9. AUXILIARY SYSTEMS

9.1 Fuel Storage and Handling

9.1.1 New Fuel Storage

9.1.1.1 Regulatory Criteria

The staff of the U.S. Nuclear Regulatory Commission (NRC) has reviewed the economic simplified boiling-water reactor (ESBWR) advanced reactor's new fuel storage capability 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 9.1.1, Revision 2, "New Fuel Storage," issued July 1981. The staff evaluation of the new fuel storage facility is contingent on compliance with the following requirements:

- General Design Criterion (GDC) 2, "Design Bases for Protection Against Natural Phenomena," as it relates to the ability of the facility and the structures housing it to withstand the effects of natural phenomena, such as earthquakes, tornados, and hurricanes.

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

- 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 62, "Prevention of Criticality in Fuel Storage and Handling," as it relates to the prevention of criticality by physical systems or processes utilizing geometrically safe configurations.

9.1.1.2 Summary of Technical Information

In Design Control Document (DCD) Tier 2, Section 9.1.1, the applicant provided the design bases, a description, and a safety evaluation 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 fuel building (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 then placed in the spent fuel pool (SFP) or in the inclined fuel transfer system (IFTS) for transfer to the reactor building (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 in the RB buffer pool hold up to 60 percent of the reactor pressure vessel (RPV) core capacity of channeled fuel assemblies. The fuel assemblies are stored in underwater storage racks located adjacent to the reactor well. The racks are side loading and are accessed using the refueling bridge. The racks have double rows of storage positions for assemblies that are side loaded into the storage racks. 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.

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 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 of all sides. The fuel channel is placed on the fuel bundle in the new fuel inspection stand. To facilitate fuel inspection, the stand is set into an inspection pit designed to allow the carriage to be lowered and raised, permitting eye-level viewing by inspecting personnel on the refueling floor.

9.1.1.3 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. In accordance with SRP Section 9.1.1, 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 noted that DCD Tier 2, does not have a statement to indicate that the spent fuel storage conforms to the industry standards of ANS 57.1 and 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 requirements. In its response, the applicant provided a markup of DCD Tier 2, Section 9.1.1 which the staff found acceptable. **RAI 9.1-39 is being tracked as a Confirmatory Item**.

Section 9.1.2 of this report further discusses the staff's evaluation as it relates to GDC 2, 61, and 62 for both new and spent fuel storage facilities since the two evaluations are very similar. **Since requests for additional information (RAIs) 9.1-3, 9.1-28, and 9.1-45 remain to be resolved and are being tracked as open items**, the staff was unable to finalize its conclusions regarding compliance with the requirements of GDC 2, 61, and 62. These open items are discussed further in Section 9.1.2 of this report.

The staff verified that the racks have been designed to preclude damage 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. Section 9.1.5 of this report discusses the staff's evaluation of the ESBWR heavy-load-handling systems and controls.

In Section 9.1.6 of DCD Tier 2, Revision 3, the applicant identified the following **combined license (COL) Action Items** relating to Section 9.1.1 of DCD Tier 2:

- The COL holder shall provide the dynamic and impact analyses of the fuel storage racks.

- The COL holder shall confirm that the fuel storage racks are designed to provide sufficient natural convection coolant flow through the rack and fuel to remove decay heat without reaching excessive water temperatures.

- The COL holder shall provide the criticality analysis described in Section 9.1.1.1 of DCD Tier 2.

The staff found that the above three COL holder items are analysis and design issues that have to be reviewed by the NRC staff before it issues a COL. The information for COL holder items will be available for review only after the license is issued. This is not acceptable because the staff will not be able to conclude, at the time the license is issued, whether the design and analysis of the spent fuel storage facility satisfy the regulatory requirements. The staff requested in RAI 9.1-40 that the applicant revise the above three COL holder items to become COL applicant items. In its response to RAI 9.1-40 the applicant stated that it will submit two topical reports 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." The applicant also stated that it would remove the COL holder items from the DCD. The staff finds this approach acceptable. However, the staff remains unable to evaluate compliance with GDC 2, 61, and 62 pending the submittal and review of the aforementioned topical reports. RAI 9.1.40 is now resolved, however, the staff will need to review NEDO-33373 and NEDO-33374 when they are submitted. **The review of NEDO-33373 and NEDO-33374 is an open item**.

9.1.1.4 Conclusion

Because of the open items that remain to be resolved for this section, including the submittal and review of NEDO-33373 and NEDO-33374, the staff was unable to complete its review of the new fuel storage facility. The staff is unable to finalize its conclusion regarding compliance with the requirements of GDC 2, 5, 61, and 62.

9.1.2 Spent Fuel Storage

9.1.2.1 Regulatory Criteria

The staff reviewed the ESBWR spent fuel storage capability in accordance with SRP Section 9.1.2, Revision 3, "Spent Fuel Storage," issued July 1981. The staff's acceptance of the spent fuel storage facility is based on compliance with the following requirements:

- GDC 2, as it relates to the ability of the facility and the structures housing it 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 ability of the facility and the structures housing it to withstand the effects of external missiles and internally generated missiles, pipe whip, jet impingement forces, and adverse environmental conditions associated with pipe breaks, such that safety functions will not be impaired.

- GDC 5, as it relates to whether shared SSCs important to safety are capable of performing required safety functions.

- GDC 61, as it relates to the facility design provisions for safe fuel storage and handling of radioactive materials.

- GDC 62, as it relates to the prevention of criticality by physical systems or processes utilizing geometrically safe configurations.

- 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.

9.1.2.2 Summary of Technical Information

The fuel storage racks provided in the SFP in the FB provide for storage of irradiated fuel assemblies resulting from 20 calendar years of plant operation plus one full-core offload. The fuel storage racks in the RB buffer pool deep pit can hold a total of 154 spent fuel assemblies.

The fuel storage racks in the RB buffer pool and in the SFP in the FB contain storage space for fuel assemblies (with channels) or bundles (without channels). A standard dynamic analysis using the appropriate response spectra is performed to demonstrate conformance to design requirements. The spent fuel storage racks are designed to withstand all credible static and seismic loadings. Furthermore, the racks are designed to protect the fuel assemblies and bundles from excessive physical damage which 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 10 CFR Part 100, "Reactor Site Criteria," 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. 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. Pool fill and drainlines enter the pool above the safe shielding water level. Redundant antisiphon 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. 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.

9.1.2.3 Staff Evaluation

Compliance with the requirements of GDC 2 depends on adherence to the guidance of Regulatory Position C.2 of RG 1.13, "Spent Fuel Storage Facility Design Basis"; the applicable portions of RG 1.29 and RG 1.117, "Tornado Design Classification"; and paragraphs 5.1.1, 5.1.3, 5.1.12, 5.3.2, and 5.3.4 of ANS 57.2, "Design Objectives for Light Water Reactor Spent Fuel Storage Facilities at Nuclear Power Stations." Compliance with the requirements of GDC 4 depends on adherence to the guidance of Regulatory Position C.3 of RG 1.13, as well as

RG 1.115, “Protection Against Low-Trajectory Turbine Missiles”; RG 1.117; and the appropriate paragraphs of 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 depends on adherence to the guidance of Regulatory Positions C.1 and C.4 of RG 1.13, the appropriate paragraphs of 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 62 depends on adherence to the guidance of Regulatory Positions C.1 and C.4 of RG 1.13, as well as the appropriate paragraphs of ANS 57.2. Finally, compliance with the requirements of GDC 63 depends on adherence to the guidance of paragraph 5.4 of ANS 57.2 and Regulatory Position C.7 of RG 1.13.

While DCD Tier 2, Section 9.1.2 provides several design bases, it does not address directly compliance with GDC 2, 4, 61, 62, and 63. In RAI 9.1-45 the staff requested the applicant to revise the DCD to address compliance with GDC 2, 46, 61, 62, and 63 and conformance to associated regulatory guides and industrial standards for spent storage in accordance with the SRP. **RAI 9.1-45 is being tracked as an Open Item**.

The staff verified compliance with the requirements of GDC 2. The applicant stated that fuel storage racks and pool liner embedments are designed to meet seismic Category I requirements. DCD Tier 2, Revision 3, Section 9.1.2.4 describes the loads applied to the rack. The applicant stated that stress analyses are performed by classical methods based upon shears and moments developed by a dynamic method. Using the given loads, load conditions and analytical methods, stresses are calculated at critical sections of the rack and compared to acceptance criteria referenced in American Society of Mechanical Engineers (ASME) Code Section III, Subsection NF.

Both the RB and the FB, which contain the fuel storage facilities, including the storage racks and pool, 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 a seismic Category II to maintain its structural integrity during a safe-shutdown earthquake (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. The racks are designed to withstand the impact force generated by the vertical free-fall drop of a fuel assembly from the maximum height expected during normal fuel handling. The applicant also stated that since there can be, at most, two fuel assemblies adjacent in this scenario, the array will remain subcritical. The staff finds this acceptable.

Because RAI 9.1-45 remains to be resolved, the staff was unable to finalize its conclusions regarding compliance with the requirements of GDC 2.

The staff verified compliance with the requirements of GDC 4. 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, Section 9.1.5.5, the applicant stated that the FB crane provides heavy-load-lifting capability for the FB floor. The main hook is used to lift new fuel shipping containers and the spent fuel shipping cask. The applicant stated that the orderly

placement and movement paths of these components by the FB crane preclude transport of these heavy loads over the SFP. The FB crane is used during refueling/servicing 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 the spent fuel shipping cask. The applicant stated that the FB crane is interlocked to prevent movement of heavy loads over the SFP.

Similarly, the applicant stated that the RB crane provides heavy-load-lifting capability for the refueling floor. The main hook is used to lift the drywell head, 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. The applicant stated that transport of these heavy loads over the spent fuel racks in the deep pit buffer pool or over the new fuel rack is prohibited by safe load paths. The RB crane is also used during refueling/servicing 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.

The staff requested in RAI 9.1-15 that the applicant describe how light-load-handling accidents would be mitigated. In its response, 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, Section9.1.2.4. The applicant also stated that the liner is seismic Category I and is designed to the acceptance criteria of ASME Code, Section III, Division 2, CC-3700. The staff found that this response insufficient and requested that the applicant provide (1) analyses demonstrating that the pool liner will retain its leaktight 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 is being tracked as an Open Item.**

Because RAIs 9.1-15 and 9.1-45 remain to be resolved, the staff was unable to finalize its conclusions regarding compliance with the requirements of GDC 4.

The staff verified compliance with the requirements of GDC 61. The guidelines in SRP Section 9.1.2 specify that the materials wetted in the SFP (e.g., spent fuel racks, fixed neutron poison, and the SFP liner) and, if applicable, the new fuel vault must be chemically compatible and stable. DCD Tier 2, Section 9.1.2, states that the spent fuel storage racks of the ESBWR are constructed in accordance with a quality assurance program to ensure that design, construction, and testing requirements are met.

In RAI 9.1-27, the staff requested that 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 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 (17-4PH) 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 guidelines in SRP Section 9.1.2 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, Section 9.1.2, was unclear whether such a program was 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.

The staff found this response insufficient and requested in RAI 9.1-28 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 is being tracked as an Open Item.**

Because RAIs 9.1-28 and 9.1-45 remain to be resolved, the staff was unable to finalize its conclusions regarding compliance with the requirements of GDC 61.

The staff verified compliance with GDC 62 by reviewing the design and arrangement of the storage racks to ensure that a subcritical array can be maintained during all storage conditions. The applicant stated that the SFP was 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. 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.

The staff finds this insufficient to conclude that measures have been taken to provide adequate cooling for high-density racks. The staff requested that the applicant provide information such as assembly dimensions, center-to-center distance, array layouts, and location within the pool. **RAI 9.1-3 is being tracked as an Open Item.**

The staff verified that the design of the storage racks is such that a fuel assembly cannot be inserted anywhere other than in a designed location. DCD Tier 2, Section 9.1.2, states 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 finds this acceptable; however, the staff requested that the applicant include this information in the DCD. In its response to RAI 9.1-4 S01 the applicant stated that it will revise the DCD accordingly. The staff finds this acceptable. **RAI 9.1-4 is being tracked as a Confirmatory Item.**

Because RAIs 9.1-3, 9.1-4, and 9.1-45 remain to be resolved, the staff was unable to finalize its conclusions regarding compliance with the requirements of GDC 62.

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

In Section 9.1.6 of DCD Tier 2, Revision 3, the applicant identified the following **COL Action Items** relating to DCD Tier 2, Section 9.1.1.

- The COL holder shall provide the dynamic and impact analyses of the fuel storage racks.

- The COL holder shall confirm that the fuel storage racks are designed to provide sufficient natural convection coolant flow through the rack and fuel to remove decay heat without reaching excessive water temperatures.

- The COL holder shall provide the criticality analysis described in Section 9.1.1.1 of DCD Tier 2.

The staff found that the above three COL holder items are analysis and design issues that have to be reviewed by the NRC staff before a COL is issued. The information for COL holder items will be available for review only after the license is issued. This is not acceptable because the staff will not be able to conclude, at the time the license is issued, whether the design and analysis of the spent fuel storage facility satisfy the regulatory requirements. In RAI 9.1-40, the staff requested that the applicant revise the above three COL holder items to become COL applicant items. In its response to RAI 9.1-40, the applicant stated that it will submit NEDO-33373 and NEDO-33374 to provide fuel rack. The applicant also stated that it would remove the COL holder items from the DCD. The staff finds this approach acceptable. However, the staff remains unable to evaluate compliance with GDC 2, 61, and 62 pending the submittal and review of the aforementioned topical reports. RAI 9.1-40 is now resolved, however, the staff will need to review NEDO-33373 and NEDO-33374 when they are submitted. **The review of NEDO-33373 and NEDO-33374 is an open item**.

9.1.2.4 Conclusion

Because of the open items that remain to be resolved for this section, including the submittal and review of NEDO-33373 and NEDO-33374, the staff was unable to complete its review of the spent fuel storage facility. The staff is unable to finalize its conclusions regarding compliance with the requirements of GDC 2, 4, 5, 61, 62, and 63.

9.1.3 Spent Fuel Pool Cooling and Cleanup System

9.1.3.1 Regulatory Criteria

The staff 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 design of the FAPCS is acceptable if its integrated design meets the requirements of 10 CFR Part 50. Specifically, the design must meet the following requirements of Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50:

- 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 SSCs important to safety are capable of performing required safety functions.
- GDC 44, "Cooling Water," as it relates to the following:
  - the system's ability to transfer heat loads from safety-related SSCs to a heat sink under both normal operating and accident conditions;
  - suitable redundancy of components so that safety functions can be performed assuming a single active failure of a component coincident with a loss of offsite power (LOOP) event;

- the system's ability to isolate components, systems, or piping so that they do not compromise the system's safety function;

- GDC 45, "Inspection of Cooling Water System," as it relates to allowing periodic inspection of safety-related components and equipment.

- GDC 46, "Testing of Cooling Water System," as it relates to allowing operational functional testing of safety-related systems or components to ensure structural integrity and system leaktightness, operability, and adequate performance of active system components, as well as the capability of the integrated system to perform the required functions during normal, shutdown, and accident conditions.

- 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.

The FAPCS must also meet the requirements of 10 CFR 20.1101(b), as it relates to radiation doses being kept as low as is reasonably achievable (ALARA).

In addition, the staff reviewed the FAPCS emergency makeup capability to the isolation condenser (IC)/passive containment cooling system (PCCS) 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 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 PCCS can be accomplished assuming a single failure.

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 PCCS pools. A separate subsystem with its own pump, heat exchanger, and water treatment unit is dedicated for cooling and cleaning of the IC and PCCS 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 flowpaths for filling and makeup of these pools during normal plant operation and during postaccident conditions, as necessary. FAPCS is also designed, if needed, to provide the following accident recovery functions 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 other 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 MCR, except the SPC mode, which is initiated either manually or automatically on a high suppression pool water temperature signal. 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 flowpath for post-accident refilling of the IC/PCCS 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 and seismic Category I or II classification. Provisions are taken to prevent inadvertent draining of the pools.

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, C.2, C.6, and C.8 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 the requirements of GDC 5 are not applicable to the single-unit design. Compliance with the requirements of GDC 44 may be based on adherence to the recommendations of SRP Branch Technical Position (BTP) Auxiliary Systems Branch (ASB) 9-2, "Residual Decay Energy for Light Water Reactors for Long-Term Cooling," for calculating the heat loads, the assumptions set forth in Criterion III.1.h of SRP Section 9.1.3, and the pool temperature limitations identified in Criterion III 1.d of SRP Section 9.1.3. 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 12.1.4.4. This is discussed in Chapter 12 of this report. The staff finds this acceptable.

The staff verified compliance 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 3, Table 3.2-1 states that the RB and FB are designed to seismic Category I requirements. Consistent with Regulatory Position C.2 of RG 1.13 and the design criteria specified in SRP Section 3.5.3, Revision 2, "Barrier Design Procedures," the staff finds this acceptable to meet GDC 4 for those portions of the system located inside these buildings. Section 3.5 of this SER discusses protection against external missiles.

DCD Tier 2, Revision 3, Table 9.1-3 states 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) 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 flowpath for post-accident refilling of the IC/PCCS and SFPs with emergency water supplies.

Consistent with Regulatory Position C.8 of RG 1.13 and Criterion III.1.f of SRP Section 9.1.3, the staff verified that the ESBWR design provides a seismic Category I makeup system and an appropriate backup method to add coolant to the SFP. DCD Tier 2, 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, 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 is designed to Quality Group D or lower and the fire pump enclosure is nonseismic. The staff requested that the applicant address quality

classification and seismic qualification 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 fire pumps are mounted on a seismic Category 1 concrete slab and the enclosures are classified as seismic Category II. The staff did not find this acceptable and requested that the applicant classify 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 is being tracked as a Confirmatory Item.**

In its response to RAI 9.1-12, the applicant also stated that the FPS provides the makeup water capability from 72 hours to 7 days, following an accident, after which time either other onsite or offsite makeup sources can be used. The applicant stated that this function of the FPS is considered to be a regulatory treatment of non-safety systems (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. Not withstanding, 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 staff found this acceptable. However, in its response to RAI 9.1-12 S01, the applicant stated that it will keep the Quality Group D designation as a result of further investigation of RTNSS quality assurance requirements. The applicant stated that this classification is identified in DCD Tier 2, Table 1.9-9, 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 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 is expected for pools that rely on active components for providing makeup during this period. Notwithstanding, the staff has not been able to determine the adequacy of the design to perform these functions 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 is being tracked as an Open Item.**

DCD Tier 2, Revision 3, Section 9.1.3 identifies the FAPCS piping and components required for emergency makeup safety-related and not as RTNSS. 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 requested 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 is being tracked as an Open Item.**

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.

Because RAIs 9.1-42 and 9.1-44 remain to be resolved, the staff was unable to finalize its conclusions regarding compliance with the requirements of GDC 2 and GDC 4.

The staff verified the design to ensure compliance with the requirements of GDC 44. The staff verified that essential portions of the system are correctly identified and are isolable from the nonessential portions of the system. DCD Tier 2, 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 flowpath during FAPCS operation. One loop is used for the SFP and auxiliary pools, and the other loop for the GDCS pools and suppression pool and for injecting water to the drywell spray sparger and reactor vessel via RWCU/SDC and feedwater pipes. The use of manifolds with proper valve alignment and separate suction-discharge piping loops serves two purposes—(1) it allows operating one train independently of the other to permit online maintenance or dual mode operation using separate trains, if necessary, and (2) it prevents inadvertent draining of the pool and mixing of contaminated water in the SFP with cleaner water in other pools.

The staff requested in RAI 9.1-8 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, the applicant stated that the SFP is designed to dissipate fuel decay heat through heat up and boiling of the pool water. The applicant stated that pool water performs the safety-related heat removal function stipulated in GDC 44. Upon loss of power, the FB heating, ventilation, and air conditioning system isolates the FB as described in DCD Tier 2, Section 9.4.2.5. Steam generated by boiling of the SFP is released to the atmosphere (the ultimate heat sink) through a relief panel in the FB.

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 is being tracked as an Open Item.**

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 of time 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 finds the applicant's response insufficient 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 is being tracked as an Open Item.**

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 BTP ASB 9-2.

The applicant 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).

The staff requested in RAI 9.1-10 that the applicant specify how adequate decay heat removal capacity will be demonstrated for normal operating (i.e., nonaccident) conditions. In its response, the applicant stated that SFP heat power as a function of time after shutdown is calculated based on a computer code developed using the standards in American National Standards Institute (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 finds that the applicant’s response to RAI 9.1-10 is insufficient to determine the acceptability of the FAPCS C/C as it relates to GDC 44 (and 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, and that GDC 44 is satisfied by passive pool boiling for 72 hours and subsequent makeup. The staff does not agree with this statement; GDC 44 and 61 require 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 that the applicant provide a summary heat balance of the FAPCS, including initial assumptions and performance requirements. **RAI 9.1-10 is being tracked as an Open Item.**

Because RAIs 9.1-8, 9.1-9, and 9.1-10 remain to be resolved, the staff was unable to finalize its conclusions regarding compliance with the requirements of GDC 44.

The staff confirmed that design provisions exist to permit appropriate inservice inspection and functional testing of system components. DCD Tier 2, Section 9.1.3.4, states that the FAPCS is

designed to permit surveillance testing and inservice inspection of the safety-related components in accordance with ASME Code Section XI. Additionally, the FAPCS is designed to permit leak-rate testing of its components that are required to perform a containment isolation function, in accordance with Appendix J, "Primary Water Containment Leakage Testing for Water-Cooled Power Reactors," to 10 CFR Part 50. The staff finds this acceptable to meet the requirements of GDC 45 and 46. Sections 6.6 and 3.9.6 of this report discuss inservice inspection and inservice testing, respectively.

The staff verified that the SFP and cooling systems meet the requirements of GDC 61. The requirements may be met in part by adherence to 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 top of the active fuel. The applicant 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. Pool fill and drainlines enter the pool above the safe shielding water level. Redundant antisiphon 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.

The staff requested in RAI 9.1-11 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 antisiphoning provisions as described above. Antisiphon 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 antisiphon 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 GDCS pools, suppression pool, and IC/PCCS pools. The applicant stated that these lines will also have antisiphoning provisions. The applicant stated that suction lines cannot have holes at the normal water level, therefore the antisiphon holes will be included on all suction lines at the elevation of the minimum water level for each respective pool. The staff finds this acceptable. However, 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. **RAI 9.1-11 is being tracked as a Confirmatory Item**.

The staff requested in RAI 9.1-13 that the applicant clarify how the redundancy requirements of the GDC 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/PCCS and SFPs. This change ensures that onsite water sources remain available as makeup for the IC/PCCS and SFPs for the first 7 days, even if a single active failure were to occur. The addition of these parallel valves ensures that the ICS and PCCS condensers can provide sufficient heat removal capability at and beyond 72 hours to satisfy GDC 34 and 38 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, the staff requested that the applicant address failure of the diesel fire pump and demonstrate 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 is being tracked as an Open Item.**

The staff requested in RAI 9.1-14 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/PCCS 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, to include a requirement for performance of this test, and provided the proposed inspection, test, analysis acceptance criteria (ITAAC). However, the staff found the applicant's response insufficient to conclude that the design will accommodate the required flow rate. Pursuant to 10 CFR 52.47, "Contents of Application," the staff requested that the applicant indicate makeup line pipe size and analyses demonstrating that the pipe size is adequate to provide the required flow to each pool for limiting conditions. **RAI 9.1-14 is being tracked as an Open Item.**

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, which identifies the maximum heat conditions as resulting from 20 years of operations. 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 staff found the response acceptable; therefore, this issue is closed. However, the applicant has not provided the cooling requirements to the staff and they are the subject of **RAI 9.1-10, which is being tracked as an Open Item**.

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, Section 9.1.3, states that the spent fuel cleanup contains a prefilter, a demineralizer, and a poststrainer.

The staff requested in RAI 9.1-29 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 GDCS pools. The capacity of the FAPCS is sufficient to achieve two water changes per day of all the pools served by the system. The water quality requirements vary depending on the pool. Therefore, the specific water quality requirements for the FAPCS F/D units are determined using guidance from several sources, including RG 1.13, SRP Section 9.1.3, and the Electric Power Research Institute's "Advanced Light Water Reactor Utility Requirement Document," Revision 8, Volume III, Section 2.2.3.2. The staff finds the applicant's description acceptable because it is a typical system configuration.

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. DCD Tier 2, Section 9.1.3, 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.

Because RAIs 9.1-10, 9.1-13, and 9.1-14 remain to be resolved, the staff was unable to finalize its conclusions regarding compliance with the requirements of GDC 61.

DCD Tier 2, Section 9.1.3, states that safety-related level instrumentation is provided in the SFP and IC/PCCS pools to detect a low water level that would indicate a loss of decay heat removal ability, in accordance with GDC 63.

The staff confirmed that the system includes 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. DCD Tier 2, Section 9.1.3.5, states that 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 signal is used to initiate low- and high-water-level alarms and to operate the makeup water control valve for the skimmer surge tank. The IC/PCCS pool has two local, panel-mounted, safety-related level transmitters. Both transmitter signals are indicated on the safety-related displays and sent through the gateways for 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/PCCS pool. The SFP has 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. All other pools (suppression pool, upper transfer pool, buffer pool, reactor well, dryer and separator storage pool) have local, non-safety-related, panel-mounted level transmitters to provide signals for high/low-level alarms in the MCR.

The applicant stated that water temperatures are monitored in the fuel and auxiliary pools with 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/PCCS pool, each condenser vault also has temperature elements and local panel-mounted temperature transmitters that send signals to the MCR for water temperature indication and high-temperature alarms. The upstream and downstream piping of the two heat exchangers in the C/C trains

have temperature elements and local panel-mounted temperature transmitters that send signals to the MCR.

The applicant stated that 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. 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 staff requested in RAI 9.1-18 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. The applicant stated that, 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 found the applicant's response insufficient and requested that the applicant provide a description of the SFP water level instrumentation, including its location relative to the top of the fuel, and to describe how the operators will respond to a low-level alarm when normal makeup is unavailable. In its response to RAI 9.1-18 S01, the applicant stated that the instrumentation indication spans from the normal water level to the top of active fuel, 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 top of active fuel. The staff finds this response insufficient. The amount of water between the top of active fuel and the SFP low-level alarm must be specified to ensure that the operators are able to detect a condition that may result in loss of decay heat removal or excessive radiation levels, as required by GDC 63. Since the applicant stated that no operator actions are needed for 72 hours, the staff requested that the applicant demonstrate that the low-level setpoint is set such that there are at least 72 hours before the top of active fuel is reached, assuming a loss of forced cooling during the maximum decay heat load conditions. **RAI 9.1-18 is being tracked as an Open Item.**

In addition, the staff requested in RAI 9.1-41 that the applicant describe how the performance of this instrumentation as required during boiling conditions in the associated pools is verified. In its response, the applicant indicated that boiling of water in the SFP may introduce some inaccuracy in water level, but any errors would be conservative. The staff is not clear as to how a decrease in the density of water in the SFP will result in a conservative water-level measurement. The staff requested 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 top of active fuel, 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 is being tracked as an Open Item.**

The staff requested in RAI 9.1-17 that the applicant describe how potentially radioactive leakage from the fuel storage pools and the FAPCS is collected and processed. In its response, the applicant stated that leakage channels are provided behind each weld of the fuel pool liners to

collect leakage. The applicant also stated that all leakages are channeled to headers and drainlines 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. The applicant stated that 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, Section 9.3.3. The staff finds this acceptable; however, the staff requested that the applicant include this information in the DCD. The staff reviewed the applicant's response to RAI 9.1-17 S01 and found it acceptable. The staff confirmed that the response to RAI 9.1-17 was acceptably incorporated in Revision 3 of the DCD and therefore the RAI is resolved.

Because RAIs 9.1-18 and 9.1-41 remain to be resolved, the staff was unable to finalize its conclusions regarding compliance with the requirements of GDC 63.

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/PCCS pool for long-term cooling, in accordance with SRP Sections 5.4.7 and 6.2.2.

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

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, 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 net positive suction head 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. 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 is being tracked as an Open Item.**

In accordance with SRP Section 6.2.2, the staff verified compliance 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 PCCS can be accomplished assuming a single failure.

Criterion III.20 of SRP Section 6.3, Revision 2, issued April 1984, 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/PCCS and SFPs. In DCD, Revision 2, the applicant revised Figures 2.6.2-1 in Tier 1 and 9.1-1 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 has 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-loss-of-coolant accident and confirm conformance to the SRP. **RAI 9.1-21 is being tracked as an Open Item.**

The staff requested in RAI 9.1-22 that the applicant clarify how the long-term cooling function of the PCCS is satisfied assuming a single active or passive failure affecting the makeup line from the FAPCS. In its response, the applicant stated that there are two large expansion pools on either side of the RB, which are normally isolated from each other. These two pools are divided into three separate compartments. The three compartments of both expansion pools are interconnected by valves, which are locked open, and the three compartments of each 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 dryer/separator 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/PCCS pools to utilize the inventory in the dryer/separator pool and reactor well. The staff requested in RAI 9.1-32 that the applicant clarify how many lines actually discharge into the IC/PCCS pools since the expansion pools are not normally connected. In its response, the applicant stated that, during an accident in which pool water is boiling off, a low-level setpoint in either of the IC/PCCS expansion pools causes safety-related connections to the dryer/separator pool to open. These connections allow water to flow freely between the expansion pools as well as the dryer/separator pool and reactor well. The staff finds this acceptable.

The applicant stated that within both expansion pools there are five smaller subcompartments—three for PCCS 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 larger 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.

Because RAIs 9.1-19, 9.1-20, and 9.1-21 remain to be resolved, the staff was unable to finalize its conclusions regarding compliance with the requirements of GDC 34 and GDC 38.

DCD Tier 2, 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 that the applicant provided the (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.

The staff requested in RAI 9.1-23 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 F016 and F017. The staff finds this acceptable.

9.1.3.4 Conclusions

Because of the open items that remain to be resolved for this section, the staff was unable to finalize its conclusions regarding the compliance of the FAPCS with the requirements of GDC 2, 4, 5, 34, 38, 44, 45, 46, 61, and 63.

9.1.4 Light-Load Handling System (Related to Refueling)

SRP Section 9.1.4, 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).

9.1.4.1 Regulatory Criteria

The staff reviewed the LLHS in accordance with SRP Section 9.1.4, Revision 2, “Light Load Handling System (Related to Refueling),” issued July 1981. 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.

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 ESBWR design includes the equipment designed to facilitate the periodic refueling of the reactor, specifically the FB crane, RB crane, refueling machine, fuel-handling machine, IFTS, fuel preparation machine, new fuel inspection stand, dryer and chimney/separator 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.

Fuel Building 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. Its 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 II; however, it is designed to withstand an SSE so as to prevent structural failure which could lead to degradation of a seismic Category I structure, system, or equipment to an unacceptable safety level.

Reactor Building 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 II.

The RB crane consists of two parallel girders along which the trolley traverses their span. It is classified as seismic Category II; however, it is designed to withstand an SSE so as to prevent structural failure with SSE loads, leading to degradation of the functioning of a seismic Category I structure, system, or equipment to an unacceptable safety level.

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 embedded tracks in 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 is 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 interlock grapple hook with hoist load and hoist-up power,
- ensure correct sequencing of the transfer operation in the automatic or manual mode.

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 in 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 non-safety-related seismic Category II; however, it is designed to withstand an SSE.

Fuel-Handling Machine

The fuel-handling machine 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 refueling floor. A telescoping mast and grapple suspended from a trolley system is 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 is 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 interlock grapple hook with hoist load and hoist-up power,
- ensure correct sequencing of the transfer operation in the automatic or manual mode.

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 in 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 non-safety-related seismic Category II; however, it is designed to withstand an SSE.

Fuel Transfer System

The ESBWR is equipped with an IFTS. The arrangement of the IFTS consists of a terminus at the upper end in the RB buffer pool that allows the fuel to be tilted from a vertical position to an inclined position before transport to the SFP. There is a means to lower the transport device (i.e., a carriage), a means to seal off the top end of the transfer tube, and a control system to effect transfer. The ESBWR has a lower terminus in the FB storage pool, and is able to tilt the fuel into a vertical position allowing it to be removed from the transport cart. Controls contained in local control panels effect the transfer. The carriage and valves may be manually operated, in the event of a power failure, to allow completion of an initiated fuel transfer. 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 drainline connection, and the lower SFP terminus equipment (tube, valve, support structure, and bellows) are designated as non-safety-related and seismic Category I. The remaining equipment is designated as non-safety-related and nonseismic.

The IFTS penetrates the RB at an angle down to the IFTS pit in the fuel storage pool. The lower terminus of the IFTS, which is anchored to the bottom of the inclined fuel transfer pool, allows for thermal expansion (axial movement 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 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, is also 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 channeling and rechanneling new and 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 and Chimney Head/Separator Strongback

The dryer and chimney head/separator strongback is a lifting device used for transporting the steam dryer or the chimney head with 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. This strongback or a similar device is also used to remove the chimney partition assembly if this option is implemented to facilitate refueling operations.

The strongback has been designed such that one hook pin and one main beam of the cruciform is capable of carrying the total load (160 tons), and no single component failure could cause the load to drop or swing uncontrollably out of the safety-related level attitude. The strongback conforms to the provisions of NUREG-0612, “Control of Heavy Loads at Nuclear Power Plants” issued January 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 four-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 four 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, will carry four 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 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 the design to ensure compliance 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 applicant stated that standard dynamic analyses using the appropriate response spectra are performed to demonstrate compliance with design requirements. In RAI 9.1-33, the staff requested that the applicant provide the dynamic analyses for fuel-handling system components. **RAI 9.1-33 is being tracked as an Open Item.**

The staff’s evaluation of the General Electric Hitachi Nuclear America, LLC (GEH) standard dynamic analyses will be in accordance with Sections 3.7.2 and 3.7.3 of the SRP, Revision 3, issued March 2007, and will be 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 interlocks and physical stops to prevent raising the fuel assembly above the normal stop position required for safe handling of the fuel. The operator can observe the exact grapple position over the core via a display screen at the operator console.

In DCD Tier 2, Revision 3, Section 9.1.4.12, the applicant stated that the IFTS is designed with sufficient redundancy and diversity in equipment and controls to prevent loss of load (carriage with fuel is released in an uncontrolled manner). The applicant also stated that no modes of operation will allow simultaneous opening of any set of valves 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, Section 2.5.10. In RAI 9.134, 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 is being tracked as an Open Item.**

In addition, 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 drainline connection, and the lower SFP terminus equipment (tube, valve, support structure, and bellows) are designated as non-safety-related and seismic Category I. The remaining equipment is non-safety-related and nonseismic.

The staff has not been 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 is being tracked as an Open Item.**

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 is being tracked as an Open Item.**

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, GEH has not described how the design of the chimney head/separator strongback and the head strongback/tensioner has 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 is being tracked as an Open Item.**

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

Because RAIs 9.1-33, 9.1-34, 9.1-35, 9.1-36, and 9.1-37 remain to be resolved, the staff was unable to finalize its conclusions regarding compliance with 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.

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.

Sections 15.7.4 and 15.7.5 of this report discuss the staff's evaluation of radiological consequences of fuel-handling accidents and spent fuel cask drop accidents, respectively.

DCD Tier 2, 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 causing a fuel assembly to be 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 preestablished heavy-load boundary zones, thereby limiting the travel of the refueling and fuel-handling machines. The staff finds the applicant's response to be acceptable.

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, the applicant stated that administrative controls are applied to the tabulated listing of the cranes and refueling equipment provided in Table 9.1-6. 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 items be changed to COL applicant 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 combined license. In its response, the applicant confirmed that these will indeed be provided by the COL applicant, and included a markup of **COL Action Items 9.1.6.4** (fuel handling operations) and **9.1.6.5** (handling of heavy loads), which the staff finds acceptable. **RAI 9.1-25 is being tracked as a Confirmatory Item.**

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

In 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. The staff found it acceptable and confirmed the changes in DCD Tier 2, Revision 2.

Compliance with the requirements of GDC 61 and GDC 62 depends on adherence to the guidance of ANSI/ANS 57.1. However, Section 9.1.4 does not contain a statement to indicate that the fuel-handling system conforms to the industry standards of 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. **RAI 9.1-43 is being tracked as an Open Item**.

9.1.4.4 Conclusion

Because the open items remain to be resolved for this section, the staff was unable to finalize its conclusions regarding the compliance of the LLHS with the requirements of GDC 2, 61, and 62.

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 and Section 2.5 of NUREG-0554. Compliance with the requirements of GDC 4 is based in part on Regulatory Position C.5 of RG 1.13. The ESBWR design is a single-unit station, 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 drywell servicing equipment, the lower drywell servicing equipment, the main steam tunnel servicing equipment, and other servicing equipment.

In DCD Tier 2, Revision 3, Section 9.1.5.3, the applicant 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 required 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 required through the equipment hatch and the precise positioning requirements 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.

In DCD Tier 2, Revision 3, Section 9.1.5.3, the applicant stated 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 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 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.

Fuel Building Crane

The FB is a reinforced concrete structure enclosing the SFP, cask-handling and cleaning facility, and other equipment. The FB crane provides heavy-load lifting capability for the FB floor. The main hook (160-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 the spent fuel shipping cask. The FB crane is interlocked to prevent movement of heavy loads over the SFP.

Reactor Building Crane

The RB is a reinforced concrete structure enclosing the reinforced concrete containment vessel, the refueling floor, the new fuel storage buffer pool, the buffer pool deep pit pool for spent fuel storage, the dryer and chimney head/separator, and other equipment. The RB crane provides heavy-load lifting capability for the refueling floor. The main hook (150-ton capacity) is used to lift the drywell head, RPV head insulation, RPV head, dryer, chimney head/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.

Upper Drywell Servicing Equipment

The upper drywell arrangement provides servicing access for the main steam isolation valves (MSIVs), feedwater isolation valves, safety/relief valves (SRVs), depressurization valves (DPVs), PCCS valves, GDCS valves, and drywell cooling coils, fans, and motors. Access to the space is from the RB through either the upper drywell personnel lock or the equipment hatch. Equipment is removed through the upper drywell equipment hatch. Platforms are provided for servicing the feedwater isolation valves and MSIVs, the SRVs, and the drywell cooling equipment 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.

GEH identifies several features which prevent the upper drywell servicing equipment from handling critical loads. Drywell maintenance activities are only performed during a plant outage. The PCCS and GDCS piping and valves need to be protected from inadvertent load drops. PCCS and GDCS piping and valves are spatially separated to preclude an inadvertent load drop that breaks more than one pipe or valve of the PCCS or GDCS. In addition, a piping support structure and equipment platform separates and shields the PCCS and GDCS piping from heavy-load transport paths.

Lower Drywell Servicing Equipment

The lower drywell 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 360° 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.

Main Steam Tunnel Servicing Equipment

The main steam tunnel is a reinforced concrete structure surrounding the main steamlines and feedwater lines. The safety-related valve area of the main steam tunnel is located inside the RB. Access to the main steam tunnel is 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, all of the piping and valves are not required to operate.

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.

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 of the DCD:

> 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 9602, NUREG-0554, ANSI N14.6, ANSI B30.9, ANSI B30.10 and NUREG-0612 Subsection 5.1.1(4) or 5.1.1(5). Cranes and hoists are also designed to criteria and guidelines of NUREG-0612 Subsection 5.1.1(7), ANSI B30.2 and CMAA-70 specifications for electrical overhead traveling cranes, including ANSI B30.11, and ANSI B30.16 as applicable.

DCD Tier 2, 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 the above cited GDC, and how industry codes and standards are applied to specific components. **RAI 9.1-38 is being tracked as an Open Item.**

DCD Tier 2, Section 9.1.5.5, 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 does 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 confirmed that this information will be provided by the COL applicant, and included a markup of **COL Action Items 9.1.6.4** (fuel-handling operations) and **9.1.6.5** (handling of heavy loads), which the staff found acceptable. **RAI 9.1-2 is being tracked as a Confirmatory Item.**

9.1.5.4 Conclusion

Because of the open item that remains to be resolved for this section, the staff was unable to finalize its conclusions regarding the compliance of the OHLHS with the requirements of GDC 1, 2, and 4.

9.2 Water Systems

In DCD Tier 2, Revision 3, Section 3.1.1.5, GE 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 plant service water system (PSWS) 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 9.2.1, Revision 4, "Station Service Water System," issued June 1986. Staff acceptance of the design is based on its meeting the requirements of 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.

9.2.1.2 Summary of Technical Information

In the DCD Tier 2, Section 9.2.1, “Plant Service Water System,” the applicant described the PSWS. The system is designed to remove heat from non-safety-related components in the RB and turbine building (TB) to the environment. The PSWS consists of two independent and 100 percent-redundant open trains that continuously recirculate raw water through the reactor component cooling water system (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 reference design uses a natural draft cooling tower for the NPHS and mechanical draft cooling towers for the AHS. DCD Tier 2, Table 3.2.1, describes the PSWS as a non-safety-related system located in the RB and service building. The PSWS has RTNSS functions to provide post-72-hour cooling to the RCCWS and TCCWS. It is designated as quality group “D” and seismic Category “NS.”

In the event of a loss of preferred power (LOPP), the PSWS supports the RCCWS in bringing the plant to cold-shutdown condition in 36 hours assuming the most limiting single active failure. Tables 9.2-1 and 9.2-2 in DCD Tier 2 tabulate the PSWS design heat loads and component design characteristics.

9.2.1.3 Staff Evaluation

The staff reviewed the design of the PSWS in accordance with SRP Section 9.2.1. Staff acceptance of the PSWS is based on the design’s meeting the requirements of GDC 2, 4, 5, 44, 45, and 46. The PSWS for the ESBWR differs from that of the traditional boiling-water reactor (BWR) designs in that the ESBWR PSWS is a non-safety-related system because the PSWS removes heat only from the RCCWS and TCCWS, which are not safety-related systems. Therefore, the portions of SRP Section 9.2.1 that apply to safety-related systems do not apply to the PSWS. Sections 9.2.2 and 9.2.8 of this report contain the staff evaluations of the RCCWS and TCCWS.

As for the non-safety-related PSWS meeting 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.2 of RG 1.29, “Seismic Design Classification,” regarding non-safety-related systems. Regulatory Position C.1 of RG 1.29 for safety-related portions is not applicable for the PSWS. The ESBWR

design is a single-unit station, and the requirements of GDC 5 are not applicable to the single-unit design. Conformance to GDC 4 is based on the fact that the PSWS is designed to accommodate the dynamic effects associated with water hammer.

Conformance to GDC 44, 45, and 46 is based on the fact that the design of PSWS is capable of removing heat from SSCs important to safety to a heat sink and that design provisions are available for inspection and operational testing.

The staff asked the applicant to provide additional information in RAIs 9.2-6 through RAI 9.2-12. The staff reviewed the applicant's responses and its evaluation of the responses is discussed below.

In RAI 9.2-6, the staff requested that the applicant include piping and instrumentation diagrams (P&IDs) in the DCD for the PSWS and RCCWS showing system function, major equipment, components, piping classes, instrumentation, and interface systems. In its response, the applicant did not provide P&IDs in the DCD for the PSWS and RCCWS. The PSWS and RCCWS were recently scoped into RTNSS. Because these systems are important, the applicant should supplement or replace existing DCD Tier 2, Figures 9.2-1 and 9.2-2 to provide additional details. **RAI 9.2-6 is being tracked as an Open Item.**

In DCD Tier 2, Revision 1, Table 1.9.9 and Section 9.2.1, the applicant determined that GDC 44, 45, and 46 were not applicable for the PSWS. In RAI 9.2-7, the staff questioned this determination. In DCD Tier 2, Revision 3, Section 9.2.1.1, the applicant demonstrated 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.

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 do not apply to the PSWS. The staff agrees that the design of the PSWS satisfies GDC 44, 45, and 46 based on the above design provisions. Therefore, RAI 9.2-7 as related to PSWS is resolved.

In RAI 9.2-8, 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. In its response, 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; radioactive leakage is highly unlikely; the PSWS design includes provisions for grab sampling; and the COL holder will make provisions for sampling cooling tower blowdown as referenced in DCD Tier 2, Table 11.5-5.

The staff did not find the responses acceptable for the following reasons. First, the radiation monitoring and sampling provisions provided in DCD Tier 2, Table 11.5-5, for the PSWS are not consistent with Table 2 of SRP Section 11.5, which indicates provisions for continuous radiation monitoring/sampling for the service water system. Secondly, DCD Tier 2 does not identify the COL holder item. In RAI 9.2-8 S01, the staff indicated that the radiation monitoring and sampling provisions in the DCD are not consistent with SRP Section 11.5 with respect to the continuous radiation monitoring/sampling for the service water system. In its response, the applicant stated that the DCD requires continuous effluent monitoring, either directly on the effluent of PSWS or on another downstream process effluent. The applicant also discussed the reasons that radioactive leakage into the PSWS is highly unlikely. Furthermore, the response states that the PSWS design includes provisions for obtaining a grab sample in the event that there is a RCCWS radiation monitor alarm, and it indicates that the COL holder will also make provisions for monitoring, sampling, or analyzing the cooling tower blowdown to ensure monitoring before release to the environment.

The staff is unable to confirm in the DCD markups for Sections 9.2.4 and 11.5.7.3 that the DCD requires continuous radiation monitoring either directly on the effluent of the PSWS or on another downstream process effluent. The staff requested that the applicant clarify where the DCD states that it requires continuous radiation monitoring. **RAI 9.2-8 is being tracked as an Open Item.**

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 3, Section 9.2.1.2, the applicant stated that the COL applicant would determine material selection and make provisions to preclude long-term corrosion and fouling of the PSWS based on site water quality analysis. The staff confirmed that DCD Tier 2, Revision 3, Section 9.2.1.6, identifies this **COL Action Item.** Therefore, RAI 9.2-9 is resolved.

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. The staff reviewed DCD Tier 2, Revision 3, Section 9.2.1.5, and finds it acceptable. Therefore, RAI 9.2-10 is resolved.

In RAI 9.2-11, 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. 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.

Furthermore, DCD Tier 2, Revision 3, Section 9.2.1, states that the PSWS meets GDC 4 with respect to water hammer.

The staff finds the above responses acceptable. However, the applicant has not identified a COL applicant item in the DCD to address the procedures discussed above. The staff also reviewed DCD Section 13.5.3, a COL information item for plant operating procedures. 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. 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 may not cover the non-safety-related systems such as PSWS and RCCWS in the ESBWR. If GEH decides to refer to the generic COL information in DCD Section 13.5.3 as the resolution to RAI 9.2-11, it will need to clarify or modify DCD Section 13.5.3.4 to ensure the general plant operating procedures will include the PSWS and RCCWS. **RAI 9.2-11 is being tracked as an Open Item.**

A non-safety-related system has 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 regarding non-safety-related systems. In RAI 9.2-12, 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 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. Chapter 22 of this report provides the staff evaluation of the RTNSS systems and the associated requirements. The staff reviewed the above RAI responses and DCD Tier 2, Revision 3, 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 non-safety-related systems because the failure of the non-safety-related portions of the systems does not impact any safety-related SSCs. The portions of RAI 9.2-12 relating to the PSWS are resolved.

DCD Tier 2, Revision 3, Section 9.2.1, states that the PSWS has RTNSS functions to provide post-72-hour cooling to the RCCWS and TCCWS. Performance of RTNSS functions is ensured by applying requirements for redundant trains, physical and electrical separation of trains, seismic requirements, ability to withstand Category 5 hurricane missiles, and flood protection. Appendix 19A to DCD Tier 2, Revision 3, provides the level of oversight and additional requirements. Chapter 22 of this report documents the review of RTNSS.

In RAI 14.3-69, the staff requested that the applicant provide more detail in DCD Tier 1, Section 2.12.7, for the PSWS, ITAAC to include a system description and system drawing, design commitment, and ITAAC scope. 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 3, Section 2.12.7, GEH provided details as requested in RAI 14.3-69. Therefore, RAI 14.3-69 is resolved.

DCD Tier 2, Revision 3, 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.

Since the PSWS is not a safety-related system, the staff agrees with the applicant's determination that the above generic issues are not applicable to the ESBWR.

DCD Tier 2, Table 1C-2, identifies Inspection and Enforcement (IE) Bulletin 81-03, "Flow Blockage of Cooling Water to Safety System Components by Corbicula Sp. (Asiatic Clam) and Mytilus Sp. (Mussel)," dated April 10, 1981, as not applicable. Even though the PSWS is not safety-related, the staff believes the issue needs to be addressed. The design of the PSWS has a self-cleaning duplex strainer. Pressure drop across the strainer is locally indicated, and a high-pressure drop is annunciated in the control room. In addition, the staff asked the applicant in RAI 9.2-9 about the measures to preclude long-term corrosion and organic fouling that would degrade PSWS performance. The applicant stated that a COL applicant will make provisions to preclude long-term corrosion and fouling of the PSWS. DCD Tier 2, Section 9.2.1.6, identifies this as a **COL Action Item.** Based on the above information, the staff finds that the ESBWR applicant has adequately addressed IE Bulletin 81-03.

In DCD Tier 2, Table 1.11-1, the applicant identified Task Action Item A-1, "Water Hammer," as being addressed in meeting the requirements of several SRP sections. SRP Section 9.2.1 is among the sections discussed. The summary of RAI 9.2-11 above discusses the staff evaluation of the water hammer issue as related to the PSWS. RAI 9.2-11 is not resolved; therefore, Task Action Item A-1 is not resolved. **RAI 9.2-11 is being tracked as an Open Item.**

In DCD Tier 1, Revision 3, Section 4.4, the applicant stated that the PSWS cooling towers and basins are not within the scope of the certified design. A specific design for this portion of the PSWS must be selected for any facility that has adopted the certified design. The plant-specific portion of the PSWS must meet the interface requirements defined in DCD Tier 1, Section 4.4. Verification of compliance with the interface requirements is achieved by inspections, tests, and analyses that are similar to those provided for the certified design. The COL applicant referencing the certified design should 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 Item.**

In DCD Tier 1, Revision 3, Section 4.5, 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 4.5. Verification of compliance with the interface requirements is achieved by inspections, tests, and analyses that are similar to those provided for the certified design. The COL applicant referencing the certified design should 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 Item.**

9.2.1.4 Conclusion

Because of the open items that remain to be resolved for this section, the staff was unable to finalize its conclusions regarding the compliance of the PSWS with the requirements of GDC 2, 4, 44, 45, and 46, and 62, "Prevention of Criticality in Fuel Storage and Handling."

9.2.2 Reactor Component Cooling Water System

9.2.2.1 Regulatory Criteria

The staff reviewed the RCCWS in accordance with SRP Section 9.2.2, Revision 3, "Reactor Auxiliary Cooling Water System," issued June 1986. Staff acceptance of the design is based on meeting the requirements of GDC 2, 4, 5, 44, 45, and 46:

- GDC 2, as it relates to the capability of the design to maintain and perform its safety function following an earthquake.
- GDC 4, as it relates to the dynamic effects associated with water hammer.
- GDC 5, as it relates to the capability of shared systems and components important to safety to perform required safety functions.
- GDC 44, as it relates to transferring heat from SSCs important to safety to a heat sink.
- GDC 45, as it relates to the design provisions to permit inspection of components and equipment.
- GDC 46, as it relates to the design provisions to permit operational testing of components and equipment.

9.2.2.2 Summary of Technical Information

In DCD Tier 2, Section 9.2.2, "Reactor Component Cooling Water System," GEH described the RCCWS. The system is designed to provide cooling water to plant auxiliary equipment during normal power operation, plant cooldown, and shutdown operations. The RCCWS consists of two 100-percent-capacity independent and redundant trains. RCCWS cooling water is continuously circulated through various auxiliary equipment heat exchangers and rejects the heat to the PSWS. DCD Tier 2, Table 3.2.1, indicates that the RCCWS is a non-safety-related system located in the RB and TB. The RCCWS has RTNSS functions to provide post-72-hour

cooling to chillers and diesel generators. It is designated as quality group "D" and seismic Category "NS."

In the event of a LOPP, the RCCWS supports the FAPCS and the reactor water cleanup/shutdown cooling system 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 and 9.2-4 of DCD Tier 2 tabulate the RCCWS design heat loads and component design characteristics.

9.2.2.3 Staff Evaluation

The staff reviewed the design of the RCCWS in accordance with SRP Section 9.2.2. However, the RCCWS for the ESBWR differs from that of the traditional BWR designs in that the ESBWR RCCWS is a non-safety-related system because the RCCWS 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 RCCWS.

As for the non-safety-related RCCWS's meeting the requirements of GDC 2 as they relate to structures and system's being capable of withstanding the effects of natural phenomena, acceptance depends on meeting the guidance of the portions of Regulatory Position C.2 of RG 1.29 regarding non-safety-related systems. Regulatory Position C.1 of RG 1.29 for safety-related portions is not applicable to the RCCWS. The ESBWR design is a single-unit station, and the requirements of GDC 5 are not applicable for the single-unit design. Conformance to GDC 4 is based on the fact that the RCCWS is designed to accommodate the dynamic effects associated with water hammer. Conformance to GDC 44, 45, and 46 is based on the fact that the design of the RCCWS is capable of removing heat from SSCs important to safety to a heat sink and that design provisions are available for inspection and operational testing.

The staff asked the applicant to provide additional information in RAIs 9.2-6, 9.2-11, 9.2-12, and 9.2-13. The staff reviewed the applicant's responses and its evaluation of the responses is discussed below.

In RAI 9.2-6, the staff requested that the applicant include P&IDs for the PSWS and RCCWS in the DCD showing system function, major equipment, components, piping classes, instrumentation, and interface systems. As discussed in Section 9.2.1.3 of this report, the GEH response is not acceptable and RAI 9.2-6 is not resolved. **RAI 9.2-6 is being tracked as an Open Item.**

In DCD Tier 2, Revision 3, Section 9.2.2.1, the applicant demonstrated that RCCWS satisfied 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.

In a design-basis accident, heat is transferred to the IC/PCC pools. The portions of the GDC 44 requirements that apply to a safety-related system do not apply to the RCCWS. The staff agrees that the design of the RCCWS satisfies GDC 44, 45, and 46 based on the above design provisions.

In RAI 9.2-11, 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. 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.

DCD Tier 2, Revision 3, Section 9.2.2, states that the RCCWS meets GDC 4 with respect to water hammer. The staff finds the above responses acceptable. However, the applicant has not identified a COL applicant item in the DCD to address the procedures discussed above. Furthermore, the applicant has not included the last two provisions (slow valve actuation time and use of check valves) in DCD Tier 2, Section 9.2.2. **RAI 9.2-11 is being tracked as an Open Item.**

A non-safety-related system has to meet the requirements of GDC 2 relating to structures and systems being capable of withstanding the effects of natural phenomena. SRP Section 9.2.2 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. In RAI 9.2-12, 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. Chapter 22 of this report provides the staff evaluation of the RTNSS systems and the associated requirements. The staff reviewed the above RAI responses and DCD Tier 2, Revision 3, 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 systems does not impact any safety-related SSCs. The portions of RAI 9.2-12 relating to the RCCWS are resolved.

In RAI 9.2-13, 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. In its response, the applicant stated that intersystem leakage in the RCCWS is monitored through four methods. The discussion of the third and forth methods is not clear. In RAI 9.2-13 S01, the staff asked the applicant to clarify its third and fourth methods. In its response, the applicant intended to restate RAI 9.2-13 and its initial response but misused the content from RAI 9.2-9 instead of RAI 9.2-13. The staff asked the applicant to clarify whether the grab sampling method is performed following a high-level alarm from the surge tank. **RAI 9.2-13 is being tracked as an Open Item.**

DCD Tier 2, Revision 3, Section 9.2.2.1, states that the RCCWS has RTNSS functions to provide post-72-hour cooling to the nuclear island chillers and diesel generators. Performance of RTNSS functions is ensured by applying requirements for redundant trains, physical and electrical separation of trains, seismic requirements, ability to withstand Category 5 hurricane missiles, and flood protection. Appendix 19A to DCD Tier 2, Revision 3, provides the level of oversight and additional requirements. Chapter 22 of this report documents the review of RTNSS.

In DCD Tier 1, Revision 3, Section 2.12.3, the applicant revised the RCCWS ITAAC to remove the system description and system drawings, design commitment, and scope of ITAAC. DCD Tier 2, Section 14.3.7.3, indicates that RTNSS systems shall have Tier 1 inputs that include design descriptions and ITAAC. The staff finds that the removal of RCCWS ITAAC in Tier 1 is not consistent with Tier 2, Section 14.3.7.3. This revision is not acceptable. In RAI 22.5-1, the staff requested that the applicant review and revise DCD Tier 1 to consistently include all RTNSS systems in Tier 1 for ITAAC. **RAI 22.5-1 is being tracked as an Open Item.**

9.2.2.4 Conclusion

Because of the open items that remain to be resolved for this section, the staff was unable to finalize its conclusions regarding the compliance of the RCCWS with the requirements of GDC 2, 4, 44, 45, and 46.

9.2.3 Makeup Water System

9.2.3.1 Regulatory Criteria

The staff reviewed the makeup water system (MWS) in accordance with 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 GDC 2 and 5:

- GDC 2, as it relates to the capability of the design to maintain and perform its safety function following an earthquake.
- GDC 5, as it relates to the capability of shared systems and components important to safety to perform required safety functions.

9.2.3.2 Summary of Technical Information

In DCD Tier 2, Section 9.2.3, “Makeup Water System,” GEH 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 reference design that is dependent on the site-specific water quality of the available source water. 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. Based on local weather conditions, freeze protection is provided for the demineralized water storage tank and piping exposed to freezing conditions. 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 flowpath of the storage and transfer subsystem of the MWS is from the MWS demineralized water storage tank, through an 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. The MWS provides water for condensate storage tank (CST) makeup; standby liquid control system makeup; liquid waste system chemical addition and line flushing; solid waste system for line flushing; RCCWS makeup; TCCWS makeup; CWS makeup; process sampling system process use; auxiliary boiler system makeup; postaccident sampling station flushing; heating, ventilation, and air-conditioning (HVAC) makeup; and IC/PCC pool normal makeup water.

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 non-safety-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 non-safety-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.

A non-safety-related system has to meet the requirements of GDC 2 relating to structures and systems being capable of withstanding the effects of natural phenomena. SRP Section 9.2.3 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. 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.

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 3, 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. The portions of RAI 9.2-12 relating to the MWS are resolved, and the MWS meets the requirements of GDC 2.

DCD Tier 2, Table 9.2-6, states that the MWS is designed to provide makeup water for the RCCWS, TCCWS, and CWS. The response to RAI 14.3-69 identifies these systems as RTNSS systems. The staff requested in RAI 22.5-19 that the applicant clarify whether the makeup to the RCCWS and TCCWS provided by the MWS is required to satisfy RTNSS selection criterion B. The staff also requested that, if the MWS is selected for RTNSS, the applicant specify the corresponding requirements, including ITAAC. **RAI 22.5-19 is being tracked as an Open Item.**

In DCD Tier 1, Revision 3, Section 4.6, the applicant stated that the plant-specific MWS demineralizer subsystem will meet the interface requirements defined in DCD Tier 1, Section 4.6. Verification of compliance with the interface requirements will be achieved by inspections, tests, and analyses that are similar to those provided for the certified design. The COL applicant referencing the certified design will develop these inspections, tests, and analyses, together with their associated acceptance criteria. The staff reviewed and agrees that it is a **COL Interface Item.**

9.2.3.4 Conclusion

Because of the open items that remain to be resolved for this section, the staff was unable to finalize its conclusions regarding the compliance of the MWS with the requirements of GDC 2 and 5.

9.2.4 Potable and Sanitary Water Systems

DCD Tier 1, Section 4.3, states that the plant-specific potable and sanitary water system will not be connected to any SSCs or any systems that could contain radioactive material. This is a **COL Interface Item.**

DCD Tier 2, Section 9.2.4, "Potable and Sanitary Water Systems," states that the COL applicant will provide the design of the potable and sanitary systems:

> The COL applicant will provide the design of the potable and sanitary water system that provides sufficient supply and is designed to provide a minimum of 12.6 l/s (200 gpm) of potable water during peak demand periods.
>
> The COL applicant will provide design of wastewater effluent systems that properly disposes of sanitation wastes.

These are **COL Action Items.** Therefore, the staff will review these systems during the review of the COL application.

9.2.5 Ultimate Heat Sink

9.2.5.1 Regulatory Criteria

The staff reviewed the ultimate heat sink (UHS) in accordance with SRP Section 9.2.5, Revision 2, "Ultimate Heat Sink," issued July 1981. Staff acceptance of the design is based on meeting the requirements of GDC 2, 5, 44, 45, and 46:

- GDC 2, as it relates to the capability of the design to maintain and perform its safety function following an earthquake.
- GDC 5, as it relates to the capability of shared systems and components important to safety to perform required safety functions.
- GDC 44, as it relates to transferring heat from SSCs important to safety to a heat sink.
- GDC 45, as it relates to the design provisions to permit inspection of components and equipment.
- GDC 46, as it relates to the design provisions to permit operational testing of components and equipment.

9.2.5.2 Summary of Technical Information

DCD Section 9.2.5, "Ultimate Heat Sink," describes the UHS. 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. DCD Tier 2, Section 5.4.6, discusses the isolation condenser system (ICS). DCD Tier 2, Section 6.2.2, discusses the passive containment cooling system (PCCS).

9.2.5.3 Staff Evaluation

The staff reviewed the design of the UHS in accordance with SRP Section 9.2.5. Staff acceptance of the UHS is based on meeting the requirements of GDC 2, 5, 44, 45, and 46.

In DCD Tier 2, Revision 3, 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.

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

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 FPS provides postaccident 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 RB and 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. The 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, Section 9.2.5.1, identifies this **COL Action Item.** The staff will review this information during the COL application process.

DCD Tier 1, Revision 3, Section 4.1, identifies an interface requirement that the UHS external water source provide a backup source of water, in sufficient quantity and flowrate, for IC/PCC pool makeup and reactor water inventory control through a connection to the FAPCS. The applicant stated that providing site-specific SSCs that are technologically similar to the SSCs for the certified ESBWR design will satisfy the interface requirements defined in DCD Tier 1, Section 4.1. Verification of compliance with the interface requirements is achieved by inspections, tests, and analyses that are similar to those provided for the certified design. These inspections, tests, and analyses, together with their associated acceptance criteria, will be developed on a site-specific basis. **This is a COL Interface Item.**

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.

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) in accordance with SRP Section 9.2.6, Revision 2, "Condensate Storage Facilities," issued July 1981. Staff acceptance of the design is based on meeting the requirements of GDC 2, 5, 44, 45, and 46:

- GDC 2, as it relates to the capability of the design to maintain and perform its safety function following an earthquake.
- GDC 5, as it relates to the capability of shared systems and components important to safety to perform required safety functions.
- GDC 44, as it relates to transferring heat from SSCs important to safety to a heat sink.
- GDC 45, as it relates to the design provisions to permit inspection of components and equipment.
- GDC 46, as it relates to the design provisions to permit operational testing of components and equipment.

9.2.6.2 Summary of Technical Information

In DCD Tier 2, Section 9.2.6, "Condensate Storage and Transfer System," GEH described 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 CST.
- Provide a distribution system to supply condensate-quality water to equipment.

- Provide one transfer pump sized to meet the requirements for filling the condensate and feedwater system in approximately 16 hours, plus recirculation of a minimum of 25 percent of the pump capacity.

- 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 Title 10, Part 20, "Standards for Protection Against Radiation," of the *Code of Federal Regulations* (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 RPV makeup following a nuclear steam supply system isolation event. The CS&TS equipment and associated piping are fabricated from stainless steel to prevent contamination of the system water.

The CST is the normal source of water for makeup to selected plant systems. The condensate transfer pumps take their suction from the CST and provide makeup water for various services in the RB, TB, 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, and 46.

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

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. A non-safety-related system has to meet the requirements of GDC 2, relating to structures and systems being capable of withstanding the effects of natural phenomena. SRP Section 9.2.6 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. 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 3, 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.

In DCD Tier 2, Revision 3, Section 9.2.6.1, the applicant stated that GDC 44 is not applicable to the CS&TS. The staff agrees with the applicant because the CS&TS does not have the function to provide makeup water to safety-related cooling systems. The CS&TS meets the requirements of GDC 45 and 46 by providing design provisions to permit inspection and operational testing.

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, 45, and 46.

9.2.7 Chilled Water System

9.2.7.1 Regulatory Criteria

The staff reviewed the CWS in accordance with the applicable portions of SRP Section 9.2.2. Staff acceptance of the design is based on meeting the requirements of GDC 2, 4, 5, 44, 45, and 46:

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

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

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

9.2.7.2 Summary of Technical Information

In DCD Tier 2, Section 9.2.7, "Chilled Water System," GEH describes the CWS. The CWS consists of the nuclear island chilled water subsystem (NICWS) and the balance-of-plant chilled water subsystem. The CWS provides chilled water to the cooling coils of air-conditioning units and other coolers in the RB; the TB; the control, radwaste, service, electrical, and FBs; the technical support center; and the hot machine shop. Water in the CWS is pumped through chiller-evaporators where the heat is transferred to the RCCWS and the TCCWS. The CWS consists of two identical chiller units of at least 100 percent of NICWS capacity. DCD Tier 2, Table 9.2-11, lists the CWS component design characteristics. DCD Tier 2, Figure 9.2-3, shows the CWS simplified diagram.

DCD Tier 2, Section 9.2.7.1, states that, except for the containment isolation valves, the CWS does not perform any safety-related function.

9.2.7.3 Staff Evaluation

The staff reviewed the CWS in accordance with the applicable portions of SRP Section 9.2.2. The ESBWR design is a single-unit station, and the requirements of GDC 5 are not applicable for the single-unit design. The chilled water absorbs the rejected heat from coolers and is pumped through the chiller-evaporators, where the heat is transferred to the RCCWS and TCCWS. GDC 44, 45, and 46 are not applicable because the CWS does not transfer heat from any safety-related SSCs.

Section 6.2.4 of this report evaluates containment isolation valves. A non-safety-related system has to meet the requirements of GDC 2, relating to structures and system's being capable of withstanding the effects of natural phenomena. SRP Section 9.2.2 indicates that the acceptance depends on meeting the guidance of the portions of Regulatory Position C.2 of RG 1.29 regarding non-safety-related systems.

The staff asked the applicant to provide additional information in RAI 9.2-12. 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 CWS meets GDC 2. In its response, the applicant stated that the CWS 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 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 3, 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 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, and therefore the CWS satisfies GDC 2. RAI 9.2-12 relating to the CWS is resolved.

DCD Tier 2, Revision 3, Section 9.2.7.1, states that the CWS has RTNSS functions to provide post-72-hour cooling for HVAC. Performance of RTNSS functions is ensured by applying requirements for redundant trains, physical and electrical separation of trains, seismic requirements, ability to withstand Category 5 hurricane missiles, and flood protection. Appendix 19A to DCD Tier 2, Revision 3, provides the level of oversight and additional requirements. Chapter 22 of this report documents the review of RTNSS.

The response to RAI 14.3-69 identifies the CWS as an RTNSS system. Electrical power is assumed to be unavailable for 72 hours and then returned to service for RTNSS systems. Restarting the CWS presents an opportunity for 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. **RAI 9.2-15 is being tracked as an Open Item.**

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. DCD Tier 2, Section 14.3.7.3, indicates that RTNSS systems will have Tier 1 inputs that include design descriptions and ITAAC. The staff finds that the removal of CWS ITAAC information in Tier 1 is not consistent with Tier 2, Section 14.3.7.3. This revision of Tier 1 information is not acceptable. In RAI 22.5-1 the staff requested that the applicant review and revise DCD Tier 1 to consistently include all RTNSS systems in Tier 1 for ITAAC. **RAI 22.5-1 is being tracked as an Open Item.**

DCD Tier 2, Revision 3, Chapter 1, discusses the following generic issues related to CWS:

- IE Bulletin 80-24, "Prevention of Damage due to Water Leakage Inside Containment (October 17, 1980 Indian Point Event," dated November 21, 1980, is identified in DCD Tier 2, Table 1C-2. The applicant stated in the DCD that it is not applicable for the ESBWR because the CWS is a closed-loop system. The staff agrees with the applicant on its determination.

- New Generic Issue 143, "Availability of Chilled Water System and Room Cooling," is identified in DCD Tier 2, Table 1.11-1. The applicant stated in the DCD that the failure of CWS does not compromise any safety-related system or component, nor does it prevent a safe shutdown of the plant. The staff finds that the CWS has RTNSS functions to provide post-72-hour cooling for HVAC. The proper level of oversight and additional requirements described in Appendix 19A to DCD Tier 2, Revision 3, are adequate to address this issue.

9.2.7.4 Conclusion

Because of the open items that remain to be resolved for this section, the staff was unable to finalize its conclusions regarding the compliance of the CWS with the requirements of GDC 2, 4, 44, 45, and 46.

9.2.8 Turbine Component Cooling Water System

9.2.8.1 Regulatory Criteria

The staff reviewed the TCCWS in accordance with SRP Section 9.2.2. Staff acceptance of the design is based on meeting the requirements of GDC 2, 4, 5, 44, 45, and 46:

- GDC 2, as it relates to the capability of the design to maintain and perform its safety function following an earthquake.
- GDC 4, as it relates to the dynamic effects associated with water hammer.
- GDC 5, as it relates to the capability of shared systems and components important to safety to perform required safety functions.
- GDC 44, as it relates to transferring heat from SSCs important to safety to a heat sink.
- GDC 45, as it relates to the design provisions to permit inspection of components and equipment.
- GDC 46, as it relates to the design provisions to permit operational testing of components and equipment.

9.2.8.2 Summary of Technical Information

In DCD Tier 2, Section 9.2.8, "Turbine Component Cooling Water System," GEH described the TCCWS. The TCCWS is a single-loop system and consists of one surge tank, one chemical addition tank, two pumps, two 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.

During normal power operation, one of the two 100-percent-capacity TCCWS pumps circulates corrosion-inhibited demineralized water through one side of one of the two 100-percent-capacity 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.

A non-safety-related system has to meet the requirements of GDC 2, relating to structures and systems' being capable of withstanding the effects of natural phenomena. SRP Section 9.2.2 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.

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

Conformance to GDC 44, 45, and 46 is based on the fact that the design of TCCWS is capable of removing heat from SSCs important to safety to a heat sink and that design provisions are available for inspection and operational testing.

As a part of RAI 9.2-12, the staff asked the applicant to demonstrate that the TCCWS meets GDC 2. In its response, the applicant stated that the TCCWS does not have any piping in the control room or interface with any safety-related components. Failure of the TCCWS will not compromise any safety-related system or component, nor will it prevent a safe shutdown. The response to RAI 19.1.0-2 identified the TCCWS as the RTNSS system to provide post-72-hour cooling functions to the TB HVAC. It will be designed to seismic requirements to be specified in Appendix 19A to DCD Tier 2. Chapter 22 of this report provides the staff evaluation of the RTNSS systems and the associated requirements. The staff reviewed the above RAI responses and DCD Tier 2, Revision 3, Section 9.2.8 and Table 3.2-1. Based on the above, the staff finds that the TCCWS 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 system does not impact any safety-related SSCs, and therefore the TCCWS satisfies GDC 2. RAI 9.2-12 relating to the TCCWS is resolved.

In DCD Tier 2, Revision 1, Table 1.9-9 and Section 9.2.8, the applicant determined that GDC 44, 45, and 46 were not applicable for the TCCWS. In DCD Tier 2, Revision 3, Section 9.2.8.1, the applicant revised its position and demonstrated that the TCCWS satisfies the above criteria because the design of the TCCWS includes the following provisions:

- capability to transfer heat loads from SSCs to a heat sink under normal and accident conditions.
- component redundancy so the system will remain functional assuming a single failure coincident with a 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.

In a design-basis accident, heat is transferred to the IC/PCC pools through the PCCS. The portions of GDC 44 requirements that apply to a safety-related system do not apply to the TCCWS. The staff finds that the design of the TCCWS satisfies GDC 44, 45, and 46 based on the above design provisions.

In DCD Tier 2, Revision 3, the applicant identified the TCCWS as an RTNSS system. GEH stated in its response to RAI 9.2-12 S01 that after a reevaluation of the RTNSS, GEH determined that the TCCWS is not an RTNSS system because it does not remove heat from any safety-related systems or from other RTNSS systems. GEH will update this information in DCD Tier 2, Revision 4. The applicant stated that the TCCWS is no longer an RTNSS system since the TB HVAC is now cooled by the CWS. The applicant identified the CWS as an RTNSS system. The staff requested clarification since the CWS is composed of two independent systems (the nuclear island and the balance of plant), which are in turn cooled by two separate closed cooling systems, the RCCWS and the TCCWS, respectively. The RCCWS and TCCWS are designed to cool independent portions of the TB HVAC, as described in DCD Tier 2, Section 9.2.7. The applicant stated that Figure 20.4-1 in NEDO-33201, Revision 2, "Probabilistic Risk Assessment," shows that the RTNSS post-accident monitoring function cooled by the TB HVAC is cooled by the nuclear island CWS and RCCWS. Therefore, the TCCWS is not required for post-72-hour cooling, and, accordingly, the TCCWS is not an

RTNSS system. Because the TCCWS is not a safety-related system or an RTNSS system, GDC 4 is not applicable. Therefore, RAI 9.2-12 S01 is resolved.

9.2.8.4 Conclusion

Based on the above discussion, the staff finds the design of the TCCWS acceptable.

9.2.9 Hot Water System

9.2.9.1 Regulatory Criteria

No SRP guidelines are directly applicable to the review of the hot water system (HWS); therefore, the staff based its review of the HWS on relevant regulatory guidance and industry standards.

9.2.9.2 Summary of Technical Information

In DCD Tier 2, Section 9.2.9, "Hot Water System," GEH described the HWS. The HWS is a closed system that provides heating water to hot water coils in the air-handling units and fan coil units of the HVAC systems for building heating during cold weather operation or during cold weather plant outages. During cold weather plant operation or during cold weather plant outages, the HWS maintains acceptable design ambient air temperatures in various areas throughout the ESBWR.

The design of the HWS for the ESBWR standard plant is based on bounding ambient environmental conditions as defined in DCD Tier 2, Chapter 2. The HWS consists of two 100-percent-capacity heat exchangers, two 100-percent-capacity pumps, an expansion tank, a chemical addition tank, piping, valves, and instrumentation. Water is heated in the heat exchangers using steam from the auxiliary boiler system. The heated water is pumped in a closed-loop system to hot water coils in the HVAC systems. During cold weather plant operation or during cold weather plant outages, the HWS supplies hot water throughout the plant to protect equipment from freezing and for personnel comfort. The heat exchanger is located in the TB.

9.2.9.3 Staff Evaluation

In DCD Tier 2, Section 9.2.9.3, the applicant stated that the HWS has no safety design basis and does not perform any safety-related function. Failure of the HWS does not affect any safety-related systems or components.

Because the HWS does not transfer heat from SSCs important to safety, the staff concludes that the requirements of GDC 44, 45, and 46 do not apply to the HWS.

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

According to the description of the HWS in DCD Tier 2, Section 9.2.9.2 and Table 3.2-1, the HWS is a non-safety system extending thoughout the plant, including the containment, without being seismically qualified. Following a seismic event, a failure of the system components and piping could be assumed because the system is non-seismic. In RAI 9.2-14, the staff requested that the applicant clarify the following aspects of the system:

- whether the system piping penetrates the containment and meets the requirements of the containment penetration and isolation.

- whether the portions of the system inside the containment have proper seismic design .

- how the system meets GDC 2 as related to Regulatory Positions C.1 and C.2 of RG 1.29.

RAI 9.2-14 is being tracked as an Open Item.

9.2.9.4 Conclusion

Because of the open item that remains to be resolved for this section, the staff was unable to finalize its conclusions regarding the compliance of the HWS with the requirements of GDC 2, 4, 44, 45, and 46.

9.2.10 Station Water System

9.2.10.1 Regulatory Criteria

No SRP guidelines are directly applicable to the review of the station water system; therefore, the staff based its review of the system on relevant regulatory guidance and industry standards.

9.2.10.2 Summary of Technical Information

In DCD Tier 2, Section 9.2.10, "Station Water System," GEH described 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 PSWS cooling tower basins
- feedwater to the MWS
- fill water to the FPS

The station water system consists of the following subsystems:

- plant cooling tower makeup system
- pretreated water supply system

The plant cooling tower makeup system provides makeup water to the cooling tower basins for both the PSWS and the CIRC. The supply of water makes up for losses resulting from evaporation, drift, and blowdown from the cooling towers. In addition, the plant cooling tower makeup system provides makeup water to replace water used for PSWS strainer backwash.

The pretreated water supply system filters and chemically pretreats water supplied to the MWS for further treatment for use as demineralized water. The pretreated water supply system also supplies water to the FPS for filling the primary firewater tanks and for maintaining pressure in the yard loop. In addition, the pretreated water supply system provides PSWS cooling tower makeup as an alternate to the plant cooling tower makeup system.

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 staff concludes that GDC 2, 5, 44, 45, and 46 are not applicable for the station water system.

In DCD Tier 2, Section 9.2.10.6, the applicant identified as a **COL Action Item** that the COL applicant will provide the design of the station water system, including the following:

- components and design capacities for each subsystem,
- a simplified diagram of the station water system, showing the plant cooling tower makeup system and the pretreated water supply system.

9.2.10.4 Conclusion

Based on the above COL action item, the staff concludes that the design of the station water system is acceptable.

9.3 Process Auxiliaries

9.3.1 Compressed Air System

9.3.1.1 Regulatory Criteria

The staff of the NRC reviewed the compressed air system for the ESBWR in accordance with the guidance of NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants" (hereafter referred to as the SRP), Section 9.3.1, Revision 2, "Compressed Air Systems," issued July 1981. The staff's acceptance of the compressed air system is based on the conformance of the design to the requirements of the following GDC:

- 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.
- 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.

Compliance with GDC 2 is based on meeting the guidance of Regulatory Positions C.1 and C.2 of RG 1.29, "Seismic Design Classification."

9.3.1.2 Summary of Technical Information

The compressed air system 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 3, Sections 9.3.6, 9.3.7, 9.3.8, and 9.4.9, 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 compressed air system, and issued RAIs concerning the specific 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 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 response GEH 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, GEH also stated that safety-related and non-safety-related separation is at the accumulator check valve.

Based on its review, the staff finds the GEH 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. Therefore, the staff's concern described in 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, GEH referred to the schematic diagram provided in the response to RAI 9.3-33. The staff finds the GEH 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. Therefore, the staff's concern described in RAI 9.3-34 is resolved.

RAI 9.3-35

In RAI 9.3-35, the staff stated the following:

> DCD Section 9.3 states that the accumulators and valves associated with the main steam isolation, automatic depressurization, and isolation condenser isolation valves are part of the respective systems. However, the DCD sections describing those systems do not include drawings or detailed descriptions regarding the safety-related pressurized gas supplies for operation of those valves. Describe how the piping, valves and pressure vessels that provide the essential pneumatic pressure for operation of safety-related valves are protected against dynamic effects associated with design basis accidents such that, concurrent with a postulated single active failure, the necessary number of safety-related valves actuate to the correct position.

In 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, GEH referred to DCD Tier 2, Section 3.6, which addresses the protection provided for safety-related SSCs against dynamic effects associated with design-basis accidents.

Section 3.6 of this report addresses the staff's evaluation of the protection provided for safety-related SSCs and systems subject to the RTNSS against the dynamic effects associated with the postulated rupture of piping.

With regard to the effects of a postulated single active failure of safety-related valves, GEH stated that single active failures are addressed through redundant modes of actuation. MSIVs will close by spring actuation and/or pneumatic actuation. Safety relief valves (SRVs) are lifted open by pneumatic actuation or lifted by steam pressure at a prescribed setpoint. For isolation condenser system (ICS) isolation valves, two pairs of valves are inside containment (one on the steam inlet and one on the condensate return lines), and each valve within a pair has a different mode of actuation from the other valve. For example, one is a nitrogen rotary-motor-operated valve and the other is a nitrogen piston-operated valve.

The staff finds the GEH response to RAI 9.3-35 acceptable because GEH has described how the valves cited above are operated with diversity of driving forces and thus will mitigate the effects resulting from a postulated single active failure of safety-related valves. Therefore, the staff's concern described in RAI 9.3-35 is resolved.

Section 9.3.6 of this report addresses the staff's evaluation of the IAS.

Section 9.3.7 of this report addresses the staff's evaluation of the SAS.

Section 9.3.8 of this report addresses the staff's evaluation of the HPNSS.

The CIS is intended to provide an inerted containment in compliance with Title 10, Section 50.44(c)(2), of the *Code of Federal Regulations* (10 CFR 50.44(c)(2)). Section 6.2.5 of this report addresses the staff's evaluation of compliance with 10 CFR 50.44(c)(2).

9.3.1.4 Conclusion

The staff's conclusions for each of the subsystems of the compressed air system appear in the respective subsections of the SER.

9.3.2 Process Sampling System

9.3.2.1 Regulatory Criteria

The staff reviewed the process sampling system (PSS) description in accordance with SRP Section 9.3.2, Draft Revision 3, "Process and Post-Accident Sampling Systems," issued April 1996. Staff acceptance of the design is based on the conformance of the design with the following requirements:

- GDC 13, "Instrumentation and Control," as it relates to the monitoring variables that can affect the fission process, the integrity of the reactor core, and the reactor coolant pressure boundary (RCPB).
- GDC 14, "Reactor Coolant Pressure Boundary," as it relates to ensuring RCPB integrity.
- GDC 60, as it relates to the capability of the PSS to control the release of radioactive materials to the environment.
- GDC 63, as it relates to detecting conditions that may result in excessive radiation levels in the fuel storage and radioactive waste systems.
- GDC 64, as it relates to monitoring the containment atmosphere and plant environs for radioactivity.
- 10 CFR 20.1101(b), as it relates to providing engineering controls based upon sound radiation protection principles to achieve occupational doses and doses to members of the public as low as is reasonably achievable (ALARA).
- Three Mile Island (TMI) Action Plan Item III.D.1.1 in NUREG-0737, as it relates to the provisions for a leakage control program to minimize the leakage from those portions of the PSS outside of the containment that contain or may contain radioactive material following an accident. 10 CFR 50.34(f)(2)(xxvi) provides equivalent requirements for those applicants subject to 10 CFR 50.34(f).
- Three Mile Island (TMI) Action Plan Item II.B.3 in NUREG-0737, as it relates to the provisions for the capability in promptly obtaining and analyzing samples from the reactor coolant system and containment without exceeding occupational radiation exposure dose limits. 10 CFR 50.34(f)(2)(viii) provides equivalent requirements for those applicants subject to 10 CFR 50.34(f)..

9.3.2.2 Summary of Technical Information

The PSS is designed to collect for analysis representative water and gaseous samples contained in the reactor coolant system (RCS) and associated auxiliary system process streams during all plant operational modes, including transients and postulated accidents. The proposed design includes permanently installed sample lines, sampling panels with analyzers and associated sampling equipment, provisions for local grab sampling, and permanent shielding. 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 PSS can provide information on the following parameters:

- pH
- iron
- silica
- iodine-131
- sulfate
- copper
- sodium
- chloride
- isotopics
- conductivity
- total anions
- gross activity
- dissolved oxygen
- organic impurities
- 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 the grab samples taken at the sampling stations have a chemical fume hood to preclude the exposure of operating personnel to contamination hazards. In addition, all sampling lines contain process isolation block valves.

9.3.2.3 Staff Evaluation

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 has reviewed the capability of the PSS to collect and deliver samples of fluids for analysis from systems needed to address GDC 13 and 14. According to SRP Section 9.3.2, in order to meet GDC 13 and 14, the PSS should obtain samples from the reactor coolant, standby liquid control system (SLCS) tank, and condensate polishing system.

The PSS design includes the following sample stations:

- RB sample station—processes samples from the reactor water cleanup/shutdown cooling system

- local grab sampling stations—processes streams requiring intermittent sampling, including in the SLCS

- condensate polishing sample station—processes samples from the condensate and feedwater system and the condensate purification system

Based on this information, the staff concludes that the PSS design meets the requirements of (1) GDC 13 as it relates to monitoring variables that can affect the fission process, the integrity of the reactor core, and the RCPB, and (2) GDC 14 as it relates to ensuring the integrity of the RCPB.

According to SRP Section 11.5, the sampling system should include engineering provisions for purging sample lines, minimizing process and effluent volumes to waste collection systems, controlling leakage and spillage or radioactive materials, limiting radiation exposures and doses to workers, and avoiding uncontrolled and unmonitored releases to unrestricted areas. The staff has reviewed the capability of the PSS to collect and deliver samples of fluids for analysis from plant systems. The staff concludes that the PSS meets the requirements of GDC 60, 63, and 64 and 10 CFR Part 20.1101(b).

Section 20 of this report addresses the staff's evaluation of TMI Action Plan Items III.D.1.1 and II.B.3.

9.3.2.4 Conclusion

The staff has determined that the proposed PSS meets (1) the requirements of GDC 13 and 14 to monitor variables that can affect the RCPB and to ensure a low probability of abnormal leakage and rapidly propagating failure and gross rupture by sampling the reactor coolant and the condensate, respectively, for chemical impurities that can affect the RCPB; (2) the requirements of GDC 13 to maintain the reactor core subcritical under cold conditions in the event that the control rod system is inoperable, by sampling the SLCS tank for boron concentration; and (3) the system includes engineering provisions for purging sampling lines, minimizing process and effluent volumes to waste collection systems, controlling leakage and spillage or radioactive materials, limiting radiation exposures and doses to workers, and avoiding uncontrolled and unmonitored releases to unrestricted areas.

DCD Tier 2, Revision 3, Section 11.5, Tables 11.5-5 and 11.5-6 identify provisions for sampling liquid and gaseous process and effluent streams. DCD Tier 2, Revision 3, Sections 12.1 and 12.5 describe design and operational considerations for the purpose of maintaining occupational radiation exposures and doses ALARA. DCD Tier 2, Revision 3, Section 12.6 describes facility measures for minimizing radioactive contamination of plant structures, systems, and components.

Based on this information, the staff concludes that the PSS design for the ESBWR is acceptable.

9.3.3 Equipment and Floor Drain System

9.3.3.1 Regulatory Criteria

The staff reviewed the equipment and floor drain system (EFDS) for the ESBWR in accordance with the guidance of SRP Section 9.3.3, Revision 2, "Equipment and Floor Drainage System," July 1981. The staff's acceptance of the EFDS is based on the design in conformance with the requirements of the following GDC:

- GDC 2 in part requires that SSCs important to safety shall be designed to withstand the effects of natural phenomena such as earthquakes.

- GDC 4, "Environmental and Dynamic Effects Design Bases," in part requires that SSCs important to safety shall be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents.

- GDC 60, "Control of Releases of Radioactive Materials," in part requires that the nuclear power plant unit design shall include means to control the release of radioactive materials in gaseous and liquid effluents suitably and to handle radioactive wastes produced during normal reactor operation, including anticipated operational occurrences. Sufficient holdup capacity shall be provided for the retention of gaseous and liquid effluents containing radioactive materials.

Compliance with GDC 2 is based on meeting the guidance of Regulatory Positions C.1 and C.2 of RG 1.29 concerning seismic classification.

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. It does not have any safety-related function other than containment isolation functions. Failure of the EFDS does not prevent any safety-related equipment from performing its safety-related functions. Each EFDS sump pump discharge line that penetrates the containment is provided with redundant valves for containment isolation following a loss-of-coolant accident (LOCA). Section 6.2.4 of this report addresses the staff's evaluation of the containment isolation valves for EFDS pump discharge lines.

The EFDS collects liquid wastes from their point of origin and transfers them 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 3, Subsection 3.4.1, no credit is taken for the EFDS system in the flooding analysis. Section 3.4.1 of this report addresses the staff's evaluation of the effects 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). Each system has its own sumps, pumps, isolation valves, and instrumentation and piping. So that the liquid wastes from various sources can be segregated and processed separately for each

specific type of impurity and chemical content, the RWDS is further divided into the following subsystems:

- 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

The CDS collects liquid wastes in sumps by gravity from the clean non-radioactive equipment and floor drains 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 liquid waste management system (LWMS) for processing and disposal. Capability is provided to sample the liquids collected in each sump. Section 11.2 of this report addresses the staff's evaluation of the LWMS.

The LCW drain subsystem collects and transfers by gravity liquid wastes from equipment drains in potentially contaminated systems in sumps located in the drywell and other areas. The drywell LCW drain, which is monitored for activity, and the liquid wastes collected in the LCW sumps are pumped to the LCW collection tank of the LWMS for processing and disposal.

The HCW drain subsystem collects and transfers by gravity liquid wastes from floor drains in potentially contaminated areas to sumps located in the drywell and other areas. The drywell HCW drain, which is monitored for activity, and the liquid wastes collected in the HCW sumps are pumped to the HCW collection tank of the LWMS for processing and disposal.

The detergent drain subsystem collects potentially contaminated wastes from the personnel decontamination stations, laundry, and shower facility drains and transfers them to the detergent drain collection tank of the LWMS for processing and disposal.

The chemical waste drain subsystem collects liquid wastes containing potentially contaminated chemicals and corrosive substances from washdown areas, laboratory drains, the hot maintenance shop, and other miscellaneous sources in the plant. These liquid wastes are transferred to the chemical drain collection tank of the LWMS for processing and disposal.

Dedicated sumps in the EFDS will collect vent and drain water from the closed loop RCCWS and will be directed to the RB cooling water drain subsystem. The size of this subsystem will accommodate the draining of the largest isolable cooling water pipe segment in the RB. The sump contents will be evaluated for radioactivity and water quality. If the cooling water is radioactively contaminated, it will be directed to the LWMS where it can be processed. If not, the cooling water may be recycled through a line tied back to the cooling water system.

Safety divisions are provided with a separate drain line connecting to the main drainage piping and leading to an appropriate sump in both the RB and control building. Each drain line is provided with a normally closed manual valve to prevent flooding of multiple safety divisions because of back flow. Watertight walls, floors, and doors on safety-related compartments also prevent flooding of multiple safety-related compartments.

As stated above, liquid wastes from various floors and equipment drains are drained by gravity to the appropriate sumps. 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. Leaks in the drywell are detected by monitoring the rate of the increase of sump levels. Leak detection in other areas is accomplished by monitoring the frequency and duration of sump pump operation.

9.3.3.3 Staff Evaluation

During the course of the DCD review, the staff identified areas in which it needed additional information to complete the evaluation. Subsequently, the staff issued three RAIs regarding the draining of floodwater that requested clarifications or more detailed descriptions of flood protection measures. In response, GEH 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 found the GEH responses to be acceptable.

The EFDS 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. The staff, therefore, determined that the requirements of GDC 2 and 4 do not apply.

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 liquid radwaste processing 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. In addition, 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 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. In addition, the ESBWR design does not use the EFDS to provide defense-in-depth capabilities for any safety function. The EFDS 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,” issued March 1994.

9.3.3.4 Conclusion

Based on its review of the EFDS and the evaluation discussed above, the staff agreed with GEH that the EFDS 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. In addition, the

EFDS is not considered as a candidate for RTNSS. Therefore, the staff concluded that the EFDS is acceptable.

9.3.4 Chemical and Volume Control System

This section does not apply to the ESBWR.

9.3.5 Standby Liquid Control System

9.3.5.1 Regulatory Criteria

The ESBWR includes an SLCS that provides backup capability for reactivity control independent of the control rod system. The SLCS functions by injecting a boron solution into the reactor to effect 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 2, Revision 3, Section 9.3.5, "Standby Liquid Control System," for the ESBWR, in accordance with SRP Section 9.3.5, "Standby Liquid Control System." 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, Revision 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 cooldown in the normal manner. Its function is to shut down the reactor and keep the reactor from going critical again during cooldown. 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 SLCS system 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)),

- 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 system; 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 15.6 of this report provides a more detailed discussion of the basis for this conclusion.

The NRC staff reviewed DCD Tier 2, Revision 3, Section 9.3.5, to evaluate compliance with the remaining regulatory criteria discussed in Section 9.3.5.1 of this SER. The NRC 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 NRC staff reviewed to determine that the design of the SLCS is completed in accordance with the applicable regulatory requirements. The NRC staff reviewed the diagram and determined that the design of the system, as shown in the figure, complies 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 NRC 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 is being tracked as an Open Item.**

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, vent 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 inservice 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. RAI 9.3-9 requests GEH 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 is being tracked as an Open Item.**

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, a motor-operated valve 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. Documentation provided by the applicant in indicates that, at a reactor pressure as high as 8.63 MPA (1250 psig), boron solution discharge from the SLCS occurs at a volumetric rate of 1.1 cubic meters per minute (m^3/min) (291 gallons per minute (gal/min)) and a velocity of 34 meters per second (100 feet per second). These flow rates are averages for the first 5.40 m^3 (1426 gal) of boron solution flow for each of two trains. The applicant indicated in the inspections, test, analyses, and acceptance criteria (ITAAC) for the SLCS that the boron concentration will be 12.5 percent, at a minimum, and that the boron will be enriched to 94 percent in the isotope boron-10.

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 Agencywide 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:

$$\frac{Q}{86} \times \frac{M_{251}}{M} \times \frac{C}{13} \times \frac{E}{19.8} \geq 1 \quad ,$$

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 an 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 staff evaluated this relationship for the ESBWR SLCS, where the injection flow is 291 gal/min, the concentration is 12.5 weight percent, and the enrichment is 96 percent. The

mass of the water inside the ESBWR reactor vessel with a 278-inch diameter, assuming a fluid volume of 534 m^3 (DCD Tier 2, Revision 3, figure 5.1-1, "Coolant Volumes"), is 1,200,000 pounds. (M251, the mass of the water in a 251-inch BWR/6 vessel, is 615,100 pounds). The result is that the design meets the requirements of the ATWS rule by a factor of 8.

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) developed by the Oak Ridge National Laboratory. 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 is being tracked as an Open Item.**

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 SER. **RAI 9.3-12 is being tracked as an Open Item.**

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 15.6 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 effect 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.

Topical Report NEDO-33083, “TRACG Application for ESBWR Anticipated Transient Without Scram Analyses,” issued January 2006, provides additional information about the applicant’s analysis of boron injection into the reactor vessel. The report includes information about the SLCS configuration and geometry, as well as the applicant’s analysis of SLCS injection behavior. The staff’s review of NEDO-333083 is discussed in Chapter 21 of this report.

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 construction projects. RAI 9.3-25 is being tracked as an Open Item pending completion of staff's review of the application of the TRACG code for ATWS analyses, as discussed in Chapter 21 of this report.

It may be noted that during the ATWS/MSIV closure scenario, the applicant took credit for high-pressure control rod drive system flow. Control rod drive flow may provide an active means of recirculating small amounts of water through the core and preventing flow stratification in the lower vessel head.

The NRC 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 is reviewing 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 is being tracked as an Open Item.**

The applicant's DCD does not describe the boron injection path to the core. In RAI 9.3-6, the staff requested GEH 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 GEH to describe the positions of the injection points relative to the active length of the core. The applicant provided the requested information in response to RAI 21.6-53. The acceptability of the SLCS boron injection path will be reviewed 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. Therefore, **RAI 9.3-6 is being tracked as an Open Item.**

9.3.5.3.6 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 18.9 m^3(5000 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 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 licensee stated that the Section I GDC (1–5) apply to the SLCS design. The evaluation of the SLCS with regard to these GDC appears in Section 6.3.3 of this report.

9.3.5.3.7 Inspections, Tests, Analyses, and Acceptance Criteria

The staff has reviewed the information contained in Tier 1 of the applicant's DCD. The ITAAC reflect a complete set of the information required to support safe operation of the ESBWR, subject to one open item.

The staff reviewed both Table 2.2.4-1 and the ITAAC presented in Table 2.2.4-2. In RAI 9.3-15, the staff requested that GEH 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 is being tracked as an Open Item.**

9.3.5.4 Conclusion

Because of the open items that remain to be resolved for this section, the staff was unable to finalize its conclusions regarding the SLCS.

9.3.6 Instrument Air System

9.3.6.1 Regulatory Criteria

The staff reviewed the ESBWR IAS in accordance with the guidance of SRP Section 9.3.1, Revision 2, issued July 1981. The staff's acceptance of the IAS is based on the design in conformance with the requirements of the following GDC:

- GDC 1 in part requires that SSCs important to safety shall be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the function to be performed. Where generally recognized codes and standards are used, they shall be identified and evaluated to determine their applicability, adequacy, and sufficiency and shall be supplemented or modified as necessary to ensure a quality product in keeping with the required safety function.

- GDC 2 in part requires that SSCs important to safety shall be designed to withstand the effects of natural phenomena such as earthquakes.

- GDC 5 requires that SSCs important to safety shall not be shared among nuclear power units unless it can be shown that such sharing will not significantly impair their ability to perform their safety functions, including, in the event of an accident in one unit, an orderly shutdown and cooldown of the remaining units.

Compliance with GDC 2 is based on meeting the guidance of Regulatory Positions C.1 and C.2 of RG 1.29 concerning seismic classification.

9.3.6.2 Summary of Technical Information

The IAS is a non-safety-related system and has no safety design basis. 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 consists of two identical 100-percent-capacity trains in parallel, one normally operating and the other in standby. The primary components of the IAS are air intake filter/silencers; compressors; after-coolers; the receiver; filtering and drying units; a common receiver/surge tank that both trains share; and instrumentation, valves and piping. Each IAS compressor takes suction through an air intake filter/silencer. Two air receivers are interconnected to service both

IAS compressors. Compressed air leaving the air receivers passes through one of the two parallel instrument air filtering and drying units and a receiver/surge tank before being distributed to the instrument air piping system. Both IAS compressor trains are connected to a common header that distributes instrument air to the radwaste building, turbine building, and RB. The IAS has piping connections outside containment to the HPNSS to serve as a manual backup to the HPNSS, and it supplies compressed air to the HPNSS loads inside containment via the HPNSS piping during refueling operations.

During normal operation, the IAS provides filtered, dry, and oil-free compressed air for plant instrumentation, control systems, and pneumatic valve/damper actuators located outside of the containment. One compressor is selected for continuous operation, while the other remains on standby and starts automatically if the continuously operating compressor cannot meet system demand. The operating compressor automatically loads or unloads in response to the IAS demand as determined by pressure changes in the air receivers. The IAS is automatically switched to the standby alternating current power source during a loss of preferred power and is backed up by the SAS during failure of both IAS air compressors.

System operational tests for components normally closed to airflow are performed periodically 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 Instrument Society of America (ISA) Standard 7.0.01. Pneumatic isolation valves are capable of operational integrity testing.

In DCD Tier 2, Revision 3, Section 9.3.6.4, GEH stated that the IAS is a non-safety-related system and meets the requirements of GDC 1 as it pertains to minimum instrument air quality standards by meeting ISA Standard 7.0.01-1996, "Quality Standard for Instrument Air." The IAS also meets the requirements of GDC 1 as it pertains to the guidance of RG 1.68.3, "Preoperational Testing of Instrument and Control Air Systems," related to operational testing of the IAS.

Components of the IAS are designed to meet 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

The IAS is a non-safety-related system. It does not have any safety-related function, and its failure does not prevent any safety-related equipment from performing its safety-related functions. The staff, therefore, determined that the requirements of GDC 1 and 2 do not apply to the IAS.

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

As stated in Section 9.3.1 of this report, during the course of the DCD review, the staff identified areas for which it needed additional information to complete the evaluation of the compressed air system, which contains the IAS, SAS, and HPNSS. Subsequently, the staff issued RAIs concerning the specific issues that are common and apply to the IAS, SAS, and HPNSS. The staff reviewed the applicant's responses to these RAIs and found them acceptable. Section 9.3.1 of this report addresses the staff's evaluation of the applicant's responses to these RAIs.

9.3.6.4 Conclusion

Based on its review of the IAS and the evaluation discussed above, the staff agrees that the IAS is non-safety-related, is not credited in any safety analysis, and is not required to achieve or maintain safe shutdown of the plant. In addition, the IAS is not considered as a candidate for RTNSS. Therefore, the staff concluded that the IAS is acceptable.

9.3.7 Service Air System

9.3.7.1 Regulatory Criteria

The staff reviewed the ESBWR SAS in accordance with the guidance of SRP Section 9.3.1, Revision 2, issued July 1981. The staff's acceptance of the compressed air system is based on the design in conformance with the requirements of the following GDC:

- GDC 1 in part requires that SSCs important to safety shall be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the function to be performed. Where generally recognized codes and standards are used, they shall be identified and evaluated to determine their applicability, adequacy, and sufficiency and shall be supplemented or modified as necessary to ensure a quality product in keeping with the required safety function.

- GDC 2 in part requires that SSCs important to safety shall be designed to withstand the effects of natural phenomena such as earthquakes.

- GDC 5 requires that SSCs important to safety shall not be shared among nuclear power units unless it can be shown that such sharing will not significantly impair their ability to perform their safety functions, including, in the event of an accident in one unit, an orderly shutdown and cooldown of the remaining units.

Compliance with GDC 2 is based on meeting the guidance of Regulatory Positions C.1 and C.2 of RG 1.29 concerning seismic classification.

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 and service air outlets located outside of the containment. The SAS also provides breathing air. During IAS system failure, the SAS can supply compressed air to the IAS loads. It does not have any safety-related function other than containment isolation functions. Failure of the SAS does not prevent any safety-related equipment from performing its safety-related functions.

The SAS consists of two identical trains in parallel—one normally operating and the other in standby. The primary components of the SAS are air intake filter/silencers; compressors; after-coolers; moisture separators; air receivers; breathing air purifiers; and instrumentation, valves, and piping.

During normal operation, one compressor train is selected for continuous operation, while the other remains on standby and starts automatically if the continuously operating compressor cannot meet system demand. The operating compressor, which 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 air compressor trains are connected to a common header that distributes air to the breathing air purifiers, radwaste building, turbine building, and RB. The SAS line that penetrates the containment is provided with redundant valves for containment isolation following a LOCA. The SAS also serves as a backup to the IAS.

The breathing air purifiers consist of two identical 100-percent trains in parallel—one normally operating, and the other in standby. The SAS supplies service air to the breathing air purifiers. Each breathing air purifier is equipped with a prefilter, twin desiccant towers, a catalyst, and an activated charcoal filter. One air receiver with a bypass line serves both purifiers. Both breathing air purifiers are connected to a common header that supplies the radwaste building, turbine building, and RB. The breathing air line that penetrates the containment is provided with redundant valves for containment isolation following a LOCA.

Components of the SAS are designed to 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.

The major components of the breathing air purifiers are designed to meet ASME Boiler and Pressure Vessel Code, Section VIII, Division I; ASME Piping Code B31.3; and Occupational Safety and Health Administration breathing air requirements.

9.3.7.3 Staff Evaluation

The SAS is a non-safety-related system. It does not have any safety-related functions, and its failure does not prevent any safety-related equipment from performing its safety-related functions. The staff, therefore, determined that the requirements of GDC 1 do not apply to the SAS. In addition, the staff found that, other than the provision for safety-related containment penetrations and isolation valves, the requirements of GDC 2 do not apply to the SAS.

Section 6.2.4 of this report addresses the staff's evaluation of the containment isolation valves for the SAS and the breathing air supply lines that penetrate the containment.

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

As stated in Section 9.3.1 of this report, during the course of the DCD review, the staff identified areas in which it needed additional information to complete the evaluation of the compressed air system, which contains the IAS, SAS, and HPNSS. Subsequently, the staff issued RAIs concerning the specific issues that are common and apply to the IAS, SAS, and HPNSS. The staff reviewed the applicant's responses to these RAIs and found them acceptable. Section 9.3.1 of this SER addresses the staff's evaluation of the applicant's responses to these RAIs.

9.3.7.4 Conclusion

Based on its review of the SAS and the evaluation discussed above, the staff agrees that the SAS is non-safety-related, is not credited in any safety analysis, and is not required to achieve or maintain safe shutdown of the plant. In addition, the SAS is not considered as a candidate for RTNSS. Therefore, the staff concluded that the SAS is acceptable.

9.3.8 High-Pressure Nitrogen Supply System

9.3.8.1 Regulatory Criteria

The staff reviewed the ESBWR HPNSS in accordance with the guidance of SRP Section 9.3.1, Revision 2, issued July 1981. The staff's acceptance of the HPNSS is based on the design in conformance with the requirements of the following GDC:

- GDC 1 in part requires that SSCs important to safety shall be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the function to be performed. Where generally recognized codes and standards are used, they shall be identified and evaluated to determine their applicability, adequacy, and sufficiency and shall be supplemented or modified as necessary to ensure a quality product in keeping with the required safety function.

- GDC 2 in part requires that SSCs important to safety shall be designed to withstand the effects of natural phenomena such as earthquakes.

- GDC 5 requires that SSCs important to safety shall not be shared among nuclear power units unless it can be shown that such sharing will not significantly impair their ability to perform their safety functions, including, in the event of an accident in one unit, an orderly shutdown and cooldown of the remaining units.

Compliance with GDC 2 is based on meeting the guidance of Regulatory Positions C.1 and C.2 of RG 1.29 concerning seismic classification.

9.3.8.2 Summary of Technical Information

The HPNSS distributes clean, dry, oil-free nitrogen gas to containment nitrogen users from the CIS. Nitrogen loads include the nuclear boiler system (NBS) automatic depressurization subsystem (ADS) SRV accumulators, the isolation condenser (IC) steam and condensate line isolation valve accumulators, the MSIV accumulators, and other pneumatically operated valves. The HPNSS provides a means for switchover from the CIS to the nitrogen bottle backup during low CIS supply pressure. The HPNSS provides a means for a manual switchover from the HPNSS supply to the IAS during refueling outages.

The HPNSS consists of the distribution piping between the CIS and the containment nitrogen users. The HPNSS also provides bottled high-pressure nitrogen gas that is clean, dry, and oil free to the NBS ADS SRV accumulators, IC line isolation valve accumulators, MSIVs, and other nitrogen loads if the CIS fails to maintain the required nitrogen supply pressure. The CIS normally supplies all containment nitrogen loads. The CIS nitrogen supply line branches into two HPNSS distribution lines. One branch line supplies the low-pressure nitrogen loads (i.e., MSIVs, instruments, and pneumatic-operated valves) while the second branch supplies the high-pressure nitrogen loads (i.e., NBS ADS SRV accumulators and the IC piping isolation valve accumulators).

The HPNSS provides the distribution piping from the CIS to the nitrogen loads in the containment. To meet the requirements of the accumulators, a pressure-reducing station initially reduces the CIS nitrogen gas pressure. One of the two branch lines distributes high-pressure nitrogen to the accumulators. The remaining branch line incorporates a pressure-

reducing station to lower the CIS nitrogen supply to the required pressure. The HPNSS incorporates containment isolation valves where the HPNSS supply lines penetrate the containment. An air-operated isolation valve is located outboard of the containment penetration while a check valve, which also functions as an isolation valve, is provided inboard.

The HPNSS backup consists of two bottle-rack trains in parallel. Each bottle-rack train consists of eight or more bottles, each bottle with a pigtail and station valve, which are piped to a manifold. Each bottle rack manifold interfaces with the nitrogen supply distribution header through an isolation valve. The supply header incorporates a pressure-reducing station to depressurize the bottled nitrogen gas to the required supply distribution pressure. The nitrogen bottle racks each have about an 8-hour supply for the containment nitrogen loads upon loss of normal CIS supply. The piping and valves of the HPNSS are designed to meet ASME Power Piping Code B31.1. It is in the non-safety-related safety class.

The HPNSS distributes CIS nitrogen supply to the loads in the containment during normal operation. An upstream pressure control valve on each branch modulates the CIS nitrogen supply to provide the required nitrogen supply pressure to the nitrogen loads. If the CIS fails to maintain the required nitrogen supply pressure, the HPNSS provides uninterrupted nitrogen gas supply from the nitrogen storage bottles. When the nitrogen gas pressure in the main header drops below the set pressure, the manifold isolation valve automatically opens to provide nitrogen gas from the storage bottles to all nitrogen loads. One bottle rack train and one pressure-reducing station are used to maintain design nitrogen supply as required. The nitrogen bottle station valves and manifold isolation valve on one train are kept open, while the standby train bottle station valves and manifold isolation valves are kept closed. Switchover from one bottle rack train to the other is a manual operation. The HPNSS bottled nitrogen normally remains on standby, through an isolation valve located upstream of the pressure-reducing station. During low nitrogen supply pressure in the main supply header, the isolation valve automatically opens to allow nitrogen gas supply from the HPNSS nitrogen bottles to all system loads. The normal CIS supply line is isolated following a set time delay. Restoration of the HPNSS to the CIS occurs through manual operation of the isolation valves from the MCR.

9.3.8.3 Staff Evaluation

The HPNSS is a non-safety-related system. It does not have any safety-related functions, and its failure does not prevent any safety-related equipment from performing its safety-related functions. The staff, therefore, determined that the requirements of GDC 1 do not apply to the HPNSS. In addition the staff found that, other than the provision for safety-related containment penetrations and isolation valves, the requirements of GDC 2 do not apply to the HPNSS.

Section 6.2.4 of this report addresses the staff's evaluation of the containment isolation valves for the HPNSS supply lines that penetrate the containment.

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

As stated in Section 9.3.1 of this report, during the course of the DCD review, the staff identified areas in which it needed additional information to complete the evaluation of the compressed air system, which contains the IAS, SAS, and HPNSS. Subsequently, the staff issued RAIs concerning the specific issues that are common and apply to the IAS, SAS, and HPNSS. The staff reviewed the GEH responses to these RAIs and found them acceptable. Section 9.3.1 of this report addresses the staff's evaluation of the GEH responses to these RAIs.

In addition, the staff issued RAI 14.3-91 regarding ITAAC for the HPNSS.

RAI 14.3-91

RAI 14.3-91 stated the following:

> DCD Tier 1, Table 2.4.1-1, Item 12, lists a test and the associated acceptance criteria for the capacity of the accumulators for the isolation condenser isolation valves. However, DCD Section 5.4.6 does not clearly describe the basis for the specified capacity, and DCD Tier 1, Table 2.1.2-2, does not include similar ITAAC regarding the design capability of the compressed gas accumulators for the MSIV and the safety relief valves. Provide specific ITAAC regarding the capability of each safety-related portion of the compressed gas systems to perform its safety function and the design basis for the capability.

In response to RAI 19.1.0-2 regarding RTNSS, the applicant identified in DCD Tier 1, Revision 3, Table 1, that the HPNSS is a safety system credited in the PRA sensitivity study; however, the applicant had neither revised DCD Tier 2, Revision 3, Section 9.3.8, to classify the HPNSS as a safety-related system nor included it in Table 3 as RTNSS. Subsequently, the staff issued supplemental RAI 22.5-3.

RAI 22.5-3

RAI 22.5-3 stated the following:

> In MFN 07-066 (response to RAI 19.1.0-2), Enclosure 1, Table 1, the High Pressure Nitrogen Supply System (HPNSS) is identified as a safety system credited in the PRA sensitivity study. However, DCD Tier 2 Section 9.3.8, Revision 3, and Section 19A. 6.1.2.1, Revision 3, identify HPNSS as a non-safety-related system. Please clarify the safety/non-safety designation of HPNSS and describe any RTNSS related functions and interfaces.

The applicant stated the following in its response, dated August 2, 2007, to RAI 22.5-3:

> The High Pressure Nitrogen Supply System (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, HPNSS does not provide a RTNSS function.
>
> Revision 3 of DCD Tier 2 Section 19 was corrected to reflect the fact that HPNSS does not meet RTNSS criteria.

Based on its review, the staff finds the applicant's response to RAI 22.5-3 acceptable. Therefore, the staff's concern described in RAI 14.3-91 is resolved.

9.3.8.4 Conclusion

Based on its review of the HPNSS information provided in DCD Tier 2, Revision 3, Section 9.3.8, the applicant's responses to the staff's RAIs, and the staff's evaluation discussed above, the staff agrees that the HPNSS is non-safety-related, is not credited in any safety analysis, and is not required to achieve or maintain safe shutdown of the plant. In addition, the HPNSS is not considered as a candidate for RTNSS. Therefore, the staff concludes that the HPNSS is acceptable.

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 3, 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 adding hydrogen to the feedwater at the suction of the feedwater pumps. The system includes monitoring systems to track the effectiveness of the HWCS. Given that the hydrogen and oxygen supply systems are site dependent, the combined license (COL) applicant will specify the use and installation of an HWCS. The COL applicant will also furnish necessary test and inspection information. **These are identified as COL Action Items in DCD Tier 2, Revision 3, Section 9.3.9.6**

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 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. The staff requested

clarification from the applicant in RAI 9.3-37. In its response, the applicant stated that the COL applicant is responsible for determining whether to install an HWCS, and the RAI was resolved.

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 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, general 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. 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 to RAI 9.1-38, 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. The COL applicant defines the site storage requirements based on oxygen demand and hydrogen water chemistry requirements (if the HWCS is implemented). **This is identified as a COL Action Item in DCD Tier 2, Revision 3, Section 9.3.10.6.** 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 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. 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 is being tracked as an Open Item.**

The staff finds that the EPRI guidelines describe a satisfactory means for storing and handling oxygen for the ESBWR design.

9.3.10.4 Conclusion

Because of the open item that remains to be resolved for this section, the staff was unable to finalize its conclusions for the OIS.

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 3, 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. 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 9.3-38, 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. **These are identified as COL Action Items in DCD Tier 2, Revision 3, Section 9.3.11.6.** 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. The SRP, through March 2007, does not include a section specifically addressing the auxiliary boiler/steam system. The staff reviewed the ABS 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.

9.3.12.2 Summary of Technical Information

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
- hot water generator of the hot water system (if required)
- preoperational testing of off-gas system equipment
- evaporation of liquid nitrogen for inerting of the containment

The auxiliary boilers burn diesel fuel oil to boil demineralized water to produce steam during plant startup, shutdown, and offline operation when main steam or gland steam evaporator steam is unavailable. Fuel oil transfer pumps are provided in the ABS to transfer fuel oil from the diesel generator (DG) fuel oil storage tank to the auxiliary boilers. The fuel oil piping for the auxiliary boilers will be connected to the DG oil storage tank by means of a fuel tank nozzle situated above the minimum fuel oil level required for the DG system. The makeup water system provides makeup feedwater to the ABS.

9.3.12.3 Staff Evaluation

Revision 3 of the DCD does not contain the information needed to determine that the failure of the ABS as a result of a pipe break or malfunction of the system would not adversely affect safety-related systems or components. The staff requested in RAI 9.3-40 that the applicant identify (1) whether the ABS would interface directly with any nuclear process systems, (2) the location of the auxiliary boiler, and (3) whether the ABS lines would pass through areas where safety-related equipment is located. **RAI 9.3-40 is being tracked as an Open Item.**

9.3.12.4 Conclusion

On the basis of the above review, the staff has not been able to find that the ABS is acceptable because GEH has not 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.

9.4 Air Conditioning, Heating, Cooling, and Ventilation Systems

9.4.1 Control Building Heating, Ventilation, and Air Conditioning (HVAC) System

9.4.1.1 Regulatory Criteria

The staff of the NRC reviewed the Control Building HVAC system (CBVS) 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." Conformance with the SRP acceptance criteria forms the basis for concluding whether the CBVS satisfies the following requirements:

- General Design Criterion (GDC) 2, "Design Bases for Protection Against Natural Phenomena," as it relates to SSCs important to safety being designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches without loss of capability to perform their safety functions

- GDC 4, "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 the means to control suitably the release of radioactive materials in gaseous and liquid effluents and to handle radioactive solid wastes produced during normal reactor operation, including anticipated operational occurrences

- Title 10, Section 50.63, "Loss of All Alternating Current Power," of the *Code of Federal Regulations* (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 (SBO) event

- 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," which provides guidance on the design and testing of engineered safety feature (ESF) postaccident air filtration cleanup systems

9.4.1.2.1 Summary of Technical Information

The 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 consists of two systems, the control room habitability area HVAC subsystem (CRHAVS) and the Control Building general area HVAC subsystem (CBGAVS).

The CRHAVS serves the MCR and associated support areas that comprise the control room habitability area (CRHA). The CRHA envelope can be isolated and protected during emergency modes of operation. When alternating current (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 (AHUs). Radiological protection is provided from a redundant set of safety-related emergency filter units (EFUs).

When ac power is not available, the CRHA is cooled passively via heat transfer to the CRHA passive heat sink, and the safety-related EFUs, which are powered from the safety-related 1E battery power source, continue to provide radiological protection. The portions of the CRHAVS that are safety-related include the EFUs and their associated fans, ductwork, instrumentation and controls; the CRHA boundary envelope; and the CRHA isolation dampers and associated ductwork. All remaining CRHAVS equipment is 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 electrical power (SBO) or during a radiological event.

The CBGAVS serves the area outside the CRHA. The CBGAVS is non-safety-related. The subsystem is made up of two subsets, each of which contain two redundant 100-percent capacity AHUs and associated fans and ductwork.

The CRHAVS safety-related functions monitor the CRHA air supply for smoke and radioactive particulate and/or iodine concentrations, isolate the normal CRHA air supply and restroom exhaust, 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. The safety-related functions also isolate the normal CRHA air supply and restroom exhaust upon detection of smoke in the CRHA normal air supply.

The portions of the CRHAVS that penetrate the CRHA envelope are safety-related and designed as seismic Category I to isolate 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.

EFUs are automatically actuated upon radiological isolation of the CRHA envelope or an extended loss of ac power.

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
- EFUs and related components

- Ductwork from the CRHA boundary envelope up to the CRHA isolation dampers and the CRHA isolation dampers

The CBVS provides a safety-related means to passively maintain habitable conditions in the CRHA following a DBA (radiological event concurrent with an SBO). A redundant set of safety-related EFUs, including high-efficiency particulate air (HEPA) and carbon filters, serve the CRHA envelope. Radiation detected in the CRHA outside air inlet causes the following actions:

- The normal outside air inlet and restroom exhaust dampers will close.
- An EFU fan will start automatically.

The CRHA is isolated during SBO 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.

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) that ensure habitability in the CRHA envelope are redundant. 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 ranges noted in DCD Tier 2, Revision 3, Table 9.4-1 by the non-safety-related CRHAVS AHU. During emergency operation, with an SBO, a non-safety-related CRHA AHU, powered from the non-safety-related uninterruptible ac power supply system, maintains the CRHA within the normal operating temperature range for 2 hours. This allows the continued operation of certain high heat producing non-safety-related MCR distributed control and information system (DCIS) electric loads. These non-safety-related MCR DCIS electrical loads automatically deenergize should the redundant CRHA AHUs not be available. In the event the SBO duration extends beyond 2 hours, the reduced CRHA heat load is passively cooled by the CRHA heat sink, which consists of the internal and external concrete walls, floor, and ceiling, such that the CRHA temperature rise is no greater than 8.3 °C (15 °F).

The CBVS provides the following:

- a controlled environment for personnel comfort and safety which includes sufficient outside air to meet the ventilation requirements for acceptable indoor air quality found in American Society of Heating, Refrigeration and Air Conditioning Engineers (ASHRAE) 62-2001, “Ventilation for Acceptable Indoor Air Quality,” Table 2. ASHRAE 62-001 is included in DCD Tier 2, Revision 3, Table 1.9-22, “Industrial Codes and Standards Applicable to the ESBWR.) DCD Tier 2, Revision 3, Table 9.4.1 provides, area design temperature and humidity design parameters,

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

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 radiologically isolated.

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's seismic Category I structural areas.

The CRHAVS is configured as a recirculation system and contains the entire supply and return AHU airflow inside the CRHA, incorporating a common supply duct for introducing outside air to the CRHA. The normal and emergency (EFU) outside air intake flows are adjusted as required to maintain a 31 pascal (Pa) (1/8" waster gauge (w.g.)) minimum positive pressure in the CRHA. The intake design and location are in accordance with RG 1.194, "Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants." Intake design, location, and control also include considerations that minimize the introduction of radiological material, toxic gases, hazardous chemicals, smoke, dust, and other foreign material. Ductwork, housings, access openings, and the like are constructed in such a manner as to minimize 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. The automatically controlled humidifier adds moisture to the air stream, if required, to maintain minimum humidity levels in the CRHA. 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.

Each EFU consists of a medium-efficiency filter (40 percent minimum credited efficiency), a HEPA filter (99.97-percent dioctyl phthalate (DOP) minimum), a carbon adsorption filter (99-percent credited efficiency), and a postfilter downstream of the carbon filter (95-percent DOP minimum). The EFUs operate only during a radiological emergency and are able to function while powered from an offsite ac source, an onsite ac source, or an onsite safety-related direct current (dc) source.

The EFUs are monitored by instrumentation that detects a loss of airflow and radiation downstream of the EFU filters. Upon such detection, the operating EFU is isolated and the standby EFU is automatically placed in service. Exhaust air from the restroom is ducted to the exhaust fan and exhausted to the outside atmosphere.

The CRHA AHUs provide cooling to the CRHA whenever offsite or onsite ac power is available. Furthermore, an AHU fan and a chilled water recirculation pump are battery powered during the first 2 hours of an SBO by the non-safety-related battery supply. Cooling of the CRHA for the 2-hour duration during which the non-safety-related heat loads in the CRHA are powered by the non-safety-related battery supply is achieved by circulating chilled water from the chilled water storage tank through an AHU.

With ac power available, an AHU can operate indefinitely during a CRHA isolation event. If the AHUs are not available during the SBO, the non-safety-related heat loads are deenergized automatically.

The non-safety-related uninterruptible ac power supply system provides power for the CRHA AHUs. An EFU provides sufficient breathing-quality air to maintain positive pressure in the CRHA when the CRHA envelope is radiologically isolated. An EFU is automatically actuated when the CRHA envelope is radiologically isolated. Controls to manually isolate the CRHA envelope and to manually actuate the EFUs are also provided.

The CBGAVS serves nondivisional equipment rooms, corridors, and other miscellaneous rooms in the Control Building general areas. Set A serves Division I and IV areas. Set B serves Division II and III 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. Low- and high-efficiency filters remove particulates from the air. 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 following discussion summarizes the staff's review of the CBVS as described in Revision 3 of the DCD, focusing on compliance with each of the aforementioned GDC.

GDC 2 requires that SSCs important to safety be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches without loss of capability to perform their safety functions. The staff review determined that the CRHAVS meets the acceptance criteria of GDC 2. The CRHA envelope comprises seismic Category I structures and components that are protected from postulated tornados, hurricanes, tsunamis, seiches, and seismic events. The CRHAVS components are designated as seismic Category II, with the exception of the safety-related CRHA envelope; isolation dampers; and the EFUs and associated fans, dampers, ductwork, and instrumentation and controls; which are designated as seismic Category I. The Control Building 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.

GDC 4 requires that SSCs important to safety be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant accidents (LOCAs). The staff review determined that the CRHAVS meets the acceptance criteria of GDC 4 by ensuring that SSCs important to safety are designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including LOCAs. The safety-related CRHA envelope; isolation dampers; and the EFUs and associated fans, dampers, ductwork, and instrumentation and controls; are designed to be protected from all postulated environmental and dynamic effects.

The remaining portion of the CBVS is the CBGAVS which serves the area outside the CRHA and is non-safety-related. GDC 4 does not apply to the CBGAVS since this system and its components are not considered important to safety.

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

GDC 19 requires that a control room be provided from which actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions, including LOCAs. It also requires that adequate radiation protection be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 0.05 Sv (5 rem) total effective dose equivalent (TEDE) as defined in 10 CFR 50.2 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 RAI 9.4-5, 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. The staff found that the applicant's list did not clarify which components are designed to which codes and standards. In RAI 9.4-5 S01, the staff asked the applicant to identify which codes are specific to components or structures in each of the systems and to eliminate those codes that are not applicable to that system. **RAI 9.4-5 is being tracked as an Open Item.**

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 EFU, and pressurization of the control room to minimize unfiltered in leakage. 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 in the event of SBO, LOCA, and high radiation signals. DCD Tier 2, Revision3, Section 9.4.1 states in the description of the CRHAVS that, "The portions of the CRHAVS which penetrate the CRHA envelope are non-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 staff believes that this may be a misprint and that the portions penetrating the CRHA envelope should be safety-related. In RAI 9.4-37, the staff requested that the applicant clarify whether these portions 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. If so, the staff requested that the applicant make the appropriate change to the DCD. **RAI 9.4-37 is being tracked as an Open Item.**

The safety-related EFU provides a source of ventilation for the control room during periods of isolation. The flow rate of the system was established to ensure an adequate supply of fresh air for the inhabitants, provide protection from airborne releases by filtration with HEPA and carbon adsorber filters, and maintain control room pressurization. The system operates from either the safety-related battery source or an ac source, if available. The rated flow for the EFU is 424 cubic feet per minute (cfm). In RAI 9.4-29, the staff requested additional information on the adequacy of the flow rate (424 cfm), the assumptions on which it is based, the ability to detect oxygen and/or carbon dioxide levels, and the ability to increase flow to the MCR if needed. **RAI 9.4-29 is being tracked as an Open Item.**

The staff is concerned about recirculation of air in the MCR during isolation. The recirculation AHU is not safety-related and would not be operating. The EFU air could be at maximum outdoor air temperature and might thermally stratify in the MCR. As such, the air in the operator

breathable zone (3 to 6 feet from the floor) may not be properly refreshed. In RAI 9.4-30, the staff requested that the applicant discuss the features that provide assurance that the air in the operator breathable zone would be adequately refreshed in the first 72 hours. **RAI 9.4-30 is being tracked as an Open Item.**

The EFU is designed to operate from the plant's safety-related battery power source for a minimum of 72 hours. The staff is concerned about the operational reliability of this power source for the 72-hour period, given changes in flow over time as batteries become depleted or as new loads are added. The staff is also concerned about the source of power after 72 hours. In RAI 9.4-31, the staff requested that the applicant provide information on the small, portable ac power generator that is identified as a power source for the EFU system in DCD Tier 2, Revision 3, Section 19A.3.1.3. **RAI 9.4-31 is being tracked as an Open Item.**

Many facilities have used pressurization of control rooms to reduce unfiltered in-leakage. Tracer gas results at a number of facilities have shown that, despite control room pressurization, significant unfiltered in-leakage did exist. DCD Tier 2, Revision 3, Table 15.4-5, "Loss-of-Coolant Accident Parameters," specifies an assumed control room unfiltered air in-leakage rate of 1.13×10^{-2} cubic meters per minute (0.4 cfm). The staff is concerned that this may be an error in the DCD or that this low in-leakage would be unachievable. In RAI 15.4-30, the staff requested that the applicant include this assumed control room unfiltered air in-leakage rate in (1) DCD Tier 1, Revision 3, Table 2.16.2-1, "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, Revision 3, Chapter 16, "Technical Specifications," Section 3.7.2, "Control Room Habitability Area Heating, Ventilation, and Air Conditioning Subsystem," as surveillance requirements, in accordance with guidance provided in Technical Specification Task Force (TSTF)-448, "Control Room Habitability," dated July 1, 2003. **RAI 15.4-30 is being tracked as an Open Item.**

In DCD Tier 2, Revision 3, Section 9.4.1, and as shown on Figure 9.4.1, the normal outdoor air supply for the CRHAVS is 424 cfm. With the leaktight structure, the MCR is essentially a confined entry vessel and differs significantly from the structures evaluated in ASHRAE 62-2001. The MCR operates at a positive pressure and depends on cracks, crevices, and door seals leakage to remove most of the surplus air resulting from the normal outside air supply. The staff is concerned that the air leaving the ESBWR MCR might not carry with it the contaminants, odors, and fumes that would be carried out by a direct exhaust equivalent to the outside air being added.

In RAI 15.4-49, the staff requested that the applicant provide additional information on the applicability of ASHRAE 62-2001 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 is being tracked as an Open Item.**

Temperature control is required for the proper operation of both personnel and equipment. The normal alignment with chilled water being supplied to the recirculation AHU cooling coils is reasonable. Loss of power that eliminates the cooling water supply poses a challenge to the control of temperature.

The applicant states that for the first 72 hours of an SBO, heat will be dissipated through the CRHA heat sink, which is essentially the walls, floors, and ceilings of the building, and that the temperature rise will be limited to 8.3 °C (15 °F). Although the building structures are massive,

the applicant needs to show that the rate of heat buildup in the air spaces of the CRHA is significantly reduced by the loss of heat to the structures, ensuring that the air temperature for personnel and equipment cooling stays within acceptable limits. In performing this analysis, the applicant should consider that the outside air entering the CRHA may be 35 °C (95 °F) and that possible solar loads on the building structures may reduce heat transfer rates.

The staff identified three RAIs to address these issues. In RAI 9.4-32, the staff requested the applicant to do the following:

A. Provide additional information justifying the use of a non-safety-related power source to provide cooling to non-safety-related heat loads in the first 2 hours of accident mitigation. The applicant should include in the information the source and magnitude of these heat loads and the impact on control room temperatures and accident mitigation if cooling is not available and/or these non-safety-related heat loads are isolated.

B. Identify any operator actions that may be required to isolate these heat loads during the first 2 hours of an accident.

In RAI 9.4-33, the staff requested that the applicant provide additional information on how the temperature in the control room will change with time during the 72-hour period following accident initiation. The information should be based on a detailed thermal heat transfer study that considers the temperature of outside air entering the control room; the heat loads in the control room; the rate of heat transfer from the heat-producing equipment to control room air; the rate of heat transfer from the control room air to building structures and components that make up the heat sink; the impact of other barriers to heat transfer, such as carpets, vinyl layers, false ceilings and floors; and the impact of heat sources on exterior CRHA surfaces, outside air conditions, and other environmental factors on the building heat sink.

In RAI 9.4-34, the staff requested that the applicant provide information on the changes to the control room temperature during a DBA. The staff also requested information on corresponding temperature changes in electrical cabinets and their effects on component performance. **RAIs 9.4-32, 9.4-33, and 9.4-34 are being tracked as Open Items.**

GDC 60 requires that the nuclear power unit design include the means to control suitably the release of radioactive materials in gaseous and liquid effluents and to handle radioactive solid wastes produced during normal reactor operation, including anticipated operational occurrences. 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. As such, the CBVS, including the CRHAVS, meets the acceptance criteria of GDC 60.

The CRHAVS EFUs meet the guidance of RG 1.52 as it relates to the design, testing, and maintenance criteria for atmosphere cleanup system and normal ventilation system air filtration and adsorption units. The staff noted that in DCD Tier 2, Revision 3, Chapter 16, Subsection 5.5.13, “Ventilation Filter Testing Program,” the applicant specified an in-place 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 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 are being tracked as Open Items.**

In RAI 9.4-50, the staff requested the applicant to label the AHU's listed in DCD, Tier 2, Revision 3, Table 9.4.2 as Recirculation AHU's to avoid confusion and the terminology should be consistent in the text, tables and figures of the DCD. **RAI 9.4-50 is being tracked as an Open Item.**

9.4.1.4 Conclusions

Because of the open items that remain to be resolved for this section, the staff was unable to finalize its conclusions regarding the acceptability of the CBVS.

9.4.2 Fuel Building HVAC System

9.4.2.1 Regulatory Criteria

The staff reviewed the Fuel Building HVAC system (FBVS) in accordance with SRP Section 9.4.2, "Spent Fuel Pool Area Ventilation System." Conformance with the SRP acceptance criteria forms the basis for concluding whether the FBVS satisfies 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.

9.4.2.2 Summary of Technical Information

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

The safety-related portion of the FBVS is the isolation dampers and ducting penetrating the FB boundary. The remainder of the system is non-safety-related.

The FBVS maintains space design temperatures, quality of air, and pressurization in the FB. The system consists of two subsystems, the FB general area HVAC subsystem (FBGAVS) and the FB fuel pool area HVAC subsystem (FBFPVS). The FBGAVS serves the general areas of the FB. The FBFPVS serves the SFP and equipment areas of the FB. Recirculation AHUs provide supplementary cooling for selected rooms in the FB.

The FBGAVS is a once-through ventilation system with redundant AHUs, exhaust fans, and FB boundary isolation dampers. Each AHU includes low- and medium-efficiency filters, heating and cooling coils, and an AHU supply fan. Outside air is filtered and heated or cooled before being distributed by the AHU in service. A common supply duct system is incorporated to distribute conditioned air to the general areas of the FB. The exhaust fan discharges the air to the outside atmosphere through the monitored vent stack where the exhaust air is monitored for radioactivity. The exhaust air may be manually diverted to the RB HVAC purge exhaust filter unit. Electric unit heaters provide supplementary heating. A recirculation AHU provides supplementary cooling for the fine motion control rod drive (FMCRD) room. The hot water system and chilled water system (CWS) provide heating and cooling water for the FBGAVS AHUs. The instrument air system (IAS) provides instrument air for the pneumatic actuators.

The FBGAVS AHUs are located in the FB HVAC equipment area. The FBGAVS exhaust fans are located in the RB. The FMCRD recirculation AHU is located in the FB.

The FBFPVS is a once-through ventilation system with redundant AHUs and exhaust fans. Each AHU includes low- and medium-efficiency filters, heating and cooling coils, and an AHU supply fan. Outside air is filtered, heated or cooled, and distributed across the SFP surface and to the equipment areas. Air is exhausted from the SFP, through redundant FB boundary isolation dampers, to the outside atmosphere through the vent stack. During high radiation conditions, the exhaust air may be manually diverted to the RB HVAC purge exhaust filter unit. The exhaust fans are also used for smoke removal. Electric unit heaters provide supplementary heating. The CWS provides cooling for the FBFPVS AHUs. Instrument air is provided for the pneumatic actuators. The FBFPVS AHUs are located in the FB HVAC equipment area. The FBFPVS exhaust fans are located in the RB. During high radiation conditions, the FB boundary isolation dampers close automatically and the supply AHU and exhaust fan shut down automatically in both subsystems.

During normal operation, both the FBGAVS and FBFPVS subsystems are fully operable. Each subsystem operates with one supply AHU and one exhaust fan in service. The redundant AHU and fans are maintained on standby. In the event of low airflow in an exhaust duct, the standby exhaust fan starts. Simultaneously, because of a loss of negative pressure in the area, the AHU supply fan serving the area stops. The AHU supply fan restarts upon reestablishment of the required negative pressure. In the event of a fan failure, the failed fan automatically shuts down and the standby fan automatically starts.

Upon detection of high radiation, the process radiation monitoring system provides a signal that trips the FBGAVS and FBFPVS subsystems. Each subsystem's supply AHU and exhaust fan shut down and their associated dampers close. Exhaust air from either subsystem may be manually diverted to the RB HVAC purge exhaust filter unit. It is then exhausted to the plant vent stack by the RB 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.

If smoke is detected in a supply air duct, the affected subsystem is shut down. In the event of a fire, fire dampers close to isolate the fire area. Following a recovery from a fire, a subsystem exhaust fan removes smoke from the area by exhausting to the outside. The FMCRD room AHU fan is started and stopped locally. A room thermostat modulates the chilled water valve in response to the room temperature.

9.4.2.3 Staff Evaluation

The staff review focused on the safety-related function of the FBVS to isolate the FB in the event of a radiological accident, as described in Revision 3 of the DCD. The safety-related components are the isolation dampers and the adjoining ducts. The staff is concerned about the functioning of the safety-related isolation valves in the event of a fuel-handling accident.

In DCD Tier 2, Section 9.4.2, the applicant states that the safety-related FB boundary isolation dampers automatically close in the event of a fuel-handling accident or other radiological accident. In RAI 9.4-51, the staff requested that the applicant identify whether the FB is isolated during the movement of irradiated fuel or whether it can be opened for maintenance or other refueling activities such that closure of these valves would be ineffective at containing radioactive release. The staff also requested that the applicant clarify the purpose of the isolation dampers. **RAI 9.4-51 is being tracked as an Open Item.**

GDC 2 requires that SSCs important to safety be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches without loss of capability to perform their safety functions. The FBVS complies with the requirements of GDC 2, in that the safety-related portions of the system are designed to comply with the guidance of RG 1.29, "Seismic Design Classification," Regulatory Position C.1, which requires a seismic Category I design. The remainder of the system is classified as non-safety-related and is designed to seismic Category II per RG 1.29, Regulatory Position C.2, to ensure that the failure of the non-safety-related portions of the system cannot affect safety-related components. The FB is a seismic Category I structure, except for the penthouse that houses HVAC equipment, which is seismic Category II. All FBVS components are designed as seismic Category II, with the exception of the safety-related isolation dampers and associated controls. The FBVS maintains its structural integrity after a safe-shutdown earthquake (SSE).

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

GDC 60 requires that the nuclear power unit design include the means to control suitably the release of radioactive materials in gaseous and liquid effluents and to handle radioactive solid wastes produced during normal reactor operation, including anticipated operational occurrences. The FBVS complies with the requirements of GDC 60 through the use of redundant safety-related isolation dampers, ducts, and associated instrumentation that would contain any release of radioactive materials within the FB. The design includes the capability of directing the system exhaust air to the RB HVAC purge exhaust filter unit during periods of high radioactivity. This unit is not a safety-related system and is tested in accordance with RG 1.140, "Design, Inspection, and Testing Criteria for Air Filtration and Adsorption Units of Normal Atmosphere Cleanup Systems in Light-Water-Cooled Nuclear Power Plants."

GDC 61 requires that the fuel storage and handling, radioactive waste, and other systems that may contain radioactivity be designed to ensure adequate safety under normal and postulated accident conditions. The FBVS complies with the requirements of GDC 61 by providing containment of radioactive releases in the FB by safety-related dampers and providing the capability of processing the release through the RB HVAC purge exhaust filter units.

In DCD Tier 2, Section 9.4.2, the applicant states that the FBVS is not required to operate during an SBO. The staff is concerned that, during an SBO, temperatures could rise in the FB and that equipment could malfunction. In RAI 9.4-52, the staff requested that the applicant

identify any components in the FB that could be affected by increases in temperature and to describe the impact of their failure. **RAI 9.4-52 is being tracked as an Open Item.**

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 FB HVAC are classified as RTNSS systems and which components require cooling in the post 72-hour period. **RAI 9.4-39 is being tracked as an Open Item.**

SFP cooling relies on pool boiling as an emergency cooling method. 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 is being tracked as an Open Item.**

In RAI 9.4-5, 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. The staff found that the applicant's list did not clarify which components are designed to which codes and standards. In RAI 9.4-5 S01, the staff asked the applicant to identify which codes are specific to components or structures in each of the systems and to eliminate those codes that are not applicable to that system. **RAI 9.4-5 is being tracked as an Open Item.**

9.4.2.4 Conclusions

Because of the open items that remain to be resolved for this section, the staff was unable to finalize its conclusions regarding the acceptability of the FBVS.

9.4.3 Radwaste Building Heating, Ventilation and Air Conditioning System

9.4.3.1 Regulatory Criteria

The staff reviewed the Radwaste Building HVAC system (RWVS) in accordance with SRP Section 9.4.3, "Auxiliary and Radwaste Area Ventilation System." Conformance with the SRP acceptance criteria forms the basis for concluding whether the RWVS satisfies 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.

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 RWVS consists of two sub systems, the Radwaste Building control room HVAC subsystem (RWCRVS) and the Radwaste Building general area HVAC subsystem (RWGAVS).

The RWCRVS maintains the Radwaste Building control room (RWCR) area temperature and maintains the control room areas at a slightly positive pressure relative to adjacent areas to minimize infiltration of air. Redundant components are provided to increase system reliability, availability, and maintainability.

The RWCRVS is a recirculating air conditioning system to provide filtered, heated or cooled, and humidified air to the RWCR area to maintain the required design ambient conditions and pressurization. The RWCR consists of two 100-percent capacity AHUs and a common outside air intake louver.

Each AHU contains prefilters, high-efficiency filters, a humidifier, a chilled water cooling coil, a hot water heating coil, and a supply fan section. Conditioned air is supplied to the RWCR area through ducts, dampers, and registers. The RWCRVS is capable of once-through operation for smoke removal using a separate 100-percent capacity exhaust fan.

The RWGAVS maintains the conditions in the Radwaste Building general area as provided in DCD Tier 2, Revision 3, Table 9.4-6. The RWGAVS maintains the Radwaste Building general area at a slight negative pressure relative to adjacent areas and the outside atmosphere to prevent the exfiltration of air to adjacent areas. Adequate exhaust from the trailer bays is provided to maintain the flow of air from the outside when the truck doors are open.

The RWGAVS comprises supply and exhaust subsystems to maintain direction of airflow from personnel occupancy areas toward 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 mobile radwaste processing systems. All exhaust air from the Radwaste Building general area is discharged to the plant vent stack. Redundant components are provided as necessary to increase system reliability, availability, and maintainability. Filtered outdoor air is provided at a minimum air exchange rate of two volume changes per hour. Smoke control, in accordance with National Fire Protection Association (NFPA) 92A, “Recommended Practice for Smoke-Control Systems,” is provided for areas without sprinklers where the fire hazards analysis identifies a potential for heavy smoke or heat conditions.

The RWGAVS limits the release of airborne radioactive particulates to the atmosphere by HEPA filtering of the exhaust air from the building before discharge to the atmosphere. The exhaust air is monitored for radiation before discharge to the atmosphere.

The RWGAVS is a once-through air conditioning system to provide filtered, heated or cooled, and humidified air to the Radwaste Building general area. The RWGAVS supply consists of two 100-percent capacity AHUs connected to a common supply distribution ductwork and a common outside air intake louver. Isolation dampers are provided at the inlet and outlet of each unit. Each AHU contains prefilters, high-efficiency filters, a chilled water cooling coil, a hot water heating coil, a humidifier, a centrifugal fan, and an automatic damper.

The RWGAVS exhaust consists of three 50-percent capacity filtration trains, each with a medium-efficiency prefilter and a HEPA filter, a 50-percent capacity exhaust fan, and a check valve/backdraft damper. Exhaust capacity is greater than the supply capacity in order to maintain the required negative pressure in the Radwaste Building general area.

Each filtration train is connected to a common exhaust collection duct and a common exhaust duct discharging to the plant vent stack. The RWGAVS exhaust subsystem is capable of once-through operation for smoke removal. The HEPA filters are bypassed in the smoke removal mode.

9.4.3.3 Staff Evaluation

The staff's review focused on compliance with the GDC relevant to this non-safety-related system, as described in Revision 3 of the DCD.

GDC 2 requires that SSCs important to safety be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches without loss of capability to perform their safety functions. The RWVS complies with the requirements of GDC 2, in that the system is classified as non-safety and is designed to seismic Category II in accordance with RG 1.29, Regulatory Position C.2, to ensure that the failure of the non-safety-related portions of the system cannot affect safety-related components. The RWVS is designed to maintain its structural integrity after an SSE.

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

GDC 60 requires that the nuclear power unit design include the means to control suitably the release of radioactive materials in gaseous and liquid effluents and to handle radioactive solid wastes produced during normal reactor operation, including anticipated operational occurrences. The RWVS complies with the requirements of GDC 60 as to the system's capability to suitably control release of gaseous radioactive effluents to the environment. The system achieves this by incorporating redundant isolation dampers, ducts, and associated instrumentation that contain the contamination within the Radwaste Building. The design includes the capability of directing the system exhaust air to the Radwaste Building general area exhaust filtration units. These units are designed, tested, and maintained in accordance with RG 1.140.

In RAI 9.4-5, 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. The staff found that the applicant's list did not clarify which components are designed to which codes and standards. In RAI 9.4-5 S01, the staff asked the applicant to identify which codes are specific to components or structures in each of the systems and to eliminate those codes that are not applicable to that system. **RAI 9.4-5 is being tracked as an Open Item.**

9.4.3.4 Conclusions

Because of the open item that remains to be resolved for this section, the staff was unable to finalize its conclusions regarding the acceptability of the RWVS.

9.4.4 Turbine Building Heating, Ventilation and Air Conditioning System

9.4.4.1 Regulatory Criteria

The staff reviewed the Turbine Building HVAC system (TBVS) in accordance with SRP Section 9.4.4, "Turbine Area Ventilation System." Conformance with the SRP acceptance criteria forms the basis for concluding whether the TBVS satisfies 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.

9.4.4.2 Summary of Technical Information

The TBVS includes the Turbine Building supply air fans and associated filter trains and the Turbine Building exhaust fans and associated filter trains. The TBVS also includes the various fan-coil units for local area heating and cooling within the Turbine Building.

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 is designed to minimize exfiltration by maintaining a slightly negative pressure in the Turbine Building by exhausting more air than is supplied to the 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-percent standby cooling capacity is provided in areas where a loss of cooling could cause degraded equipment performance. The Turbine Building ventilation systems and subsystems required for normal plant operation are provided with redundant fans with automatic start logic.

The TBVS provides temperature control and air movement control for personnel comfort, optimizes equipment performance by the removal of heat dissipated from plant equipment, and provides a sufficient quantity of filtered fresh air for personnel.

To control contamination, the TBVS provides for air movement from areas of lesser potential airborne radioactivity to areas of greater potential airborne radioactivity before final exhaust, minimizes the possibility of exhaust air recirculation into the air intake, and minimizes the escape of potential airborne radioactivity to the outside atmosphere during normal operation. For smoke removal and control, the TBVS is capable of exhausting smoke, heat, and gaseous combustion products in the event of a fire, in accordance with NFPA 92A for those areas without sprinklers where the fire hazards analysis identifies a potential for heavy smoke or heat conditions.

The TBVS maintains the Turbine Building minimum-required exhaust capability, as well as the fan-coil units of the reactor component cooling water system (RCCWS), the nuclear island subsystem of the CWS, and the IAS rooms in operation during a loss of preferred power (LOPP). The TBVAS also provides an adequate level of standby cooling capacity in areas where loss of cooling could cause degraded equipment performance.

The non-safety-related TBVS consists of the following subsystems 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 subsystem supplies outside air to the Turbine Building's noncontaminated areas and removes air from potentially contaminated rooms.

The main stairwells are pressurized to prevent infiltration of smoke from other Turbine Building areas in the event of a fire.

The TBE subsystem exhausts air from low potential airborne contamination Turbine Building areas and component vents, except lube oil areas, to the atmosphere. Turbine Building exhaust air is directed to the plant vent stack where it is monitored for radiation before being discharged to the atmosphere.

The TBCE subsystem collects, filters, and discharges air from potentially high airborne contamination Turbine Building areas or component vents to the atmosphere.

The TBLOE subsystem discharges the exhaust air directly to the atmosphere through the plant vent stack, passing first through the air filtration unit of the TBE subsystem. One fan is designed to continuously exhaust air 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-percent capacity exhaust fan, a HEPA filter, an isolation damper, and associated controls. The air exhausted from the TBDRE, once filtered, passes through the air filtration unit of the TBE subsystem and is finally released to the atmosphere through the plant vent stack, except during smoke removal. The TBDRE subsystem includes a 100-percent capacity filter bypass duct for purging smoke in the event of a fire.

The TBVS unit coolers and unit heaters are provided as required in various locations within the Turbine Building. The unit coolers are supplied with chilled water from the balance of plant subsystem of the CWS and the unit heaters are supplied with heated water from the hot water system. Temperature controls for the unit coolers and unit heaters are located in the unit inlet air path or are installed locally.

The cooling coils of the RCCWS, the nuclear island subsystem of the CWS, and the IAS rooms are fed from the corresponding nuclear island subsystem of the CWS train.

9.4.4.3 Staff Evaluation

The staff's review focused on compliance with the GDC relevant to this non-safety-related system, as described in Revision 3 of the DCD.

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

The TBVS complies with the requirements of GDC 2, in that the system is classified as non-safety-related and is designed to seismic Category NS in accordance with RG 1.29, Regulatory Position C.2, to ensure that the failure of the non-safety-related portions of the system cannot affect safety-related components. The Turbine Building is a seismic Category II non-safety-related structure.

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

GDC 60 requires that the nuclear power unit design include the means to control suitably the release of radioactive materials in gaseous and liquid effluents and to handle radioactive solid wastes produced during normal reactor operation, including anticipated operational occurrences. The TBVS 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 TBVS filtration units. These units are designed, tested, and maintained in accordance with RG 1.140.

The TBVS has adequate provision for maintaining a suitable environment for personnel access and equipment by providing recirculation and exhaust capabilities with adequate heating and cooling that are locally controlled as needed. Provision has been made to control contamination and gaseous discharges through filter systems and exhaust paths that are monitored before a release to the environment. The system also has the provision to exhaust smoke in the event of a fire consistent with the smoke management features of DCD Tier 2, Revision 3, Section 9.5.1.11.

The TBVS has adequate instrumentation, controls, and alarms in the MCR for adverse radiological conditions, temperature conditions, differential pressure for filters, and airflow. Provisions exist for testing key parameters and inspecting components to ensure operating conditions and system integrity.

DCD Tier 2, Revision 3, 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 and which components require cooling in the post-72 hour period. **RAI 9.4-39 is being tracked as an Open Item.**

DCD Tier 2, Revision 3, Figure 9.4-8 shows two filter units for the TBE subsystem, whereas Table 9.4-15 states that there are five filter units. In RAI 9.4-40, the staff requested that the applicant 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 is being tracked as an Open Item.**

In RAI 9.4-5, 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. The staff found that the applicant's list did not clarify which components are designed to which codes and standards. In RAI 9.4-5 S01, the staff asked the applicant to identify which codes are specific to

components or structures in each of the systems and to eliminate those codes that are not applicable to that system. **RAI 9.4-5 is being tracked as an Open Item.**

9.4.4.4 Conclusions

Because of the open items that remain to be resolved for this section, the staff was unable to finalize its conclusions regarding the acceptability of the TBVS.

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 3, 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 Reactor Building HVAC system (RBVS) in accordance with SRP sections for similar systems. Conformance with the SRP acceptance criteria forms the basis for concluding whether the RBVS satisfies 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.

9.4.6.2 Summary of Technical Information

The RBVS has the safety-related function of building isolation. The RBVS performs no safety-related function except for automatic isolation of the RB boundary during accidents. The isolation dampers and ducting penetrating the RB boundary and the associated controls that provide the isolation signal are safety-related. The remainder of the RBVS system is classified as non-safety-related.

The RBVS provides a controlled environment for personnel comfort and safety and for proper operation and integrity of equipment. For normal operation, the RBVS maintains potentially contaminated areas at a negative pressure to minimize exfiltration of potentially contaminated air. Under accident conditions the RBVS is isolated.

The RBVS maintains clean areas of the building, except for the battery rooms, at a positive pressure to minimize infiltration of outside air. Airflow from areas with a lower potential for contamination is directed to areas with a greater potential for contamination. The system is provided with redundant active components to increase its reliability, availability, and maintainability.

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. Smoke control conforms to

NFPA 92A for those areas without sprinklers where the fire hazards analysis identifies a potential for heavy smoke or heat conditions.

The RBVS shuts down during radiological events and isolates the RB boundary to prevent uncontrolled releases to the outside atmosphere. This function is considered to be safety-related.

The RBVS provides the capability to manually divert exhaust air for processing through the RB HVAC purge exhaust filter units. This feature is not considered in accident mitigation since the RBVS is shut down and isolated during accidents and has no safety-related function except for isolation. The RBVS also provides pool sweep ventilation air over the refueling area pool surface.

The RBVS is designed to maintain its structural integrity after an SSE and is designed such that any failure of the system does not compromise or otherwise damage safety-related equipment.

The RBVS is provided with (1) shutoff dampers on the inlet and outlet of fans and AHUs, (2) shutoff valves at the inlet and outlet of cooling and heating coils, (3) access doors for AHUs, fans, filter sections, and duct-mounted dampers, and (4) the capability for manual control of system fans to facilitate testing and maintenance.

In addition, the RBVS maintains the hydrogen concentration levels in the battery rooms below 2 percent by volume, replaces the primary containment inerted atmosphere with conditioned air during a refueling operation, and provides filtered exhaust capability for the FB through the RB HVAC purge exhaust filter unit.

The RBVS consists of three subsystems. The RB contaminated area HVAC subsystem (CONAVS) serves the potentially contaminated areas of the RB. The refueling and pool area HVAC subsystem (REPAVS) serves the refueling area of the RB. The Reactor Building clean area HVAC subsystem (CLAVS) serves the clean (i.e., nonradiological controlled) areas of the RB.

The CONAVS distributes conditioned air to the potentially contaminated areas of the RB. Air is exhausted from the potentially contaminated areas of the RB by the operating exhaust fan and discharged to the plant vent stack. During primary containment deinerting 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. The CONAVS also includes redundant RB HVAC purge exhaust filter units and exhaust fans. In the event of a fire, fire dampers close to isolate the fire area.

The REPAVS distributes conditioned air 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 plant vent stack. During a radiological event, exhaust air from the refueling area may be manually diverted through the RB HVAC purge exhaust filter units. The CWS provides cooling water for the REPAVS AHUs. The IAS provides instrument air for the pneumatic actuators. In the event of a fire, fire dampers close to isolate the fire area.

The CLAVS is a recirculating ventilation system that supplies a mixture of outside and return air which is filtered and heated/cooled before being distributed by the AHU in service. A common supply and return/exhaust air duct system distributes conditioned air to and from the RB clean areas. Return air not directed back to the AHU is exhausted directly outdoors. An economizer

cycle is used, when outside air conditions are suitable, to reduce mechanical cooling operating hours. The economizer cycle provides all outside air, or a mixture of outside air and return air, to the RB clean areas. The temperature of the air provided is at or below the supply air design temperature. The CLAVS AHU supplies air to the battery rooms. The air is then exhausted from the battery rooms by the battery room exhaust fans which discharge directly to the plant vent stack.

For each of the subsystems, in the event smoke is detected in the supply air duct, the affected subsystem is shut down. 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 because of high radiation in the exhaust ducts.

9.4.6.3 Staff Evaluation

The staff's review focused on compliance with the GDC relevant to this system, which has a safety-related isolation function. The remainder of the system is classified as non-safety-related. The staff's review is based on Revision 3 of the DCD.

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

The RBVS complies with the requirements of GDC 2, in that the isolation dampers, associated instrumentation, and duct are classified as safety-related and are designed to seismic Category I in accordance with RG 1.29, Regulatory Position C.1. The remainder of the system is classified as non-safety-related and is designed to seismic Category II in accordance with RG 1.29, Regulatory Position C.2, to ensure that the failure of the non-safety-related portions of the system cannot affect safety-related components.

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

GDC 60 requires that the nuclear power unit design include the means to control suitably the release of radioactive materials in gaseous and liquid effluents and to handle radioactive solid wastes produced during normal reactor operation, including anticipated operational occurrences.

The RBVS complies with the requirements of GDC 60 as to the system's 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 RB HVAC purge exhaust filtration units. The RB HVAC purge exhaust units are designed, tested, and maintained in accordance with RG 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.

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 systems of the RB HVAC are classified RTNSS and which components require cooling in the post-72-hour period. **RAI 9.4-39 is being tracked as an Open Item.**

The staff reviewed the equipment tables and figures provided with DCD Tier 2, Revision 3, Section 9.4.6 and identified issues needing clarification on safety-related dampers, release points, and inconsistencies.

DCD Tier 2, Revision 3, Table 9.4-10 states there are four safety-related isolation dampers for the REPAVS. Figure 9.4-11 (a simplified diagram) only shows two building isolation dampers. In RAI 9.4-41, the staff requested that the applicant show all four of the safety-related dampers on the figure and mark them as safety-related. **RAI 9.4-41 is being tracked as an Open Item.**

DCD Tier 2, Revision 3, Table 9.4-9 does not list safety-related isolation dampers. The staff asked the applicant to add this item to the table with the appropriate explanatory information. Figure 9.4-9 has a variety of dampers. In RAI 9.4-42, the staff requested that the applicant identify the isolation dampers that are safety-related and coordinate this information with Table 9.4-9. **RAI 9.4-42 is being tracked as an Open Item.**

DCD Tier 2, Revision 3, Table 9.4-9 shows two battery room exhaust fans. In RAI 9.4-43, the staff requested that the applicant include additional information on the ventilation of the battery rooms associated with Table 9.4-9. **RAI 9.4-43 is being tracked as an Open Item.**

DCD Tier 2, Revision 3, Table 9.4-11 states there are four safety-related isolation dampers for the CONAVS. Figure 9.4-10 (a simplified diagram) only shows two building isolation dampers. In RAI 9.4-44, the staff requested that the applicant show all four of the safety-related dampers on the figure and mark them as safety-related. **RAI 9.4-44 is being tracked as an Open Item.**

DCD Tier 2, Revision 3, Table 9.4-11 lists a main steam tunnel AHU, a recirculation AHU, and a refueling machine control room recirculation AHU as part of the CONAVS equipment. In RAI 9.4-45, the staff asked the following:

A. Identify these items on the Figure 9.4-10 with respect to other CONAVS equipment.

B. The main steam tunnel AHUs are located in the Turbine Building. Are there safety-related dampers for isolation at the RB interface? Include them on the appropriate diagrams and equipment list.

C. The CONAVS AHUs are located in the FB. Are there safety-related dampers at the building interface? Include them on the appropriate diagrams and equipment list.

RAI 9.4-45 is being tracked as an Open Item.

DCD Tier 2, Revision 3, Figure 9.4-9 indicates that the smoke purge is exhausted directly to the environment. In RAI 9.4-46, the staff requested that the applicant include the building isolation dampers and note whether they are safety-related in 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 is being tracked as an Open Item.**

DCD Tier 2, Revision 3, Figure 9.4-9 shows that the CLAVS exhausts air directly outdoors. In RAI 9.4-47, the staff requested that the applicant identify how the CLAVS exhaust air is monitored for radiation. **RAI 9.4-47 is being tracked as an Open Item.**

The RBVS has adequate provision for maintaining a suitable environment for personnel access and equipment by providing recirculation and exhaust capabilities.

The system also has provision to exhaust smoke in the event of a fire consistent with the smoke management features of DCD Tier 2, Revision 3, Section 9.5.1.11. The RBVS has adequate instrumentation, controls, and alarms in the MCR for adverse radiological conditions, temperature conditions, differential pressure for filters, and airflow. Provisions exist for testing key parameters and inspecting components to ensure operating conditions and system integrity.

In RAI 9.4-5, 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. The staff found that the applicant's list did not clarify which components are designed to which codes and standards. In RAI 9.4-5 S01, the staff asked the applicant to identify which codes are specific to components or structures in each of the systems and to eliminate those codes that are not applicable to that system. **RAI 9.4-5 is being tracked as an Open Item.**

9.4.6.4 Conclusions

Because of the open items that remain to be resolved for this section, the staff was unable to finalize its conclusions regarding the acceptability of the RBVS.

9.4.7 Electrical Building HVAC System

9.4.7.1 Regulatory Criteria

The staff reviewed the Electrical Building HVAC System (EBVS) in accordance with SRP Section 9.4.3, which discusses auxiliary and radwaste ventilation systems. Conformance with the SRP acceptance criteria forms the basis for concluding whether the EBVS satisfies 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.
- NUREG-0696, "Functional Criteria for Emergency Response Facilities," for guidance in establishing the habitability requirements of the Technical Support Center (TSC).

9.4.7.2 Summary of Technical Information

The EBVS does not perform or ensure any safety-related function, has no safety design basis, and is classified as non-safety-related.

The 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 HVAC subsystem (EERVS), the Technical Support Center HVAC subsystem (TSCVS), and the diesel generators HVAC subsystem (DGVS).

The EERVS provides conditioned air to maintain acceptable temperatures for equipment and personnel comfort and habitability, provides a sufficient quantity of filtered fresh air for personnel, and limits the buildup of hydrogen in the non-safety-related battery rooms to less than 2-percent hydrogen by volume.

The EERVS provides a controlled environment for the Electric Equipment Building switchgear, cable spreading, 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 electric and electronic rooms. The recirculating system includes an AHU with filter, heating and cooling coils, and two redundant fans. Building air is returned/exhausted by two redundant fans. Dedicated exhaust fans are provided for the seven battery rooms.

The TSCVS provides a controlled environment for personal comfort and safety and for the proper operation and integrity of equipment in the TSC. The system also maintains the TSC at a slightly positive pressure with respect to the adjacent rooms and outside environment to minimize the infiltration of air. The TSCVS automatically switches to the recirculation mode if toxic gases or smoke are detected in the outside intake air. In this case, there may be no differential pressure between the TSC and the surrounding areas. The TSCVS detects and limits the introduction of airborne hazardous materials (radioactivity, smoke, chlorine gas, or other site-specific toxic gas) 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 onsite diesel generators provide electrical power to the TSCVS in case of a LOPP.

The DGVS 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 dissipates their concentration in the fuel oil day tank room.

9.4.7.3 Staff Evaluation

The staff's review focused on compliance with the GDC relevant to this system, as described in Revision 3 of the DCD. The system is classified as non-safety-related. In addition, the staff considered the guidance of NUREG-0696 and Section 4.6.6 of the Electric Power Research Institute's utilities requirement document.

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

The EBVS complies with the requirements of GDC 2 by incorporating the guidance of RG 1.29, Regulatory Position C.2 for the 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 ESBWR design is a single-unit station, and the requirements of GDC 5 are not applicable to the single-unit design.

The requirements of GDC 19 also apply to the design of a TSC. The ESBWR design is committed to providing a TSC that meets the requirements of Section 4.6.6 of the EPRI utilities requirement document, which states that a TSC must be provided that complies with the requirements of NUREG-0696. NUREG-0696 requires the TSC to supply the same level of radiological protection as that required by GDC 19 for the MCR; however, the TSC is not specifically committed to providing a safety-related environment in full compliance with GDC 19, which defines the control room habitability acceptance criteria. GDC 19 requires that a control room be provided from which actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions, including LOCAs. Adequate radiation protection must be provided to permit access and occupancy of the control room under accident conditions without personnel's receiving radiation exposures in excess of 0.05 Sv (5 rem) total effective dose equivalent (TEDE) as defined in 10 CFR 50.2 for the duration of the accident.

The staff has concerns about the degree to which the TSC complies with the provisions of NUREG-0696 and its ability to protect TSC personnel under accident conditions. In RAI 9.4-5 S01, the staff requested that the applicant clarify its compliance with the recommendations of NUREG-0696. According to the guidance in NUREG-0696, the TSC should be habitable under accident conditions. Therefore, it is necessary to verify the ability of the TSCVS to maintain the TSC at positive pressure with respect to the areas around it. It is also necessary to confirm that a filter system has been installed and that it meets appropriate test requirements. Thus in RAI 14.3-61, the staff requested the applicant to add an ITAAC to verify this design commitment. In RAI 14.3-61, the staff also requested that the applicant discuss the adequacy of flow and cooling design to meet normal and accident conditions both pre-72 hours and post-72 hours so that the staff can determine with reasonable assurance that sufficient system capability exists. The staff requested the applicant to include the design details in DCD Tier 2, Revision 3, Section 9.4.7 including identification of any RTNSS systems required to perform TSC habitability functions. **RAIs 9.4-25 and 14.3-61 are being tracked as Open Items.**

GDC 60 requires that the nuclear power unit design include the means to control suitably the release of radioactive materials in gaseous and liquid effluents and to handle radioactive solid wastes produced during normal reactor operation, including anticipated operational occurrences.

The EBVS 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 Diesel Generator Building HVAC has no source of radioactive materials in either particulate or gaseous form.

The EBVS has adequate provision for maintaining a suitable environment for personnel access and equipment by providing fresh air and temperature control. The system also has the provision to exhaust smoke in the event of a fire. The EBVS has adequate instrumentation that alarms in the MCR for adverse conditions, temperature conditions, differential pressure indicators, air flow indicators, and controls. Provisions exist for testing key parameters and inspecting components to ensure operating conditions and system integrity.

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 is being tracked as an Open Item.**

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 require postaccident cooling. **RAI 9.4-39 is being tracked as an Open Item.**

In RAI 9.4-5, 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. The staff found that the applicant's list did not clarify which components are designed to which codes and standards. In RAI 9.4-5 S01, the staff asked the applicant to identify which codes are specific to components or structures in each of the systems and to eliminate those codes that are not applicable to that system. **RAI 9.4-5 is being tracked as an Open Item.**

DCD Tier 2, Revision 3, Section 9.4.7.6, notes that combined license (COL) applicants are responsible for the amount and location of possible sources of toxic chemicals in or near the plant and for TSC toxic gas monitoring, as required. COL applicants also are responsible for verifying that TSC toxic gas protection equipment, procedures, and training for TSC habitability are consistent with NUREG-0696, RG 1.78, "Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release," and the intent of Generic Issue 83, "Control Room Habitability," as applicable. RG 1.78 addresses control room protection for toxic chemicals and evaluation of offsite toxic releases (including the potential for toxic releases beyond 72 hours) in order to meet the requirements of TMI Action Plan Item III.D.3.4 (associated with control room habitability) and GDC 19. These are **COL Action Items**.

9.4.7.4 Conclusions

Because of the open items that remain to be resolved for this section, the staff was unable to finalize its conclusions regarding the acceptability of the EBVS.

9.4.8 Drywell Cooling System

9.4.8.1 Regulatory Criteria

The staff reviewed the drywell cooling system (DCS) in accordance with SRP Section 9.4.3, which discusses auxiliary and radwaste ventilation systems. Conformance with the SRP acceptance criteria forms the basis for concluding whether the DCS satisfies 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.

9.4.8.2 Summary of Technical Information

The DCS maintains the thermal environment within the drywell to specified conditions during normal reactor operation, hot standby, and refueling using fan cooling units (FCUs).

The cooling medium of the FCUs is water from the CWS. Separate FCUs service the upper and the lower drywell regions.

The DCS is classified as a non-safety-related, 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 the temperature in the upper and lower drywell spaces within specified limits, accelerates drywell cooldown during the period from hot shutdown to cold shutdown, and aids in complete purging of nitrogen from the drywell during shutdown. The DCS also maintains a habitable environment for plant personnel during plant shutdowns for refueling and maintenance and limits the drywell temperature during a 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 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 drywell heat loads are transferred to the nuclear island subsystem of the CWS circulating 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. Each upper drywell FCU has a cooling capacity of 50 percent of the upper drywell design heat load during normal plant operating conditions. Both FCUs are normally operating. Each FCU comprises a cooling coil and two fans downstream of the coil. The nuclear island subsystem of CWS train A supplies one FCU, and the nuclear island subsystem of CWS train B supplies the other. One of the fans operates while the other is on standby status. The fan on standby automatically starts upon loss of the lead fan.

Cooled air/nitrogen leaving the FCUs enters a common plenum and is distributed to the various zones in the upper drywell through distribution ducts. Return ducts are not provided. The FCUs draw air/nitrogen directly from the upper drywell.

Each lower drywell FCU has a cooling capacity of 50 percent of the lower drywell design heat load. Each FCU comprises a cooling coil and two fans downstream of the coil. One of the fans operates while the other is on standby status. The fan on standby automatically starts upon loss of the lead fan.

The nuclear island subsystem of CWS train A supplies one FCU, while the nuclear island of subsystem CWS train B supplies the other. Cooled air/nitrogen is supplied below the RPV and in the RPV support area through supply ducts. Return ducts are not provided. The 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 lower drywell is piped to a leak detection and isolation system (LD&IS) flow meter to measure the condensation rate contribution to unidentified leakage. The piping for trains A and B of the nuclear island subsystem of the CWS independently penetrate the containment. The cooling coils of one FCU in the upper drywell and one FCU in the lower drywell are piped in parallel to the nuclear island subsystem of CWS train A, and the remaining two are piped in parallel to the nuclear island subsystem of CWS train B.

The system is designed so that both FCUs in the upper drywell and both FCUs in the lower drywell are always operating during normal plant operation, assuming the loss of a single electrical group or 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 testing.

9.4.8.3 Staff Evaluation

The staff's review focused on compliance with the GDC relevant to this system, as described in Revision 3 of the DCD. The system is classified as non-safety-related.

GDC 2 requires that SSCs important to safety be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches without loss of capability to perform their safety functions. The DCS complies with the requirements of GDC 2 by conforming to the guidance in RG 1.29, Regulatory Position C.2, for the non-safety-related portions of the system. The DCS components are designated as seismic Category II.

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

GDC 60 requires that the nuclear power unit design include the means to control suitably the release of radioactive materials in gaseous and liquid effluents and to handle radioactive solid wastes produced during normal reactor operation, including anticipated operational occurrences. The DCS 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 FCUs recirculate air/nitrogen inside the upper and lower drywell. The recirculated air/nitrogen is retained in the primary containment structure. The liquid condensate from the fan cooling coils is collected and measured by the LD&IS to determine the condensation rate contribution to the unidentified leakage.

The DCS has adequate provision for maintaining acceptable temperatures in the drywell. The DCS has adequate instrumentation that alarms in the MCR in the event of a temperature rise in the drywell, failure of a fan, or a low flow alarm. Provisions exist for leak testing of cooling coils and confirming the integrity of the system.

In RAI 9.4-5, 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. The staff found that the applicant's list did not clarify which components are designed to which codes and standards. In RAI 9.4-5 S01, the staff asked the applicant to identify which codes are specific to components or structures in each of the systems and to eliminate those codes that are not applicable to that system. **RAI 9.4-5 is being tracked as an Open Item.**

9.4.8.4 Conclusions

Because of the open item that remains to be resolved for this section, the staff was unable to finalize its conclusions regarding the acceptability of the DCS.

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 3, 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,” of NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants” (hereafter referred to as the SRP). The applicant’s fire protection program (FPP) is acceptable if it meets the regulatory requirements applicable to the plant FPP. These requirements and the regulatory guidance for meeting them are as follows:

- Title 10, Section 50.48(a), of the *Code of Federal Regulations* (10 CFR 50.48(a)) requires that the holders of a combined license issued under Part 52 have a fire protection plan that meets 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”; that the plan describe specific features necessary to its implementation; and that the licensee retain the plan and all changes to it as records until the (NRC or the Commission) terminates the reactor license.

- Under 10 CFR 52.47(a)(9) the application must include an evaluation of the facility against the SRP revision in effect 6 months before the docket date of the application.

- GDC 3 requires the following:

  – SSCs important to safety be designed and located to minimize, consistent with other safety requirements, the probability and effect of fires and explosions.

  – Noncombustible and heat resistant materials be used wherever practical throughout the unit.

  – Fire detection and fighting systems of appropriate capacity and capability be provided and designed to minimize the adverse effects of fires on SSCs.

  – Fire fighting systems be designed to assure that their rupture or inadvertent operation does not significantly impair the safety capability of these SSCs.

- GDC 19, “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)(I) 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.

- 10 CFR 52.48, "Standards for Review of Applications," requires that the application for a certified design be reviewed for compliance with the standards set out in 10 CFR Part 20, 50 and its appendices, 51, 73, and 100.

- SRP Section 9.5.1, Revision 4, which includes Branch Technical Position (BTP) 9.5 1, "Guidelines for Fire Protection of Nuclear Power Plants," contains guidance and acceptance criteria for an FPP that meets the regulatory requirements described above.

- RG 1.189, "Fire Protection for Operating Nuclear Power Plants," provides additional guidance and acceptance criteria that comprise one acceptable approach for meeting the regulatory requirements for an FPP. The Commission directed this RG at currently operating plants; however, much of the guidance and acceptance criteria are applicable to new reactor designs.

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 key ESBWR fire protection design commitments by GEH that are set forth in the ESBWR DCD, Revision 3. Statements of evaluation included in this section are from the DCD; Section 9.5.1.3 of this report provides the staff's evaluation of the design.

9.5.1.2.1 Technical Design Bases—General

The FPS is the integrated complex of equipment and components that provide early fire detection and suppression to limit the spread of fires. The FPS is part of the overall FPP, which includes the plant design and layout as well as administrative controls and procedures to prevent or mitigate fires. In accordance with SRP Section 9.5.1 and RG 1.189, the FPP uses the concept of defense in depth to achieve the required degree of reactor safety through administrative controls, 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, some FPS equipment and structures have elevated seismic and quality classifications.

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" (2001 edition).

In addition to its fire protection functions, the FPS can perform a non-safety-related defense-in-depth function by serving as a 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 SFP and for reactor water inventory control following a design-basis accident. If required, the makeup function will begin approximately 72 hours after a loss-of-coolant accident (LOCA). The required minimum makeup rate will not exceed 46 cubic meters per hour (m^3/h) (200 gallons per minute (gal/min)), 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 under the RTNSS process rather than safety-related since it is not required until at least 72 hours after the LOCA.

SECY-90-016, SECY-93-087, and SECY-94-084 provide enhanced fire protection criteria for advanced reactor designs. Under these enhanced criteria, new reactor designs must ensure that safe shutdown can be achieved assuming that all equipment in any one fire area is rendered inoperable by fire and reentry into the fire area for repairs and operator actions is not possible. Because of its physical configuration, the control room is excluded from this approach, provided that the design includes an independent alternative shutdown capability that is physically and electrically independent of the control room. New reactor designs must 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 must 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 the intent of these criteria.

The ESBWR FPP design bases include the requirement 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 require any alternating current (ac) power or other support systems such as cooling water. Operation does not require 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 required 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.

9.5.1.2.2 Materials of Construction—General

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 fire hazards analysis (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.

9.5.1.2.3 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 3, 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 requirements of NUREG-1552, "Fire Barrier Penetration Seals in Nuclear Power Plants," including Supplement 1. A representative of a qualified, independent testing laboratory performs or witnesses the tests. 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 3, Sections 9.5.1.11 and 9.5.1.16 identify, this as a **COL Action Item.**

9.5.1.2.4 Water Supply and Fire Pumps

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 3, Sections 9.5.1.4 and 9.5.1.16, identify this as a **COL Action 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 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."

9.5.1.2.5 Piping

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 Action 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 3, Sections 9.5.1.5 and 9.5.1.16, identify this as a **COL Action Item.**

9.5.1.2.6 Suppression and Detection Systems—General

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) require 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 required 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.

9.5.1.2.7 Manual Suppression System—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.

9.5.1.2.8 Detection and Alarm Systems

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 are such that the failure, removal, or replacement of a detector does not affect the performance of the fire detection loop.

9.5.1.2.9 Circuit/Cable Protection from Fire

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. However, if there are alternate means of control or indication that remain unaffected by the same fire, then exception to the BTP SPLB 9.5-1 requirements for circuit routing and separation may be taken. The alternate means of control or indication are not required to be safety-related. All electrical cables (safety-related and non-safety-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 ESBWRs, relying instead on divisional separation by fire area and structural fire barriers.

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 required 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 MSIV.

- 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 requiring 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 requirement, cables for outboard containment isolation valves located in fire areas of a division different than that of the valve must not be 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.

9.5.1.2.10 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 55.6 ºC/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 regulatory expectations.

9.5.1.2.11 Spill Control Features

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 NFPA 804, "Standard for Fire Protection for Advanced Light Water Reactor Electric Generating Plants," and the guidance in RG 1.189, spill containment and drainage facilities for a given area are sized to accommodate the volume of liquid produced by all 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.

9.5.1.2.12 Miscellaneous

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.

9.5.1.2.13 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 a review of every fire area, using the defense in depth approach from NFPA 804 and RG 1.189. The aim of defense in depth is to provide a high degree of fire protection by implementing three concepts. These concepts include (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.”

GEH 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 require 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 ESBWR FHA identifies and evaluates the hazard of fires relative to maintaining the safe-shutdown capability of the plant. The FHA does not evaluate the overall FPP for an ESBWR, since many of an effective program's elements are administrative or procedural in nature, but rather assumes that an ESBWR owner will have an effective FPP in place. As described in RG 1.189, the primary objectives of an FPP at a nuclear plant are to minimize both the probability of occurrence and the consequences of fire. To meet these objectives, the FPP design should provide reasonable assurance, through defense in depth, that a fire will not prevent the performance of necessary safe-shutdown functions and that radioactive releases to the environment in the event of a fire will be minimized.

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 required for safe shutdown.

GEH 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, GEH assumed loss of offsite power for the bounding analysis for a fire in the MCR that requires evacuation.

9.5.1.2.14 Exceptions to the Standard Review Plan and RG 1.189

The ESBWR FPS design follows the recommendations of BTP SPLB 9.5-1 or RG 1.189 with the following exceptions identified in the DCD:

- Fire detection is not provided within the electrical cabinets in the MCR.
- Fire suppression systems are not provided in the office areas of the MCR complex.
- Fire suppression systems are not provided under the raised floor in the MCR.
- The capacity of the standby diesel fuel oil day tanks located in the diesel generator (DG) area may exceed 4164 L (1100 gal).
- Suppression system discharge in DG rooms may affect operation of the DG.
- Fixed automatic fire suppression is not provided in safety-related computer rooms.

9.5.1.2.15 Exceptions/Alternative Approaches to Design Bases

The following are exceptions and alternative approaches to the FPP design bases:

Steam Tunnel Barrier

The steam tunnel in the RB opens into the turbine building without a barrier wall to allow venting of steam to the turbine building in the event of a major steam line leak in the steam tunnel within the RB. There are no openings in the steam tunnel that are not protected by either fire dampers or fire doors, including the walls, floors, and ceilings. An automatic, open-head water spray system serves as a water curtain fire barrier between the turbine building and RB portions of the steam tunnel.

Division Separation Criteria

A specific analysis is prepared for each fire area in the containment and MCR that has redundant systems of safety-related equipment or electrical cables. This analysis confirms that adequate protection has been provided by means of separation by distance, physical barriers, electrical isolation, electrical circuit characteristics, or adequate backup systems. DCD Tier 2, Revision 3, Section 9A.6, “Special Cases,” includes the analyses.

Penetration Requirements

The drywell inerting system supply ductwork (piping) for the wetwell and the drywell passes through a fire barrier but does not have fire dampers. These flow paths consist of two supply lines (each 350 mm (14 in.) nominal diameter) and two exhaust lines (one 350 mm (14 in.) nominal diameter and one 400 mm (16 in.) nominal diameter). There are two containment isolation valves for each supply and exhaust piping penetration. The isolation valves are normally closed except during plant outage periods, when smoke removal could be accomplished without interruption if a fire occurs.

Fire Barrier Walls

The wall descriptions below represent anticipated alternative approaches from tested and approved 3-hour fire-resistive assemblies. The designs were previously submitted and approved in the General Electric Standard Safety Analysis Report (GESSAR) II.

The Type 1 wall design exceeds the design of the tested and approved assembly from which it was adapted. The Type 2 assembly requires a UL test.

The Type 1 wall is designed with 0.15-m (6-in.) metal studs at 0.30 m (12 in.) on center (seismic design) with three layers of 0.016-m (0.625-in.) fire code gypsum board on each side of the studs. The design is adapted from the International Conference of Building Officials standard for a 3-hour partition. The only difference from the standard is the increased gauge thickness of the structural members to meet higher seismic requirements.

The Type 2 wall is a variation of Type 1 with 0.15-m (6-in.) metal studs at 0.30 m (12 in.) on center between 0.15-m (6-in.) steel, wide-flange columns at 1.22 m (4 ft) on center. Three layers of 0.016-m (0.625-in.) fire code gypsum board line one side while 0.013-m (0.5-in.) thick steel plate for bullet resistance and two layers of 0.016-m (0.625-in.) fire code gypsum board line the other side.

Fire Doors

Certain doors throughout the facility have a multipurpose function, such as protecting against fire, tornado, pressure, missiles, or seismic events or maintaining watertightness and/or airtightness. Where possible, these doors are rated and/or labeled doors and are identified as rated doors. When other criteria require the manufacturer to design the door for some other purpose, the door is identified as equivalent to a fire-rated door. The doors, except for the RB equipment access door, must have a UL or FM label. Where the door is not constructed as a fire door, such as a containment personnel air lock, it is identified by its main function.

The use of 1.5-hour fire-rated elevator doors in 3-hour fire-rated barriers does not compromise the fire barrier. Section 6.1.2.2 of NFPA 804 specifically allows 1.5-hour fire-rated doors in elevator shafts. No other applicable codes require elevator doors to have a fire rating of more than 1.5 hours. None of the applicable codes address 3-hour fire-rated elevator shafts. It is not unusual for a door in a fire-rated wall to have a lower fire rating than the applicable firewall, because the area on both sides of the door will normally be kept free of combustible material to enable use of the door.

Basemats

In general, concrete basemats are not required to be fire rated because of the lack of any fire hazard in the ground beneath them. The substantial thickness of concrete basemats would provide a large fire rating, if one were required.

Smoke Removal

In general, the HVAC system operating in the purge or exhaust mode can exhaust smoke, heat, and products of combustion once the fire has been extinguished by the FPS.

9.5.1.2.16 Inspections, Tests, Analyses, and Acceptance Criteria

The DCD Tier 1, Revision 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.

9.5.1.3 Staff Evaluation

The staff reviewed the portions of the ESBWR DCD Revision 3, related to the ESBWR FPP included in Tier 1 and Tier 2. The FPP is primarily described in Tier 1, Section 2.16.3, “Fire Protection System,” and Tier 2, Section 9.5.1, “Fire Protection System,” including Appendix 9A, “Fire Hazards Analysis,” and Appendix 9B, “Summary of Analysis Supporting Fire Protection Design Requirements.” Additional portions of DCD Tier 2 addressing aspects of the ESBWR FPP that the staff reviewed for compliance with regulatory requirements for FPPs include the following:

- Section 1.2.2.16.3, “Fire Protection System”
- Section 1.9, “Conformance with Standard Review Plan and Applicability of Codes and Standards,” including the following tables:

- Table 1.9-9, “Summary of Differences from SRP Section 9”
- Table 1.9-20, “NRC Standard Review Plans and Branch Technical Positions Applicability to ESBWR”
- Table 1.9-21, “NRC Regulatory Guides Applicability to ESBWR”
- Table 1.9-22, “Industrial Codes and Standards Applicable to ESBWR”

- Section 1.10, “Summary of COL Items”
- Table 1.11-1, “Resolutions to NUREG-0933 Table II Task Action Plan Items, New Generic Issues, Human Factors Issues and Chernobyl Issues”
- Table 1C-1, “Operating Experience Review Results Summary—Generic Letters”
- Section 3.1.1.3, “Criterion 3—Fire Protection”
- Chapter 7, “Instrumentation and Control Systems”
- Section 8.3.1.4, “Independence of Redundant Systems”
- Section 8.3.3, “Fire Protection of Cable Systems”
- Section 8.3.4, “COL Information (Electrical)”
- Section 14.2, “Initial Plant Test Program for Final Safety Analysis Reports”
- Section 15.5.6, “Safe Shutdown Fire”
- Section 19.2.3.2, “Risk from External Events”
- Section 19.2.3.3, “Evaluation of External Event Flood”
- Section 19.2.4.1.2, “Fire During Shutdown”
- Section 19.3.1.4, “Fire Protection”
- Appendix 19A, “Regulatory Treatment of Non-Safety Systems (RTNSS)”

With the exception of the open items, discussed below, the staff has found the ESBWR FPP as described in the DCD Revision 3 in accordance with applicable regulatory requirements and guidance.

The staff concludes that the ESBWR FPP design criteria and the commitments for their implementation as described in the DCD are acceptable and meet the applicable requirements of 10 CFR Part 50 and 10 CFR Part 52, “Licenses, Certifications, and Approvals for Nuclear Power Plants”, as well as other acceptance criteria identified in Section 9.5.1.1 of this report, and are consistent with Commission policy contained in SECY-90-016, SECY-93-087, and SECY-94-084 (plants with passive safe shutdown). The staff bases this conclusion on the

applicant's meeting the guidelines of the applicable RGs and related industry standards as described in the DCD.

By meeting these guidelines, the applicant will ensure that safe shutdown can be achieved assuming that all equipment in any one fire area (excluding the control room and reactor containment) will be rendered inoperable by fire and that reentry into the fire area for repairs and operator actions is not possible. The applicant's design provides 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 the fire suppressant will not migrate into other fire areas to the extent that they 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.

The staff concludes that for differences between the licensee's FPP and the SRP acceptance criteria, three of six proposed alternatives provide an acceptable method of complying with the NRC regulations as discussed below. The remaining three are associated with open items. The staff also concludes that the exceptions and alternative approaches to the FPP design bases for the ESBWR FPP, as described in Section 9.5.1.2.15 of this report, are acceptable.

The staff agrees that because of the considerable separation 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 is not necessary. The applicant notes that this type of analysis could be performed later for an individual fire area if needed.

The ESBWR DCD 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.

9.5.1.3.1 Staff Acceptance of Alternative Approaches to SRP Acceptance Criteria

The DCD describes some potential alternative approaches to the SRP acceptance criteria for SSCs that the COL applicants will specify. These include qualified fire rating of special doors (e.g., air locks and flood-proof doors). The staff will assess the acceptability of these alternative approaches during the COL application review when specific design and qualification details are available consistent with the COL Action Items described in DCD Tier 2, Revision 3, Section 9.5.1-16.

Some other alternative approaches are tentatively acceptable to the extent that the design is described in the DCD, subject to resolution of Open Items discussed below. These alternative approaches are as follows:

- Rooms adjacent to the MCR will not have installed sprinkler protection. 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. In requests for additional information (RAIs) 9.5-45 S02 and 9.5-46 S02, the staff stated that the information in the FHA is incomplete and requested that GEH provide the FHA information identified in NFPA 804 and RG 1.189. **RAIs 9.5-45 and 9.5-46 are being tracked as Open Items.**

- The area below the raised floor in the MCR will not have installed sprinkler protection. This area contains no medium- or high-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 or very slow to spread. 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. In RAIs 9.5-45 S02 and 9.5-46 S02, the staff stated that the information in the FHA is incomplete and requested that GEH provide the FHA information identified in NFPA 804 and RG 1.189. **RAIs 9.5-45 and 9.5-46 are being tracked as Open Items.**

- Individual electrical cabinets and consoles in the MCR will not have installed smoke detectors inside the enclosures. 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. The DCD states that this alternative approach will be used unless it is identified as a significant fire hazard in the FHA. In RAIs 9.5-45 S02 and 9.5-46 S02, the staff stated that the information in the FHA is incomplete and requested that GEH provide the FHA information identified in NFPA 804 and RG 1.189. **RAIs 9.5-45 and 9.5-46 are being tracked as Open Items.**

- The DG indoor day tanks will likely exceed the limit recommended by RG 1.189 for indoor DG day tanks. However, the DGs of the ESBWR are non-safety-related and are not required 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 water-based automatic fixed suppression systems in each DG room are not designed to ensure continued operation of the DGs in the event of system discharge. Since the DGs are non-safety-related and are not required to maintain safe-shutdown conditions for the 72-hour period following a fire event and since the suppression system is a preaction type, this alternative approach is acceptable.

- ESBWR computer rooms that contain safety-related equipment do not have fixed automatic fire suppression protection. This alternative approach is acceptable based on the low voltage of the equipment in these rooms, the low combustible loading (paper

materials within computer rooms are stored in file cabinets, bookcases, or other storage locations except when in use), the ionization type smoke detectors installed in the room, and the manual fire fighting capability in these areas. Outside the MCR complex, safety-related computers are located in divisional rooms separated from each other by 3-hour fire-rated barriers.

9.5.1.3.2 Fire Barrier Separation of Electrical Divisions

In some cases, it is not feasible to separate redundant divisions by 3-hour fire barriers in the ESBWR, and cables of more than one division are in relatively close proximity. DCD Tier 2, Revision 3, Section 9A.6.4 of Appendix 9A, describes these areas in detail and provides a justification for each alternative approach. The staff has reviewed each of these alternative approaches and their justifications and, to the extent that the design is described in the DCD, accepts them based on the following, as applicable:

- It is not feasible to provide fire-rated physical barriers between redundant divisions for these cases.
- The in situ combustible loading in each area is low.
- Electrical cable is generally contained within metal raceway in these areas.
- The containment is inerted during operation.
- Administrative procedures prohibit combustibles from being stored in areas under the reactor vessel with divisional cable or within electrical equipment areas.
- The proposed alternative configurations provide reasonable assurance that should a fire occur in these areas, safe shutdown can be achieved and maintained.

DCD Tier 2, Revision3, Section 9.5.1.3, notes that, “However, if alternate means of control or indication are provided or available that remain unaffected by the same fire, then exception to the BTP SPLB 9.5-1 requirements for circuit routing and separation may be taken.” Applicants should only rely on this approach where it is not feasible to provide full separation via 3-hour barriers. Applicants should identify any exceptions in the COL application in accordance with COL Action Item 9.5.1-7, and the staff will review them.

GEH has not developed the ESBWR post-fire safe-shutdown circuit analyses for the certified design. The licensee will develop these analyses as part of the plant-specific FPP after completion of the distributed control and information system platform and detailed electrical design. The FPP and supporting analyses will be constrained by the associated DCD inputs and assumptions. In RAI 9.5-44, the staff requested GEH to revise the DCD to state that the licensee will develop the post-fire safe-shutdown circuit analyses as part of the plant-specific FPP. The staff will review the post-fire safe-shutdown circuit analyses at the COL application stage. **RAI 9.5-44 is being tracked as an Open Item.**

9.5.1.3.3 COL Action Items

DCD Tier 2, Revision 3, Table 1.10-1, and DCD Tier 2, Revision 3, Section 9A.7, list the following fire protection COL action items.

- Provide Final Quantity and Capacity of Secondary Fire Water Storage,
- Provide the Final Secondary Fire Pump Flow Rate and Pump Head,
- Provide Design Characteristics of Yard Piping,
- Provide FPS P&IDs Showing Complete Site Specific Systems,
- Provide Specific Design and Certification Testing Details for Fire Barriers and Electrical Raceway Fire Barrier Systems,
- Establish Provisions for Manual Smoke Control by Manual Actions of the Fire Brigade for All Plant Areas,
- Compliance Review of As-Built FPS Against Fire Hazards Analysis Assumptions,
- Provide Complete Description of Applicant's Fire Protection Program,
- Provide the Proposed Fire Protection Fire Zone Drawings for Yard Areas,
- Provide the Proposed Fire Protection License Condition for Making Changes to the FPS Without Prior NRC Review and Approval,
- Provide Details of QA Program for Fire Protection Program,
- Fire Hazards Analysis for Yard Areas,
- Address Deviations from Any Approved Penetration Seal Design,
- Fire Zone Drawings for Those Portions of the Yard That Are Not Associated with Turbine and Electrical Building Equipment.

Section 9.5.1.3.4 of this report describes additional COL action items that the staff has requested GEH to add to the DCD. The COL applicant's satisfactory completion and description of the action items identified above an in Section 9.5.1.3.4 will provide the staff with sufficient information to assess the acceptability of the FPP for a COL. 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.

9.5.1.3.4 Open Items

The following RAIs are being tracked as open items:

<u>RAI 9.5-44</u>

The GEH response to RAI 15.5-3 states that "The ESBWR post-fire, safe shutdown circuit analyses have not been developed at this time. These analyses will be developed later in the project life cycle as part of the plant specific fire protection program." Consequently, the DCD should be revised to state that the post-fire, safe shutdown circuit analyses will be developed by the licensee as part of the plant specific fire protection program.

RAI 9.5-45 and 9.5-46

Contrary to GEH's response that the fire hazard analysis provided in the ESBWR design certification document provides all of the information needed for a fire hazards analysis in accordance with NFPA 804 and RG 1.189, the following information listed in these two guides has not been provided to the extent described:

RG 1.189

1. Amounts, types, configurations and locations of flammable and combustible materials – in situ and transient.
2. Layout and configuration of structures, systems and components important to safety.
3. Accessibility of plant areas for manual fire fighting, location and type of manual fire fighting equipment
4. Lack of adequate access or smoke removal facilities that impede plant operations or fire extinguishment in plant areas important to safety.

NFPA 804

1. All in situ combustibles and flammable materials and their configurations should be identified. Where in situ combustibles present an exposure to nuclear safety-related systems and components, they should be uniquely identified.
2. Physical construction and layout of the buildings and equipment.
3. Description and location of any equipment necessary to ensure a safe shutdown, including cabling.
4. Analysis of smoke control system and the impact smoke can have on nuclear safety and operation for each area.
5. Analysis of the emergency planning and coordination requirements necessary for effective loss control, including any necessary compensatory measures to compensate for the failure or inoperability of any active or passive FPS or feature. As noted in the GEH response to RAI 9.5-46, RG 1.206 does acknowledge that this information may not be available even at the COL stage, but the RG also stipulates that if the information is not available, the applicant should justify the inability to provide the unavailable information in the COL application, and furnish details describing implementation plans, milestones and sequences and/or ITAAC or commitments for developing, completing, and submitting this information during the construction period, prior to fuel receipt at site. This should be identified in the DCD as a COL action item.

While this level of information may not be important in areas that provide complete 3-hour fire barrier separation between redundant post-fire safe-shutdown trains, as a minimum, the information should be developed and documented by the applicant for areas where full compliance with the criteria for enhanced fire protection is not feasible. In particular, the level of information described above should be provided for the MCR and adjacent rooms where exceptions have been taken to the regulatory guidance for fire protection, as well as for the other "Special Cases" described in DCD Section 9A.

The following RAIs are in reference to Chapter 19 of the DCD, but are for FPP issues:

RAI 19.1-150

The GEH response to RAI 19.1-150 includes seemingly contradictory statements. The response first states that "The ESBWR is designed to prevent spurious actuations that could adversely affect the capability to achieve and maintain safe shutdown" (emphasis added). Following this statement, GEH wrote that "The ESBWR design features described in DCD Tier 2, Section 7.1.3 help minimize the adverse effect on safe shutdown caused by fire-induced spurious actuations" (emphasis added). The response continues with two statements that begin with the word "Typically" and concludes that a fire in a single area cannot cause spurious actuations. This leaves open the question of what spurious actuations can occur where the typical design is not implemented. The DCD should clearly identify and address all potential fire-induced spurious actuations that could adversely affect safe shutdown or include a statement that none exist. If multiple spurious actuations can occur from a fire in a single fire area, the DCD should address how these multiple spurious actuations will be evaluated in the post-fire, safe-shutdown circuit analysis, including any assumptions with respect to mitigating the impact of each actuation.

The GEH response also states that "Since the main control room communicates with the DCIS rooms via fibers, no spurious actuation will be originated from a MCR fire." The DCD should confirm that these communication links do not include any copper or other wire conductors that could cause fire-induced spurious actuations that could adversely affect safe shutdown. It should also confirm that exposure of the distributed control and information system (DCIS) equipment to heat and smoke cannot cause spurious actuations that could adversely affect safe shutdown, including the bases for that confirmation.

RAI 19.1-151

The GEH response to this RAI states that "The performance of the compensatory manual actions for safe shutdown is not credited in the ESBWR fire PRA model for a postulated fire in the main control room." However, the GEH response to RAI 15.5-4 states that "After the operator regains the control in the remote shutdown panel (RSP), the manual action may be necessary to control Isolation Condenser System (ICS) to ensure the maximum cooldown rate not exceeding 100 ºF/hr." GEH should address the possible contradiction between these two statements, and the DCD should clearly identify any post-fire operator manual actions required for safe shutdown and state the design bases with respect to crediting specific operator manual actions.

RAI 19.1-152

Section 19A.4.2.4 states that the portion of the FPS that provides makeup to the FAPCS will "remain operable following a seismic event (Seismic Category II)." However, DCD Tier 2, Section 3.7 states that "Seismic Category II (C-II) includes all plant SSCs which perform no safety-related function, and whose continued function is not required." Since the continued function of FPC makeup to the FAPCS is required, the C-II classification would not necessarily support the performance requirement of the system. GEH should provide clarification.

9.5.1.3.5 Confirmatory Items

None.

9.5.1.4 Conclusions

Because open items remain to be resolved for this section, the staff is unable to finalize its conclusions regarding the compliance of the FPP with the requirements of 10 CFR 50.48, "Fire Protection"; GDC 3; and 10 CFR Part 52.

9.5.2 Communications System

The staff will prepare this section of the SER in coordination with Chapter 7 of the SER, "Instrumentation and Controls."

9.5.3 Plant Lighting System

9.5.3.1 Regulatory Criteria

The acceptance criteria in SRP Section 9.5.3 state that the acceptability of the normal, standby, and emergency lighting is based on the degree of similarity of the systems designed with those of previously reviewed plants with satisfactory operating experience. No GDC or RGs directly apply to the safety-related performance requirements for the 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 required for safe performance of plant operation, security, shutdown, and maintenance activities. Emergency lighting is provided in safety-related areas 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. The security lighting system is site specific and will be addressed by the COL applicant. 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 an SSE. The lighting circuits of the normal, standby, and emergency lighting subsystems are routed in separate conduits. The design of the lighting system for areas containing rotating equipment includes special provisions to eliminate the risk of stroboscopic effects caused by flicker. The circuits to the individual lighting fixtures (other than the 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 only incandescent lamps.

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 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 reflecting 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:

- MCR
- remote shutdown rooms
- operational support centers
- technical support centers
- auxiliary switchgear rooms
- Class 1E DC equipment rooms
- stairwells and aisle way
- DCIS equipment rooms
- 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.

Emergency Lighting

The emergency lighting system provides acceptable levels of illumination throughout the station, particularly in areas where emergency operations are performed, such as control rooms, battery rooms, and containment, on loss of the normal lighting system. The emergency lighting system is comprised of the following:

- control room emergency lighting

- 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 MCR emergency lighting is Class 1E seismic Category I. The standby and emergency lighting fixtures, switches, and associated cables used in the MCR are non-Class 1E.

The control room emergency lighting is integrated with standby lighting that consists of identical lighting fixtures and is supplied from the four divisions of 72-hour Class 1E uninterruptible ac power supply (UPS) system. The 72-hour Class 1E batteries and the standby onsite ac power supply system provide backup to the 72-hour Class 1E UPS. The MCR emergency lighting complies with human factor requirements by using semi-indirect, low-glare lighting fixtures.

In areas outside the MCR and remote shutdown area, emergency lighting is provided by 8-hour, self-contained battery pack, sealed-beam lighting units. These units are powered from the non-Class 1E and provide illumination for safe ingress and egress of personnel following a loss of normal lighting in areas that are required for power restoration/recovery. In addition, these units are used in areas where normal actions are required 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 requires the operation of the unit.

9.5.3.3 Staff Evaluation

DCD Tier 2, Revision 3, Section 9.5.3.3.3.2, states that each emergency lighting unit is capable of supplying sealed beam lamps for 8-hours without the charger. However, there are 2-hour-rated units and 90-minute-rated units in different applications. In RAI 9.5-58 the staff asked the applicant to clarify the discrepancy. In response the applicant stated that the 90-minute-rated units are used for exit signs only and the 8-hour-rated units are used in areas outside the MCR. The applicant clarified that 2-hour-rated units are not used in any area of the plant. The applicant will revise the first bulleted item under DCD Tier 2, Revision 3, Section 9.5.3.3.3.2, to delete the use of 2-hour-rated units. On the basis of its review, the staff finds that the applicant adequately addressed the concern. **RAI 9.5-58 is being tracked as a Confirmatory Item.**

DCD Tier 2, Revision 3, Section 9.5.3.3.3.2, states that 2-hour-rated units as a minimum are used in other areas of the plant. In RAI 9.5-59, the staff asked the applicant to clarify where the 2-hour-rated units will be used. In response the applicant stated that the 2-hour-rated units are not used in any area of the plant. On the basis of its review, the staff finds that its concern is resolved.

The staff finds that emergency lighting supplied by the 72-hour Class 1E UPS system is not used in remote shutdown areas. 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. On the basis of its review, the staff finds that the applicant adequately addressed the concern, and therefore the concern is resolved.

MCR emergency lighting is supplied from the Class 1E UPS. The lighting fixture, switches, 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, Section 9.5.3.3.3.1, is to be revised in its entirety for clarity and to add isolation requirements ("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 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 are being tracked as Open Items.**

DCD Tier 2, Revision 3, Section 9.5.3.3.3.1, states that the MCR emergency lighting is supplied from four divisions of 72-hour Class 1E UPS. In RAI 9.5-62, the staff asked the applicant to discuss the separation requirement between four divisions of UPS supplies and cables outside the MCR. In response the applicant stated that the four divisions of 72-hour Class 1E UPS are independent, 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. On the basis of its review, the staff finds that the applicant adequately addressed the concern and, therefore, the concern is resolved.

DCD Tier 2, Revision 3, Section 9.5.3.4, states that the MCR emergency lighting system is safety-related and classified as Class 1E. Also, in DCD Tier 2, Revision 3, Section 9.5.3.1, the applicant states that the MCR emergency lighting system is Class 1E. However, DC Tier 2, Revision 3, Section 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 are being tracked as Open Items.**

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 Section 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.

On the basis of its review, the staff finds that the applicant adequately addressed the concern. **RAI 9.5-64 is being tracked as a Confirmatory Item.**

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, with the exception of the open items discussed above, to be sufficient to meet the guidance of SRP Section 9.5.3.

9.5.3.4 Conclusions

Because open items remain to be resolved for this section, the staff is unable to finalize its conclusions regarding the compliance of the plant lighting system with applicable staff positions and industry standards.

9.5.4 Diesel Generator Fuel Oil Storage and Transfer System

9.5.4.1 Regulatory Criteria

The staff reviewed the ESBWR diesel generator fuel oil storage and transfer system (DGFOSTS) in accordance with the guidance of Section 9.5.4, "Emergency Diesel Engine Fuel Oil Storage and Transfer System," of the SRP, Revision 2, dated July 1981. The staff's acceptance of the DGFOSTS is based on the design's conformance with the requirements of the following GDC:

- 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 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.

Compliance with GDC 2 is based on meeting the guidance of Regulatory Positions C.1 and C.2 of RG 1.29, "Seismic Design Classification."

9.5.4.2 Summary of Technical Information

There are two redundant onsite DG units in the ESBWR design for a single unit plant to provide power to selected non-safety-related ac loads in the event of a loss of normal and preferred ac power supplies. Also, during the post-LOCA with a complete loss of ac power supplies, the DG units provide power for recharging batteries to support post-LOCA monitoring beyond 72 hours. Each DG unit is an independent system completed with its support systems, which are the DGFOSTS, jacket cooling water system, starting air system, lubrication system, and combustion air intake and exhaust system. Each DG unit is housed in a separated enclosure. The design provides adequate separation between the two DG units.

The DGFOSTS is designed to supply the day tank with sufficient fuel oil for a minimum of 8 hours of DG operation at full load and sufficient fuel oil onsite for DG operation at full load for 7 days without replenishing. The primary components of this system 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 tanks 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.

The DGFOSTS has piping connections to supply fuel oil to the auxiliary boiler system. The piping connections for the auxiliary boiler tie into the diesel oil storage tank at an elevated nozzle connection. This location ensures that fuel storage quantity stored below this level will not be affected by auxiliary boiler system usage and that the 7-day diesel fuel oil storage requirements cannot be used for any other purposes. Administrative controls ensure a minimum of fuel oil capacity is maintained on site at all times.

The DGFOSTS piping and components up to the engine skid connection are designed and constructed in accordance with the ASME Boiler and Pressure Vessel Code, Section VIII, Division 1, and ASME B31.1. The stored fuel oil meets the requirements of the ASTM D975, "Standard Specification for Diesel Fuel Oils," and the requirements of the diesel engine manufacturer. Routine testing ensures the quality of the fuel oil for the diesel engines. Each fuel oil storage tank is emptied and accumulated sediments are removed every 10 years to conform to Federal and State examination requirements.

9.5.4.3 Staff Evaluation

Each DG and its support systems, including the DGFOSTS, are non-safety-related, do not perform any safety-related function, and are not credited in any safety analysis. They are not required to achieve or maintain safe shutdown of the plant. Also, their failures do not lead to the failure of any SSCs important to safety. The staff, therefore, determined that the requirements of GDC 2, 4, and 17 do not apply to the DGFOSTS design.

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

The staff found that during a postulated post-LOCA and a complete loss of ac power supplies, the DG units are used to supply power for recharging batteries to support post-LOCA monitoring beyond 72 hours. Therefore, the DGs and their support systems meet Criteria B and C established in SECY-94-084 to apply to the passive ALWR plant design to determine the systems that are candidates for RTNSS consideration. The staff determined that they should be considered as candidates for RTNSS systems.

In RAI 19.1.0-2, the staff requested the applicant to provide documentation/analyses in support of the process used to identify RTNSS systems. In its response dated January 30, 2007, the applicant included the DG units as RTNSS systems. However, it was not clear to the staff that all the DG supporting systems would be considered as RTNSS systems. Subsequently, in RAI 22.5-4 the staff requested the applicant to clarify that all DG supporting systems were considered as RTNSS systems. In response, the applicant stated that all DG supporting systems were considered as RTNSS systems and that it will revise Section 9.5.4 of the DCD Tier 2, Revision 4, to classify the DGFOSTS as a RTNSS system.

In response to RAI 19.1.0-2 (MFN-07-066), GEH submitted DCD draft Appendix 19A, "Regulatory Treatment of Non-Safety System (RTNSS)." On July 15, 2007 (MFN-07-373), GEH submitted a revised version of draft DCD Appendix 19A. Draft Appendix 19A states that all RTNSS systems will be in the scope of the Design Reliability Assurance Program, as directed by DCD Tier 2, Revision 3, Chapter 17. On September 28, 2007, GEH submitted Appendix 19A in DCD Tier 2, Revision 4 and the staff is reviewing it. COL Action Item 17.4.13 identifies the requirements for the COL holder's Operational Reliability Assurance Program.

Draft Appendix 19A also provides an availability controls manual (ACM). Section 3.8.1 of the ACM provides availability controls for the standby DGs. The bases for Section 3.8.1 state that testing in accordance with availability control surveillance requirement (ACSR) 3.8.1.2 also demonstrates operability of the associated fuel oil transfer pump and necessary DG support system functions. **RAI 22.5-4 is being tracked as a Confirmatory Item.**

In RAI 14.3-151, the staff requested that the applicant include ITAAC for DGFOSTS. In its response, the applicant stated it would not include ITAAC for this system since it is non-safety-related. This response was not acceptable. After the applicant reclassified the DG systems as RTNSS systems, the staff again requested that the applicant include ITAAC for DGFOSTS. **RAI 14.3-151 is being tracked as an Open Item.**

In DCD Tier 2, Revision 3, Section 9.5.4.6, "COL Unit-Specific Information," GEH states that the COL applicant will describe the material and corrosion protection for the underground piping portion of the fuel oil transfer system. The COL applicant is responsible for establishing the procedural controls to ensure a minimum fuel oil capacity is maintained on site at all times. **This is a COL Action Item.**

9.5.4.4 Conclusion

Because open items remain to be resolved for this section, the staff is unable to finalize its conclusions regarding acceptability of the DGFOSTS.

9.5.5 Diesel Generator Jacket Cooling Water System

9.5.5.1 Regulatory Criteria

The staff reviewed the ESBWR diesel generator jacket cooling water system (DGJCWS) in accordance with the guidance of Section 9.5.5, "Emergency Diesel Engine Cooling Water System," of the SRP, Revision 2, dated July 1981. The staff's acceptance of the DGJCWS is based on the design's conformance with the requirements of the following GDC:

- GDC 2 requires in part that SSCs important to safety shall be designed to withstand the effects of natural phenomena such as earthquakes.

- GDC 4 requires in part that SSCs important to safety shall be designed to accommodate the effects of, and be compatible with, the environmental conditions associated with normal operation, maintenance, and testing and postulated accidents.

- GDC 5 requires that SSCs important to safety shall not be shared among nuclear power units unless it can be shown that such sharing will not significantly impair their ability to perform their safety functions including, in the event of an accident in one unit, an orderly shutdown and cooldown of the remaining units.

- GDC 17 requires in part that an onsite electric power system and an offsite electric power system shall be provided to permit functioning of SSCs important to safety.

- GDC 44, "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.

Compliance with GDC 2 is based on meeting the guidance of Regulatory Positions C.1 and C.2 of RG 1.29 concerning seismic classification.

9.5.5.2 Summary of Technical Information

Each DG unit has its own independent integrally mounted DGJCWS designed to maintain DG 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. The DGJCWS 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.

9.5.5.3 Staff Evaluation

Each DG and its support systems, including the DGJCWS, are non-safety-related, do not perform any safety-related function, and are not credited in any safety analysis. They are not required to achieve or maintain safe shutdown of the plant. Also, their failures do not lead to the failure of any SSCs important to safety. The staff, therefore, determined that the requirements of GDC 2, 4, 17, 44, 45, and 46 do not apply to the DGJCWS design.

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

As stated in Section 9.5.4 of this report, the applicant included the DG units as RTNSS. Also, in response to RAI 22.5-4, the applicant stated that all DG supporting systems were considered as RTNSS systems and that it would revise Section 9.5.5 of the DCD Tier 2, Revision 4, to classify the DGJCWS as an RTNSS system.

In response to RAI 19.1.0-2 (MFN-07-066), GEH submitted draft Appendix 19A. On July 15, 2007 (MFN-07-373), GEH submitted a revised version of draft DCD Appendix 19A. Draft Appendix 19A states that all RTNSS systems will be in the scope of the Design Reliability Assurance Program, as directed by DCD Tier 2, Revision 3, Chapter 17. On September 28, 2007, GEH submitted Appendix 19A in DCD Tier 2, Revision 4, and the staff is reviewing it. DCD Tier 2, Revision 3, COL Action Item 17.4.13 identifies the requirements for the COL holder’s Operational Reliability Assurance Program.

Draft Appendix 19A also provides an ACM. Section 3.8.1 of the ACM provides availability controls for the standby DGs. The bases for Section 3.8.1 state that testing in accordance with the surveillance requirement (ACSR 3.8.1.2) also demonstrates operability of the associated fuel oil transfer pump and necessary DG support system functions. **RAI 22.5-4 is being tracked as a Confirmatory Item.**

In RAI 14.3-151, the staff requested that the applicant include ITAAC for DGJCWS. In its response the applicant stated it would not include ITAAC for this system since it is non-safety-related. This response was not acceptable. After the applicant reclassified the DG systems as RTNSS systems, the staff again requested that the applicant include ITAAC for DGJCWS. **RAI 14.3-151 is being tracked as an Open Item.**

9.5.5.4 Conclusion

Because open items remain to be resolved for this section, the staff is unable to finalize its conclusions regarding acceptability of the DGJCWS.

9.5.6 Diesel Generator Starting Air System

9.5.6.1 Regulatory Criteria

The staff reviewed the ESBWRs diesel generator starting air system (DGSAS) in accordance with the guidance of Section 9.5.6, "Emergency Diesel Engine Starting System," of the SRP, Revision 2, dated July 1981. The staff's acceptance of the DGSAS is based on the design's conformance with the requirements of the following GDC:

- GDC 2 requires in part that SSCs important to safety shall be designed to withstand the effects of natural phenomena such as earthquakes.
- GDC 4 requires in part that SSCs important to safety shall be designed to accommodate the effects of, and be compatible with, the environmental conditions associated with normal operation, maintenance, and testing and postulated accidents.
- GDC 5 requires that SSCs important to safety shall not be shared among nuclear power units unless it can be shown that such sharing will not significantly impair their ability to perform their safety functions including, in the event of an accident in one unit, an orderly shutdown and cooldown of the remaining units.
- GDC 17 requires in part that an onsite electric power system and an offsite electric power system shall be provided to permit functioning of SSCs important to safety.

Compliance with GDC 2 is based on meeting the guidance of Regulatory Positions C.1 and C.2 of RG 1.29 concerning seismic classification.

9.5.6.2 Summary of Technical Information

The DGSAS is designed to supply sufficient compressed air for five consecutive DG starts without external power. It consists of two redundant 100-percent-capacity air compressors, an air receiver, a 100-percent-capacity air dryer, associated piping, and valves. Two redundant starting air valves, one in each engine starting air manifold, are provided for each engine. Failure of one valve does not affect the ability of the other valve to start the engine.

9.5.6.3 Staff Evaluation

Each DG and its support systems, including the DGSAS, are non-safety-related, do not perform any safety-related function, and are not credited in any safety analysis. They are not required to achieve or maintain safe shutdown of the plant. Also, their failures do not lead to the failure of

any SSCs important to safety. The staff, therefore, determined that the requirements of GDC 2, 4, and 17 do not apply to the DGSAS design.

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

As stated in Section 9.5.4 of this report, the applicant included the DG units as RTNSS. Also, in its response dated August 2, 2007, to RAI 22.5-4, the applicant stated that all DG supporting systems were considered as RTNSS systems and that it will revise Section 9.5.5 of the DCD Tier 2, Revision 4, to classify the DGSAS as an RTNSS system.

In response to RAI 19.1.0-2 (MFN-07-066), GEH submitted draft Appendix 19A. On July 15, 2007 (MFN-07-373), GEH submitted a revised version of draft DCD Appendix 19A. Draft Appendix 19A states that all RTNSS systems will be in the scope of the Design Reliability Assurance Program, as directed by DCD Tier 2, Revision 3, Chapter 17. On September 28, 2007, GEH submitted Appendix 19A in DCD Tier 2, Revision 4, and the staff is reviewing it. DCD Tier 2, Revision 3, COL Action Item 17.4.13 identifies the requirements for the COL holder's Operational Reliability Assurance Program.

Draft Appendix 19A also provides an ACM. Section 3.8.1 of the ACM provides availability controls for the standby DGs. The bases for Section 3.8.1 state that testing in accordance with the surveillance requirement (ACSR 3.8.1.2) also demonstrates operability of the associated fuel oil transfer pump and necessary DG support system functions. **RAI 22.5-4 is being tracked as a Confirmatory Item.**

In RAI 14.3-151, the staff requested that the applicant include ITAAC for the DGSAS. In its response the applicant stated it would not include ITAAC for this system since it is non-safety-related. This response was not acceptable. After the applicant reclassified the DG systems as RTNSS systems, the staff again requested that the applicant include ITAAC for the DGSAS. **RAI 14.3-151 is being tracked as an Open Item.**

9.5.6.4 Conclusion

Because open items remain to be resolved for this section, the staff is unable to finalize its conclusions regarding acceptability of the DGSAS.

9.5.7 Diesel Generator Lubrication System

9.5.7.1 Regulatory Criteria

The staff reviewed the ESBWR diesel generator lubrication system (DGLS) in accordance with the guidance of Section 9.5.7, "Emergency Diesel Engine Lubrication System," of the SRP, Revision 2, dated July 1981. The staff's acceptance of the DGLS is based on the design's conformance with the requirements of the following GDC:

- GDC 2 requires in part that SSCs important to safety shall be designed to withstand the effects of natural phenomena such as earthquakes.

- GDC 4 requires in part that SSCs important to safety shall be designed to accommodate the effects of, and be compatible with, the environmental conditions associated with normal operation, maintenance, and testing and postulated accidents.

- GDC 5 requires that SSCs important to safety shall not be shared among nuclear power units unless it can be shown that such sharing will not significantly impair their ability to perform their safety functions including, in the event of an accident in one unit, an orderly shutdown and cooldown of the remaining units.

- GDC 17 requires in part that an onsite electric power system and an offsite electric power system shall be provided to permit functioning of SSCs important to safety.

Compliance with GDC 2 is based on meeting the guidance of Regulatory Positions C.1 and C.2 of RG 1.29 concerning seismic classification.

9.5.7.2 Summary of Technical Information

Each DG unit has its own independent, integrally mounted DGLS designed to supply clean filtered oil to various moving parts of the engine under controlled pressure and at proper operating temperature to maintain the lube oil properties suitable for engine lubrication. The system consists of an oil sump in the engine frame, an engine-driven oil pump, a lube oil cooler, a main header, a strainer, and a filter.

The engine-driven lube oil pump takes oil from the lube oil sump tank and passes it through the lube oil cooler and lube oil filter, through a strainer, through the main header, through the lube oil loads, and back to the lube oil sump tank. A second feed line from the strainer supplies oil to the combustion air turbochargers through the rocker lubrication system. Pressure-regulating valves maintain constant oil pressure to the main header by bypassing excess oil back to the lube oil sump tank.

9.5.7.3 Staff Evaluation

Each DG and its support systems, including the DGLS, are non-safety-related, do not perform any safety-related function, and are not credited in any safety analysis. They are not required to achieve or maintain safe shutdown of the plant. Also, their failures do not lead to the failure of any SSCs important to safety. The staff, therefore, determined that the requirements of GDC 2, 4, and 17 do not apply to the DGLS design.

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

As stated in Section 9.5.4 of this report, the applicant included the DG units as RTNSS. Also, in its response dated August 2, 2007, to RAI 22.5-4, the applicant stated that all DG supporting systems were considered as RTNSS systems and that it would revise Section 9.5.5 of the DCD Tier 2, Revision 4, to classify the DGLS as an RTNSS system.

In response to RAI 19.1.0-2 (MFN-07-066), GEH submitted draft Appendix 19A. On July 15, 2007 (MFN-07-373), GEH submitted a revised version of draft DCD Appendix 19A. Draft Appendix 19A states that all RTNSS systems will be in the scope of the Design Reliability Assurance Program, as directed by DCD Tier 2, Revision 3, Chapter 17. On September 28, 2007, GEH submitted Appendix 19A in DCD Tier 2, Revision 4, and the staff is reviewing it. DCD Tier 2, Revision 3, COL Action Item 17.4.13 identifies the requirements for the COL holder's Operational Reliability Assurance Program.

Draft Appendix 19A also provides an ACM. Section 3.8.1 of the ACM provides availability controls for the standby DGs. The bases for Section 3.8.1 state that testing in accordance with the surveillance requirement (ACSR 3.8.1.2) also demonstrates operability of the associated fuel oil transfer pump and necessary DG support system functions. **RAI 22.5-4 is being tracked as a Confirmatory Item.**

In RAI 14.3-151, the staff requested that the applicant include ITAAC for the DGLS. In its response the applicant stated it would not include ITAAC for this system since it is non-safety-related. This response was not acceptable. After the applicant reclassified the DG systems as RTNSS systems, the staff again requested that the applicant include ITAAC for the DGLS. **RAI 14.3-151 is being tracked as an Open Item.**

9.5.7.4 Conclusion

Because open items remain to be resolved for this section, the staff is unable to finalize its conclusions regarding acceptability of the DGLS.

9.5.8 Diesel Generator Combustion Air Intake and Exhaust System

9.5.8.1 Regulatory Criteria

The staff reviewed the ESBWR diesel generator combustion air intake and exhaust system (DGCAIES) in accordance with the guidance of Section 9.5.8, "Emergency Diesel Engine Combustion Air Intake and Exhaust System," of the SRP, Revision 2, dated July 1981. The staff's acceptance of the DGCAIES is based on the design's conformance with the requirements of the following GDC:

- GDC 2 requires in part that SSCs important to safety shall be designed to withstand the effects of natural phenomena such as earthquakes.
- GDC 4 requires in part that SSCs important to safety shall be designed to accommodate the effects of, and be compatible with, the environmental conditions associated with normal operation, maintenance, and testing and postulated accidents.
- GDC 5 requires that SSCs important to safety shall not be shared among nuclear power units unless it can be shown that such sharing will not significantly impair their ability to perform their safety functions including, in the event of an accident in one unit, an orderly shutdown and cooldown of the remaining units.
- GDC 17 requires in part that an onsite electric power system and an offsite electric power system shall be provided to permit functioning of SSCs important to safety.

Compliance with GDC 2 is based on meeting the guidance of Regulatory Positions C.1 and C.2 of RG 1.29 concerning seismic classification.

9.5.8.2 Summary of Technical Information

The DGCAIES is designed to supply combustion air to the DG engine and to exhaust combustion products out of the DG to the atmosphere. It includes intake and exhaust silencers to quiet engine operation.

Each engine takes combustion air from its own inlet air cubical adjacent to the DG room. Air enters the cubical through the outside wall and is filtered before entering the air intake plenum. Combustion air is drawn from the intake plenum through an air intake silencer and into the diesel engine combustion air manifold and the engine turbocharger. The engine exhaust is collected in the exhaust manifold and ducted to the roof of the DG building where the silencers are mounted. Each engine has its own exhaust system. The exhaust outlet is well removed from the air inlet to prevent any recirculation of exhaust gases. Crankcase ventilation is accomplished by means of vacuum blowers that vent the crankcase to the atmosphere.

9.5.8.3 Staff Evaluation

Each DG and its support systems, including the DGCAIES, are non-safety-related, do not perform any safety-related function, and are not credited in any safety analysis. They are not required to achieve or maintain safe shutdown of the plant. Also, their failures do not lead to the failure of any SSCs important to safety. The staff, therefore, determined that the requirements of GDC 2, 4, and 17 do not apply to the DGCAIES design.

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

As stated in Section 9.5.4 of this report, the applicant included the DG units as RTNSS. Also, in its response dated August 2, 2007, to RAI 22.5-4, the applicant stated that all DG supporting systems were considered as RTNSS systems and that it will revise Section 9.5.5 of the DCD Tier 2, Revision 4, to classify the DGCAIES as an RTNSS system.

In response to RAI 19.1.0-2 (MFN-07-066), GEH submitted draft Appendix 19A. On July 15, 2007 (MFN-07-373), GEH submitted a revised version of draft DCD Appendix 19A. Draft Appendix 19A states that all RTNSS systems will be in the scope of the Design Reliability Assurance Program, as directed by DCD Tier 2, Revision 3, Chapter 17. On September 28, 2007, GEH submitted Appendix 19A in DCD Tier 2, Revision 4, and the staff is reviewing it. DCD Tier 2, Revision 3, COL Action Item 17.4.13 identifies the requirements for the COL holder's Operational Reliability Assurance Program.

Draft Appendix 19A also provides an ACM. Section 3.8.1 of the ACM provides availability controls for the standby DGs. The bases for Section 3.8.1 state that testing in accordance with the surveillance requirement (ACSR 3.8.1.2) also demonstrates operability of the associated fuel oil transfer pump and necessary DG support system functions. **RAI 22.5-4 is being tracked as a Confirmatory Item.**

In RAI 14.3-151, the staff requested that the applicant include ITAAC for the DGCAIES. In its response the applicant stated it would not include ITAAC for this system since it is non-safety-related. This response was not acceptable. After the applicant reclassified the DG systems as RTNSS systems, the staff again requested that the applicant include ITAAC for the DGCAIES. **RAI 14.3-151 is being tracked as an Open Item.**

9.5.8.4 Conclusion

Because open items remain to be resolved for this section, the staff is unable to finalize its conclusions regarding acceptability of the DGCAIES.

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