non-safety-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, "Environmental and Dynamic Effects Design Bases," as it relates to the dynamic effects associated with water hammer. As stated in the DCD Revision 7, 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.

Water hammer considerations were a topic for discussion at the March 19-20, 2009 audit and were addressed in the applicant's response 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 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 its 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.

- The surge tank location (high point of the system) which provides a 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, "System Description," 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, "Operating and Maintenance Procedures," the applicant provided a clarification to state that elements of ANSI/ANS-3.2-1994; R1999, "Administrative Controls and Quality Assurance for the Operational Phase of Nuclear Power Plants," addressing water hammer shall 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 water hammer has been adequately addressed 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 Item 13.5-4-A. Accordingly, based on the above and the applicant's response, RAIs 9.2-11 and 9.2-24 as they relates to water hammer are resolved. The staff concludes that the RCCWS meets GDC 4 as related 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 SRP 9.2.2 guidance, the staff review of the RCCWS against GDC 44, 45, and 46 is based on 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, "Summary of Differences from SRP Section 9," and Section 9.2.2, the applicant determined that GDC 44, 45, and 46 were not applicable for 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 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 loss of offsite power,

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

The staff reviewed the RCCWS 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 Section 9.2.2, and determined that RCCWS satisfies GDC 44, 45, and 46 with respect to its normal operation function. However, in a design-basis accident, decay heat is transferred to the isolation condenser/passive containment cooling (IC/PCC) 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 concludes 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 determined that the response to RAI 9.2-7 was 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 provided to detect radioactive leakage or contamination from one system to another.

In RAI 9.2-13, RAI 9.2-13, S01, and RAI 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 its response, the applicant stated that intersystem leakage in the RCCWS is monitored through three methods; radiation monitoring (reference DCD Tier 2, Sections 9.2.2.5 and Section 11.5.3.2.7), RCCWS flow rate, and high level alarm from the head tank (reference 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, inleakage 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, RCCWS radiation monitoring and system gross leakage was a topic of discussion 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 the 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 RCCWS to determine if an automatic train separation is necessary.

During the audit it was noted that DCD Tier 2, Figure 9.2-2b illustrates a radiation detector downstream of the A Train RWCU/SDC heat exchangers; however, and 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 has incorporated this RAI proposed change.

The RCCWS surge tank levels are used to monitor losses of cooling water, and detect intersystem leakage intrusions into RCCWS. The level transmitters in the surge tank standpipes in combination with low-low surge tank level automatically initiate 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. In its response, the applicant clarified how the previous response to RAI 9.2-13 S02 and RAI 9.2-24 had adequately addressed these concerns. In the RAI 9.2-27 response, it was summarized that the RCCWS has radiation monitors at the discharge of the RWCU/SDC heat exchangers and will 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 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, “Regulatory Guide 4.21 Design Objective and Applicable DCD Subsection Information,” describes similar provisions related to RCCWS 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)

The staff finds that these design provisions for the RCCWS meets the requirement of 10 CFR 20.1406 and conforms to the guidelines of RG 4.21. Section 12.3 of this report further addresses the ESBWR design in accordance with 10 CFR 20.1406. The staff determined the responses to RAI 9.2-13, RAI 9.2-27, and RAI 9.2-24 as it relates 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, RAI 9.2-13, RAI 9.2-27, and RAI 9.2-24 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, Appendix 19A, Section 19A.8.3, “Augmented Design Standards,” indicates that RTNSS Criterion C systems incorporate the DID principles of redundancy and physical separation to ensure adequate reliability and availability. Section 19A.8.3, “Augmented Design Standards,” 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, Section 19A.8.3 indicates that RTNSS Criterion C equipment is qualified to The Institute of Electrical and Electronics Engineers, Inc. (IEEE) Standard 344-1987, “Recommended Practice for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations-Description,” only to demonstrate structural integrity. Also, RTNSS C systems and components are designed to the seismic standards of IBC-2003 consistent with the above SSE ground motion.

As stated in the applicant’s response to RAI 9.2-24, the PSWS and RCCWS support plant investment protection (PIP) and defense-in-depth. DCD Section 9.2.2.2 describes 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. 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 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 FAPCS in providing a backup source of lower pressure injection and suppression pool cooling. 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 equal to two-thirds of the Certified Seismic Design Spectra. The staff concludes this graded design approach is acceptable considering the design function of the PSWS under the regulatory criteria for this non safety 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 key points that were included in this RAI included:

- 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.

In support of resolution of 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 results of the audit and the RAI response are discussed throughout the remainder of this section.

<u>(1) 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 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

The applicant through the March 19-20, 2009 audit and RAI response addressed all of the above items. At the audit, the supplied system diagrams were reviewed for piping sizes and adverse system velocities. The staff determined 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 was 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 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 FAPCS for cooling of the suppression pool to mitigate boiling. Accordingly, 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 increased velocities associated with failures of RCCWS flow paths were viewed by the applicant as a highly unlikely event; however, the potential higher system velocities will be allowed to exceed the recommended velocities.

In the RAI response, it was discussed that valves are usually provided with hard seats to withstand erosion due to 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.

RCCWS strainers were also discussed at the audit and in the RAI 9.2-24 response. The RCCWS is a closed system with clean de-mineralized water that is 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 concludes that the normal RCCWS system velocities had been adequately addressed and discussed. The staff finds them acceptable since the most limited piping velocities were approximately 4.6 meters per second (15 feet per second) or less. In the staff's experience and in accordance with general engineering practice, piping velocities between 1.2 and 4.6 meters per second (4-15 feet 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 concludes this is an unlikely event associated with failures of RCCWS flow paths considering all the design features of the RCCWS and the QA measures for this system being RTNSS Criterion C. The remaining items noted above were reviewed by the staff as part of the RAI response. The staff determined that the response to RAI 9.2-24 regarding RCCWS descriptive information and flow considerations was 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.

<u>(2) Heat Transfer</u>

In RAI 9.2-20, the staff asked the applicant to explain an inconsistency in DCD Revision 4, Tier 2, Table 9.2-3 "RCCWS Nominal Heat Loads," regarding the chilled water system (CWS) heat load of 12.3 MW (41.9 MBTU/hr) applicable for Train A only. In its response, the applicant stated that a note would be added to DCD Revision 5, Table 9.2-3 to clarify for the 12.3 MW (41.9 MBTU/hr) 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 determined that the RAI response was 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 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 9.2-3, "RCCWS Nominal Heat Loads," provides a listing of RCCWS 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 is needed in this regard and requested in RAI 9.2-24 that the applicant revise DCD Section 9.2.2 to address heat transfer and address 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/hr). 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 in order to bring the plant to cold shutdown condition within 36 hours. The most limiting condition for 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/hr) divided between two heat exchangers. Each RCCWS heat exchanger is designed for 30.6 MW (104 MBTU/hr).

Based on the above, the staff concludes that for the two bounding condition noted above that there is sufficient design margin between the RCCWS heat exchangers capacity and the maximum heat loads. In addition, for support of RTNSS only (FAPCS, CWS and diesel generator), 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 and the review at the audit, 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 RAI 9.2-24 response provided a DCD mark-up related to the clarification 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 with 36 hours. In addition, the applicant's response to RAI 9.1-20 S04 dated August 14, 2009 provided a markup to the FAPCS heat loads in Table 9.2-3 which were reduced by 1.3 MW (4.5 MBTU/hr). The staff confirmed that Revision 6 of the DCD has incorporated these RAIs proposed changes. The staff determined that the response to RAI 9.2-24 regarding heat transfer was 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% capacity trains with each train consisting of a total of three pumps and three RCCWS heat exchangers cooled by PSWS. The pumps in each train are powered from separate buses. During a LOPP, the pumps are powered from the two nonsafety-related standby diesel-generators. 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 air operated valves, they can be split out if necessary from the control room. The staff determined that clarification was needed for when off-site 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% or above 125% of its best efficiency point. RCCWS cooling water train supply valves (DC backed motor operated) automatically close upon a LOPP to prevent RCCWS pump runout and ensure sufficient cooling for the standby diesel generators. These valves are opened after the diesel generators 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 Tier 2 DCD Section 9.2.2.2 related to the DC motor-operated valves. The staff confirmed that Revision 6 of the DCD has incorporated this RAI proposed change. The staff determined this change is acceptable since the applicant clarified the signals used to close the DCD motor-operated valves.

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 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 and/or control or manual valves.

Air operated valves 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 air operated valves 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 air-operated valves are normally open, fail open valves. RCCWS heat exchanger bypass valves are fail closed 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 main control room (MCR) non-safety-related instrumentation and controls distributed controls and information system (N-DCIS). The valves are normally open and automatically close upon a train separation event and fail close. There are two automatic train separation signals 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 Tier 2 DCD Section 9.2.2.2 related to separation signals. The staff concluded that Revision 6 of the DCD has incorporated this RAI proposed change and the staff determined this change is acceptable since the details of RCCWS cross-tie valves and train separation signals have been adequately described and added to 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 fire protection system. Extended makeup water supply additions indicate that there is a leak in the RCCWS, and the cooling water trains should be separated, and the damaged train repaired. The separation of trains due to extended makeup water supply addition is a manually initiated event. The RCCWS surge tank makeup water inlet block valves fail close.

The RCCWS diesel generator cooling water return valves are air-operated block valves (AOV), 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, not RCCWS. Control of these valves by 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 air-operated valve will not impact the other train. As described in DCD Section 9.3.6, "Instrument Air System," (IAS) this system is designed to ensure that failure does not compromise any safety related system or component nor does it prevent a safe shutdown.

Based on the staff review of the applicant's response to RAI 9.2-24, the staff concludes that single failure consideration have been properly addressed due to the redundancy of the design, components emergency power supply availability, and components failure position on a LOPP. The design redundancy of the RCCWS provides for adequate system reliability. In addition, train independence ensures that single failure of any air-operated valve will not impact the other train. The staff determined that the response to RAI 9.2-24 regarding single failure and backup power was acceptable since the applicant clarified how the single-failure and backup power attributes of the RCCWS are included in the DCD. Based on the above, the applicant's responses and DCD changes, the single failure aspects of RAI 9.2-24 are resolved.

(4) RCCWS Pump Net Positive Suction Head (NPSH)

As described in DCD Tier 2, 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. Makeup to the RCCWS inventory is from the makeup water system (MWS) through an automatic level control valve. A manual valve provides a backup source of makeup from the fire protection system.

The staff requested in RAI 9.2-23 and RAI 9.2-24 that the applicant address NPSH and additional questions including addressing design alarms features in the main control room available to the operators.

In its 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 initiates 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 shall 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 its response to RAI 9.2-24, the applicant modified Revision 6 of the DCD Tier 2, Section 9.2.2.2 and 9.2.2.5 to add a description of 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 has incorporated this RAI proposed change.

The staff determined that the responses to RAIs 9.2-23 and 9.2-24 regarding RCCWS pump NPSH were acceptable since the applicant clarified how sufficient NPSH is assured. The applicant clarified the design features of the RCCWS to assure NPSH, including the RCCWS surge tank and its system position (high point of the system), instrumentation which detections 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

Generic Letter (GL) 96-06, "Assurance of Equipment Operability and Containment Integrity During Design-Basis Accident Conditions," was issued to address the potential for (1) water hammer and/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 also 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, Section 9.2.2.4, "Testing and Inspection Requirements," 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 DID functions over the life of the plant. The RCCWS design bases indicate that provisions are included to permit inspection of components and equipment. Also, the system description indicates that valves are arranged for ease of in-service inspection. Section 9.2.1.4, "Testing and Inspection Requirements," 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 Section 9.2.2.

In the applicant's response to RAI 9.2-24, it was 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 Appendix 19A.8, "Proposed Regulatory Oversight," and 19A.8.4.9, "Component Cooling – HVAC, Cooling Water, Chilled Water, and Plant Service Water," 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 RCCWS PRA Model (NEDO 33201 Rev 3) assumes 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 standby diesel LOPP testing.

The staff determined that the RAI response was 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 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.

<u>(7) Instrumentation, Controls, and Alarms</u>

DCD Tier 2, Section 9.2.2.5, "Instrumentation Requirements," indicates that the RCCWS is operated and monitored from the main control room. Major system parameters, which include loop flow rates, 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, "Remote Shutdown System," states that control of two RCCWS trains and two PSWS trains is provided on the remote shutdown system (RSS) panel.

In RAI 9.2-6 and RAI 9.2-6 S01, the staff determined that the simplified diagrams of Figures 9.2-1 and 9.2-2 in DCD Revisions 2 did not have sufficient details and requested that the applicant include system 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 its 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 Revision 3, 4 and 5, more details in the existing simplified diagrams of 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 in RAI 9.2-24, the staff specifically asked the applicant to include the header temperature and pressure detectors in the diagrams.

The staff determined that the response to RAIs 9.2-6 and 9.2-24 as it relates to simplified diagrams was acceptable since the supplemental information in the revised Figures 9.2-1, 9.2-1a, 9.2-2b supports the RCCWS RTNSS functions. The staff also determined 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 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 in DCD Revision 6 has corrected this omission 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, there are two automatic train separation signals 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 its response to RAI 9.2-24, the applicant modified Revision 6 of the 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 Section 13.5.3; “COL Information,” COL 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 American National Standards Institute (ANSI)/ ANS-3.2. RG 1.33, “Quality Assurance Program (Operations),” endorses 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 above generic COL information item might not cover non-safety-related systems such as PSWS and RCCWS in the ESBWR. In response to RAI 9.2-11 S04, the applicant revised Section 13.5.2 of the DCD to clarify that the water hammer procedures for the RTNSS systems will be included as a part of COL Item 13.5-2-A. Therefore, the staff finds COL Item 13.5-2-A acceptable regarding procedure development for the RCCWS.

The applicant identified no other COL Information Items in Section 9.2.2.6, “COL Information.” The staff concludes that there are no relevant COLs 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 7, Appendix 19A, Section 19A.8.1, “Regulatory Oversight – Availability Controls,” regulatory oversight is applied to each system that is designated as RTNSS to ensure adequate reliability and availability to perform RTNSS functions. 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 Availability Controls Manual (ACM). Appendix 19A, Table 19A-2, “RTNSS Functions,” identifies that the PSWS and RCCWS are support systems and that the PSWS and RCCWS ‘Availability Controls’ are the ‘Maintenance Rule,’ which means that the availability of the PSWS and RCCWS is addressed by the Maintenance Rule performance monitoring rather than by a specific ACM entry.

The PSWS and RCCWS are subject to the ACM through the systems they support. Table 19A-2 classifies the PSWS and the RCCWS as support systems for the standby diesel generators (SDGs) and the NICWS. NICWS supports building heating, ventilation, and air conditioning (HVAC), which supports the fuel and auxiliary pools cooling system (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 (ACs) for the SDGs in AC 3.8.1, "Standby Diesel Generators – Operating," and AC 3.8.2, "Standby Diesel Generators – Shutdown;" and for 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 and RCCWS are support systems that are subject to the ACs that are specified for the SDGs and FAPCS.

ACM 1.1, "Definitions", 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(s) 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(s) are also capable of performing their related support function(s). Since PSWS supports RCCWS which supports NICWS, FAPCS, and SDGs, if PSWS/RCCWS becomes unavailable, then the system in which is support becomes unavailable and the applicable ACM action statement would then apply.

Based on the above, the staff finds the availability controls for the RCCWS acceptable since the RCCWS is subject to the maintenance rule and indirectly subject to the ACM via the RCCWS being a RTNSS support system and its availability being indirectly covered by the availability controls for the FAPCS and SDGs.

9.2.2.3.4 Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC)

In DCD Tier 1, Revision 3, Section 2.12.3, "Reactor Component Cooling Water System," 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, "Criteria and Application Process," indicates that RTNSS systems shall 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 RAI 22.5-1 and RAI 22.5-1 S01, the staff requested that the applicant review and revise DCD Tier 1 to include 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.

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

> 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 determined that the RAI response was acceptable since the RCCWS Tier 1 information is adequate and reasonable based on the ESBWR graded approach for this RTNSS Criterion C, non-safety related system. For the RTNSS functions of the RCCWS, flow is verified to key RTNSS equipment such as chillers, FAPCS, and SDGS, as-built verification is performed, operation of selected controls from the MCR is verified, and RCCWS system flow indication is 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

The initial test program for ESBWR is evaluated in Section 14.2 of this report, and 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 7, Section 14.2.8.1.21, "Reactor Component Cooling Water System Preoperational Test," describes the preoperational test program for the RCCWS. The staff finds the objective of the RCCWS preoperational test program to be appropriate since it is to verify proper operation of the RCCWS 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 preoperational 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 RCCWS, this approach for this non safety 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, "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 Section 14.2.8.1.21 to address the testing of the RCCWS. In its response to RAI 9.2-24, the applicant clarified the basis for its preoperational 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 determined that the RAI 9.2-24 response was acceptable since the level of testing addresses system performance, minimum NPHS, instrumentation and interlocks, and water hammer and no additional testing needs to be described in DCD Tier 2, Section 14.2. 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 concludes that the RCCWS 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 RCCWS satisfies NRC policies that have been established with respect to its RTNSS C function.

9.2.3 Makeup Water System

9.2.3.1 Regulatory Criteria

The staff reviewed the makeup water system (MWS) based on the guidance provided in NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants" (hereafter referred to as the SRP) Section 9.2.3, Revision 2, "Demineralized Water Makeup System," issued July 1981. Staff acceptance of the design is based on meeting the requirements of General Design Criteria (GDC) 2 and 5:

- GDC 2, "Design Basis for Protection Against Natural Phenomena," as it relates to the capability of the design to maintain and perform its safety function following an earthquake.

- GDC 5, "Sharing of Structures, Systems and Components," as it relates to the capability of shared systems and components important to safety to perform required safety functions.

9.2.3.2 Summary of Technical Information

DCD Tier 2 Revision 6, Section 9.2.3, "Makeup Water System," 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, 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 the portions of Regulatory Position C.1 of RG 1.29 for the safety-related portions of the system and Regulatory Position C.2 of RG 1.29 for the nonsafety-related portions of the system.

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 non-safety-related system. SRP Section 9.2.3 indicates that the requirements of GDC 2 can be met for a non-safety-related system based on meeting Regulatory Position C.2 of RG 1.29 regarding non-safety-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 its 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, nor will it 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 non-safety-related systems because the failure of the non-safety-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.

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 RAI 22.5-19 and RAI 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 fire protection system (FPS) can provide RTNSS Criterion B seismic makeup source to the RCCWS and CWS. Based on the above, the staff finds that the MWS does not need to be a RTNSS Criterion B system. The staff determined that the RAI response was 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 conceptual design information 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 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. Conceptual design information (CDI) and COL items are addressed separately in the DCD. The DCD Tier 2, Section 1.8.2 provides a summary of the BOP Interfaces and references some DCD Tier 2 Sections where CDI information could possibly be found. Also, the COL items are provided in DCD Tier 2 Section 1.10. RG 1.206 discusses the need for COL applicants to address CDI "in addition" to addressing COL 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 items to the CDI in the DCD, since the need to address this information is specified in RG 1.206. The staff determined that the response was acceptable since the justification for not having a COL 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 concludes that the design of the MWS is acceptable and meets the requirements of GDC 2. 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 NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants" (hereafter referred to as the 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 requirements of General Design Criterion (GDC) 60:

- 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.

9.2.4.2 Summary of Technical Information

DCD Tier 2, Section 9.2.4, "Potable and Sanitary Water Systems," 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 l/s (200 gpm) of potable water during peak demand periods. The potable and sanitary water systems will meet GDC 60 for provisions provided 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 above conceptual design information for the potable and sanitary water systems will be replaced with site-specific design information in the COLA FSAR.

9.2.4.3 Staff Evaluation

The applicant states that the site-specific design information will be provided in the COLA FSAR, and the DCD only provides the conceptual design information (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 system in accordance with SRP Section 9.2.4 at the COLA stage. The design will be evaluated 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 conceptual design information for the potable and sanitary water systems will be replaced with site-specific design information in the COLA FSAR. 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 determined that the response to RAI 9.1-18 was 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.

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 relating to GDC 60.

9.2.5 Ultimate Heat Sink

9.2.5.1 Regulatory Criteria

The staff reviewed the ultimate heat sink (UHS) based on the guidance provided in NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants" (hereafter referred to as the SRP) Section 9.2.5, Revision 3, "Ultimate Heat Sink," issued March 2007. Staff acceptance of the design is based on meeting the requirements of General Design Criteria (GDC) 2, 5, 44, 45, and 46:

- GDC 2, "Design Basis for Protection Against Natural Phenomena," as it relates to the capability of the design to maintain and perform its safety function following an earthquake.

- GDC 5, "Sharing of Structures, Systems and Components," as it relates to the capability of shared systems and components important to safety to perform required safety functions.

- GDC 44, "Cooling Water," 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.

9.2.5.2 Summary of Technical Information

DCD Tier 2 Revision 6, Section 9.2.5, “Ultimate Heat Sink,” Section 5.4.6, “Isolation Condenser System (ICS),” Section 6.2.2, “Passive Containment Cooling System,” describes the UHS. The UHS consists of the isolation condenser (IC) and the passive containment cooling (PCC) pools, the dryer/separator pool and reactor well, fire protection system (FPS) makeup water for the IC/PCC pools, and spent fuel pool (SFP) from the primary (seismic Category I) firewater storage tanks via the safety-related fuel and auxiliary pools cooling system (FAPCS) piping, and other water sources that are credited for providing makeup water for the IC/PCC pools and SFP after water from the firewater storage tanks has been depleted. The dryer/separator pool and reactor well provide sufficient makeup water for the IC/PCC expansion pools to support operation of the IC system and PCC system 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 upon 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 IC/PCC 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 DCD Tier 2 Figure 6.2-10c series of decay heat curves. Therefore, the minimum total makeup water flow rate beyond 72 hours, as well as beyond seven 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 Table 9.5-2. DCD Tier 2 Section 9.1.3.2 discusses the use of the FAPCS to provide water after 72 hours post-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, Revision 3, “Ultimate Heat Sink,” dated March 2007. Staff acceptance of the UHS is based on meeting the requirements of GDC 2, 5, 44, 45, and 46.

In DCD Tier 2, Revision 6, Section 9.2.5, the applicant stated that the IC/PCC 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 have 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 IC/PCC pools have reserve capacity for 72 hours of heat removal without makeup. The parts of the UHS that are relied upon for the first 72 hours following an accident are safety-related and are evaluated in Sections 5.4.6, “Isolation Condenser System (ICS),” and 6.2.2, “Passive Containment Cooling System,” of this report. The parts of the UHS that are relied upon for providing makeup water during the period from 72 hours through seven days post-accident are not safety-related, but are readily available on-site and are subject to RTNSS as discussed in Chapter 19A of the DCD, Revision 7. Section 22.5.6, “Post-72-Hour Actions and Equipment,” of this report provides the staff evaluation. The FPS provides post-accident makeup to the IC/PCC pools through safety-related FAPCS piping. DCD Tier 2, 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 reactor building (RB) and fuel building (FB) or onsite makeup sources. The external connection and emergency makeup water piping is part of the FAPCS and is discussed in Section 9.1.3 of this report.

This section evaluates the adequacy of the capability that is credited for providing makeup water to the IC/PCC pools and 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. A constant of 200 gpm at the 72 hours, but not beyond 72 hours, was specified in DCD Tier 2 Table 9.5-2. 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 determined that the response was 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 IC/PCC pools and SFP beyond 7 days following an accident. DCD Tier 2 Revision 6, Section 9.2.5.1, identifies this as COL Item, COL 9.2.5-1-A, "Post Seven day makeup to UHS," which states:

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 Item 9.2.5-1-A acceptable since makeup sources after seven days are expected to be site-specific. The staff will review this information during the COL application process.

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 concludes that the design of the UHS is acceptable and meets the requirements of GDC 2, 44, 45, and 46. The staff will review COL 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 condensate storage and transfer system (CS&TS) based on the guidance provided in NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants" (hereafter referred to as the SRP) Section 9.2.6, Revision 3, “Condensate Storage Facilities,” issued March 2007. Staff acceptance of the design is based on meeting the requirements of General Design Criteria (GDC) 2, 5, 44, 45, and 46:

- GDC 2, “Design Basis for Protection Against Natural Phenomena,” as it relates to the capability of the design to maintain and perform its safety function following an earthquake.

- GDC 5, “Sharing of Structures, Systems and Components," as it relates to the capability of shared systems and components important to safety to perform required safety functions.

- GDC 44, “Cooling Water,” 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.

- GDC 60, “Control of Releases of Radioactive Materials to the Environment,” as it relates to tanks and systems handling radioactive material in liquids.

9.2.6.2 Summary of Technical Information

DCD Tier 2, Section 9.2.6, “Condensate Storage and Transfer System,” 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, for condensate-quality liquid waste management system effluent during normal operation, and for condensate and feedwater system and condenser hotwell inventory during system maintenance outages.

The CST also provides a minimum storage capacity for the control rod drive system as a reserve water source for reactor pressure vessel (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 reactor building (RB), turbine building (TB), and fuel and radwaste buildings. 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, "Condensate Storage Facilities." Staff acceptance of the CS&TS is based on meeting the requirements of GDC 2, 5, 44, 45, 46, and 60.

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 structures and systems' being capable of withstanding the effects of natural phenomena, acceptance depends on meeting the guidance of the portions of Regulatory Position C.1 of RG 1.29 for the safety-related portions of the system and Regulatory Position C.2 of RG 1.29 for the non-safety-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 non-safety-related system. Based on SRP Section 9.2.6, a non-safety-related system satisfies the requirements of GDC 2, by meeting the guidance of the portions of Regulatory Position C.2 of RG 1.29 regarding non-safety-related systems. In its 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, nor will it 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 non-safety-related systems because the failure of the non-safety-related portions of the systems does not impact any safety-related SSCs, and 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, 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 Paragraph II.3.A, B, and 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 such as 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 Regulatory Guide 1.143. DCD Tier 2 Section 9.2.6.1 states that the CS&TS complies with RG 1.143 Position C.I.2 for provisions to prevent uncontrolled releases of radioactive materials. DCD Tier 2, Table 12.3-18, "Regulatory Guide 4.21 Design Objective and Applicable DCD Subsection Information," describes the provisions 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)

Corresponding provisions are discussed in DCD Tier 2, Section 9.2.6. 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 because it will include the means to reliably control the release of radioactive liquid effluents.

9.2.6.4 Conclusion

Based on the above, the staff concludes that the design of the CS&TS is acceptable and meets the requirements of GDC 2 and 60.

9.2.7 Chilled Water System

9.2.7.1 Regulatory Criteria

The staff reviewed the chilled water system (CWS) based on guidance provided in NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants" (hereafter referred to as the SRP) Section 9.2.2, Revision 4, "Reactor Auxiliary Cooling Water System," issued March 2007. 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:

- General Design Criteria (GDC 2), “Design Basis for Protection Against Natural Phenomena,” as it relates to the capability of the design to maintain and perform its safety functions following an earthquake.

- GDC 4, “Environmental and Dynamic Effects Design Bases,” as it relates to the dynamic effects associated with water hammer.

- GDC 5, “Sharing of Structures, Systems and Components," as it relates to the capability of shared systems and components important to safety to perform required safety functions.

- GDC 44, “Cooling Water,” as it relates to transferring heat from structures, systems, and components (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.

The CWS is a non-safety-related system; however, the system provides defense-in-depth (DID) for the ESBWR passive plant design. In addition to the SRP guidance, the NRC staff’s evaluation of DID 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, “Regulatory Treatment of Non-Safety Systems,” of this report; (2) confirming that failure of DID systems and components will not adversely impact safety-related SSCs; (3) confirming that availability controls are established as appropriate; and (4) confirming that proposed inspections, tests, analyses, and acceptance criteria (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, “Chilled Water System,” describes the CWS. The CWS consists of two independent subsystems, the nuclear island chilled water subsystem (NICWS) and the balance-of-plant chilled water subsystem (BOPCWS). The CWS provides chilled water to the cooling coils of air-handling units and other coolers in the reactor building (RB), containment vessel (CV), the turbine building (TB), the control building (CB), radwaste building (RW), electrical building (EB), and fuel building (FB). The chilled water absorbs the rejected heat from these coolers and is pumped through the chillers where the heat is transferred to RCCWS.

The NICWS consists of two 100% capacity trains, with redundancy and independence for active components. The BOPCWS consists of one 100% capacity independent loop with crossties to the NICWS chilled water piping. DCD Tier 2, Table 9.2-11, “Chilled Water System Component Design Characteristics,” lists the CWS component design characteristics. DCD Tier 2, Figure 9.2-3, “Chilled Water System Simplified Diagram,” shows the CWS simplified diagram. DCD Tier 2, Table 3.2.1, “Classification Summary,” indicates that part of the CWS (P25) is a safety related, quality group “B” and seismic category I at the containment penetration between the reactor building (RB) and the containment vessel (CV). Part of the CWS is non-safety related, quality group “D” and seismic category II which is located in the RB and CV. The balance of the CWS is located in various parts of the TB, FB, EB, CB and RW and is non-safety related, quality group “D” and non seismic.

While the NICWS is a non-safety-related system, it performs DID functions and is also subject to RTNSS as described in DCD Tier 2 Appendix 19A, “Regulatory Treatment of Non-Safety Systems.” As stated in DCD Tier 2 Section 19A.4.2, “Assessment of Uncertainties,” 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 fuel and auxiliary pools cooling system (FAPCS) that provide low pressure injection and suppression pool cooling 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. The CWS differs from that of the traditional BWR designs in that the ESBWR CWS removes heat only from non safety-related areas. Therefore, the portions of SRP Section 9.2.2 that apply to safety-related systems do not apply to the CWS.

Safety related portions of the CWS at the containment penetrations are described in DCD Table 6.2-47, “Containment Penetrations Subject To Type A, B, and C Testing,” and Section 6.2.4.3.2.1, “Influent Lines to Containment.” The staff evaluation of the containment penetration is provided in Section 6.2.4, “Containment Isolation System,” of this report.

9.2.7.3.1 System Design Considerations

On March 19-20, 2009, the staff conducted a regulatory audit of the supporting information for the ESBWR DCD, Section 9.2, “Water Systems,” including the plant service water system (Section 9.2.1), reactor component cooling water system (Section 9.2.2) and nuclear island chilled water subsystem (Section 9.2.7). The audit was primarily focused on the review of these systems in 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

Section 3.2, “Classification of Structures, Systems and Components,” specifies the classification of SSCs based on safety importance and other considerations. The staff’s evaluation of the classification designations that are specified is provided in Section 3.2 of this report, and 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 Section 3.2 and that the designations properly reflect the regulatory oversight provisions that pertain to CWS/NICWS (RTNSS Criterion C) as discussed in Appendix 19A, Section 19A.8, “Proposed Regulatory Oversight.” The staff reviewed simplified drawings, shown in Figure 9.2-3, and confirmed that the classification designations on the drawings are consistent with those that are listed for CWS in Table 3.2-1, “Classification Summary.” In particular, the following classification designations are specified in Table 3.2-1 for CWS:

- Part of the CWS/NICWS is designated as Safety Class II, Seismic Category I at the CV and RB interface which forms part of the containment boundary. This portion of the CWS is designated Quality Group B. As discussed in Section 3.2.4, "Quality Group B," 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 B&PV Code, Section III. Containment penetrations of the CWS (P25) are described in DCD Table 6.2-47, "Containment Penetrations Subject To Type A, B, and C Testing," and Section 6.2.4.3.2.1, "Influent Lines to Containment." Section 6.2 of this report evaluates CWS containment penetrations. The balance of the CWS is designated Safety Class N which is used for non-safety-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 Section 3.2.4, "Quality Group D," this quality group generally applies to non-safety-related SSCs that satisfy specified industry codes and design standards and are subject to one or more significant licensing requirement or commitment. 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.

- Part of the CWS is a non-safety-related system located in the RB and CV 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 main control room, are designated Seismic Category II. These items are designed to structurally withstand the effects of a safe shutdown earthquake (SSE). Other portions of the CWS are located in the TB, RB, FB, RW, CB and EB are designated as non seismic. 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 reactor building and it is designed to withstand an SSE, its structural failure will not affect the safety function of any safety system or the main control room occupants.

- Quality Assurance (QA) Requirement S is specified for the CWS in Revision 6 of the DCD as stated in the applicant's response to request for information (RAI), RAI 3.2-6 S02. Based on the RAI response, RTNSS components/systems that were identified under Revision 5 of the DCD as QA Requirement E are to be changed to QA Requirement S under 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 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 concludes that Revision 6 of the DCD has incorporated this RAI proposed change and the staff determined this change is acceptable.

B. GDC 2

Section 6.2.4 of this report evaluates containment isolation valves. Other than the containment isolation, the CWS is a non-safety-related system. SRP Section 9.2.2 indicates that the requirements of GDC 2 can be met for a non-safety-related system based on meeting Regulatory Position C.2 of RG 1.29 regarding non-safety-related systems.

As a part of RAI 9.2-12 and RAI 9.2-12 S01, the staff asked the applicant to demonstrate that the CWS meets GDC 2. In its response, the applicant stated that 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 a 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, nor will it 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 Position C.2 of RG 1.29 regarding the nonsafety-related portions of the CWS. Therefore the CWS satisfies GDC 2. Accordingly, ased 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, "Environmental and Dynamic Effects Design Bases," as it relates to the dynamic effects associated with water hammer. As stated in the 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 effects of missiles, jet impingement, pipe whipping and discharge fluids are addressed by the following design consideration;

- 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 its response, the applicant stated that the CWS is designed to mitigate the possibility of water hammer as addressed in its responses to RAI 9.2-15 and 9.2-15 S01.

The CWS/NICWS is a 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 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 its response, the applicant stated that proper system engineering design, along with operation and maintenance procedures are used to ensure sufficient measures are taken to avoid water hammer. Surge tanks and air separators mitigate voiding. Surge tanks are also used per DCD Tier 2, Section 9.2.7.2, "System Description," 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, "Operating and Maintenance Procedures," the applicant provided a clarification to state that elements of ANSI/ANS-3.2-1994; R1999, "Administrative Controls and Quality Assurance for the Operational Phase of Nuclear Power Plants," addressing water hammer shall be applied in the development of procedures for RTNSS systems.

Water hammer considerations were a topic for discussion at the March 19-20, 2009 audit and were addressed in the applicant's response to RAIs 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 its response, the applicant provided the following provisions to mitigate water hammer:

- System design and layout features: This includes each NICWS Train (A and B), each of which have 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/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 RAI 9.2-15 and RAI 9.2-24, the staff concludes that water hammer has been adequately addressed since the CWS/NICWS 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 Item 13.5-4-A. Accordingly, based on the above and the applicant's responses, RAI 9.2-15, 9.2-21, and RAI 9.2-24 as they relate to water hammer are resolved. The staff concludes 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 SRP 9.2.2 guidance, 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 loss of offsite power;

- Capability to isolate components so system function is not compromised; and

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

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

The staff reviewed the CWS/NICWS 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 Section 9.2.2, and determined that CWS/NICWS satisfies GDC 44, 45, and 46 with respect to its normal operation function. However, in a design-basis accident, decay heat is transferred to the isolation condenser/passive containment cooling (IC/PCC) 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 concludes 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. The staff's review criteria (SRP Section 9.2.2, Paragraph III.4.C) specify that provisions should be provided to detect radioactive leakage or contamination from one system to another.

DCD 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 Section 9.2.7.4, "Testing and Inspection Requirements," 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, minimization of contamination, since the DCD did not adequately discuss this issue or explain in detail the CWS operating pressures relative to the system coolers. In its response, the applicant clarified that the offgas cooler-condenser operates at less than 138.9 kilopascal (20 psig) and the CWS maximum operating pressure is approximately 861.8 kilopascal (125 psig) with a nominal pressure greater than 413.7 kilopascal (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, Table 11.3-3, "Equipment Malfunction Analysis." Since the OGS consists of two redundant trains of offgas cooler-condensers an OGS train could be isolated if leakage was 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 CWS. Low-low surge tank level will alarm in the main control room (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 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 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 Subsection 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 CWS/NICWS is classified as RTNSS Criterion C. Appendix 19A, Section 19A.8.3, "Augmented Design Standards," indicates that RTNSS Criterion C systems incorporate the DID principles of redundancy and physical separation to ensure adequate reliability and availability. Section 19A.8.3, "Augmented Design Standards," 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, Section 19A.8.3 indicates that RTNSS Criterion C equipment is qualified to The Institute of Electrical and Electronics Engineers, Inc. (IEEE) Standard 344-1987 "Recommended Practice for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations-Description," only to demonstrate structural integrity.

CWS/NICWS is RTNSS C and systems and components are designed to the seismic requirements of IBC-2003 consistent with the above 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, PSWS, RCCWS and CWS/NICWS supports plant investment protection (PIP) and defense-in-depth functions. DCD Section 9.2.2.2 describes that in the event of a loss of preferred power (LOPP), the RCCWS supports FAPCS and the reactor water cleanup/shut down cooling (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 and CWS/NICWS are also significant contributors to plant availability and plant investment protection, the ESBWR design is focused on ensuring 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 FAPCS in providing a backup source of lower pressure injection and suppression pool cooling. 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 equal to two-thirds of the Certified Seismic Design Spectra. The staff concludes this graded design approach is acceptable considering the design function of the NICWS under the regulatory criteria for this non safety 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 regulatory treatment of non-safety systems (RTNSS), focusing on CWS/NICWS capability and reliability. The key points that were included in this RAI included:

- 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 (to include valves, cross-tie connections between trains, instrumentation logic and installed instruments).

- CWS/NICWS pump design to include pump recirculation protection, vortex and NPSH

- CWS/NICWS water hammer consideration.

- CWS/NICWS failure modes and effects

- CWS/NICWS component testing and component reliability.

In support of resolution of 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 results of the audit and the RAI response are discussed throughout the remainder of this section. 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.

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

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 additional information was needed 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 class breaks

The applicant through the March 19-20, 2009 audit and RAI response addressed in detail each of the above noted items.

The CWS is divided in two independent chilled water subsystems, the NICWS and the BOPCWS. At the March 19-20, 2009 audit and in the 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 BOPCWS and NICWS. The manual valves may be opened to support maintenance activities. The applicant revised DCD Tier 2, Figure 9.2-3, “Chilled Water System Simplified Diagram,” to include these clarifications. The staff confirmed that Revision 6 of DCD Tier 2, Figure 9.2-3 has incorporated this RAI proposed change. The staff determined that the response was acceptable since the revised simplified diagram clarifies the relationship between the two subsystems.

NICWS Train A and Train B, and BOPCWS are each powered by separate buses. The active components in NICWS Train A and Train B chilled water trains are identical. Each train contains two 50% chillers, two 50% 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 meters per second (15 feet per second) or less.

At the audit, the staff questioned the missing ASME 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 reactor building) are classified nonsafety-related, Quality Group D and seismic category II. The applicant provided in the RAI response a markup of changes to the component in the reactor building indicating Quality Class B, and seismic category I component (containment penetration area) and Quality Class D, and seismic category II for the remaining CWS components inside containment consistent with DCD Table 3.2-1. The staff confirmed that Revision 6 of the DCD Figure 9.2-3 has incorporated this RAI proposed DCD change related to the CWS piping classification inside the containment. The staff determined this response was acceptable since the applicant clarified the seismic, safety, and quality classifications of the various portions of the CWS.

The staff concludes that the CWS/NICWS system velocities had been adequately addressed. The staff finds them acceptable since the most limited piping velocities were approximately 4.6 meters per second (15 feet per second) or less. In the staff’s experience and in accordance with general engineering practice, piping velocities between 1.2 and 4.6 meters per second (4-15 feet second) are reasonable, thus long term internal pipe wear is expected to be minimal. The remaining items noted above were reviewed by the staff as part of the RAI response. The staff determined that the response to RAI 9.2-24 regarding CWS descriptive information and flow considerations was 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, "Chilled Water System Component Design Characteristics," 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 additional information was needed in this regard and requested in RAI 9.2-24 that the applicant revise DCD 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 Table 9.2-11, listed a CWS chiller heat load of 4,850 Kw (16.55 Mbtu/hr) and total system heat load of 19,110 kW (6.5 x 10^7Btu/hr) based on conservative preliminary calculations. This CWS heat load was used to size chillers as input for turbine closed cooling water system (TCCWS) and RCCWS heat load calculations. NICWS and BOPCWS subsystem heat loads are considered bounding with final actual heat loads determined upon completion of heating, ventilation, and air conditioning (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% 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% additional margin to selected loads with a 10% 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 (41.9 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 was developed and their relationship to RTNSS:

CWS/NICWS

- Electrical building other (2 HVAC units); RTNSS
- Diesel generator room (2 HVAC units); RTNSS
- RCCWS room (2 HVAC units); RTNSS
- NICWS room (2 HVAC units); RTNSS
- Control building (2 HVAC units); RTNSS
- Reactor building (2 HVAC units); RTNSS
- Fuel building (2 HVAC units); RTNSS
- Technical support center (TSC) (2 HVAC units); non RTNSS
- Service air system (SAS) room (2 HVAC units); non RTNSS
- Drywell cooling system (DCS) area (2 HVAC units); non RTNSS

CWS/BOPCWS

- Radwaste building – non RTNSS
- Turbine building – non RTNSS
- Other loads – non RTNSS

Based on the above, the staff concludes there is sufficient design margin between the seven CWS chillers capacity and the maximum heat loads. Of these seven chillers, four are dedicated as RTNSS chillers. In addition, for RTNSS support, which includes FAPCS and the diesel generators, the maximum heat loads are bounded by design margin between the heat exchanger capacity and the maximum heat loads. The staff determined that the response to RAI 9.2-24 regarding heat transfer was 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) Single Failure and Backup Power Considerations

DCD Tier 2, Section 9.2.7.1, states that the NICWS has RTNSS functions as described in Appendix 19A of the DCD, 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:

- Closing cross-tie isolation valves
- Start up the chillers and pumps on standby
- Start up air handling units of NICWS scope
- 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 LOPP, the NICWS is automatically powered from two nonsafety-related onsite diesel generators.

Although the two NICWS trains are normally cross-connected the staff determined that clarification was needed for when off-site power is not available and requested in RAI 9.2-24 that the applicant revised DCD Section 9.2.7 to address single failure.

In its 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 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 leak detection and isolation system (LD&IS).

As described in DCD Section 9.3.6.1, "Design Bases," the instrument air system (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 fail-safe and do not rely on a continuous air supply under emergency or abnormal conditions. The importance of non-safety related compressed air supplies was evaluated relative to the criteria for special regulatory treatment of non-safety systems in DCD Tier 2 Appendix 19A and does not meet the criteria for special regulatory treatment.

Based on the staff review of the response to RAI 9.2-24, the staff concludes that single failure consideration have been properly addressed due to 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 determined that the response to RAI 9.2-24 regarding single failure was acceptable since the applicant clarified how the single-failure attributes of the RCCWS are included in the DCD. Accordingly, based on the above, the applicant's responses, and DCD changes, the single failure aspects of RAI 9.2-24 are resolved.

<u>(4) CWS/NICWS Pump Net Positive Suction Head (NPSH)</u>

As described in DCD Tier 2, Section 9.2.7.2, the surge tanks provide a constant pump suction head and allow for thermal expansion/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. Makeup to the chilled water inventory is from the makeup water system through an automatic level control valve to the surge tanks. In addition, DCD Tier 2, 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 asked additional questions 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 its 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 make-up 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 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 has incorporated this RAI proposed change. The staff determined this change is acceptable.

The staff determined that the responses to RAIs 9.2-23 and 9.2-24 regarding CWS pump NPSH were acceptable since the applicant clarified how sufficient NPSH is assured. The applicant clarified the design features of the CWS to assure NPSH which includes the CWS surge tank and its system position (high point of the system) and instrumentation which detections 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, "Resolutions To NUREG-0933 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 several nuclear plants experience with safety system components and control systems that resulted from a partial or total loss of HVAC systems. The applicant stated in the DCD that CWS is non-safety related and provides chilled water to the cooling coils of air conditioning units and other coolers in the reactor building portion of the plant, but has no safety-related function. In addition, the failure of CWS does not compromise any safety-related system or component, nor does it 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 DID principles of redundancy and physical separation to ensure adequate reliability and availability with proper level of oversight and additional requirements described in Appendix 19A to DCD Tier 2. Accordingly, the staff finds that New Generic Issue 143 is resolved for the CWS.

DCD Tier 2, Revision 5, Table 1C-1, "Operating Experience Review Results Summary – Generic Letters," discusses Generic Letter (GL) 96-06, "Assurance of Equipment Operability and Containment Integrity During Design-Basis Accident Conditions." This GL was issued 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 also lead to a loss of containment integrity. The applicant clarified its resolution of GL 96-06 in its response to RAI 6.2-170 and modified DCD Tier 2, Revision 5, Table 1C-1 to state that:

> 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 concluded that GL 96-06 does not apply to the CWS/NICWS since the system is isolated on a LOCA signal. GL 96-06 is further discussed in Section 6.2.2 of this report.

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

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 included design provisions to permit inspection and operational testing of components and equipment.

DCD Tier 2, Section 9.2.7.4, "Testing and Inspection Requirements," 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 DID functions over the life of the plant. The CWS/NICWS design bases indicate that provisions are included to permit inspection of components and equipment. Also, the system description indicates that valves are arranged for ease of in-service inspection. DCD Tier 2, Section 9.2.7.4, "Testing and Inspection Requirements," indicates that provisions are 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 Section 9.2.7.

In the applicant's response to RAI 9.2-24, it was 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 Appendix 19A.8, "Proposed Regulatory Oversight," and 19A.8.4.9, "Component Cooling – HVAC, Cooling Water, Chilled Water, and Plant Service Water," 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 system. Such SSCs may include RTNSS components

The staff determined that the RAI response was 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 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 9.2.7.5, "Instrumentation Requirements," indicates 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 CWS chiller controls and monitoring instruments and surge tank level instruments. The CWS may be controlled from the remote shutdown system 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 its 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 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 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 with the proposed changes to the surge tank alarms and to address NPSH. The staff confirmed that Revision 6 of the DCD has incorporated this RAI proposed change. The staff determined that the RAI response was acceptable since the applicant clarified the description of 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 Items in Section 9.2.7.6, "COL Information." The staff finds that there are no relevant COL 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 7, Appendix 19A, Section 19A.8.1, "Regulatory Oversight – Availability Controls," regulatory oversight is applied to each system that is designated as RTNSS to ensure adequate reliability and availability to perform RTNSS functions.
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 Availability Controls Manual (ACM). Appendix 19A, Table 19A-2, "RTNSS Functions," 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. Table 19A-2 classifies the PSWS and the RCCWS as support systems for standby diesel generators (SDGs) and the NICWS. NICWS supports building heating, ventilation, and air conditioning (HVAC), which supports the fuel and auxiliary pools cooling system (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 (ACs) for the SDGs in AC 3.8.1, "Standby Diesel Generators – Operating," and AC 3.8.2, "Standby Diesel Generators – Shutdown;" and for 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, RCCWS, and NICWS are support systems that are subject to the ACs that are specified for the SDGs and FAPCS.

ACM 1.1, "Definitions", 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(s) 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(s) are also capable of performing their related support function(s). Since PSWS supports RCCWS which supports NICWS, FAPCS, and SDGs, if PSWS/RCCWS becomes unavailable, then the system in which it supports becomes unavailable and the applicable ACM action statement would then apply.

Based on the above, the staff finds the availability controls 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 availability controls for FAPCS and SDGs.

9.2.7.3.4 Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC)

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 RAI 22.5-1 and RAI 22.5-1 S01 the staff requested that the applicant review and revise DCD Tier 1 to include 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, the RAI response and DCD changes, RAI 22.5-1 is resolved. The resolution of these RAIs is also discussed in Section 22 of this report.

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

> 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 determined that the RAI response was acceptable since the Tier 1 information is adequate and reasonable based on the ESBWR graded approach for this RTNSS Criterion C, non-safety related system. For RTNSS functions of the NICWS, flow is verified to key RTNSS equipment such as electrical building HVAC units, diesel generator room HVAC units, RCCWS room HVAC units, NICWS room HVAC units control building HVAC units, reactor building HVAC units, and fuel building 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

The initial test program for ESBWR is evaluated in Section 14.2 of this report, and 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 7, Section 14.2.8.1.24, "Chilled Water System Preoperational Test," describes the preoperational test program for the CWS. The staff finds the objective of the CWS preoperational test program to be appropriate since it is to verify the ability of the chilled water system 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 preoperational 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, this approach for this non safety 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, "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 Section 14.2.8.1.24 to address the testing of the CWS. In its response to RAI 9.2-24 and in discussions during the March 19-20, 2009 audit, the applicant clarified the basis for its preoperational 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 Figure 9.2-3, "Chilled Water System Simplified Diagram" 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 has incorporated this RAI proposed change.

As previously stated, in DCD Tier 2, Section 13.5.2, “Operating and Maintenance Procedures,” the applicant provided a clarification to state that elements of ANSI/ANS-3.2-1994; R1999 addressing water hammer shall be applied in the development of procedures for RTNSS systems.

The staff determined that the RAI 9.2-24 response was acceptable since the level of testing addresses system performance, minimum NPHS, chiller and pump performance, instrumentation and interlocks, and water hammer and no additional testing needs to be described in DCD Tier 2, Section 14.2. 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 concludes that the CWS complies with the requirements of GDC 2, 4, 44, 45 and 46. The staff also finds that the design of the CWS/NICWS satisfies NRC policies that have been established with respect to its RTNSS C function.

9.2.8 Turbine Component Cooling Water System

9.2.8.1 Regulatory Criteria

The staff reviewed the Turbine Component Cooling Water System (TCCWS) based on the guidance provided in NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants” (hereafter referred to as the SRP) Section 9.2.2 Revision 4, “Reactor Auxiliary Cooling Water System,” issued March 2007. Staff acceptance of the design is based on meeting the requirements of General Design Criteria (GDC) 2, 4, 5, 44, 45, and 46:

- GDC 2, “Design Basis for Protection Against Natural Phenomena,” as it relates to the capability of the design to maintain and perform its safety function following an earthquake.

- GDC 4, “Environmental and Dynamic Effects Design Bases,” as it relates to the dynamic effects associated with water hammer.

- GDC 5, “Sharing of Structures, Systems, and Components," as it relates to the capability of shared systems and components important to safety to perform required safety functions.

- GDC 44, “Cooling Water,” 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.

9.2.8.2 Summary of Technical Information

DCD Tier 2, Section 9.2.8, "Turbine Component Cooling Water System," 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, 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 non-safety-related PSWS. The system is designed to Quality Group D.

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 non-safety-related system because the TCCWS removes heat only from the non-safety-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 non-safety-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 non-safety-related systems.

As a part of RAI 9.2-12, RAI 9.2-22, and RAI 9.2-22 S01, the staff asked the applicant to demonstrate that the TCCWS meets GDC 2. In its response, the applicant stated that the TCCWS is a non-safety related, non-RTNSS system. TCCWS is not relied upon to transfer heat from SSCs that are safety-related or RTNSS. Its failure will not prevent the performance of any safety function or result in any incapacitating injury to occupants of the main control room. The staff determined that Position C.1 of RG 1.29 is not applicable to TCCWS. Based on the information that TCCWS is not relied upon to transfer heat from SSCs that are safety-related or RTNSS and that its failure will not prevent the performance of any safety function or result in any incapacitating injury to occupants of the main control room, the staff determined that 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, RAI 9.2-12 and RAI 9.2-22 as related to GDC 2 for the TCCWS are resolved.

The staff reviewed the TCCWS and issued RAI 9.2-12 S01, and RAI 22.5-2 to see if the applicant had properly determined whether the TCCWS is a 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 its responses to RAI 9.2-12 S01 and RAI 22.5-2 that after a reevaluation of the RTNSS, the applicant changed its determination that the TCCWS is not an RTNSS system because it does not remove heat from any safety-related systems or from other RTNSS systems. The applicant updated this information in DCD Tier 2, Revision 6 and 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 determined that the responses RAI 9.2-12 and RAI 22.5-2 were 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, RAI 9.2-12 and RAI 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 RAI 9.2-7 S01 and RAI 9.2-7 S02, the staff questioned the applicant on the TCCWS meeting GDC 44, 45, and 46 requirements. In the responses, the applicant stated that TCCWS is not required to meet NRC regulations, safety goal guidelines, and containment performance goal objectives. Additionally, TCCWS is not a system used to transfer heat from structures, systems, and components important to safety that are RTNSS or safety-related. Therefore, the requirements of GDC 44, 45, and 46 are not applicable to the design of the TCCWS. The staff reviewed DCD Tier 2 Table 9.2-12, "TCCWS Heat Loads," 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 determined that TCCWS is adequately designed for its function even though GDC 44, 45, and 46 are not applicable. The staff determined the response was acceptable since the applicant clarified that TCCWS is not a system used to transfer heat from structures, systems, and components important to safety that are RTNSS or safety-related and therefore the requirements of GDC 44, 45, and 46 are not applicable 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 TCCWS is resolved.

9.2.8.4 Conclusion

Based on the above discussion, the staff finds the design of the TCCWS acceptable and GDC 2 and 4 are satisfied.

9.2.9 Hot Water System

In DCD Revision 6 Tier 2, Section 9.2.9, "Hot Water System," the applicant states that the hot water system for the ESBWR design has been eliminated and its function replaced with electric heating coils (in-duct) for most building loads and radiant (wall mounted) heating coils for localized heating load. Therefore, the staff 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 with the design change and therefore is resolved.

9.2.10 Station Water System

9.2.10.1 Regulatory Criteria

The staff determined that no current guidance provided in NUREG-0800, "Standard Review Plan (SRP) for the Review of Safety Analysis Reports for Nuclear Power Plants" 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, March 2007. Applicable portions of GDC 2, 4, 44, 45, 46 are the potential requirements to be evaluated.

9.2.10.2 Summary of Technical Information

DCD Tier 2, Section 9.2.10, "Station Water System," describes a conceptual design of the station water system. The station water system is designed to provide a supply of water for the following services:

- makeup water to the circulating water system (CIRC) cooling tower basin
- makeup water to the plant service water system (PSWS) cooling tower basins
- feedwater to the MWS
- fill water to the fire protection system (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 Main Control Room (MCR). Pretreated station water storage tank high and low level, 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 above CDI for the station water system will be replaced with site-specific design information in the Combined License Application Final Safety Analysis Report (COLA FSAR).

9.2.10.3 Staff Evaluation

In DCD Tier 2, 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 the site-specific design information will be provided in the COLA FSAR, and the DCD only provides the conceptual design information (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 applicable portions of GDC 2, 4, 44, 45, 46 may need to be evaluated when the plant-specific design information is available.

In NRC RAI 9.2-16, the staff asked the applicant to identify a COL information item for the site-specific station water system design. In the 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 not having 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 site-specific design of the station water system will be reviewed in COL applications.

9.2.10.4 Conclusion

Based on the above, the staff concludes 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 Economic Simplified Boiling Water Reactor (ESBWR) compressed air system in accordance with NUREG-0800, "Standard Review Plan (SRP) for the Review of Safety Analysis Reports for Nuclear Power Plants," Section 9.3.1. "Compressed Air Systems," Revision 2, dated March 2007. The staff reviewed ESBWR Design Certification Document (DCD) Tier 1, Revision 6, Section 2, "Design Descriptions and ITAAC," ESBWR DCD Tier 2, Revision 6, Section 9.3.1. "Compressed Air Systems," and various parts of other DCD Tier 2 sections (i.e. Sections 19A, 22, etc.). The staff's acceptance of the compressed air system is based on meeting the relevant requirements of the following General Design Criteria (GDC) and Code of Federal Regulations (CFR):

- GDC 1, "Quality Standards and Records," in part requires that structures, systems, and components (SSC) 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, "Design Bases for Protection Against Natural Phenomena," requires in part that SSCs important to safety shall be designed to withstand the effects of natural phenomena such as earthquakes.

- GDC 5, "Sharing of Structures, Systems, and Components," 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, "Loss of all alternating current power," as to the ability of a plant to withstand for a specified duration and recover from a station blackout (SBO).

- 10 CFR 52.47(b)(1), which requires that a design certification (DC) application contain the proposed inspections, tests, analyses, and acceptance criteria (ITAAC) that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a 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 compressed air system (CAS) consists of the instrument air system (IAS), the service air system (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 7, 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 request for additional information (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 non-safety-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 its 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 non-safety-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 non-safety-related components and piping. Accordingly, ased 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 its 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.

<u>RAI 9.3-35</u>

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 its 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 design-basis accidents, the applicant referred to DCD Tier 2, Section 3.6, "Protection Against Dynamic Effects Associated With The Postulated Rupture Of Piping," which addresses the protection provided for safety-related SSCs against dynamic effects associated with design-basis accidents.

Also, as stated in the above, in the responses to RAI 9.3-33 and RAI 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 non-safety-related pneumatic system components and piping. The safety-related and non-safety-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 subsystems IAS and CIS, is non-safety-related and has no safety-related function. Failure of the CIS does not compromise any safety-related system or component, nor does it prevent a safe shutdown of the plant.

The staff determined that the RAI response was 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, "Instrument Air System," of this report addresses the staff's evaluation of the IAS.

Section 9.3.7, "Service Air System," of this report addresses the staff's evaluation of the SAS.

Section 9.3.8, "High-Pressure Nitrogen Supply System," of this report addresses the staff's evaluation of the HPNSS.

Section 6.2.5.2, "Containment Inerting System," of this report addresses the staff's evaluation of the CIS.

Section 8.4.2, "Station Blackout," of this report addresses the staff's evaluation of the ESBWR design 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 ESBWR DCD Tier 2, Section 9.3.2, "Process Sampling System," in accordance with SRP Section 9.3.2, "Process and Post-Accident Sampling System." The DCD does not include a description of a post-accident sampling program, however, the DCD identifies the sample point parameters in Table 9.3-1 and key sample locations for the post-accident monitoring program are described in Subsections 7.5.1 and 7.5.2 of the DCD. In addition, ESBWR DCD, COL 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 Table 9.3-1 of the DCD. Therefore, the post accident monitoring program is not reviewed in this report. 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 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.

- 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 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 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 9.3.2 and that the PSS is in conformance with the relevant requirements and criteria:

- 10 CFR Part 20 & 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 50.34(f)(2)(xxvi)

The DCD states that the PSS is in conformance with the following guidelines:

- Regulatory Guides (RG) 1.21, 1.26, 1.29, 1.33, 1.56, 1.97, and 8.8;
- NUREG-0737;
- ANSI/HPS N13.1; and

- EPRI BWRVIP-130: BWR Vessel and Internals Project 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
- I-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 Regulatory Guide (RG) 1.21, Position C.2, and the Electric Power Research Institute (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).

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 Tank, Condensate

Polishing System, Fuel and Auxiliary Pools Cooling System, sumps inside containment, Main Condenser Evacuation System and inlet and outlet of the Radwaste Tank.

The ESBWR PSS design includes the following sample stations:

- Reactor Building Sample Station
- Local Grab Sampling Stations
- Condensate Polishing Sample Station
- Turbine Building Sample Station
- Condenser Sample Station
- Radwaste Building Sample Station
- Auxiliary Boiler Building Sample Station

The Reactor Building Sample Station permits an operator to take continuous samples from the fuel and auxiliary pools cooling system. In addition grab samples can be taken to test for the parameters identified above.

DCD Tier 2, Table 9.3-1, “Process Sampling System Measurement” identifies sampled systems and process measurements to be taken. DCD Tier 2, Table 1.9-21, “NRC Regulatory Guide Applicability to ESBWR” 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 point 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:

- Reactor Component Cooling Water System
- Turbine Component Cooling Water System
- Plant Service Water System
- Chilled Water System
- Circulating Water System
- Standby Liquid Control System
- Makeup Water System
- Condensate Storage and Transfer System
- Equipment and Floor Drain System

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 standby liquid control system 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 9.3.2 recommends that samples are taken from the spent fuel pool (SFP). DCD Tier 2, Table 9.1-1, states that the SPF is located in the fuel building which host no sample station according to DCD Tier 2 Section 9.3.2. The staff requested in RAI 9.3-44 that the applicant identify what process sampling is being proposed for the SFP and other fuel building pools, provide the typical process measurements that will be conducted (continuous and grab) and identify where the process samples will be processed.

In its 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 Reactor Building sample station and analyzed for the species identified in DCD Tier 2, Table 9.3-1. One FAPCS cooling and cleanup 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 FAPCS at the suction of the FAPCS pump. Therefore, the spent fuel pool can be sampled both pre and post the filter and demineralizer subsystem. The sample station for FAPCS is located in the Reactor Building. This central location allows for sampling from pools in the Containment, Reactor Building and the Fuel Building thus minimizing locations of possible spillage and contamination. DCD Tier 2, Table 9.1-1 shows the various pools served by both subsystems of FAPCS. DCD Tier 2, Table 11.5-5 identifies the SFP as having provisions for being sampled and Table 9.3-1 identifies the typical process measurements taken from FAPCS. The staff determined that the response to RAI 9.3-44 was acceptable since the applicant clarified the process sampling for the SFP and other fuel building pools. Accordingly, based on the above and the applicant's response, RAI 9.3-44 is resolved. Based on this information, the staff concludes 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 should 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 American National Standards Institute/Health Physics Society (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 7, Section 9.3.2.2 states that the PSS provides sampling of all principle 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 in order to remove sediments 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 7, Section 11.5 describes provisions for sampling liquid and gaseous process and effluent streams and summarizes the scope of radiological analyses for such samples. This information is described in DCD Tier 2, Revision 7, Section 11.5, Tables 11.5-5 to 11.5-8. 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 RG 1.21 and 4.15, and NUREG-1302 in the development of a plant-specific offsite dose calculation manual and standard radiological effluent controls for BWR plants. Site-specific conformance to the recommendations of RG 1.21 and 4.15, and NUREG-1302 are addressed by the COL applicant consistent with COL Items 11.5-2-A, “Offsite Dose Calculation Manual,” and 11.5-3-A, “Process and Effluent Monitoring Program.” These COL Items are further addressed in Section 11.5 of this report.

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 7, 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 7, 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 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 control 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 at ALARA levels. 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 7, 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 7, 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 ALARA is considered in station layout and design, 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 to minimize contamination, in accordance with 10 CFR 20.1406. It was also unclear whether the applicant had performed an assessment of the personnel doses associated with the sampling of radioactive material. In order for the staff to determine if the applicant had addressed these 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 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 stations 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. In order to limit the extent of contamination in areas where the potential for spills exist, floors are sloped towards drains and curbs are provided to simplify washdown operations. The applicant stated that they had evaluated the personnel doses associated with routine use of the PSS sample stations and these doses are listed in Table 12.4-2, Occupational Dose Estimates During Operation and Surveillances. The staff determined that the response was acceptable since the applicant design features to minimize personnel does 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 to ALARA levels 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 7, Section 9.3.2 states that all sampling lines have the process isolation block valves located close as practical to the process taps. These valves can be closed if sample line rupture occurs downstream of the valves. In the event of 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. Safety relief valves, vented to the drain headers, are provided in the stations for high-temperature process streams. Continuous samples are taken and are monitored by continuously monitoring equipment. The continuously monitoring equipment transmits signals to the plant computer and alarms are provided for indicating off-normal operating conditions. ALARA is also considered in the sampling station layout and design. 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 sampling line and components is connected.

DCD Tier 2, Revision 7, 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 standards 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 concludes 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 Economic Simplified Boiling Water Reactor (ESBWR) equipment and floor drain system (EFDS) in accordance with NUREG-0800, "Standard Review Plan (SRP) for the Review of Safety Analysis Reports for Nuclear Power Plants," Section 9.3.3, "Equipment and Floor Drainage System," Revision 3, dated March 2007. The staff reviewed ESBWR Design Certification Document (DCD) Tier 1, Revision 6, Section 2, "Design Descriptions and ITAAC," ESBWR DCD Tier 2, Revision 6, Section 9.3.3, "Equipment and Floor Drainage System," and various parts of other DCD Tier 2 sections (i.e. Sections 19A, 22, etc.). The staff's acceptance of the EFDS is based on meeting the relevant requirements of the following General Design Criteria (GDC) and Code of Federal Regulation (CFR):

- GDC 2, "Design Bases for Protection Against Natural Phenomena," in part, requires that structures, systems, and components (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 Regulatory Guide (RG) 1.29, "Seismic Design Classification."

- GDC 4, "Environmental and Dynamic Effects Design Bases," in part, requires that SSCs important to safety 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, "Control of Releases of Radioactive Materials," 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), which requires that a design certification (DC) application contain the proposed inspections, tests, analyses, and acceptance criteria (ITAAC) that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a 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 non-safety-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 EFDS 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, "Containment Isolation System," of this report addresses the staff's evaluation of the containment penetration and isolation valves for EFDS pump discharge lines.

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 6, Subsection 3.4.1, "Flood Protection," no credit is taken for the EFDS system in the flooding analysis. Section 3.4.1, "Flood Protection," 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 (e.g., no cross connections), 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 liquid waste management system (LWMS) for processing and dispose. The RWDS is further divided into the following subsystems, so that the liquid wastes from various sources are 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
- Reactor component cooling water system (RCCWS) drain subsystem

Each of the above subsystem 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, "Liquid Waste Management Systems," of this report addresses the staff's evaluation of the LWMS.

The EFDS design includes provisions for sampling the drain sumps/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 in 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 (DW) is accomplished by monitoring the DW floor drain HCW and LCW sump pump activity and the DW sump level changes. Leak detection in other areas is accomplished by monitoring the frequency and duration of sump pump operation. Section 5.2.5, "Reactor Coolant Pressure Boundary (RCPB) Leakage Detection Systems" of this report addresses the staff's evaluation of the leakages 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 draining 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 its responses, the applicant stated and clarified that the floor EFDS was a non-safety-related system and was not credited for draining floodwater in the flooding analysis. The results of the flooding analysis were found to be acceptable with the assumption that the floodwater was retained in localized areas or zones, which was conservative in the determination of the resulting water level of these specific areas. The staff determined that the response was acceptable since the applicant clarified that no credit is taken for the EFDS in the ESBWR flooding analysis. 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 non-safety-related and performs no safety-related function. As stated above, Section 6.2.4 of this report addresses the staff's evaluation of the containment isolation valves for the EFDS pump discharge lines. As for the non-safety-related EFDS on meeting the guidance of the Regulatory Position C.2 of RG 1.29, the EFDS is designed to ensure that failure of the EFDS does not compromise any safety-related system or component nor does it prevent 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 does not compromise any safety-related system or component nor does it prevent 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 non-safety-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 (e.g., no cross connections), 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/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 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 non-safety-related, is not credited in the flooding analysis or any safety analysis, and is not required to achieve or maintain safe shutdown of the plant. Also, the ESBWR design does not use the EFDS to provide defense-in-depth capabilities for any safety function. Therefore, the EFDS is not considered as a candidate for regulatory treatment of non-safety system (RTNSS) system, because it does not meet any of the five criteria as described in SECY-94-084, “Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety System in Passive Plant Designs.”

The EFDS in ESBWR DCD Tier 1 has ITAAC entries. Section 2.16.4, “Equipment and Floor Drain System,” and Table 2.16.4-1, “ITAAC for The Equipment and Floor Drain System.” of the ESBWR DCD Tier 1, Revision 6, provides the design descriptions and ITAACs 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, Revision 7. Therefore, the staff concludes that 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. Section 14.2, “Initial Plant Test Program for Final Safety Analysis Reports,” of this report addresses the staff’s evaluation of the periodic inspection and testing requirement for the EFDS.

9.3.3.4 Conclusion

The staff concludes that the design of the equipment and floor drain system 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 a 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, Section 2.2.4 and Tier 2, Section 9.3.5, “Standby Liquid Control System,” for the ESBWR, in accordance with SRP Section 9.3.5, “Standby Liquid Control System,” Revision 3. Acceptability of the SLCS design, as described in the applicant’s DCD, is based on specific GDC; 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 related 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 related 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, "Reactivity Control System Redundancy and Capability," as it relates to the requirements that (1) two independent reactivity control systems of different design principles be provided, and (2) that 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 boiling-water reactor (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. The staff evaluation of the buffering function is included in Section 15.4.3.3.2.1 of this report.

The boron solution is also credited for buffering the suppression pool such 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 non-safety-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 non-safety-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 non-safety-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. Upon making the decision to initiate the SLCS, the operator starts the system by using two protected switches, which prevents inadvertent SLCS initiation by operator error. The manual initiation switches in the MCR are recessed, spring-loaded, rotate-and-push type. Both switches have a protective switch cover. Actuation requires simultaneous depression of the switches, and the switches are located such that a single operator can readily initiate the system. Procedural controls govern operation of these initiation switches.

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 pound-force per square inch gauge (psig)), or

- 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 for 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 purpose and on the staff's acceptance of the design, the staff concludes that the intent of GDC 26 has been 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 fine motion control rod drive 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, 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 secondary containment building 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 non-safety-related portions of the system are 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 its response, the applicant stated that due to its location inside the Reactor Building, with its own compartment, the SLCS is protected 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 determined that the response was 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 have been made that 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 control rod drive 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 its response, the applicant stated that the SLCS leakage can be monitored using the accumulator pressure and level instrumentation, which provide 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. Leakage is collected by the SLCS through drains and sent 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 determined the response was 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 malfunctions. 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 air operated valves 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 °C (60 °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 RB HVAC systems are not available. The Plant Investment Protection 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 Table 9.3-5 indicates that during ATWS, at a reactor pressure as high as 8.63 MP_a (1250 psig), boron solution discharge from the SLCS occurs at a volumetric rate of 1.8 cubic meters per minute (m^3/min) (475 gallons per minute (gal/min)) during the initial injection. These flow rates are averages for the first 5.40 m^3 (1427 gal) 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 Agency wide Documents Access and Management System (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 (gal/min)
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, namely that a SLCS must be capable of injecting 86 gal/min of a 13 weight percent sodium pentaborate decahydrate solution at the natural boron-10 isotope abundance into a reactor vessel with a 251-inch 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 its response to RAI 14.3-196 S01, where the injection flow is 330 gal/min, the concentration is 12.5 weight percent, and with natural abundance of Boron-10 (nonenriched). The mass of the water inside the ESBWR reactor vessel with a 278-inch diameter, based on the fluid control volumes in DCD Tier 2, Figure 5.1-1, is 823,800 pounds. (M251, the mass of the water in a 251-inch BWR/6 vessel, is 614,300 pounds). The result is that the design meets the requirements of the ATWS rule by a factor of 2.75. If 94% 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 relation 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), present uniformly in the ESBWR core, would bring the reactor to a cold-shutdown condition. This compares to the licensee's SLCS capability of 1100 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 show that the overall TRACG boron mixing and transport models result in a lower reactivity worth and, thus, are conservative. The staff performed CFD confirmatory calculations and reached similar conclusions. Additional discussion is available in the staff's Safety Evaluation for NEDE-33083P, Supplement 2, Revision 2. 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 and 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 in 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 SLC injection. Accordingly, based on the above and the applicant's response and in view of the approval of TRACG for ATWS scenarios in 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:

- 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.

- 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 by either of two key-locked, spring-return switches 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 fine motion control rod drive 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 of the SLCS requires the operator to depress two recessed, spring-loaded, rotate-and-push switches that are protected with a cover. The switches are located such that a single operator can depress both switches simultaneously, as required for SLCS initiation.

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 staff review of NEDE-33083P, Supplement 2, Revision 2 is discussed in the Safety Evaluation for 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 possibilities 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 reactor water cleanup and shutdown cooling system. 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 control rod drive system flow. This is acceptable to the staff even though the CRD system is not safety grade. Control rod drive 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 its response to RAI 21.6-42. The staff reviewed the response to RAI 21.6-42 in 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 its 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.

The applicant's DCD does 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 further requested 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 Tier 2, Table 9.3-4 in DCD Revision 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 determined that the response to RAI 9.3-6 was 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 depressurization valve 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 (4121 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.

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. The evaluation of the SLCS with regard to these GDC appears in Section 6.3.3 of this report.

9.3.5.3.6 Inspections, Tests, Analyses, and Acceptance Criteria

The staff reviewed SLCS information DCD Tier 1, Revision 3, 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 its 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 determined that the response was 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 <u>Conclusion</u>

The NRC staff has reviewed the applicant's information related to the SLC system. For the reasons set forth above, the staff concludes that the applicant has adequately demonstrated that the SLC system has the capability for reactor shutdown and core make up. The staff concludes that the SLC system meets the requirements of GDC 2, 4, 5, 26, 27 and 10 CFR 50.62.

9.3.6 Instrument Air System

9.3.6.1 Regulatory Criteria

The staff reviewed the Economic Simplified Boiling Water Reactor (ESBWR) instrument air system (IAS) in accordance with NUREG-0800, "Standard Review Plan (SRP) for the Review of Safety Analysis Reports for Nuclear Power Plants," Section 9.3.1. "Compressed Air Systems," Revision 3, dated March 2007. The staff reviewed ESBWR Design Certification Document (DCD) Tier 1, Revision 6, Section 2, "Design Descriptions and ITAAC," ESBWR DCD Tier 2, Revision 6, Section 9.3.6. "Instrument Air Systems," and various parts of other DCD Tier 2 sections (i.e. Sections 19A, 22, etc.). The staff's acceptance of the IAS is based on the design's conformance with the requirements of the following General Design Criteria (GDC) and 10 CFR 52.47(b)(1):

- GDC 1, "Quality Standards and Records," in part requires that structures, systems, and components (SSC) 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, "Design Bases for Protection Against Natural Phenomena," 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 Regulatory Guide (RG) 1.29, "Seismic Design Classification."

- GDC 5, "Sharing of Structures, Systems, and Components," 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 DC application contain the proposed inspections, tests, analyses, and acceptance criteria (ITAAC) that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a 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. These pneumatically operated devices are designed fail-safe and do not need continuous air supply under emergency or abnormal conditions. The system 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 service air system (SAS) compressors and receives pre-filtered, oil free, compressed air from the SAS. The IAS consists of two identical 100% 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 radwaste, turbine, and reactor buildings. The IAS has piping connections outside containment to the high-pressure nitrogen supply system (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 refueling operations.

IAS operational tests, including preoperational testing as described in DCD Tier 2, Revision 6, Section 14.2.8, “Individual Test Descriptions,” 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 American Society of Mechanical Engineers (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.

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 regulatory treatment of non-safety system (RTNSS). 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 preoperational 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 determined that the IAS meets the relevant requirements of GDC 1.

Section 14.2, "Initial Plant Test Program for Final Safety Analysis Reports" of this report addresses the staff's evaluation of the operational tests including preoperational testing performed for IAS components to ensure system capability and integrity.

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 its 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 and the evaluation and resolution of RAIs 9.3-33, 9.3-34, and 9.3-35 are discussed in Section 9.3.1 of this report. 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 its response to RAI 9.3-41, the applicant stated:

- Any of the SAS compressors is capable of meeting 100% demand of the IAS and each of the dryer trains is sized for 100% 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. Also, in Table 9.3.6, "Instrument Air System Requirements," of DCD Tier 2, Revision 6, the applicant 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 quality standard established 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 Generic Letter 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 considered Generic Issue 43 resolved with the issuance of Generic Letter 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 Generic Letter 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 that:

- 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 Generic Letter 88-14 also requested each licensee/applicant to provide a discussion of their program for maintaining proper instrument air quality.

In 2005, the staff assessed the effectiveness of Generic Issue 43 and Generic Letter 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, Generic Letter 88-14, design-basis reconstitution, and others, have significantly improved air system and component performance and, thereby, resulted in improved reactor safety.

- Issuance of Generic Letter 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.

The staff's concerns cited in the above Generic Letter 88-14 are covered in ANSI/ISA 7.0.01. 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.

Since the ESBWR IAS design meets the guidance of ANSI/ISA 7.0.01, and the operation of the IAS bypass occurs only in an unlikely event that both IAS dryer trains would fail at the same time and in view of the applicant's responses to the staff's RAIs, and staff's review for the ESBWR design certification of Generic Issue 43, Generic Letter 88-14, NUREG-1275, Volume 2, and NUREG-1837, the staff concludes 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, "Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety System in Passive Plant Designs."

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 concludes 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 Economic Simplified Boiling Water Reactor (ESBWR) service air system (SAS) in accordance with NUREG-0800, "Standard Review Plan (SRP) for the Review of Safety Analysis Reports for Nuclear Power Plants," Section 9.3.1. "Compressed Air Systems," Revision 3, dated March 2007. The staff reviewed ESBWR Design Certification Document (DCD) Tier 1, Revision 6, Section 2, "Design Descriptions and ITAAC," ESBWR DCD Tier 2, Revision 6, Section 9.3.7. "Service Air Systems," and various parts of other DCD Tier 2 sections (i.e. Sections 19A, 22, etc.). The staff based its acceptance of the SAS on the design's conformance with the requirements of the following General Design Criteria (GDC) and Code of Federal Regulations (CFR):

- GDC 1, "Quality Standards and Records," in part requires that structures, systems, and components (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, "Design Bases for Protection Against Natural Phenomena," 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 Regulatory Guide (RG) 1.29, "Seismic Design Classification."

- GDC 5, "Sharing of Structures, Systems, and Components," 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 DC application contain the proposed inspections, tests, analyses, and acceptance criteria (ITAAC) that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a 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 non-safety-related system that provides filtered compressed air for general plant use via service air outlets located outside of the containment and to the instrument air system (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 American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel 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 radwaste building, turbine building, and reactor building. 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 loss-of-coolant accident (LOCA).

9.3.7.2 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 regulatory treatment of non-safety system (RTNSS). The SAS components meet ASME Boiler and Pressure Vessel 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 non-safety-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 SAS supply air line. 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 its 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 and the evaluation and resolution of RAIs 9.3-33, 9.3-34, and 9.3-35 are discussed in Section 9.3.1 of this report. Accordingly, based on the above and the applicant's responses, RAIs 9.3-33, 9.3-34, and 9.3-35 are resolved.

The SAS in ESBWR DCD Tier 1 has one ITAAC entry. In Section 2.12.8, "Service Air System," and Table 2.12.8-1, "ITAAC for The Service Air System," of the ESBWR DCD Tier 1, Revision 6, the applicant provides the design descriptions and ITAAC regarding the containment penetration and isolation valves for the SAS. Therefore, the staff concludes that SAS complies with the requirements of 10 CFR 52.47(b)(1).

In ESBWR DCD Tier 2, Revision 6, Section 9.3.7, the applicant states that the system operability is demonstrated by use during normal plant operation, system components are shop inspected and tested, system operational tests for components normally closed to airflow are performed periodically to ensure system capability and integrity, and filters are periodically inspected for cleanliness. Section 14.2.8.1.19, "Instrument Air and Service Air Systems Preoperational Tests," of this report addresses the staff's evaluation of the periodic inspection and testing requirement for the SAS.

9.3.7.3 Conclusion

The staff concludes that the design of the SAS is acceptable and meet 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 Economic Simplified Boiling Water Reactor (ESBWR) high-pressure nitrogen supply system (HPNSS) in accordance with NUREG-0800, "Standard Review Plan (SRP) for the Review of Safety Analysis Reports for Nuclear Power Plants," Section 9.3.1. "Compressed Air Systems," Revision 3, dated March 2007. The staff reviewed ESBWR Design Certification Document (DCD) Tier 1, Revision 6, Section 2, "Design Descriptions and ITAAC," ESBWR DCD Tier 2, Revision 6, Section 9.3.8. "High-Pressure Nitrogen Supply System," and various parts of other DCD Tier 2 sections (i.e. Sections 19A, 22, etc.). The staff based its acceptance of the HPNSS on the design's conformance with the requirements of the following General Design Criteria (GDC) and Code of Federal Regulations (CFR):

- GDC 1, "Quality Standards and Records," 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, “Design Bases for Protection Against Natural Phenomena,” 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 Regulatory Guide (RG) 1.29, “Seismic Design Classification.”

- GDC 5, “Sharing of Structures, Systems, and Components,” 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 DC application contain the proposed inspections, tests, analyses, and acceptance criteria (ITAAC) that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a 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 containment inerting system (CIS) to the nuclear boiler system (NBS) automatic depressurization subsystem (ADS) safety relief valve (SRV) accumulators, the isolation condenser (IC) 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 pneumatic-operated valves) while the other branch supplies the high-pressure nitrogen loads (i.e., NBS ADS SRV accumulators and the IC 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 non-safety-related piping and valves of the HPNSS meet the American Society of Mechanical Engineers (ASME) 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. Pneumatic-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 as a candidate for regulatory treatment of non-safety system (RTNSS). The nonsafety-related piping and valves of the HPNSS meet ASME 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 determined 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 contains the IAS, SAS, and HPNSS. In its 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 and the evaluation and resolution of RAIs 9.3-33, 9.3-34, and 9.3-35 are discussed in Section 9.3.1 of this report. 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.

<u>RAI 14.3-91</u>

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 its response to RAI 14.3-91, the applicant addressed the compressed gas accumulators for the MSIV and the safety relief valves but did not address the compressed gas systems. In its 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 probabilistic risk assessment (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 its 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 post accident 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 determined that response to RAIs 22.5-3 and 14.3-91 were 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 safety relief valves and main steam isolation valves. The staff finds this rationale also supports the applicant's position that HPNSS does not provide a RTNSS function. Accordingly, based on the above and the applicant's responses, RAIs 14.3-91 and 22.5-3 are resolved.

In DCD Tier 1, Revision 6, Subsection 2.15.1, "Containment System," the applicant provides the design descriptions and ITAAC regarding the containment penetration and isolation valves for the HPNSS. Therefore, the staff concludes that HPNSS complies with the requirements of 10 CFR 52.47(b)(1).

Section 14.2.8.1.20, "High Pressure Nitrogen Supply System Preoperational Test," of this report addresses the staff's evaluation of the operational tests including preoperational testing performed for HPNSS components to ensure system capability and integrity.

9.3.8.4 Conclusion

The staff concludes 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 non-safety-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 there are no safety implications associated with the HWCS as described in the DCD.

9.3.9.2 Summary of Technical Information

DCD, Revision 7, 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 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 7, Section 9.3.9.6 identifies two COL items related to the HWCS. COL Item 9.3.9-1-A states that the COL applicant will determine if HWCS is to be implemented. COL Item 9.3.9-2-A states that the hydrogen and oxygen storage facility design and appropriate supply system will be provided by the COL Applicant if HWCS is selected to be installed. The staff finds COL 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 non-safety-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 Electric Power Research Institute (EPRI) Report NP-4947-SR, "BWR Hydrogen Water Chemistry Guidelines," 1987 Revision. This report provides 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 its 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 above COL item should be the responsibility of the COL applicant or the COL holder. In RAI 9.3-37, the staff requested clarification from the applicant. In its response, the applicant modified the DCD to state that the COL applicant is responsible for determining whether to install an HWCS. The staff determined that the applicant's responses were 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 identified changes were included in DCD Revision 5.

9.3.9.4 Conclusion

The staff concludes that there are no safety implications associated with the HWCS as described in the DCD. The staff finds that the EPRI guidelines presented 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 non-safety 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 non-safety-related system. The staff reviewed the OIS to ensure that there are no safety implications associated with the OIS as described in the DCD description 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 Revision 3, 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 its 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 7, Section 9.3.10.6 identifies one COL item related to the OIS. COL 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 Item 9.3.10-1-A acceptable since the use of oxygen storage facility is dependent on whether a HWCS is used, which is site dependent.

However, the staff did not find the response acceptable because it was unclear whether the OIS would still need to meet the guidelines of EPRI Report NP-5283-SR-A if the HWCS is 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 its response, the applicant revised the DCD to state that the Oxygen Injection System uses the guidelines for gaseous oxygen injection systems in EPRI Report NP-5283-SR-A, "Guidelines for Permanent Hydrogen Water Chemistry Installations-1987 Revision, September 1987." The staff determined that the response was 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. T he staff confirmed that the identified changes were incorporated into DCD Revision 5.

9.3.10.4 Conclusion

The staff concludes that there are no safety implications 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 OIS is an optional system that, if specified by the COL applicant, will inject oxygen in the condensate and feedwater system. The COL applicant shall specify, and the NRC staff shall review, any safety implications of an OIS 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 non-safety-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 SRP, through March 2007, does not include a section specifically addressing the ZIS. The staff reviewed the ZIS to ensure that there are no safety implications associated with the ZIS as described in the DCD.

9.3.11.2 Summary of Technical Information

DCD Revision 7, Section 9.3.11 contains information on the ZIS. The ZIS is a non-safety-related system that is used optionally by the COL holder to control the build up 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 has been 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/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 its response to RAI 9.3-39, the applicant stated that the COL applicant determines whether 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 7, Section 9.3.11.6 includes these as two COL items. COL Item 9.3.11-1-A states that the COL Applicant shall determine if a Zinc Injection System is required to be implemented at startup based on plant configuration and material selection. COL Item 9.3.11-2-A states that if a Zinc Injection System is to be installed, the COL Applicant shall include necessary information on System Description, Test and Inspection The staff finds COL Items 9.3.11-1-A and 9.3.11-2-A acceptable since the use of a ZIS is site dependent. The staff determined that the response to RAI 9.3-39 was acceptable since the applicant clarified that the use of a ZIS is the responsibility of the COL Applicant. 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 there are no safety implications associated with the zinc injection system as described in the DCD. The zinc injection system 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 non-safety-related system and has no safety design basis. NUREG-0800, "Standard Review Plan (SRP) for the Review of Safety Analysis Reports for Nuclear Power Plants," revised through March 2007, does not include a section specifically addressing the auxiliary boiler/steam system. However, the staff reviewed the Economic Simplified Boiling Water Reactor (ESBWR) Design Certification Document (DCD) Tier 2, Revision 6, Section 9.3.12, "Auxiliary Boiler System." against the requirements of the General Design Criterion 2 (GDC 2) to ensure that failure of the ABS as a result of a pipe break or malfunction of the system could not adversely affect any safety-related systems or components:

- GDC 2, "Design Bases for Protection against Natural Phenomena," in part, requires that structures, systems, and components (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 Regulatory Guide (RG) 1.29, "Seismic Design Classification.

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% capacity fire tube auxiliary boiler composed of two 50% capacity fuel oil boilers
- Two complete firing systems including fuel-oil burners and fans
- Two 100% capacity fuel-oil transfer pumps
- Three 50% capacity auxiliary boiler feedwater pumps
- One 100% capacity deaerator with integral storage tank
- One 100% capacity auxiliary boiler blowdown flash tank
- One 100% capacity steam separator
- Instrumentation and controls

During plant startup and shutdown and also 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 are provided to transfer fuel oil from the standby diesel generator (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 a 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 where safety-related equipment is located. RAI 9.3-40 was being tracked as an open item in the SER with open items.

In its responses to RAI 9.3-40, the applicant stated that:

- The ABS does not interface directly with nuclear process systems.
- The auxiliary boiler is located outside the turbine building, adjacent to the radwaste building.
- ABS piping is routed in the turbine building.
- Safety related RPS sensors are located in the turbine building.

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 address 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 its 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 turbine building. The turbine building included in the ESBWR standard plant design is nonsafety-related. However the turbine building 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 determined that the RAI response was acceptable because the applicant has shown 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 nor does it 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 nor does it prevent a safe shutdown.

The ABS is non-safety-related, is not relied upon to achieve or maintain safe shutdown of the plant. Also, 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 as a candidate for regulatory treatment of non-safety system (RTNSS) system, because it does not meet any of the five criteria as described in SECY-94-084, "Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety System in Passive Plant Designs."

9.3.12.4 Conclusion

The staff concludes 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 HVAC System

9.4.1.1 Regulatory Criteria

The staff reviewed ESBWR DCD (Rev 7) Subsection 9.4.1 Control Building HVAC System in accordance with NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants" (hereafter referred to as the SRP), Section 9.4.1, "Control Room Area Ventilation System." The staff's acceptance of the CBVS is based on compliance with the following requirements:

- General Design Criterion 2 (GDC 2), “Design Bases For Protection Against Natural Phenomena,” as it relates to structures, systems, and components 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, "Environmental and Dynamic Effects Design Bases,” 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, "Sharing of Structures, Systems and Components," as it relates to ensuring that sharing among nuclear power units of SSCs important to safety will not significantly impair the ability to perform safety functions, including, in the event of an accident in one unit, an orderly shutdown and cooldown of the remaining unit(s).

- GDC 19, "Control Room," as it relates to maintaining the nuclear power unit in a safe condition under accident conditions and providing adequate radiation protection.

- GDC 60, “Control of Releases of Radioactive Materials to the Environment,” as it relates to the nuclear power unit design including 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 for cope with a station blackout event.

- 10 CFR 20.1406, “Minimization of Contamination”

9.4.1.2 Summary of Technical Information

The Control Building Heating, Ventilating and Air Conditioning System (CBVS) serves all areas of the Control Building during normal operation. The CBVS maintains space design temperatures, quality of air and pressurization. It provides a controlled environment for personnel safety and comfort, and for the proper operation and integrity of equipment located in the control building.

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 Main Control Room (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 non-safety-related redundant fresh air supply fans and two redundant non-safety-related internal floor mounted air handling units (AHU). Radiological protection is provided from a redundant set of safety-related Emergency Filter Units (EFU).

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. Portions of the CRHAVS that are safety-related are 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 non-safety-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:

- Monitors the CRHA air supply for radioactive particulate and/or iodine concentrations; and

- Isolates the normal CRHA air supply and restroom exhaust, starts an EFU fan, and

- Aligns 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 design basis accident (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, instrumentation and controls. The remaining CRHAVS functions are non-safety-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 non-safety-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% 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 Non-safety-related Distributed Control and Instrumentation System (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 non-safety-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 are provided on EFU air intake openings. These dampers 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; and

- 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 design basis accident (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 to close; and
- 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 a 72 hour duration. For longer-term operation, (post-72 hour) either of two (2) ancillary diesel generators can power either EFU fan system.

The EFU delivery and discharge system is optimized to ensure that there is adequate fresh air delivered and mixed in the CRHA. This is accomplished by using multiple supply registers, which distribute the incoming supply air 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 7-9 times per day.

This diffusion design (mixing and displacement) in conjunction with the known convective air currents (due to heat loads/sinks) and personnel movement ensures that occupied zone temperature is within acceptable limits, Buildup of contaminants (e.g., CO2) minimal, and the 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 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 non-safety-related CRHAVS Recirculation AHU. During emergency operation, with a loss of normal ac power, a non-safety-related CRHA recirculation air handling unit (AHU), powered from the non-safety-related Uninterruptible ac Power Supply System, maintains the CRHA within the normal operating temperature range for two hours. This allows the continued operation of certain high heat producing non-safety-related MCR DCIS electric loads.

Anytime during a loss of normal ac power, once either ancillary diesel generator 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 due to these sources. In the event the duration of the loss of normal ac power duration extends beyond two hours, the reduced CRHA heat load is passively cooled by the CRHA heat sink. The CRHA heats sinks consist of the following: the CRHA walls, floor, ceiling, and interior walls, and 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°C (93°F) for 72 hours. For the full duration of the design basis accident, 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.

Full capacity cooling and ventilation for the CRHA, 72 hours after an accident, is by operation of the auxiliary cooling units. 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 units chilled water recirculation pumps, independent from the normal Chilled Water System (CWS).

The MCR operator starts the auxiliary cooling system in an accident scenario (post-72-hour) when ac power is being provided from the Ancillary Diesel Generator (ADG). 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 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 Subsection 19A.8.3.

The CBVS:

- Provides a controlled environment for personnel comfort and safety. Sufficient outside air is provided to meet the standards for acceptable indoor air quality (Ref ASHRAE 62.1-2007, Section 6) DCD Table 9.4.1 depicts the area design temperature and humidity design parameters.

- Provides a controlled environment for the proper operation and integrity of equipment in the Control Building during normal, startup and shutdown operations.

- Maintains higher than atmospheric (positive) pressure to minimize the infiltration of outside air. Construction materials and processes that ensure 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. There are double door airlocks in the CRHA envelope for access and egress during emergencies when the CRHA is isolated and an EFU is operating;

- Reduces 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;

- Detects and limits the introduction of airborne hazardous materials (radioactivity or smoke) into the CRHA;

- Provides 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;

- Uses smoke control and removal functions that are in accordance with National Fire Protection Association (NFPA) guidelines in Subsection 9.5.1, Fire Protection System, Subsection 9.5.1.11, Building Ventilation;

- Is designed such that failure of non-safety-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 Control Building.

The EFUs provide breathing air and pressurization to the CRHA when the CRHA envelope is isolated due to 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 Control Building 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 Control Building.

CBVS equipment, including fans, AHUs, EFUs, and the CRHA are located within the Control Building Seismic Category I structural areas.

The CRHAVS is configured as a recirculation system that 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 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 Regulatory Guide 1.194. 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, access openings, etc. 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 non-safety-related Uninterruptible ac Power Supply 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 two hours of a loss of normal ac power event from the non-safety-related battery supply. Anytime during a loss of normal ac power event, once either ancillary diesel generator is available, the power for either Recirculation AHU fan with auxiliary cooling unit can be provided via an 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 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 due to these sources.

Each EFU consists of a medium efficiency filter (40% minimum), a high efficiency particulate air (HEPA) filter (99.97%) a carbon adsorption filter (99% credited efficiency), and a post-filter downstream of the carbon filter (95%). 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 due to 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 Control Building 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 Control Building general areas.

9.4.1.3 Staff Evaluation

The staff review focused on compliance with regulatory requirements for this system. The staff has also reviewed the RTNSS functions for the CBVS as stated in chapter 19A of the DCD against guidance for selection and identification of such systems stated in Regulatory Guide 1.206 subsection C.IV.9. The staff used additional guidance documents to evaluate the CBVS passive cooling features as described below.

GDC 2, Design bases for protection against natural phenomena, requires that structures, systems, and components important to safety shall be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches without loss of capability to perform their safety functions. The staff evaluated whether the CRHAVS meets the requirements of GDC 2. The CRHA envelope is comprised of 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 non-safety-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 the applicant clarify 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 its response, the applicant revised DCD Tier 2 section 6.4.2 Design, Component Descriptions, Section 6.4.4 System Operation Procedures and Emergency Mode, Section 6.4.7, Testing, 9.4.11 Design Bases and 9.4.1.2 Detailed System Description. The applicant revised Tier 1 Table 2.16.2-3, Control Building HVAC System Safety-Related Equipment to include the CHRA 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 its 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 determined that the response, along with the changes in DCD Revision 4, was acceptable since appropriate safety-related components were identified. 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 structures, systems, and components 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 loss-of-coolant accidents. 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, 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 non-safety-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 concludes 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 non-safety-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, Sharing of structures, systems, and components, requires that structures, systems, and components 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, Control Room, 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 loss-of-coolant accidents. 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. 10 CFR 50.63, as it relates to the CRHAVS, 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 a station blackout, thereby ensuring that the core will be cooled and appropriate containment integrity will be maintained. RG 1.155 Position C.3.2.4 provides guidelines regarding evaluating habitability and environmental conditions during a station blackout.

In RAI 9.4-5 and its supplements S01 and 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 in the SER with open items. In its 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 determined that the response was 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.

The staff reviewed the CRHAVS for radiation protection and for the establishment of acceptable environmental conditions. Radiation protection is provided by isolation, use of a safety-related emergency filter unit (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 determined that the ESBWR CRHA design provides sufficient conditioned air with adequate recirculation by the non safety-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 Table 9.4-1, that the CRHA normal design temperature will be no greater than 21.1°C (74°F). This normal design temperature is within with 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), endorsed by NUREG 1242 and therefore is acceptable.

Post-Accident with no loss of ac power supply

Since the Regulatory Treatment of Non-Safety Systems (RTNSS) qualified AHUs remain operational whenever offsite or onsite ac power is available, the staff determined that temperature control for post-accident operation is adequate for such accidents. With Loss of Normal ac power supply for the first two hours after the loss of the Normal ac power supply, the CRHA isolation dampers automatically close and an EFU is automatically started. The non-safety-related Uninterruptable ac Power Supply 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 ancillary diesel generator. Since the RTNSS qualified Recirculation AHUs remain operational, the staff determined 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 two hours.

Post-Accident 0-72 hour operation-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 non-safety related ac power for the entire 72-hour period. SRP 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.

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 smoke purge, smoke exhaust, toilet exhaust and normal supply air penetrations. The isolation dampers are seismically qualified and safety-related. The dampers fail closed on SBO, LOCA, and high radiation signals. 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 design basis accident (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 subsection 9.4.1, the CRHAVS EFUs meet the guidance of RG 1.52 as it relates to the design, inspection and testing criteria for 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, Subsection 5.5.13, “Ventilation Filter Testing Program,” 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 that found in 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 2-inch bed filter and 0.5 percent for a 4-inch 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 4 inches as specified by RG 1.52. The staff determined that the responses were 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 6.4 and Regulatory Guide 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 Testing and Inspection, Inservice Testing to specify 2.3 l/s (5 cfm) instead of near-zero as the amount of unfiltered inleakage allocated for CHRA access and egress. The applicant also clarified DCD Chapter 16 Section 5.5.12, Control Room Habitability Area (CRHA) Boundary program 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 determined that the specified unfiltered inleakage allocation of 2.3 l/s (5 cfm) as proposed in the RAI is reasonable as discussed below. In addition, the change to chapter 16 section 5.5.12 clearly allocates the allowed inleakage and therefore is acceptable to the staff. Based on the applicant's response and DCD changes, RAI 6.4-22 is resolved.

SRP 6.4 and Regulatory Guide 1.197 guidelines state that the staff considers 4.6 l/s (10 cfm) as 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 RG 1.52 and RG 1.197 guidance for design of the safety-related emergency filter unit (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.

<u>Post-Accident 0-72 hour operation-Loss of AC power supply- Evaluation of CRHA 0-72 hour temperature and air quality-Introduction</u>

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 control building concrete in order 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 below documents in order to evaluate the adequacy of the unique features of the ESBWR control room habitability area for such accidents.

- NUREG-1242, NRC Review of Electric Power Research Institute's Advanced Light Water Reactor Utility Requirements Document as it applies to Advanced Light Water Reactor control room envelope atmosphere temperature limits.

- ASHRAE Standard 62.1/2007, Ventilation for Acceptable Indoor Air Quality, as it applies to CRHA indoor air quality standards and acceptance criteria.

- Human-System Interface Design Review Guidelines NUREG-0700, as it applies to the use of the wet bulb globe temperature index in evaluation of heat stress conditions.

- The SRM on SECY 94-084, June 30, 1994, and the SRM on SECY 95-132, June 28, 1995, as they apply to the regulatory treatment of non safety- systems (RTNSS) to address uncertainties as a defense in depth method.

The applicant has proposed air quality and temperature/humidity limits based on or derived from these standards. The staff has reviewed the proposed standards and acceptance criteria and has found them acceptable for use in evaluation 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 Control Building 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 a first principle model (FPM) was developed by the NRC staff 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 non-safety 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 Section 6.4.4 defines the CRHAVS design basis accident (DBA) conditions, which include a LOCA with a loss of offsite power (LOOP). This DBA also takes no credit for non-safety-related Uninterruptable ac Power Supply System operation or ancillary diesel generator operation. It assumes that ac power from non-safety sources is not restored until 72 hours after the accident. The DBA was evaluated at two summer conditions and one winter condition identified in DCD Tier 2 Table 2.0-1, "Envelope of ESBWR Standard Plant Site Parameters," for

4 Yilmaz T. P. & Paschal W. B. article: "An analytical approach to transient room temperature analysis", *Nuclear Technology,*114:135-140

ambient design temperature; the 0% exceedance summer design condition of 47°C (117°F) with a 20% RH, the 0% exceedance summer design condition of 33.3°C (92°F) with a 85%RH and the winter design condition of -40°C (-40°F).

Since the applicant has chosen the most limiting (0% 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.

CRHA Heat Loads and Heat Sinks

The staff reviewed input parameters used in the applicant's Control Building Environmental Temperature Analysis such as heat sink wall thicknesses and surface areas 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 Control Building concrete, orientation of the Control Building for highest solar radiation, a 15% margin in the assumed sensible heat load, an assumed CRHA failure 8 hours prior to the postulated accident (resulting in increased CRHA air and heat sink temperatures at the start of the analysis.) The applicant assumed the highest normal operating temperature allowed in the ESBWR technical specifications 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 the applicant clarify the need to provide cooling to non-safety-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 Subsection 9.4.1.2, CHRA non-safety-related heat loads are automatically de-energized when the CRHA AHUs are not available during the first two hours, and no operator action is needed to isolate the non-safety-related heat loads. The staff determined that the responses were acceptable since the non-safety heat loads are de-energized when CRHA AHUs are not available and the performance of the non-safety CRHA AHUs do not need to be considered in the applicant's Control Building Environmental Temperature Analysis. Based on the above and the applicant's response, 9.4-32 is resolved.

In RAI 9.4-57, the staff requested 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 CHRA will ensure key design assumptions, such as heat sink and heat source assumptions, remain valid. The applicant indicated that DCD Tier 2 Section 17.4 Reliability Assurance Program ensures relevant aspects of plant operation are maintained. COL Item 6.4.1-A directs the COL applicants to develop procedures to control such parameters for the CRHA. The staff determined that the response was acceptable since the DCD revisions provide 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 applicants 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 and are therefore acceptable.

Proposed CRHA Air Quality Acceptance Criteria

The staff reviewed the CRHAVS capability to maintain adequate CO_2 concentration in the CRHA. DCD Tier 2 Section 6.4.1.1 states that the emergency habitability system is designed to maintain the American Society of Heating Refrigeration and Air Conditioning Engineers (ASHRAE) fresh air standards for up to 21 main control room occupants (ASHRAE Standard 62.1/2007, DCD Reference 6.4-4). The emergency Filter Unit System is designed to maintain CO_2 concentration in the CRHA to less than 5000 ppm, which is the upper limit CO_2 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) cites 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 CO_2 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 such an accident, the CRHA is heated by safety-related CRHA equipment and the CRHA is passively heated through walls, floor, ceiling and interior walls.

DCD Tier 1 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 design basis accident. The acceptance criteria is that the minimum bulk average CRHA temperature will not be below 12.8°C (55°F) on a loss of active cooling for 72 hours given winter post design basis accident conditions.

The staff reviewed this criterion against NUREG-0700 section 12.1.2.1-1, guidance for control room environment temperature winter range. The staff also reviewed NUREG-0700 section 12.2.5.2-3, "Effects of Cold on Performance." While the proposed acceptance criterion is below the 20°C (68°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°C (55°F) would not significantly affect operator performance. Therefore the staff finds the CRHA minimum temperature acceptance criterion acceptable. Evaluation of the ESBWR CRHA design to meet this acceptance criterion is discussed 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 walls, floor, ceiling and interior walls. DCD Tier 2 Table 9.4-1 states that the CRHA is designed such that the maximum CRHA temperature is limited to a value of 33.9°C (93°F). DCD Tier 1 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 design basis accident. The acceptance criteria is that the CRHA maximum bulk average air temperature will not exceed 33.9°C (93°F) on a loss of active cooling for 72 hours given design basis accident 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 8.3º C (15º F) maximum at the end of the postulated 72-hour accident for a control room that has a normal temperature range maintained at from 22.8°C to 25.6°C (73 °F to 78 °F).

Based on the applicant's chosen maximum normal design temperature of 23.3°C (74°F), the ESBWR design maximum accident CRHA temperature of 33.9°C (93°F) results in a temperature rise of 10.6°C (19°F). This exceeds the temperature rise limit guidance in the EPRI URD and NUREG-1242; however it would be within the 93 º F maximum temperature allowed by the URD for a control room with normal temperature maximum value of 25.6°C (78 °F). Therefore, the staff finds the proposed CRHA maximum temperature acceptance criterion acceptable because it is in accordance with the 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.

Impact of CRHA temperature acceptance criteria on CRHA equipment

The staff evaluated whether the maximum CRHA temperature acceptance criterion value of 33.9°C (93°F) as stated in DCD Table 9.4-1, supports the mild environment equipment qualification.

In RAI 9.4-34, the staff requested 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 also requested the applicant provide additional details on how the service temperature of electrical equipment, including computer-based I&C systems, will be determined for the ESBWR. In particular the applicant was asked to provide details on this process for equipment that is planned to be located inside electrical cabinets/panels in the RB and the CB. The applicant was also asked to explain how the detailed design and testing of electrical equipment including enclosures would be carried out 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 definition of "Equipment" in DCD section 3.11.1.3 was clarified to indicate that computer-based I&C equipment is defined by the equipment plus its surrounding enclosure.

DCD section 3.11.4.3 was clarified to indicate that system testing of computer-based I&C equipment within its cabinet or enclosure is preferred.

In DCD subsection 3.11.3.1, the applicant states that the CRHA EQ equipment is to be tested at temperatures that are 10°C (18°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, which endorses Electric Power Research Institute (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 acceptable method for addressing mild-environment qualification of Programmable

Logic Controllers (PLCs). The environmental temperature limit in EPRI TR-107300 is 60°C (140°F) plus 2.7°C (5°F) margin for a total temperature of 62.7°C (145°F) for abnormal operating occurrences in a mild environment. This far exceeds the maximum mild environment temperature of 33.9°C (93°F) proposed for this zone.

In addition, DCD Tier 2 Paragraph 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 Appendix 3H tables do not show such margins. The staff noted that in DCD subsection 3.11.3.2, the applicant referenced that the program margin would be per the guidance of IEEE Standard 323. IEEE Standard 323 recommends that a peak temperature margin of +8°C (+14°F) be applied during the temperature qualification process. Since the applicant is conducting type testing with a 10°C (18°F) margin, the staff finds that the proposed margin exceeds the IEEE Standard 323 guidelines.

Thus, since CHRA computer-based I&C equipment is to be type-tested at 60°C (140°F), with margin, there is significant margin to equipment failure if actual local temperatures exceed the calculated maximum average CRHA bulk temperature of 33.9°C (93°F) by several degrees. Based on margin in the assumed normal operating temperature used in the Control Building heat up 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 concludes 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 concludes that the maximum CRHA temperature value of 33.9°C (93°F) supports mild environment equipment qualification temperature conditions in the CRHA. Based on the applicant's responses, RAI 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°C (93°F) as stated in DCD 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 DCD Figure 3H-2 Revision 6, the applicant's control building heat up analysis indicates that the CRHA dry bulb temperature would reach 30°C (86°F) in approximately 12 hours. After 12 hours, the temperature rate of change is much lower, reaching a CHRA bulk temperature of 33.5°C (92.5°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 the applicant justify use of psychrometric wet bulb temperature as a valid index to assess heat stress in the ESBWR CHRA, 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 the applicant provide a demonstration that such a criterion can be met for the ESBWR environmental footprint. The staff requested the applicant clarify associated ITAAC for this criterion.

In its response to RAI 6.4-24, the applicant revised the DCD to state that the Wet Bulb Globe Temperature (WBGT) index would be the design basis means by which a heat stress acceptance criterion would be measured. The applicant stated that the CHRA is designed such that 32.2 ^{o}C (90 ^{o}F) WBGT would not be exceeded at the end of 72 hours of passive cooling.

The staff has reviewed the proposed DCD revisions and acceptance criterion against NRC and industry guidance and has found that the applicants chosen WBGT index acceptance criterion for heat stress at the end of 72 hours of passive cooling would not be such that compensatory actions, such as stay times would 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 32.2 ^{o}C (90 ^{o}F) WBGT for low-metabolic work with normal work clothes, which is typical of work in the control room. Therefore the staff concludes that the ESBWR CHRA temperature and humidity at the end of 72 hours of passive cooling is acceptable in regard to 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 CRHVS heat sink was analyzed to ensure that the heat sink passively maintains the temperature in the CRHA within the design basis for the first 72 hours following a design basis accident. The means of verification of this design feature is by means of a Control Building Temperature Analysis using heat sink dimensions, thermal properties, exposed surface areas, and the heat loads specified in DCD Tier 2 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, including the limiting site parameters for the ESBWR design and has found them acceptable.

Temperature Evaluation-summer case

The staff evaluated the applicant's submitted control building 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 Section 3H.3.2 describes the applicant's control building environmental temperature analysis. DCD Tables 3H-14 and 3H-15, respectively identifies the input assumptions and the results of the control building environmental temperature analysis. NEDE-33536P, which is a Tier 2* reference in DCD Tier 2 Section 3H, provides a detailed description of the control building environmental temperature analysis. The results indicate that maximum bulk average temperature reached in the CRHA during the 0-72 hour period is less than 33.9°C (93°F), which satisfy the applicant's acceptance criteria.

In RAI 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 its response to RAI 9.4-33, the applicant provided the control building 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 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 thermo physical properties of lighter concrete; 1922.2 kg m^3 (120 lb ft^3) than that specified in the

DCD; 2394.8 kg m^3 (149.5 lb ft^3). In addition the applicant applied a 2000 W margin to the expected sensible heat load in the CRHA.

The staff then consolidated a number of concerns regarding the control building environmental temperature analysis in a new RAI. In RAI 9.4-55 the staff requested the applicant incorporate the Control Building environmental temperature analysis in the DCD.

In response to RAI 9.4-55, the applicant submitted the analysis, LTR NEDE-33536P, as DCD Reference 3H.4-8, and indicated in paragraph 3H.3.2.1 and Table 1.6-1 that this report is Tier 2* information.

The staff determined that the response to RAI 9.4-55 addresses the concerns for RAIs 9.4-33 and 9.4-55 since LTR NEDE-33536P provides a methodology to show that both the baseline Control Building described in the DCD and the as-built Control Building 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 regarding incorporating the Control Building 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 control building environmental temperature analysis as described in LTR NEDE-33536P as a basis for designing the main control room HVAC systems as stated in chapter 9, section 8.2.2.1 of the 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 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 non-safety-related AC power is not available, and the ESBWR CRHA meets the design guidance for maximum control room temperature in the URD and NUREG-1242.

<u>Temperature Evaluation Winter Case</u>

The staff also reviewed the impact of low temperature air at the winter design condition temperature of -40°C (-40°F), on control room operators. The applicant provided an analysis that indicated that CRHA bulk temperature will not be below 16°C (61°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 ^{o}C (61 ^{o}F) at 72 hours. The staff reviewed of the applicant's results and performed confirmatory calculations using a first principles methodology with similar input assumptions. The staff obtained similar results. Based on the analysis results, the staff concludes that the Control Building 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.

CRHA 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, primarily driven by temperature differences within the CRHA and buoyancy forces, provide adequate air circulation. The applicant also noted that air movement is also promoted by normal personnel movement reasonably expected to occur. Since control building environmental temperature analysis does not quantify air movement due to 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 CHRA as a single node in the control building 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 control building environmental temperature analysis also does not include pressure changes in the CRHA due to 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 its responses to RAI 9.4-29, the applicant clarified that the EFU flow rate is consistent with ASHRAE standards for 21 people. In order to illustrate that air movement is also expected to occur due to 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 CHRA airflow design details obtained from this analysis including a description and illustration of the airflow expected in the CRHA occupied zone in DCD Tier 2 Section 6.4. Based on 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 applicants 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 control building 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 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 its response, 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 preoperational testing as described in DCD section 6.4.7 and surveillances as described in the technical specifications, DCD Tier 2 Chapter 16 Subsection 5.5.13, verify that minimum air flow rate to the

CRHA is supplied. The staff determined that the RAI response was acceptable since the technical specifications 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 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.

Based on the safety-related EFUs being designed and tested to supply air to the CRHA at the ASHRAE supply rate that is sufficient for a conservative number of personnel in the MCR and supported by the CRHA design features to promote air mixing described in the DCD, the staff finds that the CBVS meets GDC 19 as it applies to control room air quality and air movement.

<u>Evaluation of Control Building Environmental Temperature Analysis for the ESBWR Summary</u>

In summary of the staff's review of the analytical basis for maximum and minimum temperatures, the staff concludes that the control building environmental temperature analysis adequately predicts maximum CHRA occupied zone bulk temperature within the applicant's acceptance criteria. The control building 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 the winter cases provides reasonable assurance that assumptions in the analysis remain valid. The applicant's maximum and minimum temperature acceptance criteria are adequate to assure that the CHRA would have an acceptable environment for personnel and equipment in a postulated accident. Therefore, the staff concludes that the passive cooling design and associated acceptance criteria are acceptable, and the ESBWR CBVS meets GDC19 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 ancillary diesel generators provide a defense-in-depth function for the CBVS. The non-safety-related ancillary diesels have the RTNSS function to provide ac power for active systems to cool the CRHA after 72 hours. Subsection 8.3.1.1 of the DCD states that the ancillary diesel generators automatically start upon sensing undervoltage on their respective busses. Therefore, the staff notes that the availability of the ancillary diesels generators in practice serve to minimize uncertainties in the performance of the safety-related passive CRHA design features.

<u>Post-Accident beyond 72 hours</u>

The staff reviewed the design of the CBVS to maintain satisfactory environmental conditions in the MCR in accordance with GDC 19 during the long term (post-72-hours).

DCD Tier 2, 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 the applicant clarify the power source for the EFU during the post-72-hour period. In its response to the RAI, the applicant modified the design such that the EFUs rely on ancillary diesel generators, 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 in support of post-72-hour main control room habitability. 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 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 its response, the applicant identified that the AHU's were renamed "recirculation AHUs" in DCD Tier 2 Revision 4. The staff determined that the response was 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 Regulatory Treatment of Non-Safety Systems (RTNSS) qualified AHUs are likely to be available for the post-72 hour period after the onset of a design basis accident, the staff determined that temperature control for post-accident 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 ancillary diesel generator. 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 determined 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 provides guidelines for evaluating the habitability of a nuclear power plant control room during a postulated hazardous chemical release. DCD Tier 2 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 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 guidelines of RG 1.78 have been adequately addressed 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 CRVS 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 a station blackout event, and that the guidance of Regulatory Guide 1.155, including position C.3.2.4 has been met.

Based on the above discussions, the staff finds that the CRVS in conjunction with the CRHA provides adequate protection to permit access to and occupancy of the control room under accident conditions. In addition, the CRVS 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, control of releases of radioactive materials to the environment, 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 Control Building does not contain any portion of the nuclear steam supply process or other equipment that can act as a source of radioactive material; and therefore 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.

<u>Proposed ITAAC and proposed surveillance requirements</u>

The staff reviewed the proposed ITAAC for the Control Building Habitability HVAC subsystem and associated passive design features. The applicant has proposed ITAAC in Tier 1 Table 2.16.2-4, inspections Tests, Analyses and Acceptance Criteria (ITAAC) 4i, 4ii and 4iii whereby the as-built CRHVS 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 design basis accident. The means of verification of this design commitment is a Control Building Temperature analysis using the as built heat sink dimensions, thermal properties, exposed surface areas, as built heat sink thermal properties and the as-built heat loads to confirm the results of the control room design basis control building 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 design basis accident.

In RAI 6.4-24 and it supplements, the staff requested the applicant justify use of psychrometric wet bulb temperature as a valid index to assess heat stress in the ESBWR CHRA, 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 the applicant provide a demonstration that such a criterion can be met for the ESBWR environmental footprint. The staff requested the applicant clarify associated ITAAC for this criterion.

In its response to RAI 6.4-24, the applicant revised DCD Tier 1 Table 2.16.2-4 to include an ITAAC 4iii that requires a licensee to demonstrate this via an analysis updated with as built design information.

The staff has 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 RAI 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 its response to RAI 9.4-33, the applicant provided the control building 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 still did not provide a sufficient control building heat up analysis in the DCD.

The staff then consolidated a number of concerns regarding the control building environmental temperature analysis in a new RAI. In RAI 9.4-55 the staff requested the applicant to incorporate the control building environmental temperature analysis in the DCD and revise the ITAAC to specifically refer to this analysis.

In response to RAI 9.4-55, the applicant submitted the analysis, NEDE-33536P, as DCD Reference 3H.4-8, and indicated in paragraph 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 determined 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 Control Building described in the DCD and the as-built Control Building, 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 Table 2.16.2-6, Inspections Tests, Analyses and Acceptance Criteria (ITAAC) 1 through 12i provides 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 Control Room Habitability Area (CRHA) inleakage does not exceed the unfiltered inleakage assumed by control room operator dose analysis. The method of testing (ASTM E741) included in Table 2.16.2-6 conforms to Regulatory Guide 1.197 integrated test guidance.

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, "ITAAC for the Reactor Building HVAC," of Section 2.16.2, "Heating, Ventilating, and Air-Conditioning Systems," as an ITAAC item, and (2) DCD Tier 2, Chapter 16, "Technical Specifications," Section 3.7.2, "Control Room Habitability Area Heating, Ventilation, and Air Conditioning Subsystem," 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 its 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 DCD Tier 2 Chapter Section 3.7.2. The staff determined that the response was acceptable since the applicant made the changes requested by the staff and the staff confirmed that the requested information was included in DCD Revision 4. 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 Control Building Habitability HVAC subsystem, associated passive design features, and the Emergency Filter Units acceptable.

Regulatory Treatment of Non-Safety Systems

DCD Tier 2, Section 19A.8.4.10, states that component cooling will be performed by the HVAC systems in the RB, Electrical Building, FB, Control Building, and parts of the Turbine Building. In RAI 9.4-39, part D, the staff requested that the applicant identify which parts of the CBVS are classified as RTNSS systems 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 subsection 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. DCD Tier 1 subsection 2.16.2.2 item 10 and Table 2.16.2-4 item 10 were added to provide additional ITAAC for RTNSS functions

DCD Tier 2, Table 19A-2, RTNSS functions, 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 seismic events. 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, Section 19A.3.1.3, the applicant states that the control room habitability area must have adequate temperature controls during an accident to support operator actions and 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, 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 by means of availability treatment for the system in the Availability Controls Manual. In DCD Tier 2, 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 air handling units. As stated in DCD Tier 2, 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, Section 19.A.8.3 states that RTNSS criterion B systems, such as the CRHAVS, have augmented design standards.

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, Section 19A, section 19A.8.4.4, and 19A.8.4.14, the CRHAVS air handling units, auxiliary heating and cooling units and Ancillary Diesel Generators and support systems would be subject to regulatory oversight via the Availability Controls Manual. The staff has reviewed the proposed regulatory treatment design standards and the system design basis information in Tier 2 of the DCD against the criteria for such systems as stated in Regulatory Guide 1.206 subsection C.IV.9, and SECY-95-132 and has determined that the proposed regulatory treatment of the CBVS for RTNSS conforms to this guidance and is therefore acceptable. The staff has reviewed proposed ITAAC for RTNSS functions in DCD Tier 1 Tables 2.16-2-4 and 2.16-2-6 finds the proposed ITAAC provides 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, "Minimization of Contamination," the staff reviewed the CBVS 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, Table 12.3-18, "Regulatory Guide 4.21 Design Objective and Applicable DCD Subsection Information," describes the provisions related to CBVS for:

- 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 Control Building HVAC systems have 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 guidelines of RG 4.21 since the MCR positive pressure will minimize radioactive contamination of the MCR. Section 12.3 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 concludes 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 a COL information item.

9.4.2 Fuel Building HVAC System (FBVS)

9.4.2.1 Regulatory Criteria

The staff reviewed the ESBWR DCD (Rev 7) Subsection 9.4.2 Fuel Building HVAC System in accordance with SRP Sections 9.4.2, "Spent Fuel Pool Area Ventilation System." The staff's acceptance of the 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, “Minimization of Contamination”

9.4.2.2 Summary of Technical Information

The FBVS is non-safety-related except for the isolation dampers and ducting penetrating the Fuel Building boundary. The Fuel Building boundary is automatically isolated in the event of fuel handling accident or other radiological accidents. With the above exception, the FBVS performs no safety-related functions.

The Fuel Building Heating Ventilating and Air Conditioning System (FBVS) serve the following areas of the Fuel Building:

- General areas;
- Spent Fuel Pool;
- Equipment areas.

The FBVS is non-safety-related except for the isolation dampers and ducting penetrating the Fuel Building boundary. The Fuel Building boundary is automatically isolated in the event of 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 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 Subsection 19A.8.3.

The FBVS maintains space design temperatures, quality of air, and pressurization in the Fuel Building. The system consists of two subsystems: the Fuel Building General Area HVAC subsystem (FBGAVS) and the Fuel Building Fuel Pool Area HVAC subsystem (FBFPVS). The FBGAVS serves the general areas of the Fuel Building. The FBFPVS serves the Spent Fuel Pool and equipment areas of the Fuel Building. Recirculation air handling units provide supplementary cooling for selected rooms in the Fuel Building.

The FBGAVS is a once-through air conditioning and ventilation system with air handling unit (AHU), redundant exhaust fans and Fuel Building boundary isolation dampers. The AHU includes filters, heating elements, cooling coils, and redundant AHU supply fans. Outside air is filtered and heated or cooled prior to being distributed by the AHU. A common supply duct system is incorporated to distribute conditioned air to the general areas of the Fuel Building. 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 Fuel Building HVAC Purge Exhaust Filter Unit. Electric unit heaters provide supplementary heating as necessary. A recirculation AHU provides supplementary cooling for the fine motion control rod drive (FMCRD) room. The chilled water system provides cooling water for the FBGAVS AHUs. The instrument air system provides instrument air for the pneumatic actuators.

The FBGAVS AHUs and exhaust fans are located in the Fuel Building HVAC Equipment Area.

The FMCRD maintenance room recirculation AHU is located in the Fuel Building.

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 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 Spent Fuel Pool surface and to the equipment areas. Air is exhausted from the Spent Fuel Pool area, through redundant Fuel Building 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 Fuel Building HVAC Purge Exhaust Filter Unit. The exhaust fans are also used for smoke removal. Electric unit heaters provide supplementary heating as necessary. The chilled water system 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 Fuel Building HVAC Equipment Area.

During high radiation conditions, the Fuel Building 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, due to 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.

On 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 Fuel Building HVAC Purge Exhaust Filter Unit. It is then exhausted to the RB/FB vent stack by the Fuel Building 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 General Design Criteria 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 chapter 19A of the DCD against guidance for selection and identification of such systems stated in Regulatory Guide 1.206 subsection C.IV.9.

GDC 2, Design bases for protection against natural phenomena, requires that structures, systems, and components important to safety shall be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches without loss of capability to perform their safety functions. DCD Tier 2, Section 9.4.2 states that the safety-related portions of the FBVS are designed to comply with guidance of Regulatory Guide (RG) 1.29 position C.1 which specifies a Seismic Category I design. The remainder of the system is classified as non-safety-related and is designed to Seismic Category II per Regulatory Guide (RG) 1.29 position C.2 to assure that the failure of non-safety-related portions of the system can not affect safety-related components. In addition, DCD Tier 2, Section 9.4.2 states that the FB is a Seismic Category I structure except for the penthouse that houses HVAC equipment that 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 a Safe Shutdown Earthquake (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 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 could occur during an SBO. RAI 9.4-52 was being tracked as an open item in the SER with open items. In its response, the applicant stated no components in the FB would be affected by increased temperature during an SBO. The staff reviewed the RAI response and reviewed the safety-related components that are located in the Fuel Building as stated in DCD Table 3.2-1. All electrical components on this table are listed in DCD Table 3.11-1 as environmentally qualified for harsh environments. Since the environmental conditions during a SBO are not anticipated to exceed the harsh environment conditions, the staff determined that the applicant’s response was 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, Sharing of structures, systems, and components, requires that structures, systems, and components 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, Control of releases of radioactive materials to the environment, 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, 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 Fuel Building HVAC Purge Exhaust Filter Unit during periods of high radioactivity. The Fuel Building 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 and therefore conform to the guidelines of SRP 9.4.2. Accordingly, the staff finds that the FBVS complies with the requirements of GDC 60.

GDC 61, Fuel storage and handling and radioactivity control, 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, Section 9.4.2 states that 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 RB HVAC Purge exhaust filter units. As previously noted, the Fuel Building 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 1.13 and, RG 1.140, and therefore conform to the guidelines of SRP 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 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 section 15.4.1.5, the applicant states that no credit is taken for Control Room Emergency Filter Unit (EFU) mitigation, nor is the Reactor Building or Fuel Handling Building integrity assumed for such accident. The staff determined that the applicant's response was acceptable since it clarifies that that FBVS isolation dampers close on a high radiation condition and that credit is not taken for the fuel building 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 due to fuel boiling. As stated in Tier 2 Subsection 9.4.4.3, the FBVS has no functions during an accident other than the FB boundary isolation function. After an accident, the RB purge exhaust filter unit (charcoal filter trains) can be employed to clean up the Fuel Building. The staff determined that the applicant's response was acceptable since the FBVS is a non-safety related system except for the isolation functions and the ESBWR design provides means to clean up the fuel building following the design basis boiling of the spent fuel pool. 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, Section 19A.8.4.10, states that component cooling will be performed by the HVAC systems in the RB, Electrical Building, FB, Control Building, and parts of the Turbine Building. In RAI 9.4-39, part C, the staff requested that the applicant identify which parts of the FBVS are classified as RTNSS systems 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 subsection 9.4.2.1 to state that the FBVS has RTNSS functions as described in DCD appendix 19A, with the associated RTNSS design requirements. DCD Tier 1 subsection 2.16.2.5 item 5, and Table 2.16.2-9 item 5 were added to provide additional ITAAC for RTNSS functions associated with post-72-hour cooling for the FAPCS pump motors and N-DCIS components.

RTNSS functions, DCD Table 19A-2 lists the FBVS as a system that performs functions which 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 and large release frequency. 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 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 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 has reviewed Section 19.A.8.3 and finds that the FBVS would be 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 Section 19A, 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 Tier 2 of the DCD against the criteria for such systems as stated in Regulatory Guide 1.206 subsection C.IV.9, and SECY-95-132 and has determined 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 Table 2.16-2-9 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 C is resolved.

In RAI 9.4-5 and its supplements S01 and 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 in the SER with open items. In its 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 determined that the response was 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, "Minimization of Contamination", the staff reviewed the FBVS 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, Table 12.3-18, "Regulatory Guide 4.21 Design Objective and Applicable DCD Subsection Information," describes the provisions related to the FBVS for:

- Minimizing leaks and spills (design objective 1)
- Decreasing the spread of contaminant from the source (design objective 4)
- Facilitate decommissioning by designing the facility to facilitate the removal of equipment or components that may require removal (design objective 5)
- 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 FB HVAC system maintains a negative pressure in the building to minimize the exfiltration of potentially contaminated air.

The FB HVAC system is provided with access doors for AHUs fans, filter section, and duct-mounted dampers to allow for maintenance as applicable.

On 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 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 Regulatory Guide 1.140.

The staff finds that these design provisions for the FBVS meet the requirement of 10 CFR 20.1406 and conforms to guidelines of RG 4.21. Section 12.3 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 concludes that the ESBWR CVBS design conforms to the requirements GDC 2, 60, and 61, and 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 (Rev 7) Subsection 9.4.3 Radwaste Building Heating, Ventilation and Air Conditioning System (RWVS) in accordance with SRP Sections 9.4.3, "Auxiliary and Radwaste Area Ventilation System." The staff's acceptance of the 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, "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 non-safety-related.

The non-safety-related Radwaste Building HVAC (RWVS) consists of two sub systems: the Radwaste Building Control Room (RWCR) HVAC subsystem (RWCRVS) and the Radwaste Building General Area (RWGA) HVAC subsystem (RWGAVS).

The RWCRVS maintains the Radwaste Building control room 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% 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% capacity exhaust fans.

The RWGAVS is a once-through air conditioning and ventilation system to provide filtered and heated or cooled air to the RWGA. The RWGAVS supply consists of one AHU with two (2) 100% 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% capacity air filtration units (AFU), each with prefilters and HEPA filters, a 50% 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 General Design Criteria for this important but non-safety-related system.

GDC 2, Design bases for protection against natural phenomena, requires that structures, systems, and components important to safety shall be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches without loss of capability to perform their safety functions. DCD Tier 2, Section 9.4.3 states that the RWVS is classified as non-safety and is designed to Seismic Category II per

Regulatory Guide (RG) 1.29 position C.2 to assure that the failure of non-safety-related portions of the system can not affect safety-related components. The RWVS is designed to maintain its structural integrity after a Safe Shutdown Earthquake (SSE). The staff finds that because the RWVS conforms to the guidance of RG 1.29 with respect to seismic categorization, the design conforms to the guidelines of SRP 9.4.3. Accordingly, the staff finds that the RWVS complies with the requirements of GDC 2.

GDC 5, Sharing of structures, systems, and components, requires that structures, systems, and components 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, Control of releases of radioactive materials to the environment, 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, Section 9.4.3 states that the RWVS design includes redundant isolation dampers, ducts, and associated instrumentation which contains the contamination within the radwaste building. 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 Regulatory Guide 1.140. The staff finds that the FBVS design features conform to RG 1.140 and therefore conforms to the guidelines of SRP 9.4.2. Accordingly, the staff finds that the FBVS complies with the requirements of GDC 60.

In RAI 9.4-5 and its supplements S01 and 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 its 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 determined that the response was 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, “Minimization of Contamination”, the staff reviewed the RWVS 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, Table 12.3-18, “Regulatory Guide 4.21 Design Objective and Applicable DCD Subsection Information,” describes the provisions related to RWVS 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)

- 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 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 RWVS Filtration Units. The RWVS Filtration Units are designed, tested, and maintained in accordance with Regulatory Guide 1.140.

The RWCRVS maintains the radwaste building control room areas at a slightly positive pressure relative to adjacent areas to minimize infiltration of air. The RWGAVS maintains the Radwaste Building general area 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 RWB 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 guidelines of RG 4.21. Section 12.3 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 concludes that the ESBWR CVBS design conforms to the requirements GDC 2 and 60, and 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 (Rev 7) Subsection 9.4.4 Turbine Building HVAC System in accordance with SRP Sections 9.4.4, “Turbine Area Ventilation System.” The staff’s acceptance of the TBVS 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, “Minimization of Contamination”

9.4.4.2 Summary of Technical Information

The Turbine Building HVAC system (TBVS) includes the Turbine Building supply air fans and associated filter trains, and the Turbine Building exhaust fans and associated filter trains. The various fan-coil units for local area heating and cooling within the Turbine Building are included in the TBVS.

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 non-safety-related.

The TBVS has RTNSS functions as described in 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 Subsection 19A.8.3.

The TBVS is designed to minimize exfiltration by maintaining a slightly negative pressure in the Turbine Building by exhausting more air than is supplied to the Turbine Building. 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% standby cooling capacity is provided in areas where a loss of cooling could cause degraded equipment performance. Turbine Building 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 Turbine Building areas or component vents is collected, filtered, and discharged to the atmosphere through the Turbine Building Compartment Exhaust (TBCE) system.

Exhaust air from other (low potential airborne contamination) Turbine Building areas and component vents is exhausted to the atmosphere through the Turbine Building Exhaust (TBE) system. Turbine Building exhaust 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 Turbine Building. The chiller rooms, located in the Turbine Building, meet ASHRAE-15, Safety Standard for Refrigeration Systems. They are equipped with a dedicated purge system and leak detectors with alarms.

The non-safety-related Turbine Building HVAC (TBVS) consists of the following sub systems and components:

- Turbine Building Air Supply (TBAS) subsystem
- Turbine Building Exhaust (TBE) Subsystem
- Turbine Building Compartment Exhaust (TBCE) Subsystem
- Turbine Building Lube Oil Area Exhaust (TBLOE) Subsystem
- Turbine Building 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% capacity supply fans. Two of the three fans are normally operating to supply filtered, temperature-controlled air to all levels of the Turbine Building. 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% 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% 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% 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% capacity filter bypass duct for purging smoke in the event of a fire. The air exhausted from the Turbine Building 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% capacity fans, isolation dampers, low efficiency filters, and exhaust ductwork. The TBLOE fans discharge the exhaust air directly to Turbine Building Exhaust 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, which includes one 100% 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 Turbine Building. The AHUs are supplied with chilled water from the balance of plant subsystem of the Chilled Water System (CWS) 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 2-100% redundant recirculation air handling units. 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 Reactor Component Cooling Water System (RCCWS), Nuclear

Island subsystem of the CWS, selected electrical equipment rooms and Instrument/Service Air System rooms are fed from the corresponding Nuclear Island subsystem of the CWS 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 Turbine Building, 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 Turbine Building. 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% 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 non-safety-related diesel generators. The local AHUs of the RCCWS, Nuclear Island subsystem of the CWS and Instrument/Service Air System rooms and selected electrical equipment rooms also remain in operation.

9.4.4.3 Staff Evaluation

The staff review focused on compliance with General Design Criteria for this important but nonsafety-related system. The staff has also reviewed the RTNSS functions for the TBVS as stated in chapter 19A of the DCD against guidance for selection and identification of such systems stated in Regulatory Guide 1.206 subsection C.IV.9.

GDC 2, Design bases for protection against natural phenomena, requires that structures, systems, and components important to safety shall be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches without loss of capability to perform their safety functions. DCD Tier 2, Section 9.4.4 states that the TBVS is classified as non-safety-related and is designed to Seismic Category NS per Regulatory Guide (RG) 1.29 position C.2 to assure that the failure of non-safety-related portions of the system can not affect safety-related components. The turbine building is a Seismic Category II non-safety-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 9.4.2. Accordingly, the staff finds that the TBVS complies with the requirements of GDC 2.

GDC 5, Sharing of structures, systems, and components, requires that structures, systems, and components 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, Control of releases of radioactive materials to the environment, 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, as to 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 Regulatory Guide 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. Provision has been made 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 Section 9.5.6.

The TBVS has adequate instrumentation that alarm in the main control room for adverse radiological conditions, temperature conditions, 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 9.4.4. Accordingly, the staff finds that the TBVS complies with the requirements of GDC 60.

DCD Tier 2, Section 19A.8.4.10, states that component cooling will be performed by the HVAC systems in the RB, Electrical Building, FB, Control Building, and parts of the Turbine Building. 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 subsection 9.4.4.1 to state that the TBVS has RTNSS functions as described in DCD appendix 19A. The applicant also revised DCD Tier 1 subsection 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 Turbine Building and room cooling for the Nuclear Island Chilled Water System and RCCWS pumps.

DCD Tier 2, 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 core damage frequency and large release frequency. The EBVS is a support system for the FAPCS. It provides equipment and room cooling to support RCCWS, Nuclear Island Chilled Water System, and associated N-DCIS support cooling. In DCD 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 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 has reviewed 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 Section 19A, section 19A.8.4.9, the room cooler portions of the TBVS 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 Tier 2 of the DCD against the criteria for such systems as stated in Regulatory Guide 1.206 subsection C.IV.9, and SECY-95-132 and has determined that the proposed regulatory treatment of the TBVS for RTNSS is acceptable, as described above. The staff has reviewed proposed ITAAC for RTNSS functions in DCD Tier 1 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 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. Furthermore, 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 its response, the applicant stated the TB exhaust (TBE) system design has been changed, and filter units have been removed. The applicant also modified DCD section 9.44, Table 9.4-15 and figure 9.2-8 in revision 4 of DCD to be consistent. The staff determined that the response was 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 and its supplements S01 and 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 in the SER with open items. In its 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 determined that the response was 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, “Minimization of Contamination”, 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, Table 12.3-18, “Regulatory Guide 4.21 Design Objective and Applicable DCD Subsection Information,” describes the provisions related to TBVS 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)
- 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, as to 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 Regulatory Guide 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 air handler units 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 12.3 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 concludes that the ESBWR TBVS design conforms to the requirements 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 7, Sections 6.4 and 9.4.1. The staff's evaluation of the EFUs is provided in Sections 6.4 and 9.4.1 of this report.

9.4.6 Reactor Building HVAC System

9.4.6.1 Regulatory Criteria

The staff reviewed the ESBWR DCD (Rev 7) Subsection 9.4.6 Reactor Building HVAC System in accordance with SRP Sections 9.4.3, "Auxiliary and Radwaste Area Ventilation System. For those areas that contain safety-related equipment, the staff reviewed Subsection 9.4.6 in accordance with SRP Sections 9.4.5, "Engineered Safety Feature Ventilation System". The staff's acceptance of the RBVS 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 50.63 as it relates to necessary support systems providing sufficient capacity and capability to ensure the capability for coping with a station blackout event.
- 10 CFR 20.1406, "Minimization of Contamination"

9.4.6.2 Summary of Technical Information

The Reactor Building HVAC System (RBVS) maintains the design temperature, quality of air, and pressurization in the Reactor Building (RB) spaces. The isolation dampers and ducting penetrating the Reactor Building boundary and associated controls that provide the isolation signal are safety-related. The RBVS performs the safety-related function of automatic isolation of the Reactor Building boundary during accidents.

The RBVS serves the following areas of the RB:

- The potentially contaminated areas (CONAVS);
- The refueling area (REPAVS);
- The non-radiologically controlled areas (CLAVS); and
  Containment during inerting and de-inerting operations

The RBVS has the safety-related function of building isolation. The isolation dampers and ducting penetrating the Reactor Building boundary and associated controls that provide the isolation signal are safety-related. The RBVS performs the safety-related function of automatic isolation of the Reactor Building boundary (CONAVS and REPAVS subsystems) during accidents. The RBVS has non-safety-related Reactor Building Purge Exhaust Filter Units for mitigating and controlling gaseous effluents from the Reactor Building. The RBVS has non-safety-related Reactor Building HVAC Accident Exhaust Filter Units for use post accident (>8 hours) to create a negative pressure in the Reactor Building contaminated areas and exhaust the filtered air to the RB/FB stack.

The RBVS has RTNSS functions as described in 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 Subsection 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 Reactor Building boundary (CONAVS and REPAVS subsystems) to prevent uncontrolled releases to the outside atmosphere

The RBVS provides the capability to manually divert exhaust air for processing through the Reactor Building HVAC On-line Purge Exhaust Filter Units.

Reactor Building 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% by volume in accordance with RG 1.128 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 area.

The RBVS maintains SLC accumulator room environmental conditions within temperature-limits including employing two backup heaters per room. PIP A and PIP B busses provide power for these heaters.

The RBVS provides cooling for CRD and RWCU pump motors, rooms, and/or electrical/instrument panels designed to limit the room/equipment to within its temperature environmental qualification when the building is isolated. The motor cooler heat sink is the RCCW, while Chilled Water or Direct Expansion Units are provided for electrical cabinet cooling.

The RBVS consists of three subsystems. The Reactor Building Contaminated Area HVAC Subsystem (CONAVS) serves the potentially contaminated areas of the Reactor Building. The Refueling and Pool Area HVAC Subsystem (REPAVS) serves the refueling area of the Reactor Building. The Reactor Building Clean Area HVAC Subsystem (CLAVS) serves the clean (non-radiological controlled) areas of the Reactor Building.

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 prior to distribution by the AHU in service. The chilled water system provides cooling for the CONAVS AHUs. The Instrument Air System provides instrument air for the pneumatic actuators. A common supply air duct distributes conditioned air to the potentially contaminated areas of the Reactor Building.

Air is exhausted from the potentially contaminated areas of the Reactor Building 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 then used for smoke removal with the exhaust air being monitored for radiological contamination. If contaminated, temporary portable filters may be used to exhaust the contaminated air. The building isolation dampers close and the supply and exhaust fans stop due to high radiation in the exhaust ducts. CONAVS also includes redundant Reactor Building 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 Reactor Building 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 CONVAS 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 motors coolers from RCCWS and electrical/instrument panels are provided with either Chilled Water or Direct Expansion Units designed to limit the room/equipment to within its temperature environmental qualification when the building is isolated. Electric unit heaters provide supplementary heating. The CONAVS AHUs are located in the Fuel Building HVAC Equipment Area. The CONAVS exhaust fans are located in the Reactor Building. The Reactor Building HVAC Purge Exhaust Filter Units and exhaust fans are located in the Reactor Building. The refueling machine control room recirculating AHU is located in the Reactor Building. Electric unit heaters are located in or near the areas they serve.

The REPAVS is a once-through ventilation system and consists 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 prior to 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 chilled water system provides cooling water for the REPAVS AHU. The instrument air system 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 contaminated, temporary portable filters are used to exhaust the contaminated air. The building isolation dampers close and the supply and exhaust fans stop due to high radiation in the exhaust ducts.

The REPAVS AHUs are located in the Fuel Building HVAC Equipment Area. The REPAVS exhaust fans are located in the Reactor Building. 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/cooled prior to distribution by the AHU in service. A common supply and return/exhaust air duct system distributes conditioned air to and from the Reactor Building 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 Reactor Building 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 chilled water system provides cooling for the CLAVS AHU. The instrument air system 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 in order to keep hydrogen concentration below 2%. 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 Fuel Building 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 General Design Criteria for this system which has a safety-related isolation function. The remainder of the system is classified as non-safety. In order 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 station blackout. The staff has also reviewed the RTNSS functions for the RBVS as stated in chapter 19A of the DCD against guidance for selection and identification of such systems stated in Regulatory Guide 1.206 subsection C.IV.9.

GDC 2, Design bases for protection against natural phenomena, requires that structures, systems, and components important to safety shall be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches without loss of capability to perform their safety functions. DCD Tier 2, Section 9.4.4 states that the RBVS isolation dampers, associated instrumentation and duct are classified as safety-related and are designed to Seismic Category I per Regulatory Guide 1.29 position C.1. The remainder of the system is classified as non-safety-related and is designed to Seismic Category II per Regulatory Guide 1.29 position C.2 to assure that the failure of nonsafety-related portions of the system can not 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 the 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 item in the SER with open items. In response, the applicant made several changes to the DCD Tier 2, Section 9.4.6 figures and tables, including (1) revising REPAVS system DCD tier 2 Figure 9.4-11 to be consistent with table 9.4-10, (2) revising figure 9.4-9 and table 9.4-9 to include all the REPAVS building isolation dampers, (3) revising DCD tier 2 Figure 9.4-10 to include all CONAVS building isolation dampers, and (4) revising Table 9.4-11 to identify the building isolation dampers as safety-related. The staff determined that the applicant's response was acceptable since the RBVS safety-related isolation dampers are clearly and consistently identified. The staff confirmed that the DCD changes were incorporated 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 determined that the response was acceptable since applicant included statements in the DCD for monitoring and exhausting hydrogen from the safety related battery rooms. Based on the above, the applicant's responses and DCD changes, RAI 9.4-43 is resolved. The staff confirmed that the DCD changes were incorporated into DCD Revision 4.

In RAI 9.4-45, the staff requested that the applicant make tables and figures of the main steam tunnel AHU, main steam tunnel recirculation AHU and the refuelling machine control room recirculation AHU consistent and to clarify the location of CONAVS safety-related dampers. RAI 9.4-45 was being tracked as 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 determined that the RAI response was acceptable since the appropriate AHUs are included in the DCD Tier 2 Section 9.4 figures and tables and the isolation dampers are clearly identified. 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. Since 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 open item in the SER with open items. In response to the RAI the applicant changed revised 9.4-9 to show the building isolation dampers and revised Figure 9.4-9 to show the CLAVS isolation dampers. The applicant clarified that since only clean areas are serviced by the CLAVS, radiation monitoring is not required. The staff determined that the applicant's response was acceptable since appropriate changes were made to Figure 9.4-9. In addition, since Figure 9.4-9 is for the CLAVS or the clean portion of the reactor building, 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 9.4.3. Accordingly, the staff finds that the RBVS complies with the requirements of GDC 2.

GDC 5, Sharing of structures, systems, and components, requires that structures, systems, and components 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, Control of releases of radioactive materials to the environment*,* 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, 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 of directing the system exhaust air to the Reactor Building HVAC purge exhaust Filtration Units. Reactor Building HVAC purge exhaust are designed, tested, and maintained in accordance with Regulatory Guide 1.140. Under accident conditions, the RBVS is isolated by safety-related dampers, duct, and instruments to prevent the release of contamination to the environment through the intake and exhaust pathways.

In RAI 9.4-5 and its supplements S01 and 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 in the SER with open items. In its 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 ESBWR HVAC systems. The applicant also provided a table in the RAI response showing where relevant standards are discussed throughout the DCD. The staff determined that the response was 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 RAI 9.4-47 and supplemental RAIs, the staff requested the applicant identify how the CLAVS exhaust air is monitored for radiation since 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 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 determined that the applicant's response was acceptable since with the design change of directing the CLAVS exhaust air to the RB/FB vent stack, the staff finds that there is reasonable assurance that releases from the CLAVS area of the reactor building 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 its response, the applicant clarified that no credit has been 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 post accident dose evaluations credited use of the Reactor Building HVAC Accident Exhaust Filter Units for mitigating dose consequences. The applicant also identified that the Reactor Building 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 ITAAC Table 2.16-2-2 items 11 and 12b to clarify that the filter must meet two separate tests of RG 1.140 and ASME AG-1; the efficiency as tested in the laboratory and the in place bypass leakage test which is done in the field. The staff determined that the applicant's response was acceptable since the exfiltration flow from the RB in the dose calculations does not depend on the operation of the either CONAVS or the Reactor Building HVAC Accident Exhaust Filter

Units. The ITAAC change is acceptable since it confirms that the Reactor Building HVAC Accident Exhaust Filter Units meet regulatory guidelines in RG 1.140 for testing non-safety related air filtration units. Accordingly, based on the above, the applicant's responses, and DCD changes, RAI 9.4-53 is resolved. The applicant identified that a portion of the response to RAI 9.4-53 was inadvertently deleted from DCD revision 7 and has provided a markup to include it in DCD revision 8. **RAI 9.4-53 is being tracked as a confirmatory item.**

The staff finds that the RBVS design features conforms to RG 1.140, the therefore conform to the guidelines of SRP 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 a station blackout (i.e., loss of offsite electric power system concurrent with reactor trip and unavailability of the onsite emergency ac electric power system). A station blackout 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 station blackout requirements.

RB 0-72 hour temperature-Introduction

DCD Tier 2 Section 9.4.6.2 states that the RBVS is not required to operate during a station blackout. DCD Tier 2 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 Table 3H-15 lists the results of the applicant's environmental temperature analysis for the RB.

The staff chose a room (#1720) which 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 non-safety related ac power is assumed lost. After 72 hours, RBVS and the CLAVS and the CONAVS are expected to function. As described below these subsystems support the Regulatory Treatment of Non-Safety Systems (RTNSS) function of post 72 hour cooling for the DCIS cabinets and their electrical supporting equipment.

The applicant has proposed an analytical approach, NEDE-33536, "Control Building and Reactor Building Environmental Temperature Analysis" (hereafter referred to as the Reactor Building 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 Control building portion of the Control Building and Reactor Building Environmental Temperature Analysis for the ESBWR NEDE-33536P are described in Section 9.4.1.3 of this report, and are similar to those used to review the Reactor Building portion of this report.

5 Yilmaz T. P. & Paschal W. B. article: "An analytical approach to transient room temperature analysis," *Nuclear Technology,*114:135-140

Input Assumptions

RB Heat loads and heat sinks

The staff has reviewed input parameters used in the applicants Reactor Building Environmental Temperature Analysis in NEDE-33536P, such as heat sink wall thicknesses and surface areas 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 Reactor Building concrete, a conservative assumption of internal heat loads that assume a HELB with a SBO are considered. Internal heat loads assumed are documented in NEDE-33536P, Table G-3 for each room. The applicant assumed the highest normal operating temperature allowed in each room 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 9.4-58 the staff requested the applicant incorporate the Control Building and Reactor Building Environmental Temperature Analysis for the ESBWR NEDE-33536P in the DCD and 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 reactor building heat up analysis, and revised. Tier 1 Table 2.16.2-2, to add ITAAC 13. ITAAC #13 requires an applicant to demonstrate passive heat sink performance of the Reactor Building. An applicant is to perform the design basis Reactor Building Environmental Analysis using as-built information. The staff determined that the applicant's response was acceptable since the Reactor Building Environmental Analysis uses a similar methodology to the Control Building Environmental 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 Reactor Building Environmental 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 applicants 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 RB Maximum Temperature Acceptance Criteria

The staff evaluated the proposed ESBWR RB maximum temperature criteria. 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 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°C (122°F) as stated in DCD Appendix 3H Table 3H-9.

Therefore, 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 RB temperature acceptance criteria on RB equipment

The staff evaluated the impact maximum RB temperature acceptance criterion value of 50°C (122°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 I&C systems, will be determined for the ESBWR. In particular the applicant was asked to provide details on this process for equipment that is planned to be located inside electrical cabinets/panels in the RB and the CB. The applicant was also asked to explain how the detailed design and testing of electrical equipment including enclosures would be carried out 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.

DCD section 3.11.1.3 definition of equipment was clarified to indicate that computer-based I&C equipment is defined by the equipment plus its surrounding cabinet or enclosure.

DCD section 3.11.4.3 was clarified to indicate that system testing of computer-based I&C equipment within its cabinet or enclosure is preferred.

In DCD subsection 3.11.3.1, the applicant states that the RB computer-based I&C equipment is to be type tested at temperatures that are 10°C (18°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 I&C equipment is to be qualified at the nominal temperature of 50°C (122°F) as stated in DCD Appendix 3H Table 3H-9. In addition, DCD Tier 2 Paragraph 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 Appendix 3H tables do not show such margins. The staff noted that in DCD subsection 3.11.3.2, the applicant referenced that the program margin would be per the guidance of IEEE Standard 323. IEEE Standard 323 recommends that a peak temperature margin of +8°C (+14°F) be applied during the temperature qualification process. Since the applicant is conducting type testing with a 10°C (18°F) margin, the staff finds that the applicants exceeds the IEEE Standard 323 guidelines. The staff determined that applicant's response was acceptable since the DCD was modified to state that computer-based I&C systems are tested in the enclosures and that a 10°C (18°F) test margin 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 reactor building. Based on the maximum RB temperature acceptance criterion value being the equipment qualification temperature and the 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 design basis accident. The means of verification of this design feature is by means of a Reactor Building Temperature analysis using heat sink dimensions, thermal properties, exposed surface areas, heat sink thermal properties and the heat loads specified in

DCD Tier 2 Table 3H-14. As previously discussed, the staff reviewed these input assumptions and has found them acceptable.

The staff has reviewed the results of the applicant's reactor building environmental temperature analysis as described in NEDE-33536P as a basis for demonstrating that RB can be passively cooled during the postulated accident. The staff reviewed of 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 Reactor Building environmental temperature analysis as described in NEDE-33536P, confirmed by staff calculations, which shows that the calculated Reactor Building room temperatures remain below equipment qualification temperatures, and its confirmation using as-built information, the staff finds that there is confidence that reactor building 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 Regulatory Guide 1.155, including position C.3.2.4 and therefore addresses the requirements of 10 CFR 50.63 in that necessary support systems provide sufficient capacity and capability for coping with a station blackout event.

Regulatory Treatment of Non-Safety Systems

DCD Tier 2, Section 19A.8.4.10, states that component cooling will be performed by the HVAC systems in the RB, Electrical Building, FB, Control Building, and parts of the Turbine Building. 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 subsection 9.4.6.1 to state that the RB HVAC system has RTNSS functions as described in DCD appendix 19A, with the associated RTNSS design requirements DCD Tier 1 subsection 2.16.2.1 items 7 and Items 12a and 12b, and Table 2.16.2-2 item 7 and Item 12b were added to provide additional ITAAC for RTNSS functions associated with post-72-hour N-DCIS cooling for FAPCS and Reactor Building HVAC Accident Exhaust filter efficiency.

RTNSS functions, DCD Table 19A-2 lists the RBVS Accident Exhaust Filters as a system that performs functions which falls 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 subsection 19.A.6.2.2, failure to provide adequate filtration was judged by the applicant to be an adverse system interaction. The applicant has chosen to apply regulatory oversight by means of availability treatment for the system through the Availability Controls Manual (ACM) to provide assurance that the filters will be capable of performing their function. In addition, as stated in DCD 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 has reviewed Section 19.A.8.3 and finds that the RBVS contaminated area ventilation system filters would be designed in accordance with standards for RTNSS criterion E systems.

The staff has reviewed the applicant's proposed regulatory treatment of the system and system description and finds that, as described in DCD Section 19A, the RBVS Accident Exhaust Filters are RTNSS systems, and this portion of the RBVS would be subject to regulatory oversight via the ACM and the Maintenance Rule. The staff has reviewed the proposed regulatory treatment design standards and the system design basis information in Tier 2 of the DCD against the criteria for such systems as stated in Regulatory Guide 1.206 subsection C.IV.9, and SECY-95-132 and has determined that the proposed regulatory treatment of the RBVS Accident Exhaust Filters for RTNSS conform to this guidance and is therefore acceptable. The staff has reviewed proposed ITAAC for RTNSS functions in DCD Tier 1 Table 2.16-2-2 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 A is resolved.

<u>Minimization of Contamination</u>

In consideration of 10 CFR 20.1406, "Minimization of Contamination", the staff reviewed the RBVS 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, Table 12.3-18, "Regulatory Guide 4.21 Design Objective and Applicable DCD Subsection Information," describes the provisions related to RBVS for:

- 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)

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 Reactor Building HVAC purge exhaust Filtration Units. Reactor Building HVAC purge exhaust are designed, tested, and maintained in accordance with Regulatory Guide 1.140. Under accident conditions, the RBVS is isolated by safety-related dampers, duct, 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 9.4.3.

The RBVS CONAVS subsystem 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 Reactor building HVAC Purge Exhaust Filter Units. The building isolation dampers close and the supply and exhaust fans stop due to high radiation in the exhaust ducts.

The CLAVS subsystem 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 air handler units 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 12.3 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 concludes 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 ESBWR DCD (Rev 7) Subsection 9.4.7 Electrical Building HVAC System in accordance with SRP Sections 9.4.3, “Auxiliary and Radwaste Area Ventilation System. The staff’s acceptance of the EBVS 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 19, as it applies to habitability criteria specified by NUREG-0696 for the Technical Support Center (TSC)
- GDC 60, regarding the capability of the system to suitably control release of gaseous radioactive effluents to the environment
- 10 CFR 20.1406, “Minimization of Contamination”

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

- NUREG-0696 for guidance in establishing the habitability criteria for the Technical Support Center (TSC)

9.4.7.2 Summary of Technical Information

The Electrical Building HVAC System (EBVS) maintains acceptable temperatures for equipment and personnel comfort and habitability in the Electrical Building. It consists of three subsystems: the Electric and Electronic Rooms (EER) HVAC subsystem (EERVS), the Technical Support Center (TSC) HVAC subsystem (TSCVS), and the Diesel Generators (DG) HVAC subsystem (DGVS). The EERVS and DGVS do not perform or ensure any safety-related function, and thus has no safety design basis. The TSCVS performs functions related to emergency response facilities.

The EBVS is classified as non-safety-related. The EBVS has RTNSS functions as described in 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 Subsection 19A.8.3.

The EERVS subsystem 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 non-safety-related battery rooms below 2% by volume in accordance with RG 1.128

The onsite diesel generators provide electrical power to the EERVS in case of a Loss of Preferred Power (LOPP). EBVS provides the post-72-hour cooing for safety-related electrical distribution and support for electrical power to FAPCS.

The EERVS provides a controlled environment for the Electrical Building switchgear, electronic and non-safety-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 7 battery rooms.

The TSCVS subsystem provides a controlled environment for personal comfort and safety and for the proper operation and integrity of equipment in the TSC and maintains the TSC at a slightly positive pressure with respect to the adjacent rooms and outside environment to minimize the infiltration of air. The TSC HVAC subsystem 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 Air Filtration Units (AFU) with supply fans, high efficiency particulate air (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 slight positive pressure. The recirculating AHU system includes redundant air handling units (with fans, air mixing plenum, filters, heating and cooling coils, and humidifier) to provide conditioned air through ducts, dampers, and registers to the TSC. The exhaust system includes redundant fans to direct the air from the kitchen and toilet areas into the atmosphere.

The TSCVS contains non-safety–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. Since RG 1.140 applies specifically to normal atmosphere cleanup, and since the filter units are not credited ESF units per RG 1.52, the Codes and Standards that dictate the testing of a filtration system designed for habitability are described. 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 on-site diesel generators provide electrical power to the TSC HVAC subsystem in case of Loss of Preferred Power (LOPP).

The DGVS subsystem 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, and prevents the accumulation of combustible vapors and dissipate their concentration in the fuel oil day tank room. The onsite diesel generators provide electrical power to the DGVS in case of a Loss of Preferred Power (LOPP).

9.4.7.3 Staff Evaluation

The staff review focused on compliance with General Design Criteria for this system. The system is classified as non-safety-related. In addition, the staff considered the guidance of NUREG-0696 and the EPRI Utilities Requirement Document Section 4.6.6. The staff has also reviewed the RTNSS functions for EVBS as stated in chapter 19A of the DCD against guidance for selection and identification of such systems stated in Regulatory Guide 1.206 subsection C.IV.9.

GDC 2, Design bases for protection against natural phenomena, requires that structures, systems, and components important to safety shall be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches without loss of capability to perform their safety functions.

DCD Tier 2, Section 9.4.7 states that the EBVS complies with Regulatory Guide 1.29 position C.2 for non-safety-related portions of the system. The EBVS components are designated as Seismic Category NS. The Electrical building is non safety-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 9.4.3. Accordingly, the staff finds that the EBVS complies with the requirements of GDC 2.

GDC 5, Sharing of structures, systems, and components, requires that structures, systems, and components 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, Control Room, 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 loss-of-coolant accidents. Adequate radiation protection shall be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident.

DCD Tier 2 Section 9.4.7, Revision 7 states that the ESBWR TSC, located in the electrical building, is designed to comply with the NUREG-0696 guidance. NUREG-0696 Section 2.6 provides guidance on TSC habitability, stating 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 inplant 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 a TSC ventilation system that includes high-efficiency particulate air (HEPA) and charcoal filters are needed, as a minimum.

The TSCVS design includes high-efficiency particulate air (HEPA) and charcoal filters. DCD Tier 2 Section 9.4.7.2 states that the TSCVS filter units will be designed, and tested in accordance with RG 1.140, "Design Inspection and Testing Criteria for Air Filtration and Adsorption units of Normal Atmosphere Cleanup Systems". NUREG-0696 references Standard Review Plan 6.4, "Control Room Habitability System" which states that Regulatory Guide 1.52, "Design Inspection and Testing Criteria for Air Filtration and Adsorption Units of Post-Accident Engineered Safety Feature Atmospheric Cleanup Systems in Light-Water-Cooled Nuclear Power Plants" should be referenced as guidance for ventilation system design and 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 ESF system, On this basis, and in view of the above, the staff concludes that use of RG 1.140 to meet NUREG-0696 guidance on habitability of the TSC is acceptable, and 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 Tier 1 Section 2.3 Subsection 2.3.2 and Table 2.3.1-1.

The TSCVS is supplied by a non-safety-related power source to the TSCVS. As stated in DCD Tier 2, section 9.4.7.1, the non-safety-related on-site diesel generators provide electrical power to the TSC HVAC subsystem in case of Loss of Preferred Power (LOPP). Although the supply of AC power to the TSCVS is not identified as a RTNSS function, the staff notes that availability of power to the TSCVS is enhanced by the RTNSS regulatory treatment of the non-safety-related on-site diesel generators in the Availability Controls Manual. If all ac power is lost during an accident, per 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 its 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, subsection 2.16.2.7, Electrical Building HVAC System, to include TSCVS ITAAC required to confirm that the TSC provides a habitable work environment when non-safety-related power is available. Design Description 2.16.2.7, items (3), (4) and (5) and ITAAC Table 2.16.2-10 items (3), (4) and (5) provide assurance that the TSCVS Air Filtration Unit (AFU) HEPA filters and charcoal 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 determined that the applicant's response was acceptable since as discussed above, DCD Tier 2, Revision 7 adequately addresses conformance with NUREG-0696. In addition, the EBVS ITAAC incorporates 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 the TSCVS conformance to NUREG-0696 and the corresponding ITAAC, the staff finds that EBVS complies with the requirements of GDC 19.

GDC 60, Control of releases of radioactive materials to the environment, 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 Electrical Building, 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 figure 9.4-12 to show the air inlet louvers more clearly and indicated that Table 9.4-16 lists EBVS subsystem flow rates. DCD Tier 2, Revision 4 was updated to describe EBVS more clearly. The applicant indicated that major component data is included in DCD Tier 2, Table 9.4-16. The staff determined that the RAI response was acceptable since the revised DCD Tier 2 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, Electrical Building, FB, Control Building, and parts of the Turbine Building. 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 subsection 9.4.7.1 to state that the EB HVAC system has RTNSS functions as described in DCD appendix 19A, with the associated RTNSS design requirements. DCD Tier 1 subsection 2.16.2.7, and Table 2.16.2-10 were added to provide additional ITAAC for RTNSS functions associated with post-72-hour cooling for Diesel Generators and safety-related electrical distribution, and support for electrical power to FAPCS.

RTNSS functions, DCD Table 19A-2 lists the EBVS as a system that performs functions which 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 and large release frequency. The EBVS is a support system for the FAPCS. It provides equipment and room cooling to support the standby diesel generators and PIP Buses. In DCD 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 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 has reviewed Section 19.A.8.3 and finds that the EBVS 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 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 has reviewed the proposed regulatory treatment; design standards and the system design basis information in Tier 2 of the DCD against the criteria for such systems as stated in Regulatory Guide 1.206 subsection C.IV.9, and SECY-95-132 and has determined that the proposed regulatory treatment of the EBVS for RTNSS conforms to this guidance and is therefore acceptable. The staff has reviewed the response to RAI 9.4-39, part B and proposed ITAAC for RTNSS functions in DCD Tier 1 Table 2.16-2-10 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 applicant's response and DCD revision, RAI 9.4-39, part B is resolved.

In RAI 9.4-5 and its supplements S01 and 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 in the SER with open items. In its 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 determined that the response was 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, "Minimization of Contamination," the staff reviewed the EBVS 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, Table 12.3-18, "Regulatory Guide 4.21 Design Objective and Applicable DCD Subsection Information," describes the provisions related to EBVS for:

- 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 EER, TSC and the Diesel Building HVAC systems 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 meets the requirement of 10 CFR 20.1406 and conforms to guidelines of RG 4.21. Section 12.3 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 concludes that the ESBWR EBVS design conforms to the requirements 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 concludes 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 the ESBWR DCD (Rev 7) Subsection 9.4.8 Drywell Cooling System in accordance with SRP Sections 9.4.3, “Auxiliary and Radwaste Area Ventilation System. The staff’s acceptance of the DCS 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, “Minimization of Contamination”

9.4.8.2 Summary of Technical Information

The Drywell Cooling System (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 Chilled Water System (CWS) water. There are separate FCUs for the upper and the lower drywell regions.

The Drywell Cooling System (DCS) is classified as a non-safety-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 the DCS 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 the lower drywell. Ducts distribute the cooled, recirculated air/nitrogen through diffusers and nozzles. The FCUs and the fans are redundant.

The drywell heat loads are transferred to the CWS through the cooling coils of the FCUs. 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% of 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 in 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 with an increase in the ambient temperature.

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, 50% of the lower 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 in 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 with an increase in the ambient temperature.

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 a Leak Detection and Isolation System (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 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 (ILRT).

9.4.8.3 Staff Evaluation

The staff review focused on compliance with General Design Criteria for this system. The system is classified as non-safety-related.

GDC 2, Design bases for protection against natural phenomena, requires that structures, systems, and components important to safety shall be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches without loss of capability to perform their safety functions. DCD Tier 2, Section 9.4.8 states that the DCS complies with Regulatory Guide 1.29 position C.2 for non-safety-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 9.4.3 for GDC 2. Accordingly, the staff finds that the DCS complies with the requirements of GDC 2.

GDC 5, Sharing of structures, systems, and components, requires that structures, systems, and components 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, Control of releases of radioactive materials to the environment, 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, Section 9.4.8 states that the DCS includes the capability to suitably control release of gaseous radioactive effluents to the environment. The fan cooler units 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 Leak Detection and Isolation System to determine the condensation rate contribution to the unidentified leakage. The staff finds that the DCS design features are conforms to the guidelines of SRP 9.4.3. Accordingly, the staff finds that the DCS complies with the requirements of GDC 60.

In RAI 9.4-5 and its supplements S01 and 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 in the SER with open items. In its 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 determined that the response was 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, “Minimization of Contamination,” the staff reviewed the DCS 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, Table 12.3-18, “Regulatory Guide 4.21 Design Objective and Applicable DCD Subsection Information,” describes the provisions related to DCS for:

- 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 DW purge operations is the containment air connected with the CONVAS subsystem of the RBVS. During DW purge operations, the containment purge fan can be used to discharge containment air to the CONVAS subsystem. The CONVAS system has RB HVAC Purge Exhaust Filter Units that are designed, tested and maintained in accordance with Regulatory Guide 1.140.

The staff finds that these design provisions for the DCS meets the requirement of 10 CFR 20.1406 and conform to the guidelines of RG 4.21. Section 12.3 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 concludes that the ESBWR DCS design conforms to the requirements 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 containment inerting system (CIS).

9.4.9.2 Summary of Technical Information

The CIS is described in DCD Tier 2, Revision 7, Section 6.2.5.2 and Section 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, 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 RB HVAC system exhaust before being diverted to the plant stack.

The CIS also provides nitrogen to the high pressure nitrogen supply system (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). 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 ESBWR DCD Tier 1, Section 2.16.3, "Fire Protection System," and Tier 2, Section 9.5.1, "Fire Protection System," in accordance with Section 9.5.1, "Fire Protection Program," Revision 5, of NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants" (hereafter referred to as the SRP). : The staff's acceptance of the ESBWR fire protection program (FPP) is based on meeting the relevant requirements of the following GDC and regulations:

- 10 CFR 50.48(a)(4) requires, in part, that each applicant for a design certification under Part 52 must have a description and analysis of the fire protection design features for the standard plant necessary to demonstrate compliance with General Design Criterion (GDC) 3, "Fire Protection," in Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities."

- GDC 3, "Fire Protection," requires the following:

  - Structures, Systems, and Components 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, "Control Room," 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 inspections, tests, analysis, and acceptance criteria (ITAAC) which are necessary and sufficient to provide reasonable assurance that, if 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, “Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety Systems in Passive Plant Designs,” dated March 28, 1994. 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 SER includes a summary of key ESBWR fire protection design commitments by the applicant that are set forth in the ESBWR DCD, Revision 6.

The fire protection system (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, FPSs and features, and safe-shutdown capability. The ESBWR FPS does not perform any safety-related function; however, because of non-safety-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 backup source of makeup water through a piping connection to the Fuel and Auxiliary Pools Cleaning System (FAPCS) for the Isolation Condenser/Passive Containment Cooling System (IC/PCCS) pools and the spent fuel pool and for reactor water inventory control following a design-basis accident. If necessary, the makeup function will begin no later than 72 hours after a loss-of-coolant accident (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 since 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 National Fire Protection Association (NFPA) 804, "Standard for Fire Protection for Advanced Light Water Reactor Electric Generating Plants" (2006 edition) and the International Building Code.

DCD Tier 2, Revision 7, Section 9.5.1 includes a description of FPP compliance with the International Building Code (IBC). Since the staff's evaluation of the ESBWR FPP is applicable to U.S. nuclear power plants, a review of IBC compliance is not required and was not performed.

DCD Tier 2, Revision 6, Section 9.5.1.16 and Section 9A.7 list the fire protection COL action items.

9.5.1.3 Staff Evaluation

The staff reviewed the ESBWR fire protection program in accordance with SRP Section 9.5.1 and RG 1.189.

Fire Hazards Analysis

DCD Tier 2, Appendix 9A describes the ESBWR fire hazards analysis.

The ESBWR FHA establishes and evaluates distinct fire areas for the RB, FB, control building, turbine building, radwaste building, electrical building, yard, pump house, guard house, hot machine shop, service water/water treatment building, cold machine shop, warehouse, training center, service building, auxiliary boiler building, and administration building. 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 is to provide a high degree of fire protection by implementing three concepts. These concepts are (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, "Standards for Protection Against Radiation," 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 control of combustible transient materials 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 MJ/m^2 (123,600 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 having a maximum calorific value of 29.8 MJ per kilogram (kg) (12,834 Btu/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 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, radwaste building, electrical building, yard, and turbine building do not contain any safe-shutdown components, and as such, 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. Both the turbine building and the electrical building have 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 loss of offsite power 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 fire hazards analysis 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 that safe shutdown in conformance with RG 1.189. GDC 3 requires that the fire protection program provide protection for structures, systems and components important to safety. In its responses, the applicant clarified that the structures systems and components 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 non-safety equipment necessary to achieve and maintain stable shutdown. The applicant made corresponding changes to DCD Revision 6. The staff determined that the response was acceptable since the applicant implemented the guidelines for important to safety in RG 1.189 to meet GDC 3 for the fire hazards analysis 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. Table 9A.5-6 states that both fire areas contain safety-related divisional equipment or cables for all four divisions, while the safe shutdown evaluations state that a fire in the area affects no safety-related equipment. In its response, the applicant agreed to change the wording in the safe shutdown evaluation in 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 determined that the response was acceptable since the applicant addressed the contradiction in 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 fire hazards analysis conforms to the guidelines of SRP Section 9.5.1 and RG 1.189.

Passive Fire Protection, Detection and Suppression Features

DCD Tier 2, Section 9.5.1 describes the materials of construction as it relates to the fire protection program. 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 American Society for Testing and Materials (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 finds the materials of construction are consistent with the guidelines of SRP Section 9.5.1 and RG 1.189.

DCD Tier 2, Section 9.5.1.10 describes the ESBWR fire protection program fire barriers.

Fire barriers of 3-hour fire resistance rating separate the following:

- safety-related systems from any potential fires in non-safety-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 non-safety-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 7, 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 per 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 doors that provide fire area separation that may not be fire doors that have been qualified 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,” where 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,” including Supplement 1. The documented test results for the acceptable fire-stops will be part of the plant design records.

The combined license (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 E-119; and the guidance in RG 1.189. DCD Tier 2, Revision 7, Sections 9.5.1.11 and 9.5.1.16 identify this as a COL Item.

For the reason set forth above, the staff finds that the ESBWR fire barriers are consistent with the guidelines of SRP Section 9.5.1 and RG 1.189.

DCD Tier 2, 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 megajoules per square meter (MJ/m^2) (123,600 British thermal unit per square foot (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 (MFAP) in the MCR.

DCD Tier 2, Revision 5, 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 fire protection program provide protection for structures, systems and components 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 their basis from safe shutdown equipment to equipment important to safety. In its response, the applicant clarified that the structures systems and components 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 non-safety equipment necessary to achieve and maintain stable shutdown. The applicant made corresponding changes to DCD revision 6. The staff determined that the response was 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 are consistent with the guidelines of SRP Section 9.5.1 and RG 1.189.

DCD Tier 2, Section 9.5.1.9 describes the ESBWR detection and alarm systems which includes standpipes and hose stations.

Instrumentation for the fire detection system provides signals for early detection and warning of fires. Local fire alarm panels per NFPA 72, "National Fire Alarm Code," supervise fire and smoke detectors. The local fire alarm panels are in turn connected to the MFAP via a dedicated data link. Signals transmitted include detector status (normal, alarm, supervisory, and trouble) as well as local fire alarm panel status. On receipt of a signal from any of the area fire detectors, alarms and visual indications are activated at the MFAP 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. A minimum of two detectors is 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 that safe shutdown in conformance with RG 1.189. GDC 3 requires that the fire protection program provide protection for structures, systems and components important to safety. In its response, the applicant clarified that the structures systems and components 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 non-safety equipment necessary to achieve and maintain stable shutdown. The applicant made corresponding changes to DCD revision 6. The staff determined that the response was 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 its response, the applicant clarified that manual fire alarm pull boxes are installed at each building exit of the Ancillary Diesel Building. The applicant revised Section 9A4.10 of the DCD stating that manual fire alarm pull boxes will be located at each building exit. The staff determined that the response was 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, 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 shall 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 use only. The COL applicant will provide information on the final quantity and capacity of secondary fire water storage. DCD Tier 2, Revision 7, Sections 9.5.1.4 and 9.5.1.16, identify this as a COL Item.

The primary seismic Category I fire water storage tanks provide the required emergency makeup water volume for the IC/PCCS pools and SFP to the FAPCS following the design-basis LOCA. The primary source of fire water has a minimum capacity of 3900 m^3 (1,030,000 gal). The secondary source has a minimum capacity of 2081.8 m^3 (550,000 gal) 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 loss of preferred power. 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 (2000 gal/min) and provides 100 percent of the fire water demand to the worst-case fire within the nuclear island (RB, FB, and control building) 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 (4256 gal/min) for a design-basis turbine building 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 American Society of Mechanical Engineers (ASME), Boiler and Pressure Vessel Code Section B31.1, “Power Piping.” The seismic Category I loop is designed to remain functional following a safe-shutdown earthquake (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 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 3, Section 9.5.1.5 identifies this as a COL Item. 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-meter (m) (250-foot (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, control building, and FB is designed and installed to withstand an SSE and remain operational. Fire suppression system piping in the turbine building, radwaste building, and electrical building 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 7, Sections 9.5.1.5 and 9.5.1.16, identify this as a COL Item.

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, 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-millimeter (mm) (1.5-inch (in.)) lined fire hose. The water supply pressure maintains a gauge pressure of 448.2 kilopascals gauge (kPaG) (65 pounds-force per square inch gauge (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 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 control building hose stations by opening the hose valve at each station. In its response, the applicant revised both figures to clarify that the closed valves represents a typical hose station valve by adding a note to the figures indicating that these valves represent a typical hose station valve. The staff determined that the response was acceptable since the applicant clarified the hose stations in the control 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 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, 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 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 Institute of Electrical and Electronics Engineers IEEE-1202 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 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 main steam isolation valve sensors and cabling in the turbine building 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 a main steam isolation valve closure on elevated temperature due to a fire in the area.

- Main steam line automatic depressurization valve 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; for heating, ventilation, and air conditioning (HVAC); 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 standby liquid control system 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 than 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 non-safety-related systems from either of the two remote shutdown system panels located in separate fire areas.

- Complete burnout of all safety-related devices and their cables in the turbine building 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 hydraulic control unit (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 fine motion control rod drive portion of the control rod drive system or the standby liquid control system 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 it supplements, the staff requested that the applicant describe how the ESBWR design specifically prevents or mitigates spurious actuations that could prevent safe shutdown due to the effects of fire, including smoke, and these design features included in the DCD. In it responses, the applicant provided additional description and clarification on the design features that prevent or mitigate spurious actuations in DCD Tier 2, Revision 6, Sections 7.1.3.2, 7.1.5.3 and 9.5.1.10. The staff determined that the response was acceptable since the applicant described specific features that prevent spurious actuations in its discussion of fire barriers and 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 the 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 one described in the industry guidance document for circuit analysis, NEI 00-01, "Guidance for Post-Fire Safe Shutdown Circuit Analysis". In its 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 in accordance with RIS 2005-30, "Clarification of Post-Fire Safe-Shutdown Circuit Regulatory Requirements." The staff determined that the response was acceptable since the applicant clarified the fire hazards analysis 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, 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 (RSP), manual action may be necessary to control the ICS to ensure the maximum cooldown rate does not exceed 55.6ºC/hour (100 ºF/hour). Because the controls at the RSP are identical to those in the MCR, the operator can fully control the ICS from the RSP 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 the ESBWR design provides adequate protection of safe shutdown capability in the event of a fire.

<u>Miscellaneous</u>

DCD Tier 2, Section 9.5.1 describes the additional design features that support the ESBWR fire protection program.

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.

DC 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 gal) and all tanks containing chemicals used in water/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 500 gal/min (1892.5 L/min) 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 communications systems conform to the guidelines of RG 1.189 Position 4.1.7. In its 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 determined that the response was acceptable since the fire brigade communication systems are site specific and the applicant committed to conform to the guidelines of RG 1.189, 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 the ESBWR these 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 fire protection program with the guidelines of Commission papers SECY 90-016, SECY 93-087 and SECY 94-084 which provide enhanced fire protection criteria for advanced reactor designs as follows:

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 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. A remote shutdown system, 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 isolation condenser system (ICS). This is a system employed for both hot standby and long-term core cooling modes and that can operate at full reactor coolant system 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 alternating current (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
- gravity-driven cooling system
- automatic depressurization system
- passive containment cooling system
- 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 the ESBWR program meets the enhanced fire protection criteria for advanced reactor designs.

<u>Exceptions to the Standard Review Plan and RG 1.189</u>

The ESBWR design includes specific exceptions to the overall FPP design bases for the ESBWR, as well as specific exceptions and alternatives to the NRC acceptance criteria for FPPs. These exceptions and alternatives are described in DCD Tier 2, Revision 6, Section 9.5.1.12 "Safety Evaluation," and Section 9A.6 of Appendix 9A. These sections describe and justify, in detail, each of the plant configurations and designs that deviate from the ESBWR FPP design bases or deviate from the NRC acceptance criteria for FPPs. As described below, the staff has reviewed each of these exceptions and alternative approaches and their justifications in Sections 9.5.1.12 and 9A.6 and finds them acceptable.

- Individual electrical cabinets and consoles in the main control room (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 it is identified as a significant fire hazard in the fire hazards analysis (FHA).

- Rooms adjacent to the MCR will not have installed automatic fire suppression systems as recommended in 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 it is identified as a significant fire hazard in the FHA.

- 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 Main Control Room Complex and subfloor volume is considered to be a low risk fire area, due to 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 it is identified as a significant fire hazard in the FHA.

- The standby diesel generator (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 ancillary diesel generator (ADG) fuel oil tank capacity will exceed the limit recommended by Section 6.1.8 of RG 1.189 for indoor DG day tanks. The capacity of each of the ADG day tanks will not exceed 4164 L (1100 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 ADB 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. Since the DGs are not nonsafety-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 due to 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 meters (200 feet) 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 Items for (1) a post-fire safe shutdown circuit analysis, (2) the fire hazards analysis 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. RAI 9.5-44, 9.5-45, and 9.5-46 were being tracked as open items in the SER with open items. In its responses, the applicant stated that fire hazards analysis 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 a COL Item (COL 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 determined that the response was acceptable, as augmented by the revised COL Item 9.5.1-7-A, since the COL Item addresses the fire hazards analysis in a comprehensive way such that individual elements do not need to be identified. The COL 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, RAI 9.5-44, 9.5-45 and 9.5-46 are resolved.

Inspections, Tests, Analyses, and Acceptance Criteria

The DCD Tier 1, Revision 7, 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 per ASTM E-119 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 consistent safety rather than safe shutdown in conformance with RG 1.189. GDC 3 requires that the fire protection program provide protection for structures, systems and components important to safety. In its response, the applicant clarified that the structures systems and components 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 non-safety equipment necessary to achieve and maintain stable shutdown. The applicant made corresponding changes to DCD Revision 6, Tier 1 Section 2.16.3. The staff determined that the response was acceptable since 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 Section 2.16.3 to finds that it conforms to the FPS and fire barriers design basis as described in DCD Tier 2, Section 9.5.1. Accordingly, the staff find that the FPS ITAAC complies with the requirements of 10 CFR 52.47(b)(1).

COL Items

DCD Tier 2, Revision 6, Section 9.5.1.16, "COL Information", and Section 9A.7, "COL Information" of Appendix 9A, list the following fire protection COL items

- 9.5.1-1-A Secondary Firewater Storage Source - The COL Applicant will provide the capacity of the secondary firewater source (DCD Subsection 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 (2130 gpm) with sufficient discharge pressure to develop a minimum of 107 psig line pressure at the Turbine Building / yard interface boundary (DCD Subsection 9.5.1.4).

- 9.5.1-4-A Piping and Instrument Diagrams - The COL Applicant shall provide simplified FPS piping and instrumentation diagrams showing complete site-specific systems (DCD Subsection 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 applicable section of NFPA 251 "Standard Methods of Tests of Fire Resistance of Building Construction and Materials", ASTM E-119 "Standard Test Methods for Fire

Tests of Building Construction and Materials" and guidance in RG 1.189 (DCD Subsection 9.5.1.10).

- 9.5.1-6-A Smoke Control - The COL Applicant shall establish provisions for manual smoke control by manual actions of the fire brigade for all plant areas in accordance with NFPA 804 guidelines (DCD Subsection 9.5.1.11).

- 9.5.1-7-A FHA Compliance Review - The COL Applicant shall conduct a compliance review of the final as-built design against the assumptions and requirements stated in the FHA. Based on this review, the FHA will be updated as necessary (DCD Subsection 9.5.1.12).

- 9.5.1-8-A FP Program Description - The COL Applicant shall provide a milestone for implementation of the applicant's FPP (DCD Subsection 9.5.1.15).

- 9.5.1-10-A Fire Brigade - The COL Applicant shall provide provisions for manual fire-fighting capability for all plant areas (DCD Subsection 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 Subsection 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 Turbine and Electrical Building equipment (DCD Subsection 9A.4.7).

- 9A.7-2-A FHA 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 Subsection 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, "Combined License Applications for Nuclear Power Plants (LWR Edition)," 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 items acceptable.

9.5.1.4 Conclusion

The staff concludes that the applicant's FPP design criteria are acceptable and meet the applicable requirements of 10 CFR Part 50 and 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 has met the guidelines of the applicable regulatory guides 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 concludes 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 NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants" (hereafter referred to as the SRP) Section 9.5.2 Revision 3, "Communications Systems," issued March 2007. Acceptance criteria are based on meeting the relevant requirements of the following Commission regulations:

- Appendix E to 10 CFR Part 50, "Emergency Facilities and Equipment," 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), "Provide an onsite Technical Support Center." (Three Mile Island (TMI) Action Plan Item III A.1.2).
- 10 CFR 50.47(a)(8), "Equipment and Facilities to Support Emergency Response."
- 10 CFR 50.55a, "Codes and Standards."
- General Design Criterion (GDC) 1, "Quality Standards and Records."
- GDC 2, "Design Basis for Protection Against Natural Phenomena."
- GDC 3, "Fire Protection."
- GDC 4, "Environmental and Missile Design Bases."
- GDC 19, "Control Room."
- 10 CFR 73.45(e)(2)(iii), "Communications subsystems and procedures to provide for notification to authority."

- 10 CFR 73.45(g)(4)(i), "Provide Communications Networks."

- 10 CFR 73.46(f), "Fixed Site Physical Protection Systems, Subsystems, Components, and Procedures - Communications Subsystems."

- 10 CFR 73.55(i), "Detection and assessment systems."

- 10 CFR 73.55(j), "Communications requirements."

- 10 CFR 52.47(b)(1), which requires that a design certification (DC) application contain the proposed inspections, tests, analyses, and acceptance criteria (ITAAC) that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the DC is built and will operate in accordance with the DC, the provisions of the Atomic Energy Act, and the Nuclear Regulatory Commission's (NRC) regulations.

9.5.2.2 Summary of Technical Information

DCD Tier 2, 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/onsite agencies as required by 10 CFR Part 73.55(j). 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." The communications systems consist of the following systems:

- Plant page/party-line (PA/PL) system;
- Private automatic branch exchange system;
- Plant sound-powered telephone system;
- Plant radio system;
- Evacuation alarm and remote warning system;
- Emergency offsite communications system; and
- Completely independent communications (radio) system for security purposes.

The communications systems above are described in detail in DCD FSAR Tier 2 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); and

- The communications systems are functional during a loss of offsite power.

PA/PL 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.

PABX 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 non-safety 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 non-safety 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 9.5.2.5-1-A); 2) A Health Physics Network; which provides a communications link with NRC health physics personnel (COL 9.5.2.5-3-A); 3) A Ringdown Phone System; which provides a communications link with local and state agencies (COL 9.5.2.5-4-A); 4) A Crisis Management Radio System that provides communications capability in accordance with the NUREG-0654 (COL 9.5.2.5-3-A); 5) A Fire Brigade Radio System in accordance with RG 1.189, Position 4.1.7 (COL 9.5.2.5-5-A); and 6) A Transmission System Operator Communication Link (COL 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, March 2007). DCD Tier 2, Table 1.9-9, "Summary of Difference from SER Section 9," states that there are no differences from 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 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, Section 9.5.2.2 identifies 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, Section 9.5.2.2 identifies that the PABX and plant radio systems provide offsite communications. Diverse non-safety related power supplies connected to the plant standby generators power the PA/PL telephone, PABX and plant radio systems.

DCD Tier 2, Section 9.5.2.2 identifies six emergency communications systems covered by the five COL Items: (1) Emergency Notification System (COL 9.5.2.5-1-A); (2) Health Physics Network (COL 9.5.2.5-3-A); (3) Ringdown Phone System (COL 9.5.2.5-4-A); (4) Crisis Management Radio System (COL 9.5.2.5-3-A); (5) Fire Brigade Radio System (COL 9.5.2.5-5-A); and (6) Transmission System Operator Communication Link (COL 9.5.2.5-2-A). The staff finds that the COL Items are for portions of the communications systems that are site-specific and are therefore acceptable.

The communications system is classified as non-safety 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 main control room (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 technical support center (TSC) for the facility. SRP 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 Section 13.3, "Emergency Planning," 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 10 CFR 50.34(f)(2)(xxv) requirements in regard to communications systems.

10 CFR 50.47(a)(8) requires adequate equipment and facilities to support emergency response. SRP 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 Section 9.2.2.2 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, 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 Chapter 17 describes the applicant's quality assurance program for equipment maintenance and is evaluated in Chapter 17 of this report. Emergency Planning response is evaluated in Chapter 13 of this report. Therefore, based on the above the staff finds that the design meets the requirements of 10 CFR 50.47(a)(8) in regard to communications systems.

10 CFR 50.55a requires an applicant to address codes and standards. In DCD Tier 2, Table 3.2-1, "Classification Summary," the communications systems are classified as non-safety related systems. In DCD FSAR Tier 2, Table 1.9-9, "Summary of Differences from SRP Section 9," the applicant indicates no departures from the guigance of SRP 9.5.2. DCD Tier 2 Table 1.9-20 lists SRP and BTP applicable to the ESBWR and included SRP 9.5.2. Based on the communications descriptions and the information above, the staff finds that classification is acceptable for a non-safety related system and that the design has adequately addressed the requirements of 10 CFR 50.55a in regard to communications systems.

GDC 1, "Quality Standards and Records," requires that structures, systems, and components (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, Table 3.2-1, "Classification Summary," the communications systems are classified as a non-safety related, non-seismic systems where system components are mounted to

Seismic Category II requirements in safety-related areas. DCD Tier 2, Table 1.9-20 lists SRP and BTP applicable to the ESBWR and included SRP 9.5.2 in effect at the time of filing of the DCD application. Non-safety related items are controlled by the quality assurance program described in 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, Chapters 3 and 17, the staff finds that the communications systems design satisfies the GDC 1 requirements.

GDC 2, "Design Bases for Protection Against Natural Phenomena," 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. In DCD Tier 2, Table 3.2-1, "Classification Summary," the communications systems are classified as non-safety related systems. DCD Tier 2, Table 3.2-1, states that the communications systems components are mounted in accordance with Seismic Category II requirements in safety-related areas. The evaluation of protection for natural phenomena such as earthquakes, tornadoes, hurricanes, and floods is evaluated in Chapter 3 of this report. However, DCD Tier 2 Section 9.5.2.2 states that the PA/PL, PABX, and plant radio systems are physically independent systems powered from diverse non-safety 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 standby diesel-generator-backed power supplies. Accordingly, based on these design features and given the importance of the functions of these systems (see discussion of GDC 19 below), the staff finds that the communications systems design have sufficient diversity and independence that combined with the protection discussed in Chapter 3 of this report, the requirements of GDC 2 have been adequately addressed.

GDC 3, "Fire Protection," 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. The fire protection features are evaluated in Section 9.5.1 of this report. In DCD Tier 2 Table 3.2-1, "Classification Summary," the communications systems are classified as non-safety related systems. However, two-way voice communications are used to support safe shutdown and emergency response in the event of fire. DCD Tier 2, Section 9.5.2.2 states the Plant Radio system complies with RG 1.189, 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 FSAR Tier 2 Section 9.5.2.2 states that three systems (PA/PL, PABX, and plant radio systems) are physically independent systems powered from diverse non-safety 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, the staff finds that the communications systems design has sufficient diversity and independence that, combined with the protection discussed in Section 9.5.1 of this report, the requirements of GDC 3 have been adequately addressed.

GDC 4, "Environmental and Dynamic Effects Design Bases," 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 loss of coolant accidents (LOCAs). In DCD Tier 2 Table 3.2-1, "Classification Summary," the communications systems are classified as non-safety related systems. The evaluation of protection for pipe rupture and flooding, EMI and RFI, and EQ is in Chapter 3 of this report. In the plant radio system, lower power portable radios are used to ensure there is no EMI with control and instrument circuits and vice versa. DCD Tier 2, Section 9.5.2.2 states that the PA/PL, PABX, and plant radio systems are physically independent systems powered from diverse non-safety 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. 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, "Control Room," requires that a 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 non-safety 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 FSAR Tier 2 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. On 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, 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, 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.

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 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, 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, Section 13.6 and evaluated in Section 13.6 of this report 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 9.5.2 states that Information regarding the requirements of 10 CFR 73.46(f) will be found acceptable if 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; 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 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 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 Section 9.5.2.2, the PA/PL, PABX, and plant radio system are physically independent systems powered from diverse non-safety 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 capability to support the communications required by 10 CFR 73.46(f).

10 CFR 73.55(e) ”Physical Barriers,” and 10 CFR 73.55(i), “Detection and assessment systems,” 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, Section 13.6 and evaluated in Section 13.6 of this report.

10 CFR 73.55(j), “Communication requirements,” 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, 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 and, 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 non-safety 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, 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) have been adequately addressed in regard to the communications systems design described in DCD Tier 2 Section 9.5.2.

10 CFR 52.47(b)(1) requires that a DC application contain the proposed ITAAC that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the DC is built and will operate in accordance with the DC, the provisions of the Atomic Energy Act, and the NRC's regulations. DCD Tier 1 Section 2.13.7, “Communication System,” state that no ITAAC are required for this system. The staff finds that this is acceptable since the communications systems are non-safety related and do not have any RTNSS functions. 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 concludes that the communications systems design is acceptable and meets the requirements of 10 CFR Part 50, Appendix E, 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 General Design Criteria (GDC) or Regulatory Guides (RGs) directly apply to the functions of the lighting system. However, the plant lighting system is necessary to support accident mitigation (e.g., Fire Protection Program) 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 alternating current (ac) power (i.e., during a station blackout) 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, “Guidelines for Control Room Design Review,” 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 seismic loading of a safe shutdown earthquake. 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 direct current (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 non-safety-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 volts alternating current (Vac), three-phase, four-wire, and grounded neutral system distribution panels supplied from normal 480 Vac motor control centers. The incandescent lighting fixtures on refueling platforms are powered from 480/277 Vac, 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 Vac, three-phase, four-wire, and grounded, or 480-240/120 Vac, 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 Vac, three-phase, four-wire, and grounded neutral system distribution panels normally supplied by the plant investment protection non-safety-related buses. The primary areas served by this system are as follows:

- main control room (MCR)
- remote shutdown rooms
- operational support centers

- technical support centers
- auxiliary switchgear rooms
- safety-related dc equipment rooms
- stairwells and aisle way
- distributed control and information system (DCIS) equipment rooms
- diesel generator (DG) rooms
- DG 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 plant investment protection non-safety-related buses.

<u>Emergency Lighting</u>

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, on loss of the normal lighting system. The emergency lighting system is comprised of the following:

- MCR and remote shutdown area emergency lighting, and
- Non-safety-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 uninterruptible ac power supply (UPS) system. Electrical isolation of non-safety-related emergency lighting circuits from safety-related UPS power supply is accomplished by the use of series isolation devices that are designed to coordinate with upstream 120 Vac 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 non-safety-related. Cables used for emergency lighting in the MCR and the remote shutdown area are non-safety-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 non-safety-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/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 fire and 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 Vac or 120 Vac from 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 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 non-safety-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, Subsection 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 will revise the first bulleted item under DCD Tier 2, Revision 3, Subsection 9.5.3.3.3.2, to delete the use of 2-hour-rated units. The staff determined that the response was acceptable since 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 and, hence, this confirmatory item is closed.

DCD Tier 2, Revision 3, Subsection 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 its 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 determined that the response was acceptable since 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 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 non-safety-related and provide illumination for safe ingress/egress of personnel and shutdown activities and are powered from diesel-backed busses upon loss of normal ac power. The staff determined that the response was not acceptable and 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 main control room. Also, the staff asked the applicant to provide a discussion about the emergency lighting in remote shutdown area in DCD Tier 2, Subsection 9.5.3.3.3. In its response, 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 Subsection 9.5.3.3.3.1 which states that the control room and remote shutdown area emergency lighting is supplied from safety-related UPS as shown in DCD Chapter 8, Figure 8.1-4, sheet 1 of 1. Figure 8.1-4, sheet 1 of 1 indicates that MCR emergency lighting is supplied from four divisions of safety-related UPS, while the remote shutdown area emergency lighting is supplied from Divisions 1 and 2 UPS.

In RAI 9.5-60 S03, the staff asked the applicant to provide an explanation why the emergency lighting from Divisions 1 and 2 is acceptable in the remote shutdown area. In its 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 plant investment protection (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 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, so will the PIP A and PIP B VDUs, 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 as it is provided by the eight hour battery powered lights and non-safety-related power from the PIP buses. The staff determined that the RAI responses were acceptable since the applicant clarified the emergency lighting in the remote shutdown areas and the basis for it power supplies. The staff confirmed that Subsections 9.5.3.3.3, 9.5.3.3.3.1, and 9.5.3.3.3.2 of DCD, Tier 2, Revision 5, have been revised 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 Class 1E power supply and non-Class 1E circuits. In response the applicant stated that Class 1E power supply and non-Class 1E circuits are isolated through 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, Subsection 9.5.3.3.3.1, is to be revised in its entirety for clarity and to add isolation provisions ("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, etc.) 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 its response, the applicant clarified that MCR emergency lighting fixtures are non-safety-related; hence, separation devices are necessary. In DCD Revision 5, the applicant further clarified that the lighting fixtures, circuits, and associated cables are non-safety related. The staff confirmed that DCD Tier 2, Revision 5, Subsections 9.5.3.1, 9.5.3.3.3.1 and 9.5.3.4 have been revised accordingly. The staff determined that the RAI response, with the additional DCD changes, was 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, Subsection 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, are 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 determined that the response was 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, Subsection 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, Subsection 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, etc.) 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 its response, the applicant clarified that MCR emergency lighting fixtures are non-safety-related; hence, separation devices are necessary. In DCD Revision 5, the applicant further clarified that the lighting fixtures, circuits, and associated cables are non-safety related. The staff confirmed that DCD Tier 2, Revision 4, Subsections 9.5.3.1, 9.5.3.3.3.1 and 9.5.3.4 have been revised accordingly. The staff determined that the RAI response, with the additional DCD changes, was 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 non-safety related. Based on the above, the applicant's responses, and DCD changes, RAI 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 boiling-water reactor 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 non-safety-related power source and are mounted inside the console. The supports for the lighting fixtures are seismic Category I. The applicant further stated that a new Subsection 9.5.3.3.3.3, "Panel Lighting," will be added to the 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 non-safety-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 determined that the RAI response was acceptable since the applicant added a design description of panel lighting in the MCR. Based on the above and the applicant's response, RAI9.5-64 is resolved. RAI 9.5-64 was being tracked as a confirmatory item. The staff confirmed that the applicant added a new Subsection 9.5.3.3.3.3, "Panel Lighting" in DCD Tier 2, Revision 4 and, hence, 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 in all vital areas from onsite power sources during the full spectrum of accident and/or 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 concludes 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 Economic Simplified Boiling Water Reactor (ESBWR) standby diesel generator (SDG) and ancillary diesel generator (ADG) fuel oil storage and transfer systems (FOSTS) in accordance with NUREG-0800, “Standard Review Plan (SRP) for the Review of Safety Analysis Reports for Nuclear Power Plants,” Section 9.5.4, “Emergency Diesel Engine Fuel Oil Storage and Transfer System,” Revision 3, dated March 2007. The staff reviewed ESBWR Design Certification Document (DCD) Tier 2, Revision 6, Section 9.5.4, “Diesel Generator Fuel Oil Storage and Transfer System.” The staff also reviewed DCD Tier 1, Revision 6, Section 2.0, “Design Descriptions and Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC),” and other DCD Tier 2 sections noted below. The staff’s acceptance of the FOSTS is based on the design’s conformance with the requirements of the following General Design Criteria (GDC) and Code of Federal Regulation (10 CFR 52.47(b)(1)):

- GDC 2, “Design Bases for Protection against Natural Phenomena,” requires in part that structures, systems, and components (SSC) 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 Positions C.1 for the safety-related portions and C.2 for the non-safety-related portions of Regulatory Guide (RG) 1.29, “Seismic Design Classification.”

- GDC 4, “Environmental and Dynamic Effects Design Bases,” 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, “Sharing of Structures, Systems, and Components,” 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, “Electric Power Systems,” 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) which requires that a DC application contains the proposed inspections, tests, analyses, and acceptance criteria (ITAAC) that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that 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 non-safety-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, 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 ancillary diesel generators (ADGs) to provide 480V 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 and completed 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 non-safety-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 FOSTS have the RTNSS functions, as supporting systems to provide power and are included in the plant Availability Controls Manual (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 for a minimum of 8 hours of SDG operation at full load and sufficient fuel oil onsite for its associated SDG operation at full load for 7 days without replenishing. In addition, the FOSTS has piping connections to supply fuel oil to the auxiliary boiler system, the diesel-engine driven fire protection system 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 SDG will not be affected by auxiliary boiler system usage, fire protection system 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 Combined License (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/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 at full load 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. 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 as described in the above for SDG FOSTS.

The SDG and ADG FOSTS permit periodic testing and inspection in accordance with the ACM. FOST 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 American Society for Testing Materials (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 American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section VIII, Division 1, and ASME B31.1, "Power Piping." 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 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 and ADGs and their supporting systems including the FOSTS have RTNSS functions, as supporting systems to provide AC source of power 72 hours after an abnormal event. Based on its review as discussed in Section 3.4.1, "Flood Protection," Section 3.5.1.1, "Internally Generated Missiles (Outside Containment)," Section 3.5.1.4, "Missiles Generated by Natural Phenomena," and Section 3.5.2, "Structures, Systems, and Components to be protected from externally Generated Missiles," of this report, the staff finds that the SDG FOSTS and ADG FOSTS meet the relevant requirements of GDC 2 as it pertains to Position C.2 of RG 1.29. The FOSTS also meet the requirements of GDC 2 as it pertains to 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, "Plant Design for Protection against Postulated Piping Failures in Fluid Systems Outside of Containment," the staff finds that the SDGs, the ADSs and their support systems including the FOSTS are protected against the effects of, and be 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, "Offsite Power System," and 8.3, "Onsite Power System," 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 the ESBWR DCD Revision 0, the staff found that during a postulated post-loss of coolant accident (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, "Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety Systems (RTNSS) in Passive Plant Designs," to apply to the passive advanced light water reactor (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 its response to RAI 19.1.0-2, the applicant included the SDG units as RTNSS systems. However, it was not clear to the staff that all 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 its response to RAI 22.5-4, the applicant stated that all SDG supporting systems including SDG FOSTS were considered as RTNSS systems. In Section 9.5.4 of the DCD Tier 2, Revision 4, SDG FOSTS was included and classified as an RTNSS system. The staff determined that the responses to RAIs 19.1.0-2 and 22.5-4 regarding the SDG FOSTS were 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 DCD changes were incorporated 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 they are considered as RTNSS systems. In RAI 14.3-151, the staff requested the applicant to include ITAAC for all the SDG supporting systems. In its response to RAI 14.3-151, the applicant stated that it would not include ITAAC for SDG supporting systems since they were non-safety 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, "ITAAC for The Standby OnSite Power Supply," the applicant included ITAAC for only two SDG supporting systems, the SDGFOSTS and the SDG starting air system. The staff further issued another RAI, 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 revise ESBWR DCD Revision 5 to include an ITAAC entry for each of the SDG supporting systems. The staff has determined 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, RAI 14.3-151 and 14.3-177 regarding the SDG FOSTS are resolved. The staff confirmed that the DCD changes were incorporated into DCD Revision 5.

In DCD Tier 1 Revision 6, Section 2.13.4, "Standby Onsite AC Power Supply," and Table 2.13.4-2, "ITAAC for the Onside AC Power Supply," the applicant provides the design descriptions and ITAACs for the SDGs, ADGs, and their supporting systems including the FOSTSs. The staff finds that these ITAACs, which commit to verify that the SDGs, ADGs, and their supporting systems, including the FOSTSs, are constructed and installed as described in DCD Tier 2, Revision 6. Therefore, the staff concludes 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 standby and ancillary diesel generators is addressed by 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 13.5-2-A, "Plant Operating Procedures Development Plan." In Section 13.5.2, "Operating and Maintenance procedures," of the ESBWR DCD Tier 2, Revision 6, 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.

Corrosion protection of the underground portion of the storage tank and piping for the SDGs and ADGs is addressed by COL Item 9.5.4-2-A. As stated in DCD Tier 2, Revision 6, 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. The COL 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 state, federal, and local laws for underground petroleum storage tanks, which include corrosion protection. The staff finds the provisions for corrosion protection acceptable because they ensure corrosion protection will be included in site-specific designs and submitted in COL applications to the NRC for review.

DCD Tier 2, Revision 6, Subsections 14.2.8.1.37, “Standby Diesel Generator & AC Power System Preoperational Test,” provides the initial testing provisions associated with this review item. The staff reviewed DCD Tier 2, Revision 6, Section 9.5.4 against the guidance in SRP Section 14.3.7, “Evaluation Process For Updating Design Descriptions and ITAAC,” and finds that no additional ITAAC are needed in connection with this section.

DCD Tier 2, Revision 6, Section 16, “Technical Specifications,” does not have any technical specification requirements associated with this review item. The staff reviewed DCD Tier 2, Revision 6, Section 9.5.4 against 10 CFR 50.36, “Technical Specifications,” and agrees that no technical specifications are needed in connection with this section.

Section 9.5.4.6, “Combined License (COL) Unit-Specific Information,” and Table 1.10-1, ”Summary of COL Items,” of DCD Tier 2, Revision 6 include:

- COL Action Item 9.5.4-1-A, “Fuel Oil Capacity,” specifies that the COL applicant should establish procedural controls to ensure a minimum fuel oil capacity is maintained onsite at all times for both SDGs and ADGs.

  The staff finds this COL Action Item 9.5.4-1-A acceptable because it will ensure a minimum fuel oil capacity maintained onsite at all times for SDGs and ADGs.

- COL Action Item 9.5.4-2-A, “Protection of Underground Piping,” specifies that COL applicants should 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 state, federal, and local laws for underground petroleum storage tanks, which include corrosion protection.

  The staff finds the 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, “Plant Procedures,” 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, “Regulatory Treatment of Non-Safety Systems,” of this report addresses the staff’s evaluation regarding RTNSS systems in conformance with the requirements of the SECY-94-084.

9.5.4.5 Conclusion

The staff concludes that the FOSTSs for SDGs and ADGs meet the guidelines of SRP Section 9.5.4, Revision 3. Based on the above, the staff concludes that the FOSTS for SDGs and ADGs design is acceptable and meets the relevant requirements of GDC 2, 4, and 17 and 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 standby diesel generator (SDG) and ancillary diesel generator (ADG) jacket cooling water systems (JCWS) in accordance with NUREG-0800, “Standard Review Plan (SRP) for the Review of Safety Analysis Reports for Nuclear Power Plants,” Section 9.5.5, “Emergency Diesel Engine Cooling Water System,” Revision 3, dated March 2007. The staff reviewed ESBWR Design Certification Document (DCD) Tier 2, Revision 6, Section 9.5.5, “Diesel Generator Jacket Cooling Water System.” The staff also reviewed DCD Tier 1, Revision 6, Section 2.0, “Design Descriptions and Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC),” and other DCD Tier 2 sections noted below. The staff’s acceptance of the JCWS is based on the design meeting the relevant requirements of the following General Design Criteria (GDC) and Code of Federal Regulation (10 CFR 52.47(b)(1)):

- GDC 2, "Design Bases for Protection against Natural Phenomena," requires in part that structures, systems, and components (SSCs) important to safety shall be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes and floods.

  Compliance with GDC 2 is based on meeting the guidance of Regulatory Positions C.1 for the safety-related portions and C.2 for the non-safety-related portions of RG 1.29, “Seismic Design Classification.”

- GDC 4, "Environmental and Dynamic Effects Design Bases." 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, and testing and postulated accidents.

- GDC 5, "Sharing of Structures, Systems, and Components," 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, "Electric Power Systems," 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, “Cooling Water,” requires in part that a system shall be provided to transfer heat from SSCs important to safety to an ultimate heat sink.

- GDC 45, “Inspection of Cooling Water System,” 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, “Testing of Cooling Water System,” requires in part that the cooling water system shall be designed to permit appropriate pressure and functional testing.

- 10 CFR 52.47(b)(1) which 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 non-safety-related alternating current (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 fuel oil storage and transfer system, jacket cooling water system, 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 ADGs to provide 480V 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 and completed 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 non-safety-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 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 reactor component cooling water system (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 jacket cooling water system 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 in the above, the SDGs, ADGs and their support systems are non-safety-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 ac power 72 hours after an abnormal event. Based on its review as discussed in Section 3.4.1, "Flood Protection," Section 3.5.1.1, "Internally Generated Missiles (Outside Containment)," Section 3.5.1.4, "Missiles Generated by Natural Phenomena," and Section 3.5.2, "Structures, Systems, and Components to be protected from externally Generated Missiles," of this report, the staff finds that the SDG JCWS meets the relevant requirements of GDC 2 as it pertains to Position C.2 of Reg. Guide 1.29. The SDG JCWS also meets the relevant requirements of GDC 2 as it pertains to 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, "Plant Design for Protection against Postulated Piping Failures in Fluid Systems Outside of Containment," the staff finds that the SDGs, ADGs and their support systems including the SDG JCWS are protected against the effects of, and be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents. The staff, therefore, finds that the SDG JCWS meet 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 TCWS, 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, “Offsite Power System,” and 8.3, “Onsite Power System,” 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 auxiliary heat sink.

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 loss of offsite power;

- Capability to isolate components or piping so system function is not compromised; and

- 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 the DCD Revision 0, the staff determined that during a postulated post-loss of coolant accident (LOCA) and a complete loss of AC power supplies, the SDG (acronym used for standby diesel generator in previous DCD versions) 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, “Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety Systems (RTNSS) in Passive Plant Designs,” to apply to the passive advanced light water reactor (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 its response, to RAI 19.1.0-2 the applicant included the SDG units as RTNSS systems. However, it was not clear to the staff that all 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 its response to RAI 22.5-4, the applicant stated that all SDG supporting systems including SDG JCWS were considered as RTNSS systems. In Section 9.5.5 of the DCD Tier 2, Revision 4, SDG JCWS was included and classified as RTNSS system. The staff determined that the responses to RAIs 19.1.0-2 and 22.5-4 regarding the SDG JCWS were 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 DCD changes were incorporated 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 they are considered as RTNSS systems. In RAI 14.3-151, the staff requested the applicant to include ITAAC entries for all the SDG supporting systems. In its response, to RAI 14.3-151 the applicant stated that it would not include ITAAC entries for SDG supporting systems since they were non-safety 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 requested the applicant again 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, "ITAAC for The Standby on Site Power Supply," the applicant included ITAAC for only two SDG supporting systems, the SDG fuel oil storage and transfer systems (FOSTS) and the SDG starting air system (SAS). The staff further issued another RAI, 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 revise DCD in Revision 5 to include an ITAAC entry for each of the SDG supporting systems. The staff determined that the response RAI 14.3-177, 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, RAI 14.3-151 and 14.3-177 regarding the SDG JCWS are resolved. The staff confirmed tha the DCD changes were incorporated into DCD Revision 5.

In DCD Tier 1 Revision 6, Section 2.13.4, "Standby Onsite AC Power Supply," and Table 2.13.4-2, "ITAAC for the Onside AC Power Supply," the applicant provides the design descriptions and ITAACs for the SDGs, ADGs, and their supporting systems including the SDG JCWSs. The staff finds that these ITAACs which commit to verify that the SDG and ADG units and their supporting systems, including the SDG JCWSs, are constructed and installed as described in DCD Tier 2, Revision 6. Therefore, the staff concludes that the SDGs, ADGs, and their supporting systems, including the SDG JCWSs, comply with the requirements of 10 CFR 52.47(b)(1).

DCD Tier 2, Revision 6, Subsections 14.2.8.1.37, "Standby Diesel Generator & AC Power System Preoperational Test," provides the initial testing provisions associated with this review item. The staff reviewed DCD Tier 2, Revision 6, Section 9.5.5 against the guidance in SRP Section 14.3.7, "Evaluation Process For Updating Design Descriptions and ITAAC," and finds that no additional ITAAC are needed in connection with this section.

DCD Tier 2, Revision 6, Section 16, "Technical Specifications," does not have any technical specification requirements associated with this review item. The staff reviewed DCD Tier 2, Revision 6, Section 9.5.5 against 10 CFR 50.36, "Technical Specifications," and agrees that no technical specifications are needed in connection with this section.

Section 9.5.5.6, "Combined License (COL) Unit-Specific Information," and Table 1.10-1, "Summary of COL Items," of the DCD Tier 2, Revision 6 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, “Regulatory Treatment of Non-Safety Systems,“ of this report addresses the staff’s evaluation of RTNSS systems in conformance with the guidance of the SECY-94-084.

9.5.5.4 Conclusion

Based on the above, the staff concludes that the JCWS for SDGs and ADGs 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 Economic Simplified Boiling Water Reactor (ESBWR) standby diesel generator starting air system (SDGSAS) in accordance with NUREG-0800, “Standard Review Plan (SRP) for the Review of Safety Analysis Reports for Nuclear Power Plants,” Section 9.5.6, “Emergency Diesel Engine Starting System,” Revision 3, dated March 2007. The staff reviewed ESBWR Design Certification Document (DCD) Tier 2, Revision 6, Section 9.5.6, “Diesel Generator Starting Air System.” The staff also reviewed DCD Tier 1, Revision 6, Section 2.0, “Design Descriptions and Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC),” and other DCD Tier 2 sections noted below. The staff’s acceptance of the SDGSAS is based on the design’s conformance with the requirements of the following GDC and Code of Federal Regulation (10 CFR 52.47(b)(1)):

- GDC 2, “Design Bases for Protection against Natural Phenomena,” requires in part that structures, systems, and components (SSCs) important to safety shall be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes and floods.

  Compliance with GDC 2 is based on meeting the guidance of Regulatory Positions C.1 for the safety-related portions and C.2 for the non-safety-related portions of RG 1.29, “Seismic Design Classification.”

- GDC 4, “Environmental and Dynamic Effects Design Bases.” 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, and testing and postulated accidents.

- GDC 5, “Sharing of Structures, Systems, and Components,” 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 “Electric Power Systems,” 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) which 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.

9.5.6.2 Summary of Technical Information

There are two redundant onsite Seismic Category II standby diesel generator (SDG) units in the ESBWR design for a single unit plant to provide power to non-safety-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 fuel oil storage and transfer system, jacket cooling water system, 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 ancillary diesel generators (ADGs) to provide 480V 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 and completed 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 non-safety-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 SDGSAS have RTNSS functions, as supporting systems to provide power and are included in the plant Availability Controls Manual (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% capacity air compressors, air receiver, a 100% 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; and
- 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 non-safety-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, "Flood Protection," Section 3.5.1.1, "Internally Generated Missiles (Outside Containment)," Section 3.5.1.4, "Missiles Generated by Natural Phenomena," and Section 3.5.2, "Structures, Systems, and Components to be protected from externally Generated Missiles," of this report, the staff finds that the SDGSAS meets the relevant requirements of GDC 2 as it pertains to Position C.2 of Reg. Guide 1.29. The SDGSAS also meet the relevant requirements of GDC 2 as it pertains to 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 SDGSAS to determine if the design meets the relevant requirements of GDC 4. Based on its review as discussed in Section 3.6.1, "Plant Design for Protection against Postulated Piping Failures in Fluid Systems Outside of Containment," the staff finds that the SDGs, ADGs and their support systems including the SDGSAS are protected against the effects of, and be 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, "Offsite Power System," and 8.3, "Onsite Power System," 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, "Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety Systems (RTNSS) in Passive Plant Designs," to apply to the passive advanced light water reactor (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 its response to RAI 19.1.0-2, the applicant included the SDG units as RTNSS systems. However, it was not clear to the staff that all 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 its response, dated August 2, 2007, to RAI 22.5-4, the applicant stated that all SDG supporting systems including SDGSAS were considered as RTNSS systems. In Section 9.5.6 of the DCD Tier 2, Revision 4, the SDGSAS was included and classified as a RTNSS system. The staff determined that the responses to RAIs 19.1.0-2 and 22.5-4 regarding the SDGSAS were 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 DCD changes were incorporated 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 as RTNSS systems. In RAI 14.3-151, the staff requested request the applicant to include ITAAC for all the SDG supporting systems. In its response to RAI 14.3-151, the applicant stated that it would not include ITAAC for SDG supporting systems since they were non-safety 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, "ITAAC for The Standby On-Site Power Supply," the applicant included ITAAC for only two SDG supporting systems, the SDGFOSTS and the SDG starting air system. The staff further issued another RAI, 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 revise ESBWR DCD Revision 5 to include an ITAAC entry for each of the SDG supporting systems. The staff has determined 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, RAI 14.3-151 and 14.3-177 regarding the SDGSAS are resolved. The staff confirmed that the DCD changes were incorporated into DCD Revision 5.

In DCD Tier 1 Section 2.13.4, "Standby Onsite AC Power Supply," and Table 2.13.4-2, "ITAAC for the Onside AC Power Supply," of the ESBWR DCD Tier 1, Revision 6, the applicant provides the design descriptions and ITAACs for the SDGs, ADGs, and their supporting systems including the SDGSAS. The staff finds that these ITAACs which commit to verify that the SDGs, ADGs, and their supporting systems including the SDGSAS are constructed and installed as described in DCD Tier 2, Revision 6. Therefore, the staff concludes 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 6, Subsections 14.2.8.1.37, “Standby Diesel Generator & AC Power System Preoperational Test,” provides the initial testing provisions associated with this review item. The staff reviewed DCD Tier 2, Revision 6, Section 9.5.6 against the guidance in SRP Section 14.3.7, “Evaluation Process For Updating Design Descriptions and ITAAC,” and finds that no additional ITAAC are needed in connection with this section.

DCD Tier 2, Revision 6, Section 16, “Technical Specifications,” does not have any technical specification requirements associated with this review item. The staff reviewed DCD Tier 2, Revision 6, Section 9.5.6 against 10 CFR 50.36, “Technical Specifications,” and agrees that no technical specifications are needed in connection with this section.

Section 9.5.6.6, “Combined License (COL) Unit-Specific Information,” and Table 1.10-1, ”Summary of COL Items,” of the DCD Tier 2, Revision 6 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, “Regulatory Treatment of Non-Safety Systems,“ of this report addresses the staff's evaluation of RTNSS systems in conformance with the guidance of the SECY-94-084.

9.5.6.4 Conclusion

Based on the above, the staff concludes that the SDGSAS design is acceptable and meets the relevant requirements of GDC 2, 4, and 10 CFR 52.47(b)(1).

9.5.7 **Diesel Generator Lubrication System**

9.5.7.1 Regulatory Criteria

The staff reviewed the Economic Simplified Boiling Water Reactor (ESBWR) standby diesel generator lubrication system (SDGLS) in accordance with NUREG-0800, “Standard Review Plan (SRP) for the Review of Safety Analysis Reports for Nuclear Power Plants,” Section 9.5.7, “Emergency Diesel Engine Lubrication System,” Revision 3, dated March 2007. The staff reviewed ESBWR Design Certification Document (DCD) Tier 2, Revision 6, Section 9.5.7, “Diesel Generator Lubrication System.” The staff also reviewed DCD Tier 1, Revision 6, Section 2.0, “Design Descriptions and Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC),” and other DCD Tier 2 sections noted below. The staff's acceptance of the SDGLS is based on the design's conformance with the requirements of the following General Design Criteria (GDC) and Code of Federal Regulations (10 CFR 52.47(b)(1)):

- GDC 2, “Design Bases for Protection against Natural Phenomena,” requires in part that structures, systems, and components (SSCs) important to safety shall be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes and floods.

  Compliance with GDC 2 is based on meeting the guidance of Regulatory Positions C.1 for the safety-related portions and C.2 for the non-safety-related portions of Regulatory Guide (RG) 1.29, “Seismic Design Classification.”

- GDC 4, “Environmental and Dynamic Effects Design Bases.” 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, and testing and postulated accidents.

- GDC 5, “Sharing of Structures, Systems, and Components,” 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 “Electric Power Systems,” 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) which 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.

9.5.7.2 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 non-safety-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 fuel oil storage and transfer system, jacket cooling water system, 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 ancillary diesel generators (ADGs) to provide 480V 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 and completed 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 non-safety-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 Availability Controls Manual (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 (SDGLS) 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 non-safety-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 the 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, "Flood Protection," Section 3.5.1.1, "Internally Generated Missiles (Outside Containment)," Section 3.5.1.4, "Missiles Generated by Natural Phenomena," and Section 3.5.2, "Structures, Systems, and Components to be protected from externally Generated Missiles," of this report, the staff finds that the SDG JCWS meets the relevant requirements of GDC 2 as it pertains to Position C.2 of Reg. Guide 1.29. The SDGLS also meet the relevant requirements of GDC 2 as it pertains to 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, "Plant Design for Protection against Postulated Piping Failures in Fluid Systems Outside of Containment," the staff finds that the SDGs, the ADSs and their support systems including the SDGLS are protected against the effects of, and be 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, "Offsite Power System," and 8.3, "Onsite Power System," 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 the ESBWR DCD Revision 0, the staff determined that during a postulated post-loss of coolant accident (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, "Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety Systems (RTNSS) in Passive Plant Designs," to apply to the passive advanced light water reactor (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 its response to RAI 19.1.0-2, the applicant included the SDG units as RTNSS systems. However, it was not clear to the staff that all 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 its response to RAI 22.5-4, the applicant stated that all SDG supporting systems including SDGLS were considered as RTNSS systems. In Section 9.5.7 of the DCD Tier 2, Revision 4, SDGLS was included and classified as an RTNSS system. The staff determined that the responses to RAIs 19.1.0-2 and 22.5-4 regarding the SDGLS were 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 were resolved. RAI 22.5-4 was being tracked as a confirmatory item in the SER with open items. The staff confirmed that the DCD changes were incorporated 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 as RTNSS systems. In RAI 14.3-151, the staff requested request the applicant to include ITAAC for all the SDG supporting systems. In its response to RAI 14.3-151, the applicant stated that it would not include ITAAC for SDG supporting systems since they were non-safety 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, "ITAAC for The Standby Onsite Power Supply," the applicant included ITAAC for only two SDG supporting systems, the SDGFOSTS and the SDG starting air system. The staff further issued another RAI, 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 revise ESBWR DCD Revision 5 to include an ITAAC entry for each of the SDG supporting systems. The staff has determined 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, RAI 14.3-151 and 14.3-177 regarding the SDGLS are resolved. The staff confirmed that the DCD changes were incorporated into DCD revision 5.

In DCD Tier 1 Revision 6, Section 2.13.4, “Standby Onsite AC Power Supply,” and Table 2.13.4-2, “ITAAC for the Standby Onsite AC Power Supply,” the applicant provides the design descriptions and ITAAC for the SDGs, ADGs, and their supporting systems, including the SDGLS. The staff finds that these ITAACs which commit to verify that the SDGs, ADGs, and their supporting systems, including the SDGLS, are constructed and installed as described in DCD Tier 2, Revision 6. Therefore, the staff concludes 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 6, Subsections 14.2.8.1.37, “Standby Diesel Generator & AC Power System Preoperational Test,” provides the initial testing provisions associated with this review item. The staff reviewed DCD Tier 2, Revision 6, Section 9.5.7 against the guidance in SRP Section 14.3.7, “Evaluation Process For Updating Design Descriptions and ITAAC,” and finds that no additional ITAAC are needed in connection with this section.

DCD Tier 2, Revision 6, Section 16, “Technical Specifications,” does not have any technical specification requirements associated with this review item. The staff reviewed DCD Tier 2, Revision 6, Section 9.5.7 against 10 CFR 50.36, “Technical Specifications,” and agrees that no technical specifications are needed in connection with this section.

DCD Tier 2, Revision 6, Section 9.5.7.6, “Combined License (COL) Unit-Specific Information,” and Table 1.10-1, ”Summary of COL Items,” of the DCD Tier 2, Revision 6 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, “Regulatory Treatment of Non-Safety Systems,” 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 concludes that the LS for SDGs and ADGs design is acceptable and meets the relevant requirements of GDC 2, 4, 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 Economic Simplified Boiling Water Reactor (ESBWR) diesel generator combustion air intake and exhaust system (DGCAIES) in accordance with NUREG-0800, “Standard Review Plan (SRP) for the Review of Safety Analysis Reports for Nuclear Power Plants,” Section 9.5.8, “Emergency Diesel Engine Combustion Air Intake and Exhaust System,” Revision 3, dated March 2007. The staff reviewed ESBWR Design Certification Document (DCD) Tier 2, Revision 6, Section 9.5.8, “Diesel Generator Combustion Air Intake and Exhaust System.” The staff also reviewed DCD Tier 1, Revision 6, Section 2.0, “Design Descriptions and Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC),” 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 General Design Criteria (GDC) and Code of Federal Regulation (10 CFR 52.47(b)(1)):

- GDC 2, “Design Bases for Protection Against Natural Phenomena,” requires in part that structures, systems, and components (SSCs) important to safety shall be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes and floods.

  Compliance with GDC 2 is based on meeting the guidance of Regulatory Positions C.1 for the safety-related portions and C.2 for the non-safety-related portions of RG 1.29, “Seismic Design Classification.”

- GDC 4, “Environmental and Dynamic Effects Design Bases.” 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, and testing and postulated accidents.

- GDC 5, “Sharing of Structures, Systems, and Components,” 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 “Electric Power Systems,” 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) which 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.

9.5.8.2 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 non-safety-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 fuel oil storage and transfer system, jacket cooling water system, 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 480V 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 and completed 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 non-safety-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 DGCAIES have the RTNSS functions, as supporting systems to provide power and are included in the plant Availability Controls Manual (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, 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 non-safety-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 Section 3.4.1, “Flood Protection,” Section 3.5.1.1, “Internally Generated Missiles (Outside Containment),” Section 3.5.1.4, “Missiles Generated by Natural Phenomena,” and Section 3.5.2, “Structures, Systems, and Components to be protected from externally Generated Missiles,” of this report, the staff finds that the DGCAIES meets the relevant requirements of GDC 2 as it pertains to Position C.2 of Reg. Guide 1.29. The DGCAIES also meet the relevant requirements of GDC 2 as it pertains to 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 verified 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, "Plant Design for Protection against Postulated Piping Failures in Fluid Systems Outside of Containment," the staff finds that the SDGs, the ADSs and their support systems including the DGCAIES are protected against the effects of, and be 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, "Offsite Power System," and 8.3, "Onsite Power System," 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 the ESBWR DCD Revision 0, the staff determined that during a postulated post-loss of coolant accident (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, "Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety Systems (RTNSS) in Passive Plant Designs," to apply to the passive advanced light water reactor (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 its response to RAI 19.1.0-2, the applicant included the SDG units as RTNSS systems. However, it was not clear to the staff that all 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 its response to RAI 22.5-4, the applicant stated that all SDG supporting systems including SDCAIES were considered as RTNSS systems. In Section 9.5.8 of the DCD Tier 2, Revision 4, DGCAIES was included and classified as RTNSS system. The staff determined that the responses to RAIs 19.1.0-2 and 22.5-4 regarding the DGCAIES were acceptable since the applicant clarified that all SDG supporting systems including 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 DCD changes were incorporated 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 as RTNSS systems. In RAI 14.3-151, the staff requested request the applicant to include ITAAC for all the SDG supporting systems. In its response to RAI 14.3-151, the applicant stated that it would not include ITAAC for SDG supporting systems since they were non-safety 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, "ITAAC for The Standby On-Site Power Supply," the applicant included ITAAC for only two SDG supporting systems, the SDGFOSTS and the SDG starting air system. The staff further issued another RAI, 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 revise ESBWR DCD Revision 5 to include an ITAAC entry for each of the SDG supporting systems. The staff has determined 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, RAI 14.3-151 and 14.3-177 regarding the DGCAIES are resolved. The staff confirmed that the DCD changes were incorporated into DCD Revision 5.

In DCD Tier 1 Revision 6, Section 2.13.4, "Standby Onsite AC Supply," and Table 2.13.4-2, "ITAAC for the Standby Onside AC Power Supply," 6, the applicant provides the design descriptions and ITAACs for the SDGs, ADGs, and their supporting systems. The staff finds that these ITAACs which commit to verify that the SDGs, ADGs, and their supporting systems, including the DGCAIES, are constructed and installed as described in DCD Tier 2, Revision 6. Therefore, the staff concludes 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 6, Subsections 14.2.8.1.37, "Standby Diesel Generator & AC Power System Preoperational Test," provides the initial testing provisions associated with this review item. The staff reviewed DCD Tier 2, Revision 6, Section 9.5.8 against the guidance in SRP Section 14.3.7, "Evaluation Process For Updating Design Descriptions and ITAAC," and finds that no additional ITAAC are needed in connection with this section.

DCD Tier 2, Revision 6, Section 16, "Technical Specifications," does not have any technical specification requirements associated with this review item. The staff reviewed DCD Tier 2, Revision 6, Section 9.5.8 against 10 CFR 50.36, "Technical Specifications," and agrees that no technical specifications are needed in connection with this section.

Section 9.5.8.6, "Combined License (COL) Unit-Specific Information," and Table 1.10-1, "Summary of COL Items," of the DCD Tier 2, Revision 6 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, "Regulatory Treatment of Non-Safety Systems," 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 concludes that the combustion air intake and exhaust systems for SDGs and ADGs design is acceptable and meets the relevant requirements of GDC 2, 4, and 10 CFR 52.47(b)(1).