6. ENGINEERED SAFETY FEATURES

6.1 Engineered Safety Features Materials

6.1.1 Engineered Safety Features Metallic Materials

6.1.1.1 Regulatory Criteria

In the economic simplified boiling-water reactor (ESBWR) design control document (DCD) Tier 2, Section 6.1.1, "Metallic Materials," the applicant described the selection, fabrication, and compatibility of materials with core cooling water and containment sprays for engineered safety feature (ESF) systems. The U.S. Nuclear Regulatory Commission (NRC) staff based its review of DCD Tier 2, Section 6.1.1, and its acceptance criteria on the relevant requirements set forth in Title 10, Section 50.55a, "Codes and Standards," of the Code of Federal Regulations (10 CFR 50.55a); Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," General Design Criteria (GDC) 1, "Quality Standards and Records," 4, "Environmental and Dynamic Effects Design Bases," 14, "Reactor Coolant Pressure Boundary," 31, "Fracture Prevention of Reactor Coolant Pressure Boundary," 35, "Emergency Core Cooling," and 41, "Containment Atmosphere Cleanup"; and Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," to 10 CFR Part 50.

- GDC 1 and 10 CFR 50.55a(a)(1) require that structures, systems, and components (SSCs) important to safety be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions they perform.

- GDC 4 requires that SSCs important to safety be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents (e.g., loss-of-coolant accidents (LOCAs)).

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

- GDC 31 requires that the design of the RCPB include sufficient margin to assure that, when stressed under operating, maintenance, testing, and postulated accident conditions, it will behave in a nonbrittle manner and the probability of rapidly propagating fracture is minimized.

- GDC 35 requires a system to provide abundant emergency core cooling. GDC 35 also requires that, during activation of the system, clad metal-water reaction will be limited to negligible amounts.

- GDC 41 requires that the design provide containment atmosphere cleanup systems to control fission products, hydrogen, oxygen, and other substances that may be released into the reactor containment. The staff limited its review of the ESF structural materials to ensuring that they met the requirements of GDC 41 with respect to corrosion rates related to hydrogen generation in postaccident conditions.

- Appendix B to 10 CFR Part 50 mandates that applicants establish quality assurance (QA) requirements for the design, construction, and prevent or mitigate the consequences of postulated accidents that could cause undue risk to the health and safety of the public.

6.1.1.2 Summary of Technical Information

The ESFs of the ESBWR design are those systems provided to mitigate the consequences of postulated accidents. The ESF systems are identified in DCD Tier 2, Chapter 6, and include (1) fission product containment and containment cooling systems, (2) emergency core cooling systems (ECCS), and (3) control room habitability systems.

The applicant stated that materials used in the ESF components have been evaluated to prevent material interactions that could potentially impair operation of the ESF.

The applicant selected materials to withstand the environmental conditions encountered during normal operation and postulated accidents. The applicant considered the materials' compatibility with core and containment spray water and also evaluated the effects of radiolytic decomposition products.

The design uses primarily metallic and metal-encapsulated insulation inside the ESBWR containment, except antisweat insulation used on cooling water lines. All nonmetallic thermal insulation must have the proper ratio of leachable sodium plus silicate ions to leachable chloride plus fluoride, consistent with Regulatory Guide (RG) 1.36, "Nonmetallic Thermal Insulation for Austenitic Stainless Steel," dated February 23, 1973, to minimize the possible contribution to stress-corrosion cracking (SCC) of austenitic stainless steel.

DCD Tier 2, Section 5.2.3 provides the evaluation of RCPB materials, and DCD Tier 2, Table 5.2-4 lists the principal pressure-retaining materials and the appropriate material specifications for the RCPB components. DCD Tier 2, Table 6.1-1 lists the principal pressure-retaining materials and the appropriate material specifications of the containment system and the ECCSs.

All materials of construction used in essential portions of ESF systems are resistant to corrosion, both in the medium contained and the external environment. General corrosion of all materials, except carbon and low-alloy steel, is negligible. The DCD provides conservative corrosion allowances for all exposed surfaces of carbon and low-alloy steel.

ESBWR core cooling water and containment sprays employ demineralized water with no additives. DCD Tier 2, Section 9.2.3 describes the water quality requirements. Leaching of chlorides from concrete and other substances is not significant. No detrimental effects occur on any of the ESF construction materials from allowable containment levels in the high-purity water. Thus, the materials are compatible with the post-LOCA environment.

With regard to ESF components, the ESBWR design conforms with the guidance provided in the following:

- RG 1.31, "Control of Ferrite Content in Stainless Steel Weld Metal"

- RG 1.36, “Nonmetallic Thermal Insulation for Austenitic Stainless Steel”

- RG 1.37, “Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants”

- RG 1.44, “Control of the Use of Sensitized Steel,” issued May 1973

- Generic Letter (GL) 88-01, “NRC Position on IGSCC in BWR Austenitic Stainless Steel Piping,” issued January 25, 1988

- NUREG-0313, Revision 2,“Technical Report on Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping,” dated January 31, 1988

6.1.1.3 Staff Evaluation

Materials and Fabrication

To meet the requirements of GDC 1 and 10 CFR 50.55a to ensure that plant SSCs important to safety are designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety function they perform, the applicant must identify codes and standards and maintain records. Selection of the materials specified for use in these systems must be in accordance with Section III, Division 1, of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code or RG 1.84, “Design, Fabrication, and Materials Code Case Acceptability, ASME Section III.” Section III references applicable portions of ASME Code, Section II, Parts A, B, C, and D.

DCD Tier 2, Table 6.1-1, lists the ASME Code classification and material specifications of components of the ESF systems. The staff reviewed the material specifications listed in Table 6.1-1 and verified that the aforementioned materials are acceptable for use in the ESBWR design in accordance with Section III of the ASME Code or RG 1.84. Given that DCD Tier 2, Section 6.1.1.1 states that Table 6.1-1 lists the principal pressure-retaining materials for the containment system and the ECCSs, the staff issued Request for Additional Information (RAI) 6.1-1, asking the applicant to verify that all ESF materials meet the requirements of ASME Code, Section III, or the guidance of RG 1.84.

The applicant stated that materials for these systems must comply with Section III and, therefore, will only be materials that appear in Appendix I (now Section II, Part D), and that all such materials are in accordance with ASME Code, Section II, Parts A, B, or C, or RG 1.84. The applicant further stated that the design, fabrication, and testing requirements for ESF components, and fracture toughness requirements for all ferritic ESF materials, in the ESBWR design will comply with the appropriate Section III class shown in DCD Tier 2, Section 6.1, Table 6.1-1.

In RAI 6.1-2, the staff asked the applicant to include weld filler metal specifications in Table 6.1-1. In its response, the applicant provided filler metal specifications and classifications for weld filler metal used in the ESF systems with the exception of carbon steel and low-alloy steel filler materials. Given that the specifications for carbon and low alloy steel listed by the applicant encompass a broad range of filler metal classifications, the RAI response was incomplete. In supplemental RAI 6.1-2, the staff requested that the applicant include classifications of filler materials used to join carbon steel and low-alloy steel components in ESF systems. The applicant responded and proposed a revision to Table 6.1-1.

The applicant listed weld filler material classifications E9018-B3L and ER90S-B3L for use when welding low-alloy steel. The staff notes that ASME discontinued these weld filler material classifications and replaced them with classifications E8018-B3L and ER80S-B3L. DCD Tier 2, Revision 3, Table 5.2-4, contains similar inappropriate references to discontinued classifications. To determine that the weld filler materials used in the ESBWR design meet the requirements of ASME Code, Section II, Part C, the staff issued RAI 6.1-2(a), Supplement 2, requesting the applicant to modify Tables 5.2-4 and 6.1-1 to include the correct weld filler material classifications.

The applicant's proposed revision to Table 6.1-1 lists the weld filler material that will be used to weld P5C, Group 1 (G1) materials. After reviewing the ESF material specifications provided by the applicant in the proposed revision to Table 6.1-1, the staff is unable to identify any materials that fall into the P5C, G1 category in accordance with ASME Code, Section IX, Table QW-422. To determine that the materials specifications and grades used in the ESBWR design meet the requirements of ASME Code, Section II, Parts A, B, and C, the staff issued RAI 6.1-2(b), Supplement 2, requesting the applicant to identify the P5C, G1 materials used in the ESBWR design for ESF components or else delete this information from the DCD if it does not apply. The staff noted that the same issue exists in DCD Tier 2, Revision 3, Table 5.2-4, in which the applicant references P5C, G1 materials as requiring welding, but the staff cannot identify any P5C materials in the RCPB. Therefore, the staff also requested, as part of RAI 6.1-2(b), Supplement 2, that the applicant identify the P5C, G1 materials used in the ESBWR design for RCPB components or else delete this information from Table 5.2-4 if it does not apply.

The applicant's proposed revision to Table 6.1-1 identifies shielded manual arc welding filler material E8018-G for use in welding low-alloy steel in the ESBWR design. To complete its review and evaluate the applicant's compliance with 10 CFR 50.55a, the staff issued RAI 6.1-2(c), Supplement 2, requesting the applicant to provide the complete GEH specification that will be used to purchase E8018-G for fabricating ASME Code, Section III, Class 1, 2, and 3 components. In addition, the staff requested that the applicant provide a technical justification for using the GEH specification in lieu of commercially available welding electrodes. The staff identifies the above issues regarding weld filler metal specifications and P numbers as RAI 6.1-2. **RAI 6.1-2 is being tracked as an open item.**

The isolation condenser system (ICS) in the ESBWR design includes four isolation condensers (ICs), which are ASME Code, Section III, Class 2 components. In RAI 5.4-20, the staff requested that the applicant provide detailed information on the design of the ICs. In response to this RAI, the applicant indicated that the IC tubes would be fabricated from a modified form of Alloy 600 (ASME Code Case N –580-1). However, in other portions of its submittal (i.e., Table 6.1-1), the applicant indicated that Alloy 600 would be used in the fabrication of the IC tubes. In supplemental RAI 5.4-20(D), the staff requested that the applicant clarify the material of construction for IC tubes. As part of its response to RAI 5.4-20, the applicant indicated that the IC tubes will be bent by induction. However, the applicant did not indicate what effect, if any, this would have on the material properties of the tubing, nor did it indicate what testing, if any, was performed to confirm the acceptability of the material properties following bending of the piping/tubing. In supplemental RAI 5.4-20(A), the staff requested that the applicant discuss how it has confirmed that the material properties of the most limiting bent tube remain acceptable following induction bending. The staff also requested that the applicant include a discussion of the material properties tested (e.g., hardness), the results, and the acceptance criteria.

In RAI 5.4-20, the staff also requested that the applicant provide additional details on the design of the support structures for the IC tubes, if any, on the "pool side" and their materials of construction. In its response to RAI 5.4-20, the applicant indicated that the design of the support structures of the IC tubes is not currently available. The staff notes that, depending on the design, there may be crevices between the IC tube and the support. Such crevices could result in the accumulation of chemical contaminants that could lead to corrosion. In addition, the materials of construction of the support are important because any corrosion of them could result in a loss of support for, or damage to, the IC tubes. Given that material selection and specific design attributes, such as the presence of crevices, can contribute to degradation, the staff requested, in supplemental RAI 5.4-20(B), that the applicant provide a combined license (COL) action item to require submittal of this information. **RAI 5.4-20 is being tracked as an open item.**

Because open items remain to be resolved for this section, the staff was unable to finalize its conclusions regarding whether the applicant will conform to ASME Code, Section III and RG 1.84 and that the ESF materials meet the requirements of GDC 1 and 10 CFR 50.55a.

Austenitic Stainless Steels

The ESBWR design must meet the requirements of (1) GDC 4, relative to compatibility of components with their environmental conditions; (2) GDC 14, with respect to fabrication and testing of the RCPB so as to have an extremely low probability of abnormal leakage, rapidly propagating failure, or gross rupture; and (3) the QA requirements of Appendix B to 10 CFR Part 50. Designs may meet these requirements by following the guidance of GL 88-01; NUREG-0313, Revision 2; and RGs 1.31, 1.34, 1.37, and 1.44 and by providing controls over the use of cold-worked austenitic stainless steels.

For stainless steel components in the ESF systems, DCD Tier 2, Section 6.1.1.3, refers to DCD Tier 2, Section 5.2.3 for discussion of the following on fabrication and processing of austenitic stainless steels as well as conformance with the regulatory guidance in RGs 1.31, 1.37, and 1.44; GL 88-01; and NUREG-0313, Revision 2. Section 5.2.3 of this report contains the staff's evaluation of the applicant's conformance with the aforementioned NRC documents. The staff has determined that the applicant either follows the guidance of, or has provided an acceptable alternative to, RGs 1.31, 1.37, and 1.44; GL 88-01; and NUREG-0313, Revision 2. The staff has also determined that the applicant's controls over the use of cold-worked austenitic stainless steels, as discussed in DCD Tier 2, Sections 5.2.3 and 6.1.1.3.3, are acceptable as they will provide reasonable assurance that appropriate steps will be taken to control the use of cold-worked austenitic stainless steels in the ESBWR.

Ferritic Steel Welding

To meet the requirements of GDC 1 related to general QA and codes and standards, Appendix B to 10 CFR Part 50 for control of special processes, and 10 CFR 50.55a, the amount of minimum specified preheat must meet ASME Code, Section III, Appendix D, Article D-1000, and RG 1.50, “Control of Preheat Temperature for Welding of Low-Alloy Steel,” issued May 1973, unless an alternative procedure is justified. In addition, moisture control on low-hydrogen welding materials must conform to the requirements of ASME Code, Section III.

As requested by the staff, the applicant verified that minimum preheat requirements meet ASME Code, Section III, Appendix D, Article D-1000, and follow the guidelines of RG 1.50. For the standby liquid control (SLC) accumulator tank, the preheat recommendations of ASME Code, Section III, Appendix D, Article D-1000 will be followed. The applicant specified the use of an alternative to RG 1.50. The applicant’s alternative consists of performing a postweld bakeout of welds that do not go directly from preheating temperature to postweld heat treatment. The staff concludes that the applicant’s alternative to RG 1.50 is acceptable given that it provides reasonable assurance that delayed hydrogen cracking will not occur between the completion of welding and postweld heat treatment. Section 5.2.3 of this report discusses the staff’s evaluation of the applicant’s alternative in more detail.

Dissimilar Metal Welds

The applicant provided a description of all dissimilar metal welds (DMWs) in the ESF systems and discussed the selection of filler metals, welding processes, and process controls for DMWs. The DMWs in the ESF will be performed with the same materials and process selections as the RCPB. The staff reviewed the applicant’s response and considers the applicant’s description of its selection of filler metals, welding processes, and process controls acceptable, as they will provide reasonable assurance that the DMWs in the ESBWR design will maintain structural integrity throughout the design life of the plant. Section 5.2.3 of this report contains the staff’s more detailed evaluation of this topic.

Limited Accessibility Welder Qualification

In RAI 6.1-6, the staff asked the applicant to verify that the ESBWR design related to fabrication of ESFs will follow the guidance in RG 1.71, “Welder Qualification for Areas of Limited Accessibility.” The applicant responded that RG 1.71 will be applied to ESF systems in the same manner as for the RCPB systems. The staff finds the applicant’s level of compliance with the guidelines detailed in RG 1.71 acceptable, as they will provide reasonable assurance that welds made under limited access conditions will be performed by personnel with appropriate qualifications to produce sound, high-quality welds. Section 5.2.3 of this report gives the staff’s more detailed evaluation of the applicant’s implementation of RG 1.71 for RCPB systems.

Composition and Compatibility of ESF Fluids

The core cooling water and containment sprays in the ESBWR use demineralized water with no additives. The applicant indicated that materials used in essential portions of ESF systems are resistant to corrosion, both in the medium contained and the external environment. The applicant also stated that general corrosion of all materials, with the exception of carbon and low-alloy steels, is negligible. The ESBWR design provides conservative corrosion allowances for all exposed surfaces of carbon and low-alloy steel.

The process for determining the corrosion allowance for ferritic materials is the same as that applied to RCPB materials. The corrosion allowance is primarily based on GEH internal testing. The allowances consider fluid velocity, oxygen content, and temperature, and they include a safety margin over the actual measured corrosion rates of approximately a factor of two. The designs of most operating boiling-water reactors (BWRs) (GEH design) have applied the same method, with corresponding allowances, including the certified advanced boiling-water reactor (ABWR) design. The staff considers the applicant's corrosion allowances acceptable given that the ESBWR corrosion allowances for ferritic materials are based on laboratory testing, operational experience and a safety margin of two.

To meet the requirements of GDC 4, 14, and 41, the plant design should control the water used in the ESF to provide assurance against SCC in unstabilized stainless steel components. The staff reviewed the applicant's water quality requirements for the makeup water system demineralized water storage tank (DCD Tier 2, Table 9.2-7) and makeup water system demineralizer effluent (DCD Tier 2,Table 9.2-7). The chemistry control requirements of Tables 9.2-7 and 9.2-8 for conductivity, chloride, and pH in the ESBWR design are consistent within the limits listed in Section 6.1.1 of NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants" (hereafter referred to as the SRP), and are therefore acceptable.

DCD Tier 2, Table 6.1-1 indicates that Alloy 600 is used for IC tubing and header fabrication. Alloy 600 has a history of being susceptible to SCC in light-water reactor systems. In RAI 6.1-10, the staff asked the applicant to provide a basis for the use of Alloy 600 in the IC, including material condition (i.e., mill annealed or thermally treated) as it relates to susceptibility to SCC in the reactor coolant and demineralized water environment. In response, the applicant indicated that there have been no reports of Alloy 600 cracking in BWRs in the absence of a welded crevice or a crack initiated in adjacent Alloy 182. These initiating features are absent from the ESBWR design. In addition, the material used for the IC is the same alloy as that used for reactor shroud support and stub tubes (see response to RAI 4.5-18, as discussed in Section 4.5 of this report). This alloy (see ASME Code Case N –580-1) is a significantly modified version of Alloy 600 wherein the carbon content is limited, niobium (columbium) is added as a stabilizer, and high-temperature solution heat treatment is required instead of a mill anneal. Stress corrosion resistance is very good. The alloy is approved for use by ASME Code Case N–580-1 and has been deployed in several operating BWRs, including the Kashiwazaki-Kariwa 6/7 ABWRs. Several of these units have been operating for more than 10 years. In RAI 5.4-55, the staff requested that the applicant discuss the corrosion allowances for Alloy 600 used in the ICs. In response, the applicant indicated that the Alloy 600 tubing in early BWR ICs performed satisfactorily without incident resulting from general corrosion in this application. Although general corrosion is a concern, the applicant did not address whether any other incidences of corrosion or other degradation have occurred in operating units. In supplemental RAI 5.4-55, the staff requested that the applicant discuss whether there have been any other "incidents" associated with the use of these materials in these applications. **RAI 5.4-55 is being tracked as an open item.**

Component and Systems Cleaning

The staff reviewed the ESF structural materials to ensure that the requirements of Appendix B to 10 CFR Part 50 were met, as they relate to the establishment of measures to control the cleaning of material and equipment. The controls established for cleaning of material and equipment must be performed in accordance with work and inspection instructions to prevent damage or deterioration.

The ESBWR design complies with RG 1.37 except as noted in DCD Tier 2, Table 1.9-21B. Table 2-1 of NEDO-11209-04a, “GE Nuclear Energy Quality Assurance Program Description,” Class I (nonproprietary), Revision 8, dated March 31, 1989, documents the alternative that the applicant may use. The alternative involves using methods, other than mechanical methods, to remove local rusting on corrosion-resistant alloys. The NRC approved this alternative on March 31,1989. Therefore, the applicant’s request to use this alternative is acceptable. Section 4.5.1.2.5 of this report further discusses the applicant’s level of compliance with RG 1.37. Thus, the ESBWR design satisfies the QA requirements of Appendix B to 10 CFR Part 50 for component and system cleaning.

Thermal Insulation

The type of thermal insulation used in the ESBWR containment will be primarily metallic and metal-encapsulated insulation, except antisweat insulation used on cooling water lines. In DCD Tier 2, Section 6.1.1.3.4, the applicant stated that nonmetallic thermal insulation materials used on ESF systems are selected, procured, tested, and stored in accordance with RG 1.36.

To meet the requirements of GDC 1, 14, and 31, ESF systems should be designed, fabricated, erected, and tested such that there is an extremely low probability of abnormal leakage, rapidly propagating failure, or gross rupture. The levels of leachable contaminants in nonmetallic insulation materials that come into contact with 300 series austenitic stainless steels used in fluid systems important to safety should be under careful control so as not to promote SCC. In particular, the leachable chlorides and fluorides should be held to the lowest practical levels. The staff’s position is that following the guidance in RG 1.36 is an acceptable method to control leachable contaminants in nonmetallic insulation materials. Since the applicant has stated that it will follow the guidance in RG 1.36, the staff finds this acceptable as it will meet the requirements of GDC 1, 14, and 31.

6.1.1.4 Conclusions

Because open items remain to be resolved for this section, the staff was unable to finalize its conclusions on whether the ESBWR DCD specifications for the materials to be used in the fabrication of the ESFs are acceptable and meet the relevant requirements of GDC 1, 4, 14, 31, 35, and 41; Appendix B to 10 CFR Part 50; and 10 CFR 50.55a.

6.1.2 Organic Materials

6.1.2.1 Regulatory Criteria

The staff reviewed the protective coating systems (paints) and organic materials in accordance with SRP Section 6.1.2, “Protective Coating Systems (Paints)—Organic Materials.” Staff acceptance is based on meeting the requirements of Appendix B to 10 CFR Part 50 as it relates to the QA requirements for the design, fabrication, and construction of safety-related SSCs. To meet the requirements of Appendix B to 10 CFR Part 50, the applicant can specify that the coating systems and their applications will follow the guidance of RG 1.54, Revision 1, “Service Level I, II, and III Protective Coatings Applied to Nuclear Power Plants,” issued July 2000.

6.1.2.2 Summary of Technical Information

The ESBWR design has reduced the use of coatings inside containment to a minimum. The areas in which most of the coatings are used are the following:

- internal steel structures
- carbon steel containment liner
- equipment inside drywell and wetwell

Some coatings and organic materials inside containment will not meet the criteria in RG 1.54 and ASTM D 5144-00, "Standard Guide for Use of Protective Coating Standards in Nuclear Power Plants." The exceptions are restricted to small-size equipment where, in case of a LOCA, paint debris is not a safety hazard. Exceptions include items such as electronic/electrical trim, covers, faceplates, and valve handles.

6.1.2.3 Staff Evaluation

The staff reviewed the protective coating systems (paints) and organic materials in accordance with SRP Section 6.1.2. Staff acceptance is based on meeting the requirements of Appendix B to 10 CFR Part 50 as it relates to the QA requirements for the design, fabrication, and construction of safety-related SSCs. To meet the requirements of Appendix B to 10 CFR Part 50, the applicant should specify that the coating systems and their applications will follow the guidance of RG 1.54, Revision 1. This RG references the QA standards of ASTM D 3842, “Selection of Test Methods for Coatings for Use in Light Water Nuclear Power Plants”; ASTM D 3911, “Evaluating Coatings Used in Light Water Nuclear Power Plants at Simulated Design Basis Accident (DBA) Conditions”; and ASTM D 5144-00.

RG 1.54, Revision 1, provides guidance on practices and programs that are acceptable to the NRC staff for the selection, application, qualification, inspection, and maintenance of protective coatings applied in nuclear power plants. In addition, this latest revision to the RG updates the definitions of Service Level I, II, and III coating’s locations to include both safety-related and non-safety-related regions, as set forth by the ASTM Committee and the updated ASTM guidance.

The applicant stated that the protective coating system meets the regulatory positions of RG 1.54, Revision 1, and the standards of ASTM D 5144-00, as applicable.

The applicant also stated that not all coatings inside containment will meet the criteria of RG 1.54, Revision 1, and ASTM D 5144-00. The exceptions are for small-size equipment where, in case of a LOCA, paint debris is not a safety hazard. To address this issue, the applicant included a commitment that the COL applicant is required to do the following:

- describe the approach to be taken to identify and quantify all organic materials that exist within the containment building in significant amounts that do not meet the requirements of ASTM D 5144-00 and RG 1.54, Revision 1, as per Section 6.1.2

- provide the milestone when evaluations will be complete to determine the generation rate, as a function of time, of combustible gases that can be formed from these unqualified organic materials under design-basis accident (DBA) conditions

- as part of these evaluations, provide the technical basis and assumptions used

This was identified as a COL information item in DCD Tier 2, Revision 3, Section 6.1.3.1.

Although the amount of organic materials that do not meet the requirements of Regulatory Guide 1.54 will not be available before the procurement of the components. In a request for addition information (RAI 6.1-16), the staff requested the applicant to revise the DCD (including addressing a COL action item) to ensure the COL applicant provides a bounding value for the amount of unqualified coatings and the assumptions used to determine this bounding value. In addition, the staff requested the COL applicant include plans to include the following as part of the ITAAC program:

1. Total amount of protective coatings and organic materials used inside containment that do not meet the requirements of ASTM D 5144 and Regulatory Guide 1.54.

2. Evaluation of the generation rate, as a function of time, of combustible gases that can be formed from these unqualified organic materials under DBA conditions.

3. Technical basis and assumptions used for this evaluation.

RAI 6.1-16 is being tracked as an open item.

6.1.2.4 Conclusions

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

6.2.1 Containment Functional Design

6.2.1.1 Pressure Suppression Containment

6.2.1.1.1 Regulatory Criteria

The staff reviewed ESBWR DCD Tier 2, Section 6.2.1.1, “Pressure Suppression Containment,” in accordance with SRP Section 6.2.1, “Containment Functional Design,” Revision 2, July 1981; SRP Section 6.2.1.1.C, “Pressure-Suppression Type BWR Containments,” Revision 6, August 1984; and SRP Section 6.2.1.3, “Mass and Energy Release Analysis for Postulated Loss-of-Coolant,” Revision 1, July 1981.

Acceptance criteria are based on the following GDC, which apply to the design and functional capability of a BWR pressure suppression type containment:

- GDC 4 requires that SSCs important to safety be designed to accommodate the dynamic effects (e.g., effects of missiles, pipe whipping, and discharging fluids that may result from equipment failures) that may occur during normal plant operation or following a LOCA.

- GDC 16, “Containment Design,” and GDC 50, “Containment Design Basis,” as they relate to the containment being designed with sufficient margin, require that the containment and its associated systems can accommodate, without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions resulting from any LOCA.

- GDC 53, “Provisions for Containment Testing and Inspection,” as it relates to (1) appropriate periodic inspection of all important areas, such as penetrations, (2) an appropriate surveillance program, and (3) periodic testing at containment design pressure of the leaktightness of penetrations which have resilient seals and expansion bellows.

To meet the requirements of GDC 16 and 50 regarding the design margin for the ESBWR, which is similar in design to a BWR III plant, the peak calculated values of pressure and temperature for the drywell and wetwell should not exceed the respective design values. To meet the requirement of GDC 16, provisions should be made in one of the following ways to protect the drywell and wetwell (or containment) against loss of integrity from negative pressure transients or postaccident atmosphere cooldown:

- Structures should be designed to withstand the maximum calculated external pressure.

- Vacuum relief devices should be provided in accordance with the requirements of the ASME Boiler and Pressure Vessel Code, Section III, Subsection NE, to assure that the external design pressures of the structures are not exceeded.

The maximum allowable leakage area for steam bypass of the suppression pool should be greater than the TS limit for leakage measured in periodic drywell-wetwell leakage tests to demonstrate that the design meets the requirement of GDC 53 regarding periodic testing at containment design pressure.

6.2.1.1.2 Summary of Technical Information

The containment systems for the ESBWR include a containment structure and a reactor building (RB) surrounding the containment structure and housing equipment essential to safe shutdown of the reactor. The containment is designed to prevent the uncontrolled release of radioactivity to the environment with a leakage rate of 0.5 percent by weight per day at the calculated peak containment pressure related to the design basis accident (DBA). The RB is designed to provide an added barrier to the leakage of airborne radioactive materials from the primary containment in case of an accident. ESBWR DCD Figure 6.2.1 shows the principal features of the ESBWR containment.

The ESBWR containment is designed with the following main features:

- The drywell consists of (1) an upper drywell volume surrounding the upper portion of the reactor pressure vessel (RPV) and housing the main steam and feedwater piping, gravity-driven cooling system (GDCS) pools and piping, PCCS piping, ICS piping, safety relief valves (SRVs) and piping, depressurization valves (DPVs) and piping, drywell coolers and piping, and other miscellaneous systems; and (2) a lower drywell volume below the RPV support structure housing the lower portion of the RPV, fine motion control rod drives, other miscellaneous systems and equipment below the RPV, and vessel bottom drain piping.

The upper drywell is a cylindrical, reinforced concrete structure with a removable steel head and a diaphragm floor constructed of steel girders with concrete fill. The RPV support structure separates the lower drywell from the upper drywell. There is an open communication path between the two drywell volumes via upper drywell to lower drywell connecting vents, built into the RPV support structure. Penetrations through the liner for the drywell head, equipment hatches, personnel locks, piping, and electrical and instrumentation lines are provided with seals and leak-tight connections.

The drywell, which has a net free volume of 7,206 m^3 (254,500 ft^3), is designed to withstand the pressure and temperature transients associated with the rupture of any primary system pipe inside the drywell and also the negative differential pressures associated with containment depressurization events, when the steam in the drywell is condensed by the PCCS, the GDCS, the fuel and auxiliary pools cooling system (FAPCS), and cold water cascading from the break following post-LOCA flooding of the RPV. The drywell design pressure and temperature are 310 kilopascal gauge (kPa(g)) (45 pounds per square inch gauge (psig)) and 171℃ (340℉), respectively. The design drywell-wetwell pressure differences (i.e., drywell pressure being higher/lower than the wetwell pressure) are +241 kilopascal differential (kPa(d)) (31 pounds per square inch differential (psid)) and -20.8 kPa(d) (-3.0 psid). The design drywell-RB differential pressure is -20.7 kPa(d) (-3.0 psid).

- The wetwell consists of a gas volume and a suppression pool, with a net gas volume of 5,432 m^3 (191,800 ft^3) and a minimum pool volume of 4,424 m^3 (156,200 ft^3) at low water level.

- The wetwell is designed for an internal pressure of 310 kPa(g) (45 psig) and a temperature of 121℃ (250℉).

- The suppression pool, which is located inside the wetwell annular region between the cylindrical RPV pedestal wall and the outer wall of the wetwell, is a large body of water that will absorb energy by condensing steam from SRV discharges and pipe break accidents. The pool is an additional source of reactor water makeup and serves as a reactor heat sink. The flowpath to the wetwell is designed to entrain radioactive materials by routing fluids through the suppression pool during and following a LOCA. The gas space above the suppression pool is leak tight and sized to collect and retain the drywell gases following a pipe break in the drywell, without exceeding the containment design pressure.

- Following a postulated DBA, the mass and energy released to the drywell will be transferred to the wetwell through a system of 12 vertical circular channels of a nominal diameter of 0.70 m (2.3 ft), each containing 3 horizontal vents for a total of 36 vents. The three-vent centerlines in each column are located at 1.95 m (6.4 ft), 3.32 m (10.9 ft), and 4.69 m (15.4 ft) below the suppression pool water level when the suppression pool is at the low water level.

- A spillover system provides drywell to wetwell connection to limit suppression pool drawdown and the holdup volume in the drywell following a LOCA by transferring water from the drywell annulus to the suppression pool. Spillover is accomplished by 12 horizontal holes (200-mm nominal diameter), which are built into the vent wall connecting the drywell annulus with each vertical vent module. If water ascending through the drywell annulus following a postulated LOCA reaches the spillover holes, it will flow into the suppression pool via the vertical/horizontal vent modules. Once in the suppression pool, the water can be used for accident mitigation (i.e., by restoration of RPV inventory).

- A drywell-to-wetwell vacuum breaker system protects the integrity of the diaphragm floor slab and vent wall between the drywell and the wetwell, and the drywell structure and liner, and will prevent back-flooding of the suppression pool water into the drywell. The vacuum breaker is a process-actuated valve, similar to a check valve, and is provided with redundant proximity sensors to detect its closed position. On the upstream side of the vacuum breaker, a direct current (dc)-powered, solenoid-controlled, and spring-operated backup valve, designed to fail-close, is provided. During a LOCA, the vacuum breaker opens and allows the flow of gas from wetwell to drywell to equalize the drywell and wetwell pressure. After the drywell and wetwell pressure equalizes, the vacuum breaker closes to prevent extra bypass leakage caused by the opening created by the vacuum breaker, and, therefore, to maintain the pressure suppression capability of the containment. If the vacuum breaker does not completely close, as detected by the proximity sensors, a control signal will close the upstream backup valve. Redundant vacuum breaker systems are provided to protect against a single failure of a vacuum breaker, either failure to open or failure to close when required.

Similar to an ABWR, the ESBWR containment design uses combined features of the Mark II and Mark III designs, except that the drywell consists of upper drywell and lower drywell volumes.

The vents to the suppression pool are a combination of the vertical Mark II and horizontal Mark III systems. The wetwell is similar to a Mark III wetwell.

<u>Vacuum Breakers</u>. Vacuum breakers are provided between the drywell and wetwell. The vacuum breaker is a self-actuating valve, similar to a check valve. The purpose of the drywell-to-wetwell vacuum breaker system is to protect the integrity of the diaphragm floor slab and vent wall between the drywell and the wetwell, and the drywell structure and liner, and to prevent back-flooding of the suppression pool water into the drywell. The vacuum breaker is provided with redundant proximity sensors to detect its closed position. On the upstream side of the vacuum breaker, a dc-powered, solenoid-controlled, and spring-operated backup valve designed to fail-close is provided. During a LOCA, the vacuum breaker opens and allows the flow of gas from wetwell to drywell to equalize the drywell and wetwell pressure. After the drywell and wetwell pressure equalizes, the vacuum breaker closes to prevent extra bypass leakage caused by the opening created by the vacuum breaker, and therefore, to maintain the pressure suppression capability of the containment. If the vacuum breaker does not completely close, as detected by the proximity sensors, a control signal will close the upstream backup valve.

Redundant vacuum breaker systems are provided to protect against a single failure of a vacuum breaker, either failure to open or failure to close when required. DCD Table 6.2-1 provides the design drywell-to-wetwell pressure difference and the vacuum breaker full open differential pressure.

The vacuum breaker valves are protected from pressure suppression loads by structural shielding designed for pressure suppression loads based on a Mark II/III containment design.

Steam Bypass of the Suppression Pool. The pressure suppression containment is designed such that any steam released from a pipe rupture in the primary system is condensed by the suppression pool and does not produce a significant pressurization effect on the containment. This is accomplished by channeling the steam into the suppression pool through a vent system. If a leakage path were to exist between the drywell and the suppression pool (wetwell) gas space, the leaking steam would produce undesirable pressurization of the containment.

The bounding DBA calculation assumes a bypass leakage area of 1 cm^2 ($A/\sqrt{K}$), as specified in TS SR 3.6.2.2.2. In the ESBWR design, the PCCS also condenses some of the steam released from the pipe rupture.

Loss-of-Coolant Accidents. The containment functional evaluation was based on the GEH consideration of a representative spectrum of postulated LOCAs which would result in the release of reactor coolant to the containment. These LOCAs include the following:

- liquid line breaks
  - an instantaneous guillotine rupture of a feedwater line (FWL)
  - an instantaneous guillotine rupture of a GDCS line
  - an instantaneous guillotine rupture of a vessel bottom drainline
- steamline breaks
  - an instantaneous guillotine rupture of a main steamline (MSL)

GEH used the TRACG computer program to evaluate the containment performance (NEDC-33083P-A and NEDE-32176P). The staff's safety evaluation in Section 4 of NEDC-33083P-A contains items needing confirmation during the ESBWR design certification stage. Staff's Addendum to the Safety Evaluation Report with Open Items for NEDC-33083P-A, Application of the TRACG Computer Code to the ECCS and Containment LOCA Analysis for the ESBWR Design (ML073190044) addressed these confirmatory items.

ESBWR DCD Tables 6.2-1 through 6.2-4 list key design and operating parameters of the containment system, including the design characteristics of the drywell, the wetwell, and the pressure suppression vent system and key assumptions used for the DBA analysis. ESBWR DCD Tables 6.3-1 through 6.3-4 provide the performance parameters of the related emergency safety feature systems, which supplement the design conditions of ESBWR DCD Table 6.2-1, for containment performance evaluation. ESBWR DCD Table 6.2-6 provides the nominal and bounding values for the plant initial and operating conditions for evaluating the containment performance.

GEH evaluated four cases, the three liquid line break cases and the steamline break case, using the nominal initial and operating conditions listed in Table 6.2-1. The results of the four cases showed that instantaneous guillotine ruptures of an MSL and an FWL gave the highest containment pressure. GEH then used the bounding initial and operating conditions listed in Table 6.2-1 in its evaluation of the main steamline break (MSLB) and the feedwater line break (FWLB) cases. Results of these analyses showed that an instantaneous guillotine rupture of an MSL with failure of one DPV produced the most limiting responses for the containment pressure evaluation. The second limiting case is an instantaneous guillotine rupture of an FWL with failure of one SRV. ESBWR DCD Table 6.2-5 lists the results of GEH evaluations of the four cases using the nominal initial and operating conditions and the two cases using bounding initial and operating conditions.

Negative Pressure Design Evaluation. During normal plant operation, the inerted wetwell and the drywell volumes remain at a pressure slightly above atmospheric conditions. Certain events could lead to a depressurization transient that can produce a negative pressure differential in the containment. A drywell depressurization results in a negative pressure differential across the drywell walls, vent wall, and diaphragm floor. A negative pressure differential across the drywell and wetwell walls means that the RB pressure is greater than the drywell and wetwell pressures, and a negative pressure differential across the diaphragm floor and vent wall means that the wetwell pressure is greater than the drywell pressure. If not mitigated, the negative pressure differential can damage the containment steel liner. The ESBWR design provides the vacuum relief function necessary to limit these negative pressure differentials within design values.

The following events may cause containment depressurization:

- Post-LOCA drywell depressurization is caused by the ECCS (e.g., GDCS, CRD) flooding of the RPV and cold water spilling out of the broken pipe or cold water spilling out of the broken GDCS line directly into the drywell.

- The drywell sprays are inadvertently actuated during normal operation or during the post-LOCA recovery period.

- The combined heat removal of the ICS and PCCS exceeds the rate of decay heat steam production.

GEH expects drywell depressurization following a LOCA to produce the most severe negative pressure transient condition in the drywell. The results of the MSLB analysis showed that the containment did not reach negative pressure relative to the RB and the maximum wetwell-drywell differential pressure was within the design capability. This calculation assumed one available vacuum breaker with an area of 0.2 m^2, which is conservative with respect to the planned installed vacuum breaker area. To prevent excessive negative pressure, the drywell spray flow rate must be less than 227 m^3/h (1000 gpm).

6.2.1.1.3 Staff Evaluation

For pressure-suppression type BWR plant containments, the staff review covers the following areas:

- the temperature and pressure conditions in the drywell and wetwell that result from a spectrum (including break size and location) of postulated LOCAs

- suppression pool dynamic effects during a LOCA or following the actuation of one or more reactor coolant system safety/relief valves, including vent clearing, vent interactions, pool swell (PS), pool stratification, and dynamic symmetrical and asymmetrical loads on suppression pool and other containment structures

- the consequences of a LOCA occurring within the containment (wetwell or outside the drywell)

- the capability of the containment to withstand the effects of steam bypassing the suppression pool

- the external pressure capability of the drywell and wetwell and systems that may be provided to limit external pressures

- the effectiveness of static and active heat removal mechanisms

- the pressure conditions within subcompartments and acting on system components and supports as a result of high-energy line breaks (HELBs)

- the range and accuracy of instrumentation provided to monitor and record containment conditions during and following an accident

- the suppression pool temperature limit during reactor coolant system safety/relief valve operation, including the events considered in analyzing suppression pool temperature response, assumptions used for the analyses, and the suppression pool temperature monitoring system

- the reactor coolant system safety/relief valve in-plant confirmatory test program

- the evaluation of analytical models used for containment analysis.

The ESBWR DCD does not describe a chronology of progression of a LOCA, how it affects the containment and its systems, or how containment systems operate to mitigate the consequences of a LOCA. GEH needs to add this information to the ESBWR DCD. **RAI 6.2-175 is being tracked as an open item.**

Table 6.2-1 of this report reproduces DCD Table 6.2-6. ESBWR DCD Table 6.2-6 lists the RPV nominal water level as “NWL.” However, NWL is not defined in the Global Abbreviations and Acronyms List, and its value is not given in DCD Tier 2. **RAI 6.2-174 is being tracked as an open item.**

Table 6.2-1 Plant Initial and Operating Conditions Considered in the Containment Performance Evaluation Cases

No.	Plant Parameter	Nominal Value	Bounding Value
1	RPV Power	100%	102%
2	Wetwell relative humidity	100%	100%
3	PCC pool level	4.8 m (15.7 ft)	4.8 m (15.7 ft)
4	PCC pool temperature	43.3℃ (110℉)	43.3℃ (110℉)
5	Drywell pressure	101.3 kPa (14.7 psia)	110.3 kPa (16.0 psia)
6	Drywell temperature	46.1℃ (115℉)	46.1℃ (115℉)
7	Wetwell pressure	101.3 kPa (14.7 psia)	110.3 kPa (16.0 psia)
8	Wetwell temperature	43.3℃ (110℉)	43.3℃ (110℉)
9	Suppression pool temperature	43.3℃ (110℉)	43.3℃ (110℉)
10	GDCS pool temperature	46.1℃ (115℉)	46.1℃ (115℉)
11	Suppression pool level	5.45 m (17.9 ft)	5.50 m (18.0 ft)
12	GDCS pool level	6.60 m (21.7 ft)	6.60 m (21.7 ft)
13	Drywell relative humidity	20%	20%
14	RPV pressure	7.17 MPa (1040 psia)	7.274 MPa (1055 psia)
15	RPV water level	NWL	NWL + 0.3 m (0.9 ft)

Vacuum Breakers. Section B.3.b of Appendix A to SRP Section 6.2.1.1.C specifies that the operability of all vacuum valves should be tested at monthly intervals to assure free movement of the valves. Operability tests are conducted at plants of earlier BWR designs using an air-actuated cylinder attached to the valve disk. The air-actuated cylinders have proven to be one of the root causes of vacuum breakers failing to close. Free movement of the vacuum breakers in the ESBWR design has been enhanced by eliminating this potential actuator failure mode, improving the valve hinge design, and selecting materials that are resistant to wear and galling. Therefore, GEH considers this requirement for monthly testing unnecessary for the ESBWR. However, the vacuum breakers will be tested for free movement during each outage. The operability of the vacuum beakers is verified per Technical Specification (TS) 3.6.1.6, “Suppression Wetwell-to-Drywell Vacuum Breakers.”

The staff determined that testing ESBWR vacuum breakers during each outage is acceptable for several reasons. First, proximity sensors are provided to detect if a vacuum breaker is not fully closed. Second, on the upstream side of the vacuum breaker, a dc-powered, solenoid-controlled, and spring-operated backup valve, designed to fail-close, is provided. Third, the containment analysis assumed that only two of three vacuum breakers would operate following a LOCA thereby providing a level of redundancy to address potential failure of a vacuum breaker (DCD Tier 2, Section 6.2.1.1.3.1).

ESBWR DCD Tier 2, Revision 3, does not provide the vacuum breaker opening and closing differential pressure settings used in the TRACG containment analysis of the DBA. Therefore, in RAI 6.2-99, the staff asked GEH to provide this information. In response, GEH provided the information. However, this information still needs to be added to the DCD. **RAI 6.2-99 is being tracked as an open item.**

In response to RAI 6.2-59, GEH stated that "[t]he ESBWR design uses 3 vacuum breakers. Assuming one vacuum breaker is out of service for the LOCA analyses, there should be 2 vacuum breakers available for the LOCA transient." Making three vacuum breakers available during a LOCA appears to be more conservative considering that a higher rate of noncondensable gas flow from the wetwell to drywell would degrade the passive containment cooling system (PCCS) more than when only two vacuum breakers are available. Therefore, in RAI 6.2-142, the staff requested that GEH explain this apparent nonconservative modeling of only two of three vacuum breakers being available during a LOCA. In response, in a letter dated June 14, 2007, GEH stated that vacuum breakers open during the early phase of the transient, and the maximum containment pressure for the period of 72 hours following a LOCA occurs at the end of this period. Therefore, two versus three vacuum breakers opening was expected to have a minimal impact on the PCCS performance in the long term and thus on the maximum containment pressure. The staff determined that the GEH response is acceptable. RAI 6.2-142 is closed.

Steam Bypass of the Suppression Pool. The potential exists for steam to bypass the suppression pool by leakage through the vacuum breakers or directly from leak paths in the drywell-to-suppression chamber vent pipes, the diaphragm-wall seal around diaphragm penetrations, or cracks in the concrete diaphragm. In response to RAI 6.2-12, GEH stated that a sensitivity analysis showed that the peak drywell pressure of an FWLB accident would approach the design pressure of 45 psig at 72 h after the pipe break if the leakage size were increased to $(A/\sqrt{K}) = 100$ cm^2 ($0.107 ft^2$). In RAI 6.2-147 the staff requested GEH to add this information to the DCD. In response, in a letter dated June 7, 2007, GEH stated that the latest containment analysis results included in DCD Tier 2, Revision 3, Section 6.2, indicate that the bounding LOCA break is an MSLB instead of an FWLB as reported in DCD Tier 2, Revision 2, Section 6.2. GEH referred to the containment analysis of an MSLB described in DCD Tier 2, Revision 3, Section 6.2.1.1.5.1, which states that the containment pressure remains below the design capability of the drywell with a bypass leakage of 2 cm^2 (2.16×10^{-3} ft^2) $(A/\sqrt{K})$. Therefore, the bypass leakage of 100 cm^2 ($0.107 ft^2$) $(A/\sqrt{K})$ is no longer limiting, and RAI 6.2-147 is closed.

DCD Tier 2, Revision 2, Section 6.2.1.1.5.1, states that the bounding design-basis calculation assumed a bypass leakage of 1 cm^2 $(A/\sqrt{K})$. This value is significantly lower than the design capacities of Mark I, II, and III containments, which are 18.6, 46.5, and 929 cm^2 $(A/\sqrt{K})$, respectively (SRP Section 6.2.1.1.C, Revision 6, August 1984).

DCD, Tier 2, Revision 2, Section 6.2.1.1.5.4.3, states that the acceptance criterion for the bypass leakage area for the leakage tests will be 10 percent of 1 cm^2 $(A/\sqrt{K})$ (i.e., 0.1 cm^2 $(A/\sqrt{K})$). The staff is concerned that this may be a low bypass leakage value, which plants may find difficult to confirm. Therefore, in RAI 6.2-145, the staff asked GEH to explain this statement.

In response GEH proposed an alternate acceptance criterion for the bypass leakage area for the leakage tests—the leakage which is analytically required to keep the containment below design pressure, 2 cm^2 (2.16×10^{-3} ft^2) (A/√K). GEH argued that the ability of the containment to tolerate degraded (increased) leakage up to ultimate strength had been determined to be more than a factor of 5 above the design capability. In RAI 6.2-145 supplement 1, the staff stated that its position that the containment design pressure but not the containment ultimate pressure should be used for determining design margins. The staff stated that GEH's proposed bypass leakage criterion is unacceptable to the staff and requested GEH to propose an acceptable bypass leakage acceptance criterion. RAI 6.2-145 addresses these issues. **RAI 6.2-145 is being tracked as an open item.**

Loss-of-Coolant Accidents. The staff reviewed the information provided in Section 6.2.1.1 of DCD Tier 2. During the review, the staff performed confirmatory containment analyses using the MELCOR computer code and, on December 11-15, 2006, performed an audit of GEH containment analysis.

Melcor computer code references:

R.O. Gauntt, et al. "MELCOR Computer Code Manuals: Primer and User's Guide," NUREG/CR-6119, Version 1.8.5, Vol. 1, Revision 2, December 2000.

R.O. Gauntt, et al. "MELCOR Computer Code Manuals: Reference Manuals," NUREG/CR-6119, Version 1.8.5, Vol. 2, Revision 2, December 2000.

R.O. Gauntt, et al. "MELCOR Computer Code Manuals: Demonstration Problems," NUREG/CR-6119, Version 1.8.5, Vol. 3, May 2001, September 2001.

Treatment of Noncondensable Gases

The stratification and holdup of noncondensable gases in the drywell during the blowdown phase of the LOCA and their later release can affect the performance of the PCCS. If the performance of the PCCS during the long-term cooling phase of the LOCA is degraded because of the presence of noncondensable gases which were not purged during the blowdown, then the steam that is not condensed in the PCCS will be vented to the suppression pool. This raises the temperature of the suppression pool and increases the containment pressure.

The NRC approved approach addressed uncertainties in the ability of TRACG to account for mixing and stratification in the drywell (NEDC-33083P-A). The NRC approved TRACG model consisted of a "tee" model to control the release of noncondensable gases from the lower drywell (NEDC-33083P-A and NEDE-32176P). The DCD model does not have such a "tee" model to control noncondensable gases, and the DCD does not describe the behavior of noncondensable gases. It appears that a newer model was used for the containment analysis presented in the DCD. Therefore, in RAI 6.2-52, the staff requested that GEH provide justification for the modeling changes and a discussion of containment response to the limiting DBA with respect to noncondensable gas holdup, movement, mixing, and stratification throughout the containment. The staff needs this information to determine whether noncondensable gas mixing and stratification in the containment are appropriately modeled in the evaluation of the ESBWR containment performance. In response, GEH described the modeling changes and the results of tieback calculations performed to determine the effect of the modeling change; the impact on containment performance from the modeling changes was minimal. GEH described the behavior of noncondensable gases in the containment adequately. However, in a supplemental request to RAI 6.2-52, staff requested GEH to provide the justification including the results of the tie-back calculations in the DCD or in a supplement to NEDC-33083P-A, "TRACG Application for ESBWR." **RAI 6.2-52 is being tracked as open item.**

In RAI 6.2-59, staff requested GEH provide (in graphical form) the GDCS double ended guillotine (DEG) break, the vessel bottom drain line DEG break and the MSLB DEG using licensing analysis assumptions to conservatively maximize the containment pressure or temperature response for each case. In response to RAI 6.2-59, GEH stated that it had found an error with the time step used for the TRACG analysis, and after correction of this error, the limiting DBA became an MSLB. In a supplemental requested to RAI 6.2-59, staff requested GEH to include the input error corrections and model enhancement information as a supplement to NEDC-33083P-A, "TRACG Application for ESBWR," March 2005. **RAI 6.2-59 is being tracked as an open item.**

The ESBWR DCD does not discuss the containment response to the limiting DBA with respect to the movement of noncondensable gases and mixing and stratification in the containment. This information is needed to review the containment performance in response to the limiting DBA. Therefore, in RAI 6.2-53, the staff requested that GEH provide this information. In response, GEH provided the results of nominal analysis for the limiting DBA. The staff makes its determination on containment performance based on bounding analysis but not on nominal analysis. Therefore, in RAI 6.2-98, the staff asked GEH to update its response to RAI 6.2-53 using bounding analysis. **RAI 6.2-98 is being tracked as an open item**.

Further, because the limiting DBA changed from the FWLB to the MSLB as discussed in RAI 6.2-59 (above), staff requested in a supplement to RAI 6.2-53 to reanalyze the containment response to the limiting DBA with respect to the movement of noncondensable gases and mixing and stratification in the containment using the MSLB as the limiting DBA. **RAI 6.2-53 is being tracked as an open item.**

Treatment of Non-Safety-Related Systems

DCD Tier 2, Section 19A.3.1.2, describes the ESBWR treatment of non-safety systems. The safety-related ICS and the safety-related PCCS provide the safety function of removing reactor decay heat from the core and containment. These systems are capable of removing decay heat for at least 72 hours without the need for active systems or operator actions. After 72 hours, makeup water is needed to replenish the boil-off from the upper containment pools. The ESBWR design includes permanently installed piping in the FAPCS that connects directly to a diesel-driven makeup pump system. This connection enables the upper containment pools and spent fuel pools to be filled with water from the fire protection system (FPS), which provides onsite makeup water to extend the cooling period from 72 hours to 7 days. The dedicated FPS equipment for providing makeup water and the flowpaths to the pools are classified as non-safety-related. A dedicated external connection to the FAPCS line allows for manual hook-up of external water sources, if needed, at 7 days for either upper containment pool replenishment and for spent fuel pool makeup. These functions are manually actuated from the yard area and can be performed without any support systems. The components within the scope of regulatory treatment of non-safety systems (RTNSS) are the diesel-driven makeup pump system, FAPCS piping connecting to the diesel-driven makeup pump system, and the external connection.

The ESBWR DCD was not clear on taking credit for the non-safety systems for the containment analysis. Therefore, in RAI 6.2-57, the staff asked GEH to discuss the effect of the non-safety systems in releasing the mass and energy releases into the containment and how these systems would respond during the DBAs analyzed (FWLB, MSLB, GDCS line break, and bottom drainline break). In response, GEH stated that ESBWR took no credit for the non-safety systems for the ECCS and containment analyses. GEH summarized the non-safety systems and described their functions and impact on the LOCA responses, if they are available. These systems are the high-pressure control rod drive (CRD) system, reactor water cleanup (RWCU)/shutdown cooling system, FAPCS in suppression pool cooling mode, FAPCS in drywell spray mode, and FAPCS in low-pressure coolant injection mode. GEH agreed to update the DCD to include this information. The staff determined that the GEH response addressed the concern and is acceptable. RAI 6.2-57 is being tracked as a confirmatory item.

The staff noticed that the containment pressure predicted for the limiting DBA continued to increase until the end of the calculation time of 72 hours following a LOCA, with a possibility of exceeding the containment design pressure after 72 hours.

In RAI 6.2-140, the staff requested that GEH justify its position that the containment pressure does not exceed the design pressure after 72 hours. **RAIs 6.2-140 and 6.2-177 are being tracked as open items.**

Single Failures Considered

ESBWR DCD Tier 2 did not describe the active single failures considered when analyzing the containment performance under DBAs. As stated in RG 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants (LWR Edition)," Section 6.2.1.4, a failure mode and effects analysis should be performed to determine the most severe single active failure for each break location for the purpose of maximizing the mass and energy released to the containment and the containment pressure response. The analysis should consider, for example, the failure of a steam or feedwater isolation valve, the feedwater pump trip, and containment heat removal equipment. Therefore, in RAI 6.2-58, the staff asked GEH to discuss the active single failures considered for each break type (FWLB, MSLB, GDCS line break, and vessel bottom line break) and to provide the resulting peak pressure and temperature for each case evaluated using appropriate licensing analysis assumptions to conservatively maximize the containment pressure or temperature response for each case.

In response, GEH stated that DCD Tier 2, Table 6.3-6, summarizes the single, active failures considered in the ECCS performance analysis. The assumed single failures are one DPV, one SRV, and one GDCS injection valve. Other postulated failures are not specifically considered, because they all result in at least as much ECCS capacity as one of the above failures. The assumed single failures for the containment analysis are one DPV and one SRV. Results of double-ended guillotine (DEG) pipe break analyses at four different locations show that an instantaneous guillotine rupture of an MSL with failure of one DPV produces the most limiting responses for the containment pressure evaluation. The second limiting case is an instantaneous guillotine rupture of an FWL with failure of one SRV.

The GEH response states that various single active failures were considered in regards to the ECCS analysis but not with respect to peak containment pressure. For example, an MSLB or FWLB with a failure of a shutoff valve in one of the standby liquid control system (SLCS) trains was not considered for peak containment pressure and temperature analysis. DCD Tier 2, Section 9.3.5.2, states that the operation of the accumulator vent could limit the amount of nitrogen injected into the reactor vessel by assisting in reducing accumulator pressure. However, if a shutoff valve in one of the SLCS trains fails, nitrogen could be transported to the reactor vessel until the accumulator tank depressurizes (with the assistance of the accumulator vent).

The effect of this event on the peak ESBWR containment pressure was not analyzed. Therefore, in supplemental RAI 6.2-58, the staff requested that GEH describe the active single failures considered with respect to peak containment pressure. **RAI 6.2-58 is being tracked as an open item.**

Initial Containment Conditions

DCD Tier 2, Table 6.2-2, lists the average drywell temperature during normal operation as 57.2℃ (135℉). However, DCD Tier 2, Table 6.2-6, lists the initial temperature used in analyzing the containment DBA cases as 46.1℃ (115℉). In RAI 6.2-64, the staff asked GEH to justify its position that the lower than average drywell temperature during normal operation used in the containment analysis would provide conservative results. GEH responded that the expected operating range of drywell temperature is from 46.1℃ (115℉) to 57.2℃ (135℉). Results from a previous sensitivity study on simplified boiling-water reactor (SBWR) design (Figure 4.3-2, NEDE-32178, Rev. 1) showed that increasing initial drywell temperature caused a decrease in the long-term drywell pressure. Cooler initial temperature represents more initial inventory for the noncondensable gases and consequently higher long-term containment pressure. Therefore, the reported DBA analyses were performed at 46.1℃ (115℉) to ensure conservative peak drywell pressure. GEH agreed to update the DCD to include this response. The staff determined that the GEH response is acceptable. RAI 6.2-64 is being tracked as a confirmatory item.

DCD Tier 2, Table 6.2-2, lists the average drywell relative humidity during normal operation as 50 percent. However, DCD Tier 2, Table 6.2-6, lists the initial relative humidity used in analyzing the containment DBA cases as 20 percent. In RAI 6.2-65, the staff asked GEH to justify its statement that the lower than average drywell relative humidity during normal operation used in the containment analysis would provide conservative results. GEH responded that the lower bound on the relative humidity in the drywell is 20 percent. It selected the lower-bound value because a lower initial drywell relative humidity results in more noncondensable gases available to be transferred to the wetwell and higher containment pressures following the LOCA. GEH agreed to update the DCD to include this response. The staff determined that the GEH response is acceptable. RAI 6.2-65 is being tracked as a confirmatory item.

DCD Tier 2, Table 6.2-2, lists the suppression pool temperature in hot standby as 130℉, while DCD Tier 2, Table 6.2-6, lists the initial suppression pool temperature used for the DBA analyses as 110℉, which is lower than the hot standby temperature. In RAI 6.2-67, the staff requested GEH (1) to justify that the suppression pool initial temperature used for the containment analysis would provide conservative results and (2) to describe the impact of operating the reactor at less than 100-percent power with respect to the stored energy and mass in the primary system which would be released to containment during a DBA.

In response to part (1) of the request, GEH stated that the suppression pool average temperature during normal operation was less than 110℉, and the maximum pool temperature of 110℉ was used in the safety analyses. According to the TS (DCD Tier 2, Chapter 16, Section 3.6.2.1), the reactor is required to reduce thermal power to less than 1 percent of rated thermal power when the suppression pool temperature is greater than or equal to 110℉, and the reactor will be switched to shutdown mode immediately when the suppression pool temperature is greater than or equal to 120℉. The staff determined that the GEH response to part (1) of the request is acceptable because it adequately addresses the staff's concern.

TRACG Modeling Parameters

In the “Pre-application Model,” as described in Section 3.3.1.1.1 of NEDC-33083, GEH conservatively modeled the suppression pool by forcing energy entering the pool to mix with and heat only the portion of the pool above the level of entry. This was accomplished by restricting the flow area of the suppression pool cells below the source of energy addition. The DCD was not clear as to whether the same model was used for the analysis presented in the DCD. Therefore, in RAI 6.2-55, the staff requested clarification from GEH. In response, GEH stated that it had used the same approach for all the DCD calculations, except for FWLB. Following an FWLB, energy addition from the spillover pipe continues in the long-term heatup, so the flow area restriction is not applied. The staff determined that considering the long-term energy addition to the pool by spillover flow following an FWLB, the exception for FWLB is acceptable. GEH agreed to update the DCD to reflect its response. RAI 6.2-55 is being tracked as a confirmatory item.

The ESBWR DCD does not provide information on energy sources available in the containment. This information is an input to TRACG analysis, which needs to be reviewed by the staff, and is also needed for the staff’s confirmatory containment analysis performed using the MELCOR computer code. Therefore, in RAI 6.2-63, the staff asked GEH to provide (1) the energy source information identified in RG 1.70, Table 6.9, for the limiting FWLB and limiting MSLB cases and (2) energy removal by the PCCS. In response, GEH provided the information which is acceptable to the staff, however the information needs to be added to the DCD. Therefore, in a supplement to RAI 6.2-63, staff requested GEH to update the DCD. **RAI 6.2-63 is being tracked as an open item.**

Previous versions of the DCD did not have information on how GEH evaluated the various containment volumes to ensure a conservative evaluation of the containment response to DBAs. These volumes include gas space in the drywell, wetwell, and GDCS pool and water volume in the suppression and GDCS pools. Therefore, in RAI 6.2-69, the staff asked GEH to provide this information.

In response, GEH stated that it had calculated the net drywell gas space volume by subtracting the displaced volumes occupied by equipment and structures located inside the drywell from the gross drywell volume. The gross drywell volume is calculated from the available arrangement drawings. GEH calculated the displaced volumes of equipment and structures, including the RPV, reactor shield wall, GDCS pool structures, RPV support brackets, fine motion control rod drives, and the protective layer on basemat, from the design drawings. GEH assumed, based on engineering judgment, that the other piping, equipment, and miscellaneous structures would displace a total of 1 percent of the gross volume.

GEH calculated the net wetwell gas space volume by subtracting the displaced volume occupied by the equipment hatches that are located in this region from the gross volume. GEH assumed that the displaced volume occupied by the equipment hatch was 0.1 percent of the total gross volume. GEH calculated the net gas space volume above the GDCS pools from the gross volume, assuming insignificant volume compared to the total gross volumes for other equipment and structures located in these regions. GEH calculated the gross wetwell volume from the available arrangement drawings. GEH calculated the net GDCS pool water volumes (total volume and nondrainable volume) from the available arrangement drawings and GDCS drain pipe suction elevation.

GEH calculated the net suppression pool water volume from the available arrangement drawings and assumed insignificant volume as compared to the total gross volumes for other equipment and structures located in these regions. GEH agreed to update the DCD to include this information. The staff determined that the GEH response was acceptable because it addressed the staff's concern. RAI 6.2-69 is being tracked as a confirmatory item.

Previous versions of the DCD did not have information on how GEH evaluated the various primary system volumes and heat structures (piping, RPV, etc.). DCD Tier 2, Table 6.2-6, provided the reactor power and reactor pressure for the bounding case but not the reactor temperature. The staff needs this information to determine whether these values were conservatively evaluated. Therefore, in RAI 6.2-70, the staff requested that GEH provide this information. In response, GEH described how it evaluated primary system volumes and heat structures using the available design drawings. Regarding the reactor temperature used for the containment analysis, GEH stated that the reactor dome temperature corresponds to the saturation temperature at the specified dome pressure. Staff found GEH's response acceptable. RAI 6.2-70 is being tracked as a confirmatory item.

GDCS Air Space

ESBWR DCD Tier 2, Section 6.2.1.1.10.2, states that the GDCS pools are placed above the RPV with their air space connected to the drywell, and that once the GDCS pools are drained, the total volume of the GDCS pools is added to the volume of the drywell airspace. The staff believes that adding volume to the drywell airspace was not possible because the water removed from the GDCS pools would occupy the drywell volume. In RAI 6.2-152, the staff requested an explanation from GEH. In response, GEH concurred that the statement was misleading because there was no net gain of drywell airspace resulting from the draining of the GDCS pools. GEH agreed to delete the statement from the DCD in a later revision. The staff determined that GEH had addressed the concern. RAI 6.2-152 is being tracked as a confirmatory item.

TRACG Modeling

The TRACG model used for the analysis presented in the DCD has an additional axial node in the upper wetwell that is not in the model used in preapplication, which was reviewed by the staff. In the preapplication TRACG model, the treatment of the upper wetwell limited mixing to conservatively assess the wetwell gas space temperature. In RAI 6.2-54, the staff requested GEH to (1) provide the rationale for adding the additional axial node, (2) state whether the same conservative approach used in the preapplication TRACG model was used in the DCD TRACG model, and (3) state whether the gas space temperature was treated conservatively. In response, GEH stated that there are 24 I-beams located at the top of the wetwell to support the diaphragm floor, and an additional axial node was added to the wetwell to refine the simulation of the trapped gas space between the I-beams. GEH stated that it had used the same conservative approach described in the preapplication model in the DCD TRACG model. GEH stated that the gas space temperature was treated in a conservative manner as described in the preapplication report. It applied an irreversible loss coefficient at the interface between the cells in the top two gas space levels to introduce forced stratification, thereby restricting flow between cells in the top two gas space levels. The staff determined that the GEH response addressed its concerns. GEH agreed to add this information in a supplement to DCD Section 6.2. RAI 6.2-54 is being tracked as a confirmatory item.

The ESBWR DCD does not provide information on passive heat sinks used in the containment analysis. The staff needs this information to perform confirmatory containment analysis. Therefore, in RAI 6.2-62, the staff asked GEH to provide this information as listed in RG 1.70, Table 6-11, per SRP Section 6.2.1.1.C. In response, GEH provided information acceptable to the staff except that this information needs to be added to the DCD. **RAI 6.2-62 is being tracked as an open item.**

The DCD identified the systems modeled as part of the DCD version of the TRACG model but did not show them in the nodal scheme. The staff needs a more complete nodalization, including, for example, the ICS, the SLCS, and the feedwater system, to review the TRACG model. Therefore, in RAI 6.2-72, the staff requested that GEH provide this information. In response, GEH provided the TRACG nodalization schematic diagrams for the ICS and feedwater system, which were later added to the DCD. GEH stated that the SLCS was simulated via a FILL component (FILL0037) that injected boric liquid into the RPV at the mid-elevation of the outer bypass (RPV axial Level # 5, Ring # 3). GEH agreed to update the DCD to include this information. The staff determined that the GEH response addressed its concerns. RAI 6.2-72 is being tracked as a confirmatory item.

The DCD was not clear as to (1) how GEH applied the ± 2σ uncertainty to the choked flow in lines, SRVs, DPVs, and both sides of breaks and (2) what critical flow models were used for choked flowpaths. The staff needs this information for its review. Therefore, in RAI 6.2-73, the staff asked GEH to provide this information. In response to part (1) of the request, GEH provided information that needs to be updated as a result of the staff's request in RAI 6.2-141. In response to part (2) of the request, GEH stated that the TRACG critical flow model was applied to all flowpaths at locations where the choking calculation was specified in the input model. These choked paths included the SRVs, DPVs, FWLB (RPV side), FWLB (balance-of-plant (BOP) side), and drywell main vents. GEH updated the DCD to include this information. The staff determined that GEH's response to RAI 6.2-73 is acceptable. RAI 6.2-73 is resolved.

TRACG Results

Previous versions of the DCD provided the results in graphic form only for FWLB but not for GDCS line break, vessel bottom line break, or MSLB. The staff needs the results for these other breaks for its review of containment response to DBAs. Therefore, in RAI 6.2-59, the staff asked GEH to provide this information. In response, GEH stated that it had discovered an erroneous result for FWLB (i.e, an early peak in drywell pressure), because the FWLB was sensitive to time step size. GEH found that the pressure disturbance was the result of a numerical problem, commonly known as "water packing." Water packing generally occurs when steam is condensing in the subcooled water in a confined volume. Usually, this numerical problem can be avoided by using smaller time step size during the time period when the water packing problem is likely to occur. Lowering the time step from 0.05 to 0.025 corrected this problem. GEH also stated that it had corrected three input errors in vacuum breaker flow area, SLCS flow input table, and axial power input into part-length fuel rods and enhanced models for vapor additive friction loss coefficients. GEH needs to supplement NEDC-33083P-A (NEDC-33083) to include this information. GEH provided error-corrected and model- enhanced results for FWLB, GDCS line break, vessel bottom line break, and MSLB and updated the DCD. The staff reviewed the results and found them acceptable. RAI 6.2-59 addresses the issue of supplementing NEDC-33083P-A. **RAI 6.2-59 is being tracked as an open item.**

ESBWR DCD Tier 2 states that only DEG breaks were analyzed. However, the DCD also states that a spectrum of sizes was evaluated but does not describe the results. The information on containment analysis for breaks smaller than DEG breaks is needed to confirm that the four DEG breaks analyzed (FWLB, GDCS line break, vessel bottom line break, and MSLB) were limiting DBAs. Therefore, in RAI 6.2-60, the staff requested that GEH (1) confirm whether only four DEG breaks with different locations and sizes were analyzed, (2) provide the results of sensitivity analyses for smaller than DEG break sizes for FWLB and MSLB to ensure that DEG breaks were limiting, and (3) provide the results of sensitivity analyses for MSLB at high and low locations in the containment to justify that the MSLB analyzed was limiting. In response to part (1) of the request, GEH clarified that it had performed containment design-basis calculations for a spectrum of four DEG pipe break sizes and locations to ensure that it had identified the worst case and updated the DCD to include this clarification. The staff determined that the response was acceptable. In response to part (2) of the request, GEH provided and described the results of parametric analyses performed with different break areas (40 percent, 60 percent, 80 percent, and 100 percent of the DEG break area) for FWLB and MSLB. These analyses showed that the breaks with 100 percent of the DEG break areas were limiting. This confirmed that the assumed 100-percent DEG break size for the DBA MSLB analysis was limiting. GEH needs to update the DCD to include a summary of results. In response to part (3) of the request, GEH provided and described results of the base-case calculation performed for a break occurring in the drywell at level 34 as shown in DCD Tier 2, Figure 6.2-7, and parametric calculations for breaks occurring at levels 31, 25, and 23. The base case with the highest break location generated the highest maximum drywell pressure. This confirmed that the base-case break location assumed for the DBA MSLB analysis was limiting. GEH needs to update the DCD to include a summary of results. RAI 6.2-60 addresses issues related to parts (2) and (3) of the request. **RAI 6.2-60 is being tracked as an open item.**

For the DBAs analyzed, ESBWR DCD Tier 2 does not provide mass and energy release data, mass inventories for systems modeled, and gas and pool stratification data. The staff needs this information for its review of TRACG containment analysis. Therefore, in RAI 6.2-61, Part 1, the staff asked GEH to provide mass and energy release data from the RPV side and from the BOP side of the break for the limiting FWLB and limiting MSLB.

In RAI 6.2-61, Part 2, the staff requested that GEH provide, for the limiting FWLB and limiting MSLB, (a) mass and energy release from the safety valves and DPVs, (b) mass flow through GDCS, PCCS, ICS, SLCS, hydraulic control units (HCUs), drywell main vents, wetwell to drywell vacuum breakers, and drywell leakage, (c) RPV water level-collapsed and two-phase, drywell pool level, suppression pool level, GDCS water level, PCCS/ICS upper pool level, noncondensable partial pressure in the drywell and wetwell, (d) local gas and pool temperatures in the drywell, wetwell, and RPV to reveal regional stratification for selected nodes, and (e) suspended liquid water masses for the RPV steam dome, drywell, and wetwell volumes.

In response to RAI 6.2-61, Part 1, GEH provided graphic results for the break flow mass and energy release and the effective break area for FWL and MSL cases. GEH agreed to update the DCD to include these results.

In response to RAI 6.2-61, Part 2, GEH provided graphic results for FWL and MSL cases. GEH agreed to update the DCD to include the results provided in response to items (a), (b), and (c) of the request.

During an NRC audit conducted between December 11 and December 15, 2006, resuming for the period between December 19 and December 20, 2006., GEH stated that it had made several changes to the TRACG containment model. GEH agreed to identify these changes in an appendix to DCD Tier 2, Section 6. GEH later added DCD Tier 2, Revision 3, Appendix 6A to identify the changes. GEH made a design configuration change to designate feedwater isolation as safety grade, which made MSLB the limiting DBA for containment performance. GEH supplemented its response to RAI 6.2-61 by providing nominal and bounding analyses for the MSLB**.** The staff used the information provided in response to RAI 6.2-61 for performing confirmatory containment performance analysis using the MELCOR computer code. RAI 6.2-61 is resolved.

Previous versions of the DCD provided predictions for containment temperature in graphs of temperature versus time for DBAs analyzed. However, GEH did not provide information on how the temperatures were combined to determine the values shown in the graphs because the DCD version of the TRACG model was nodalized for the free volumes and pool regions. The staff needs this information to compare its confirmatory containment analysis results with the GEH results. Therefore, in RAI 6.2-71, the staff requested that GEH provide this information. In response, GEH stated that the temperatures provided represent the maximum envelope of the corresponding temperatures from all the cells residing in the region of interest.

GEH stated that individual cell temperatures would better describe the response to thermal stratification (such as that in the suppression pool and in the wetwell). GEH agreed to (1) replace or supplement the graphs in the DCD with those showing individual cell temperatures and (2) describe the response in the DCD. The staff determined that the GEH response was acceptable because it addressed the concerns. RAI 6.2-71 is being tracked as a confirmatory item.

In figure titles, DCD Tier 2 incorrectly refers to noncondensable gas as “air.” For example, see “Figure 6.2-14d1. Main Steam Line Break (Bounding Case)—Drywell and GDCS Air Pressures (72 hrs).” In RAI 6.2-176, the staff requested that GEH correct this. **RAI 6.2-176 is being tracked as an open item.**

Post 72-Hour Containment Pressure Control

ESBWR DCD Tier 2 provided TRACG results for up to 72 hours following the initiation of a LOCA. The maximum drywell pressure predicted by TRACG for the limiting DBA of MSLB is 384.2 kPa (55.8 psia), which is 29.0 kPa (4.2 psi) below the containment design pressure of 413.7 kPa (60 psia). However, the maximum drywell pressure is predicted to occur at 72 hours, when the calculation ends, and the pressure increases continually with a possibility of exceeding the design pressure post-72 hours. GDC 50 requires in part that the containment and its associated systems can accommodate, without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions resulting from any LOCA. The staff was unable to conclude that the ESBWR design meets the requirements of GDC 50 because the ESBWR DCD does not contain a prediction of containment pressure after 72 hours following a LOCA. Therefore, in RAI 6.2-140, the staff asked GEH to justify that the containment pressure does not exceed the containment design pressure post-72 hours. **RAI 6.2-140 is being tracked as an open item.**

Staff Audit of TRACG Containment Analysis

The staff audited the GEH TRACG containment analysis between December 11 and December 15, 2006, resuming for the period between December 19 and December 20, 2006. The following is a summary of the staff's observations and their resolution.

The amount of noncondensables in the GDCS airspace is sensitive to whether a single pipe node or a double pipe node is used in modeling the junction between the GDCS airspace and the drywell. GEH later changed the TRACG nodalization to use a double pipe junction for bounding DBA containment analyses.

The staff requested that GEH update the TRACG LOCA application to the ESBWR with consideration of the modeling changes that have been made since the original approval. GEH agreed and later provided this information as Appendix A to DCD Tier 2, Chapter 6.

DCD Tier 2, Revision 2, Section 6.3, assumed the availability of the containment back pressure in determining the minimum water level in the RPV following a LOCA. The depressurization of the RPV and thus the initiation of GDCS depends on the assumptions used for determining the containment back pressure. However, the GEH analyses are inconsistent with SRP Section 6.2.1.5, "Minimum Containment Pressure Analysis for Emergency Core Cooling System Performance Capability Studies," Revision 2, July 1981, and the associated Branch Technical Position CSB 6-1, "Minimum Containment Pressure Model for PWR ECCS Performance Evaluation." Although, CSB 6-1 was developed to evaluate the performance of the ECCS of a pressurized-water reactor, most of its guidance also applies to determining the performance of the GDCS for the ESBWR. Specifically, the input information for the model, active heat sinks (e.g., FAPCS operating in drywell spray mode) and passive heat sinks affect the containment back pressure. During the audit, the staff requested GEH to justify the containment back pressure used for determining the minimum RPV water level considering Branch Technical Position CSB 6-1. The staff later issued RAI 6.2-144 to formally request this information. In response, GEH evaluated the impact of containment back pressure on the ECCS performance and presented in ESBWR DCD Tier 2, Revision 4, Appendix 6C. The staff reviewed GEH's evaluation and determined that the minimum chimney collapsed level is not sensitive to the changes in the containment back pressure expected for the ESBWR design under LOCA conditions. RAI 6.2-144 is resolved.

Staff Confirmatory Analysis

The staff used the MELCOR computer code to perform confirmatory analysis for the ESBWR DBA containment performance evaluation for the bounding MSLB scenario as presented in DCD Revision 3. The MELCOR model was set up using the bounding initial and model parameters and biases as described in the DCD and GEH responses to staff's RAIs. The MELCOR model used a well-mixed drywell volume resulting in minimal noncondensable gas trapping.

Table 6.2-2 lists a sequence of events and compares the predicted timing of events. Automatic depressurization system (ADS) actuation agreement is within a few seconds between the DCD reported time and those times calculated with the MELCOR model. MELCOR predicted that the expansion/PCC tank reflood would occur 34,376 s (9.55 h) earlier than predicted by TRACG. However, the reflood timing has a small impact on containment pressure responses since the PCCS system efficiency is not notably affected by the relatively small amount of tube length uncovered before reflood (about one-fourth uncovered).

The difference in the reflood timing is the result of differences between the TRACG and MELCOR models relative to the trapping of drywell gases, and subsequently, the rate of release of those gases to the PCCS. The agreement between TRACG and MELCOR event timings is reasonable.

Table 6.2-3 presents a summary of maximum pressures calculated and their safety margins for the bounding MSLB scenario using TRACG and MELCOR computer codes. Both TRACG and MELCOR predicted the maximum pressure occurring at 72 hours following an MSLB. The comparisons of pressure profiles between the DCD and MELCOR calculation for the bounding MSLB case are quite good if the blowdown period can be excluded.

However, as the licensing focus moves from blowdown to later times, such as the GDCS recovery period and long-term cooling, the pressures reported in the DCD and calculated with MELCOR are essentially equivalent. At 72 hours, the DCD-reported drywell pressure of 384.2 kPa and the MELCOR-calculated pressure of 370.5 kPa provide reasonable confirmation of the certification analysis presented in the DCD. Safety margins for the DCD and MELCOR calculation are 9.4 and 13.8 percent, respectively.

Table 6.2-3 also presents the results of the MELCOR calculations performed to address the long-term pressurization sensitivity to radiolytic gas source and bypass leakage area. The doubling of the bypass leakage capacity reduced the margin to the design pressure from 13.8 to 4.5 percent. These results indicated that the impact of bypass leakage capacity on the containment pressure is significant. The bypass leakage capacity is discussed above.

Table 6.2-2 Sequence of Events for MSLB (Bounding Case) with Failure of One DPV

Event	Time (s)	
	DCD Tier 2 Revision 3	MELCOR
Guillotine break of MSL inside containment	0	0
Main vent clearing time Top vent: Middle vent: Bottom vent:	 1.8 2.3 3.1	 1.1 1.6 2.8
Reactor isolated	13	13
Level 1 is reached	496	482
Level 1 signal confirmed; ADS/GDCS/SLCS timer initiated; SRV actuated	506	492
DPV actuation begins at 50 s after confirmed Level 1 signal; SLCS flow starts	556	542
GDCS flow into vessel begins	726	686
SLCS flow depleted	856	832
PCC pool drops below the elevation of 29.6 m; water from dryer/storage pool flows into expansion pool	126,776	92,400
DW pressure peak	259,000 ~72 h (384.2 kPa)	259,000 ~72 h (370.5 kPa)

Table 6.2-3 Summary of Peak Pressures Calculated for the Bounding MSLB Scenario Using TRACG and MELCOR Computer Codes

Case	TRACG (DCD Rev. 3)		MELCOR	
	Pressure (kPa)	Margin to Design Pressure (%)*	Pressure (kPa)	Margin to Design Pressure (%)
Reference	384.2	9.7	370.5	14.3
Radiolytic gas generation terminated at 12 h	---	---	347.	22
Bypass leakage doubled	---	---	400.	4.5

* Margin to Design Pressure (%) = (DPL- maximum pressure)/(DPL - initial pressure) * 100 where design pressure limit (DPL) = 413.7 kPa; initial pressure 110.3 kPa

Negative Pressure Design Evaluation. ESBWR DCD Tier 2, Revision 3, Section 6.2.1.1.4, states that the MSLB will not result in unacceptable results, but it does not indicate if other LOCAs were evaluated to conclude that this is the limiting case. In RAI 6.2-11, the staff requested that GEH discuss how the limiting cases were identified for both the drywell and wetwell. In response, GEH provided adequate information; however, staff needs to verify the information is included in the next revision of the DCD. RAI 6.2-11 is being tracked as a confirmatory item.

6.2.1.1.4 Conclusions

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

6.2.1.2 Containment Subcompartments

6.2.1.2.1 Regulatory Criteria

The staff reviewed ESBWR DCD Tier 2, Section 6.2.1.2, “Containment Subcompartments,” in accordance with SRP Section 6.2.1.2, “Subcompartment Analysis,” Revision 2, July 1981.

The acceptance criteria given below apply to the design and functional capability of subcompartments in the primary containment.

- GDC 4, as it relates to the environmental and missile protection provided to assure that SSCs important to safety are designed to accommodate the dynamic effects (e.g., effects of missiles, pipe whipping, and discharging fluids that may result from equipment failures) that may occur during plant normal operations or during an accident

- GDC 50, as it relates to the subcompartments being designed with sufficient margin to prevent fracture of the structure because of pressure differential across the walls of the subcompartment.

When performing analyses to demonstrate compliance with the requirements of GDC 50, the following assumptions and modeling schemes should be used to ensure that the results are conservative.

- The initial atmospheric conditions within a subcompartment should be selected to maximize the resultant differential pressure.

- Subcompartment nodalization schemes should be chosen such that there is no substantial pressure gradient within a node (i.e., the nodalization scheme should be verified by a sensitivity study that includes increasing the number of nodes until the peak calculated pressures converge to small resultant changes). The guideline of Section 3.2 of NUREG-0609, "Asymmetric Blowdown Loads on PWR Primary Systems, issued January 1981, should be followed, and a nodalization sensitivity study should be performed which includes consideration of spatial pressure variation, e.g., pressure variations circumferentially, axially, and radially within the subcompartment, for use in calculating the transient forces and moments acting on components.

- If vent flowpaths are used that are not immediately available at the time of pipe rupture, the following criteria apply:

  – The vent area and resistance as a function of time after the break should be based on a dynamic analysis of the subcompartment pressure response to pipe ruptures.

  – The validity of the analysis should be supported by experimental data or a testing program should be proposed at the construction permit or design certification stage that will support this analysis.

  – In meeting the requirements of GDC 4, the effects of missiles that may be generated during the transient should be considered in the safety analysis.

- The vent flow behavior through all flowpaths within the nodalized compartment model should be based on a homogeneous mixture in thermal equilibrium, with the assumption of 100-percent water entrainment.

In addition, the selected vent critical flow correlation should be conservative with respect to available experimental data. Currently acceptable vent critical flow correlations are the "frictionless Moody" (Moody, F.J., "Maximum Flow Rate of a Single Component, Two-Phase Mixture," Journal of Heat Transfer, Trans. ASME, Series C, Vol. 87, p. 134, February 1965) with a multiplier of 0.6 for water-steam mixtures and the thermal homogeneous equilibrium model for air-steam-water mixtures.

6.2.1.2.2 Summary of Technical Information

The design of the containment subcompartments was based on a postulated DBA occurring in each subcompartment.

For each containment subcompartment in which high-energy lines are routed, mass and energy release data corresponding to a postulated double-ended line break were calculated. The mass and energy release data, subcompartment free volumes, vent path geometry, and vent loss coefficients were used as input to an analysis to obtain the pressure/temperature transient response for each subcompartment. In addition to the drywell and the wetwell, the containment has two subcompartments, the drywell head region and the reactor shield annulus (RSA).

Drywell Head Region

The drywell head region is covered with a removable steel head, which forms part of the containment boundary. The drywell bulkhead connects the containment vessel flange to the containment and represents the interface between the drywell head region and the drywell. There are no high-energy lines in the drywell head region.

Reactor Shield Annulus

The RSA exists between the reactor shield wall (RSW) and the RPV. The RSW is a steel cylinder surrounding the RPV and extending up close to the drywell top slab, as shown in DCD Figure 6.2-2. The opening between the RSW and the drywell top slab provides the vent pathway necessary to limit pressurization of the annulus resulting from a high-energy pipe rupture inside the annulus region. The shield wall is supported by the reactor support structure. Several high-energy lines extend from the RPV through the RSW. There are also penetrations in the RSW for other piping, vents, and instrumentation lines. The RSW is designed for transient pressure loading conditions from the worst high-energy line rupture inside the annulus region.

GEH used the TRACG computer program to perform the RSA subcompartment evaluation.

6.2.1.2.3 Staff Evaluation

The staff reviewed ESBWR DCD Tier 2, Section 6.2.1.2, and plans to perform confirmatory containment subcompartment analyses. During its review, the staff issued the RAIs described below.

DCD Tier 2 does not provide a synopsis of the piping break analyses performed and a justification for the selection of the DBA (break size and location), nor does it describe whether the leak-before-break was assumed to limit the pipe break area. The staff needs this information for its review of the GEH subcompartment analysis. Therefore, in RAI 6.2-13, the staff requested that GEH provide this information. In response, GEH stated that RSA was the only subcompartment, in addition to drywell and wetwell subcompartments, requiring assessment of pipe breaks. GEH assessed four types of pipe break (MSL, FWL, GDCS injection line, and bottom drainline) for the drywell and wetwell compartments. GEH assessed two types of pipe break (FWL and RWCU) for the RSW. GEH selected the break locations to maximize the mass and energy release into the subcompartment. The break locations are usually the pipe segments on any flowpath with the largest cross-section in the containment. GEH did not assume leak-before-break to limit the break area because it postulated DEG breaks for all pipe breaks. Staff finds this acceptable except that GEH needs to update the DCD to include this information. RAI 6.2-13 is being tracked as a confirmatory item.

ESBWR DCD Tier 2 does not state whether pipe restraints are used to limit the break area of the pipe ruptures. In response to RAI 6.2-14, GEH stated that it took no credit in the analysis to limit the break area because of the presence of pipe restraints. The staff asked GEH to provide this information in the DCD. **RAI 6.2-14 is being tracked as an open item.**

ESBWR DCD Tier 2, Section 6.2.1.2, states that a factor of 1.4 is applied to the peak differential pressure calculated for the subcompartment, structure, and the enclosed components. However, DCD Tier 2, Section 6.2.1.2.1, states that at least 15 percent margin above the analytically determined pressure is applied for structural analysis. Therefore, in RAI 6.2-15, the staff requested that GEH clarify this apparent discrepancy. **RAI 6.2-15 is being tracked as an open item.**

The staff was unable to determine from the information provided in ESBWR DCD Tier 2 whether possible insulation collapsing in the containment subcompartment affects the vent areas used in the analyses. In response to RAI 6.2-17, GEH stated that the RSA subcompartment vent areas in ESBWR containment are always open, and no insulation collapsing would occur in this subcompartment. The staff asked GEH to provide this information in the DCD. **RAI 6.2-17 is being tracked as an open item**.

ESBWR DCD Tier 2, Section 6.2.1.2.3, states that the mass release rates are determined with Moody's frictionless critical flow model. This section also states that, when analyzed with TRACG, the peak subcompartment pressure responses were found to be below the design pressure for all postulated pipe break accidents.

ESBWR DCD Tier 2, Section 6.2.1.2, states that a factor of 1.4 was applied to the peak differential pressure calculated for the subcompartment, structure, and enclosed components for use in the design of the structure and the component supports. ESBWR DCD Tier 2, Section 6.2.1.2.1, states that at least 15-percent margin above the analytically determined pressures was applied for structural analysis. The DCD does not specifically state which margin was used in the design and how it relates to the factor of 1.4 and 15-percent margins. In RAI 6.2-18, the staff requested clarification from GEH. **RAI 6.2-18 is being tracked as an open item.**

ESBWR DCD Tier 2, Section 6.2.1.2.3, states that the TRACG computer code was used for the ESBWR containment subcompartment analysis. However, ESBWR DCD Tier 2 does not provide information on the conservatism of the blowdown model with respect to the pressure response of the subcompartment and a justification for using TRACG for subcompartment analysis. Therefore, in RAI 6.2-19, Supplement 1, the staff requested GEH to provide this information.

In response, in a letter dated June 5, 2006, GEH stated that TRACG was qualified for analysis of SBWR and ESBWR reactor system and containment in NEDC-32725 and NEDC-33083. GEH provided results of time step sensitivity analyses on peak maximum pressures and provided the sizes of the smallest nodes that are located around the postulated break. GEH agreed to provide this information in a proprietary licensing report for reference in the DCD. GEH stated that it had performed sensitivity studies to assess the effects of annulus volume, RSW vent flow area, and annulus hydraulic diameters and found the effects to be minor. The staff determined that the GEH response addressed its concerns. RAI 6.2-19 is being tracked as a confirmatory item.

Section 6.2.1.2.3 states that the containment subcompartment analysis assumed that the reactor is operating at full power and the containment is filled with dry air at atmospheric pressure and 100℃ when the postulated pipe break occurs. ESBWR DCD Tier 2 does not state whether the reactor power was adjusted to account for measurement uncertainties and does not justify using air while the ESBWR containment is inerted with nitrogen. Therefore, in RAI 6.2-20, Supplement 1, the staff asked GEH to provide this information. **RAI 6.2-20 is being tracked as an open item.**

ESBWR DCD Tier 2 did not describe and justify the subsonic and sonic flow models used in vent flow calculations and did not state and justify the degree of entrainment assumed for the vent mixture. The staff needs this information to evaluate the ESBWR subcompartment loading. Therefore, in RAI 6.2-21, the staff requested that GEH provide this information. In response, GEH stated that it used the frictionless Moody critical mass flux correlation to model the break flow and that the model assumed critical velocity at the break and therefore was conservative. The frictionless Moody critical mass flux correlation is acceptable to the staff as stated in SRP Section 6.2.1.2. GEH stated that it used the TRACG interfacial shear model described in a paper by F.J. Moody ("Maximum Flow Rate of a Single Component, Two-Phase Mixture," Journal of Heat Transfer, Trans. ASME, Series C, Vol. 87, p. 134, February 1965). GEH agreed to update the DCD to include this information. The staff determined that the GEH response is acceptable because it addressed the staff's concern. RAI 6.2-21 is being tracked as a confirmatory item.

ESBWR DCD Tier 2 does not provide information on the containment subcompartment nodalization. In response to RAI 6.2-23, GEH provided nodal data but stated without specifics that it calculated large pipe and vessel support structure volumes and hydraulic diameters and accounted for the additional obstructions by applying a 10-percent reduction factor in the annulus volume for cells where a specific obstruction is not modeled. The staff needs the details of nodalization to perform its confirmatory analysis, and these details should be provided in the DCD. In supplemental RAI 6.2-23, the staff requested this information from GEH. **RAI 6.2-23 is being tracked as an open item.**

ESBWR DCD Tier 2 does not provide graphs of the pressure responses of subnodes within a subcompartment as functions of time. This information is need for evaluations of the effect on structures and component supports. Therefore, in RAI 6.2-24 Supplement 1, staff requested GEH to add this information to the DCD. **RAI 6.2-24 is being tracked as an open item.**

ESBWR DCD Tier 2 does not provide the mass and energy release data for the postulated pipe breaks. In response to RAI 6.2-25, GEH provided the method used to calculate mass and energy release date but not the actual data. The staff requested the actual data. GEH provided this information. RAI 6.2-25 is resolved.

ESBWR DCD Tier 2 did not state the flow conditions (subsonic or sonic) for vent flowpaths up to the time of peak pressure. This information is needed to evaluate ESBWR subcompartment loads per SRP Section 6.2.1.2 and RG 1.70, Section 6.2.1.2. Therefore, in RAI 6.2-26, the staff requested GEH to provide this information. In response, GEH stated that before the time of peak pressure, the vent flow was subsonic.

GEH agreed to update the DCD to provide this information. The staff determined that the GEH response was acceptable because it addressed the staff's concern. RAI 6.2-26 is being tracked as a confirmatory item.

ESBWR DCD Tier 2 does not provide information on the time step and nodalization study, code validation, and comparison to approved methods. This information is needed to justify modeling and using the TRACG computer program for the containment subcompartment analysis. Therefore, in RAI 6.2-29, Supplement 1, the staff asked GEH to provide this information. **RAI 6.2-29 is being tracked as an open item.**

6.2.1.2.4 Conclusions

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

6.2.1.3 Mass and Energy Release Analysis for Postulated Loss-of-Coolant Accidents

Staff's review of the DCD to meet SRP, 6.2.1.3, "Mass and Energy Release Analysis for Postulated Loss-of-Coolant Accidents," is located in Section 6.2.1.1 of this report.

6.2.1.4 Mass and Energy Release Analysis for Postulated Secondary System Pipe Ruptures

SRP 6.2.1.4, "Mass and Energy Release Analysis for Postulated Secondary System Pipe Ruptures," applies to PWRs and thus is not applicable to the ESBWR.

6.2.1.5 Minimum Containment Pressure Analysis for Emergency Core Cooling System Performance Capability Studies

SRP 6.2.1.5, "Minimum Containment Pressure Analysis for Emergency Core Cooling System Performance Capability Studies," applies to PWRs and thus is not generally applicable to the ESBWR. However, during a December 2006 audit, the staff raised an issue with possible implications of the minimum containment pressure on the initiation timing of GDCS injection, and thus on the ECCS performance. As described in Section 6.2.1.1 of this report, this issue was resolved by issuing RAI 6.2-144. GEH added DCD Tier 2, Revision 4, Appendix 6C to provide an evaluation of the impact of containment backpressure on the ECCS performance.

6.2.1.6 Suppression Pool Dynamics Loads

6.2.1.6.1 Regulatory Criteria

To meet the requirement of GDC 4, regarding the dynamic effects associated with normal and accident conditions, calculation of dynamic loads should be based on appropriate analytical models and supported by applicable test data. The calculations should consider loads on suppression pool retaining structures and structures that may be located directly above the pool, as a result of pool motion during a LOCA or following actuation of one or more reactor coolant system safety/relief valves.

6.2.1.6.2 Summary of Technical Information

GEH submitted DCD Tier 2, Appendix 3B, to define the containment hydrodynamic load definitions for the ESBWR. The methodology used to develop these load definitions and the justification for their applicability to ESBWR is given in a proprietary report, NEDE-33261P, “ESBWR Containment Load Definition,” issued May 2006.

The NEDE-33261P report provides a description and load definition methodology for hydrodynamic forces acting on the ESBWR primary containment during a postulated LOCA and/or SRV or DPV actuation. The load definition methodology used for the ESBWR containment design is similar to that used for earlier BWR containment designs and particularly similar to that used and approved for the ABWR design.

The geometries of the pressure suppression systems in the ABWR and ESBWR designs are similar. Key differences between the two containment designs are listed in Table 6.2.1.6-1.

Table 6.2.1.6-1

Parameter	ESBWR	ABWR
Number of vertical vents	12	10
Suppression pool angular sector per vertical vent (deg)	30	36
Pool depth (m)	5.5	7.1
Top vent submergence (m)	2.0	3.6
Distance from vent exit to outer containment wall (m)	9.0	6.85
Pool surface area per vent (m^2)	66.6	50.7
Vertical vent distance between drywell entrance and top vent entrance (m)	9.35	17.0

In both ABWR and ESBWR designs, the drywell and the annular suppression pool are connected by a set of circular vertical vents of the same diameter, each with three circular horizontal vents, also of the same diameter, and at the same elevations, extended into the suppression pool to the same distance.

Since there is a high degree of geometric similarity between the ESBWR and ABWR containments, the physical phenomena associated with the postulated DBA events during the first few minutes into the accidents are identical for both designs. The following is a description of these phenomena, based on the NUREG-1503, "Final Safety Evaluation Report Related to the Certification of the Advanced Boiling Water Reactor Design," issued July 1994, and the NEDE-33261P report.

LOCAs and SRV discharges, as well as the DPV actuation, are the events that can impose dynamic loads on the suppression pool. SRVs discharge steam from the reactor pressure valve (RPV) through discharge piping that is routed into the suppression pool and fitted at the pool end with a quencher to enhance heat transfer between the hotter SRV discharge fluid (steam and air) and the cooler suppression pool water. DPV discharges the mass and energy to the containment, increasing the mass flux through the main vents. However, this additional mass flux is bounded by the LOCA vent mass flux and, therefore, the containment hydrodynamic loads calculated for DBA LOCA are used for the design.

Since the ESBWR design has no recirculation line, the largest postulated pipe breaks are FWLB and MSLB. The dynamic loads in the suppression pool caused by these events can be characterized by several phenomena that occur in the order of (1) vent clearing, (2) PS, (3) high steamflow, and (4) chugging (CH). After an FWLB or MSLB, with sufficient pressurization of the drywell, water in the vents is forced out into the pool. This vent water clearing causes submerged jet-induced loads on nearby structures and the pool basemat. After vent clearing, an air and steam bubble flows out of the vents. The air component, originating from the drywell, expands in the pool causing a rise in pool surface level, referred to as PS, and imposing loads on submerged structures and pool boundaries. After PS, a period of high steamflow occurs, and steam is condensed in the pool vent exit area, causing pressure oscillations in the pool. This phenomenon, referred to as condensation oscillation (CO), produces oscillatory and steady loadings on the containment structure. Later, as vent steamflow decreases, a steam bubble may occur, and its sudden collapse creates oscillatory loads. This process (CH) imposes significant vent and suppression pool boundary loads.

The CO experiments (e.g., NEDC-31393, "Containment Horizontal Vent Confirmatory Test, Part I," Revision 0 (proprietary), March 1987) indicate that there are also local effects on wall, liner, and submerged structures within two vent diameters of each horizontal vent. The methodology, as presented in NEDE-33261P, includes this phenomenon.

One of the ESBWR unique design features is the PCCS (see Section 6.2.2). Its operation, which immediately follows a LOCA, would mitigate to some extent the PS loads calculated for the scenario described above, although the LOCA analysis did not credit the performance of the PCCS for the first several minutes of the postulated accident.

Other postulated LOCAs, intermediate and small, lead to similar scenarios and the resulting PS, CO, and CH loads are bounded by those calculated for the DBA LOCA.

For certain reactor transients, the pressure relief is through activation of the SRV. For these events, the steam discharge into the suppression pool consists of three phases, water clearing, air clearing, and steamflow. The discharge pipe standing column of water first is pushed out, or cleared, into the pool by blowdown steam pressure. Water clearing creates SRV pipe pressure and thermal loads, pipe reaction forces, drag loads on structures submerged in the pool, and pool boundary loads. After water clearing, air clearing occurs as air above the water column in the pipe is forced out of the pipe and into the pool. The air-clearing phase generates expanding bubbles in the pool that cause transient drag loads on a submerged structure as a result of both the velocity and acceleration fields and oscillating pressure loads on the pool boundary. Finally, the steam-flow phase creates pipe reaction forces, quencher thrust forces, structure thermal loads, and oscillating pool boundary loads as a result of steam jet condensation at the quencher.

The ESBWR SRV discharge is directed to the suppression pool through X-quenchers that GEH has stated are identical to the quenchers used for the Mark III designs. GEH also stated that the calculation methodology used for establishing the ESBWR quencher discharge loads is the same as previously used for ABWR, Mark II, and Mark III containments. In brief, the methodology is based on empirical correlations derived from the test of various scales. Therefore, GEH concluded that the hydrodynamic load methodology developed for the Mark II and Mark III designs was applicable to both the ESBWR suppression pool geometry and the X-quencher configuration.

During the ABWR review, the staff raised an issue of the SRV loads that would result from a second opening of the SRV while the SRV tailpipe is still hot from the initial discharge, referred to by the staff as "subsequent actuation" or "consecutive actuation" in NUREG-0802, "Safety/Relief Valve Quencher Loads: Evaluation for BWR Mark II and III Containments," issued October 1982. The concern was that a subsequent SRV actuation could generate higher loads on the structure. However, the subsequent actuation effect is considered in the methodology as described in NEDE-33261P. Therefore, the staff accepted the GEH position that the methodology it used to calculate hydrodynamic loading on SRV discharge piping resulting from the initial and subsequent SRV actuations is consistent with the methodology used for earlier BWR (Mark II and III) designs.

The ESBWR suppression pool configuration is similar to that of the ABWR, as shown in Table 6.2.1.6-2.

Table 6.2.1.6-2

Design Feature	ESBWR	ABWR
Reactor power, MWt	4500	4000
Drywell volume, m^3 (ft^3)	7206 (~254,520)	7350 (~259,500)
Wetwell gas space volume, m^3 (ft^3)	5432 (~191,800)	5960 (~210,000)
Vertical vents (total), m^2 (ft^2)	13.6 (146)	11.6 (125)
Pool surface only, m^2 (ft^2)	799 (~8600)	507 (~5450)

Potentially, a slightly higher power and a slightly smaller drywell volume may increase the hydrodynamic forces. However, these negative effects are more than offset by a larger vent area, a larger pool volume, and a larger pool surface area.

Based on these similarities, GEH considers the methodology used to evaluate the pool response to a postulated accident (i.e., pool boundary loads resulting from bubble formation, the PS velocity and acceleration, the pool surface elevation, and the wetwell gas space pressure) for the ABWR design to be equally applicable to the ESBWR containment.

Adjustments for ESBWR application

Although the ESBWR and ABWR pressure suppression systems are similar, there are some differences in specific dimensions. These differences were accounted for as described below.

For PS, the methodology approved for the ABWR required no adjustment. One difference is that there are no vacuum breaker or upward diaphragm loads since, during the PS phase (0–5 s), the wetwell pressure is always lower than the drywell pressure. As this conclusion is based on analyses for the six postulated cases, it needs to be confirmed for an as-built ESBWR plant.

For CO loads, an additional pressure time history was added by compressing the time scale of the time history with the highest frequency content. The frequency was increased by the ratio of ESBWR-to-ABWR vertical distance from the vent entrance to the top vent (approximately 1/1.8). This additional pressure signature is to account for any possible influence of vent acoustic modes on the CO frequency.

For CH loads, to adjust the ABWR CH frequency to the ESBWR, the frequency was increased by the ratio of ESBWR-to-ABWR pool depth ratio (approximately 1/1.3).

For both CO and CH loads, the pressure amplitude was increased by a factor of 1.2. Although, given the ESBWR pool geometry, this additional conservatism is not necessary, it is included as part of the initial design assumptions.

For SRV loads, the X-quencher methodology, as described and reviewed in NUREG-1503, is used without adjustment.

Effect of unique ESBWR features

The PCCS, described in Section 6.2.2, receives a steam-gas mixture directly from the drywell. Most, if not all, steam is condensed in the tubes, and the remaining gas, primarily noncondensables, is deposited in the suppression pool. These PCCS characteristics reduce the CO loads and prevent the occurrence of the CH loads. In addition, the small venting area and low submergence of the vent line minimize the effect of PS, bounded by the LOCA loads.

The GDCS pools, described in Section 6.3, are equipped with spillover pipes to direct potential water overflow to the entrance of the main vents. In Revision 1 of the NEDE-33261P report (October 2007) GE stated that these pipes have no impact on containment thermal-hydraulic loads.

The largest postulated pipe breaks in the ESBWR are FWLB and MSLB since there is no recirculation line. Because of more rapid pressurization during the MSLB, the FWLB PS loads are bounded by the MSLB loads. For CO and CH loads, both breaks need to be evaluated. The review of thermal-hydraulic conditions revealed that the predicted steam mass fluxes for the ESBWR MSLB and FWLB are well below the values measured during the horizontal vent tests used for the ABWR load definition. Therefore, the ABWR CO and CH load definitions are applicable to the ESBWR design.

The ESBWR pool-to-vent area ratio is about 58, for the ABWR the ratio is about 40, for Mark II the ratio is typically 20.0, and for Mark III it is typically 12.0. GEH believes that the larger pool relative to the vent area will cause the pool hydrodynamic loads to be reduced. NUREG-0808, "Containment Program Evaluation and Acceptance Criteria," issued August 1981, supports this position.

The shallower and wider ESBWR pool and two additional vents tend to produce lower pressure amplitude, while a lower mass flow rate produces frequencies in the lower range of the existing experimental database.

6.2.1.6.3 Staff Evaluation

The staff believes that GEH needs to give adequate consideration to the differences between the Mark II, Mark III, and ABWR databases to determine that the suppression pool wall pressures for ESBWR do not exhibit any unusual characteristics when compared to the Mark III wall pressures. A high degree of similarity between the ABWR and ESBWR suppression pool designs upholds a concern (NUREG-1503) regarding the scaling loads used by GEH for developing the load definition. The ABWR-specific subscale (SS) and partial full-scale (FS) tests appear to be adequate representations of the ESBWR main vents for predicting the suppression pool hydrodynamic response for unstable CO and CH loads. However, DCD Revision 3 does not appear to discuss the applicability of the SS and FS tests to the ESBWR design. (The SS facility has a single horizontal pipe, and the FS facility has two horizontal pipes, while the ESBWR has three horizontal vent pipes extended into the suppression pool.) Also, the DCD should include a discussion of the Mark III data from the pressure suppression test facility blowdown tests, reported in NUREG-0978, "Mark III LOCA-Related Hydrodynamic Load Definition," issued

August 1984, which were conducted with FS vent lengths and all three horizontal vents. RAI 6.2-158 requests GEH to address the above issues. **RAI 6.2-158 is being tracked as an open item.**

As currently implemented in the Mark I, II, and III designs, the suppression pool temperature limits involve a three-tier approach. The lowest temperature threshold requires the operator to take such actions as activating pool cooling to reduce the suppression pool temperature. The plant, however, can continue to operate at power during this time. The intent of this threshold is to ensure that the operator acts to reduce pool temperature. This temperature is typically 35℃ (95 ℉). Operation can continue until the suppression pool reaches 43℃ (110℉). At this temperature, an automatic scram on high suppression pool temperature occurs. Finally, if the pool reaches 49 ℃ (120℉), the TSs require depressurization of the reactor coolant system and initiation of cold shutdown conditions. This approach ensures that suppression pool temperature limits for reactor scram and reactor depressurization, as defined in NUREG-0783, "Suppression Pool Temperature Limits for BWR Containment," issued November 1981, will not be reached. ESBWR TS 3.6.2.1, "Suppression Pool Average Temperature," includes temperature thresholds of 110℉, 120℉, and 130℉, respectively. It is not clear if these temperature thresholds are consistent with the NUREG-0783 guidance. The DCD should describe the effect of pool temperature on the SRV load evaluation in Tier 2 of the ESBWR DCD. RAI 6.2-159 requests GEH to address the above issue. **RAI 6.2-159 is being tracked as an open item.**

It is implied in the NEDE-33261P report that GE used the PICSM computer code to compare Mark III suppression pool swell test data from the pressure suppression test facility (PSTF) with analytical predictions. The code is described in GE technical report NEDE-21544P. GE validated test data generated for the Mark II design; however, the code was not reviewed and approved by the staff. The staff is concerned with potential liquid and froth impacts on the vacuum breaker valves. RAI 6.2-160 requests GEH to address this issue. **RAI 6.2-160 is being tracked as an open item.**

GEH applies the Mark II hydrodynamic loads to the ESBWR design. The staff documented its evaluation of the definition of the Mark II design containment hydrodynamic load in NUREG-0808. In the evaluation of the PS phenomena (discussed in Section 2.1 of NUREG-0808), the staff relied on comparisons, and a substantial amount of data from tests conducted by both GEH and the Japan Atomic Energy Research Institute. These tests were directly applicable to the Mark II design. GEH developed a computer program PSAM (NEDO-21061) to be used as part of the Mark II hydrodynamic load evaluation program. The staff has reviewed the Mark II program and approved the methodology and PSAM in NUREG-0808. However, it did not find the GEH methodology within PSAM acceptable. Rather, the staff based its acceptance on the favorable comparisons with the database. As a result, the use of the program for configurations other than those encompassed by the test data are not acceptable without further comparisons with applicable test data. The DCD should include a discussion of the applicability of the Mark II hydrodynamic loads to the ESBWR design in Tier 1 and Tier 2 of the ESBWR DCD. RAI 6.2-161 requests GEH to address the above issue. **RAI 6.2-161 is being tracked as an open item.**

Both Tier 1 and Tier 2 of the ESBWR DCD need to include a description of the ADS, including the final arrangement of the SRV, SV, and DPV. Also, NEDE-33261P (page 7-2) describes 12 drywell spillover (drain) pipes located in the inner suppression pool wall connecting the lower drywell airspace to the bottom of the suppression pool. Tier 1 and Tier 2 of the ESBWR DCD do not appear to discuss these pipes, so the staff asked that GEH include this discussion. In addition, Tier 2 of the ESBWR DCD does not include a COL action item, and Tier 1 of the

ESBWR DCD does not include acceptance criteria for the as-built internal structures to withstand the hydrodynamic loads and structural vibrations. RAI 6.2-162 and RAI 6.2-164 address these issues. **RAI 6.2-162 and RAI 6.2-164 are being tracked as open items.**

6.2.1.6.4 Conclusions

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

The staff reviewed the methodology presented in NEDE-33261P and used for evaluation of the ESBWR hydrodynamic loads. The analytical models of the involved physical phenomena are the same as those used for the safety evaluation of the approved ABWR design. The review included evaluation of the applicability of the rationale the staff used in the ABWR design approval process. Also, the staff reviewed the relevant database from previous BWR research programs.

In a separate evaluation, the staff reviewed and approved the application of the TRACG code for the ESBWR pool dynamic analysis (Letter from W.D. Beckner (NRC) to L.Quintana (GEH), "Safety Evaluation Report Regarding the Application of GENE's TRACG Code to ESBWR LOCA Analyses," August 19, 2004). The staff also acknowledges that, compared to the approved ABWR design, the shallower and wider ESBWR pool and the two additional vents tend to produce lower pressure amplitude, while a lower mass flow rate produces frequencies in the lower range of the existing experimental database. Therefore, the staff finds the methodology presented in NEDE-33261P to be acceptable. However, its application to an as-built ESBWR design must consider the issues discussed in Section 6.2.1.6.3 of this report.

6.2.1.7 <u>Containment Debris Protection for ECCS Strainers</u>

6.2.1.7.1 Regulatory Criteria

GDC 35, GDC 38, and GDC 41 require that systems be provided to perform emergency core cooling, containment heat removal, and containment atmosphere cleanup following a postulated DBA.

RG 1.82, "Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident," Revision 3, issued November 2003, contains guidance on the sizing criteria for ECCS strainers. The following NRC bulletins (BL) provide additional guidance:

- BL 93-02, "Debris Plugging of Emergency Core Cooling Suction Strainers," dated May 11, 1993

- BL 93-02, Supplement 1, "Debris Plugging of Emergency Core Cooling Suction Strainers," dated February 18, 1994

- BL 95-02, "Unexpected Clogging of a Residual Heat Removal (RHR) Pump Strainer While Operating in Suppression Pool Cooling Mode," dated October 17, 1995

- BL 96-03, "Potential Plugging of Emergency Core Cooling Suction Strainers by Debris in Boiling-Water Reactors," dated May 6, 1996

- BL 98-04, “Potential for Degradation of the Emergency Core Cooling System and the Containment Spray System after a Loss-of-Coolant Accident Because of Construction and Protective Coating Deficiencies and Foreign Material in Containment,” dated July 14, 1998.

6.2.1.7.2 Summary of Technical Information

ESBWR DCD Tier 2, Section 6.3.2.7.2, states that suppression pool equalization lines have an intake strainer to prevent the entry of debris material into the system that might be carried into the pool during a large-break LOCA. The GDCS pool airspace opening to the drywell will be covered by a perforated steel plate to prevent debris from entering the pool and potentially blocking the coolant flow through the fuel. The holes in the perforated steel plate will be smaller than the orifice holes in the fuel support castings.

6.2.1.7.3 Staff Evaluation

The ESBWR GDCS or PCCS does not have active pumps that are required for core cooling or containment heat removal during the 72 hours following a design-basis LOCA. The staff reviewed the DCD to determine that latent or LOCA-generated debris will not clog the GDCS or PCCS flowpaths. The staff issued the below RAI 6.2-6, Supplement 1:

> DCD, Tier 2, Revision 3, Section 6.2.1.1.2 states that “[t]here is sufficient water volume in the suppression pool to provide adequate submergence over the top of the upper row of horizontal vents, as well as the passive containment cooling system (PCCS) return vent, when water level in reactor pressure vessel (RPV) reaches one meter above the top of active fuel and water is removed from the pool during post-loss-of-coolant accident (LOCA) equalization of pressure between RPV and the wetwell.” If the ESBWR design relies on the suppression pool equalization line to maintain one meter depth of water above active fuel in RPV, the suppression pool equalization line should be designed as such. In response to RAI 6.3-40, GEH stated that the suppression pool equalization line will not open for 72 hours and beyond for all design basis LOCA scenarios. DCD, Tier 2, Revision 3, Section 6.3.2.7.2 states that “[s]uppression pool equalization lines have an intake strainer to prevent the entry of debris material into the system that might be carried into the pool during a large break LOCA.” Please provide information on how the intake strainer is designed to prevent the entry of debris material into the system.

ESBWR DCD Tier 2, Section 6.2.1.1.2, states that there is sufficient water volume in the suppression pool to provide adequate submergence over the top of the upper row of horizontal vents, as well as the PCCS return vent, when water level in the RPV reaches 1 m above the top of active fuel (TAF) and water is removed from the pool during post-LOCA equalization of pressure between the RPV and the wetwell. Water inventory, including the GDCS, is sufficient to flood the RPV to at least 1 m above the TAF.

In response to RAI 6.2-6, GEH stated that during the post-LOCA period, the suppression pool equalization line will open, allowing water to flow from the suppression pool to the RPV.

ESBWR DCD Tier 2, Section 6.3.2.7.2, states that suppression pool equalization lines have an intake strainer to prevent the entry of debris material into the system that might be carried into the pool during a large-break LOCA. DCD Tier 2 does not describe how this strainer is designed to address issues with debris clogging.

In response to RAI 6.3-40, GEH stated that during 72 hours and post-72 hours after a LOCA, the RPV water level would stay above the level setpoint for equalization line initialization logic. In a supplement to RAI 6.2-6, the staff requested GEH to clarify. A discussion of RAI 6.2-6 is provided above. **RAI 6.2-6 and RAI 6.3-40 are being tracked as open items.**

ESBWR DCD Tier 2, Revision 1, Section 6.3.2.7.2, states that the GDCS pool airspace opening to the drywell will be covered by a mesh screen or the equivalent to prevent debris from entering the pool and potentially blocking the coolant flow through the fuel. Although a mesh screen could protect GDCS pools from the entrance of some debris, it will not stop debris smaller than the mesh size from entering. Debris that enters the GDCS pool could flow with the GDCS injection flow into the vessel and could potentially block the coolant flow through the fuel.
Therefore, in RAI 6.3-41, the staff asked GEH to explain what action it would take to prevent such debris blockage. In response, GEH stated that it would use a perforated steel plate instead of a mesh screen to protect the GDCS pool from the entrance of debris and that the holes in the perforated steel plate will be smaller than the orifice holes in the fuel support castings. In RAI 6.3-41, Supplement 1, the staff requested the specific dimensions of the perforated plate holes, fuel assembly inlet orifice diameter, and the minimum GDCS line diameter. The staff needed this information to confirm that the holes in the perforated plate are small enough to prevent the entrance of debris that could block the fuel inlet orifice. In response, GEH provided the requested information, which needs to be added to the DCD.

ESBWR DCD Tier 2, Revision 3, Section 6.3.2.7.2, states that the GDCS injection system consists of one 200-mm (8-in.) pipe mounted with a temporary strainer. The staffs was concerned that the temporary strainer could clog with debris and impede the GDCS injection flow. Therefore, in RAI 6.3-41, Supplement 1, the staff asked GEH to explain the effect of the temporary strainer on the GDCS injection flow. In response, GEH stated that the temporary strainer was not intended to remain as part of the system configuration and that the strainer will be removed after initial flushing of the GDCS injection lines. GEH agreed to update the DCD to include this information. The staff determined that this response addressed the staff’s concerns and was acceptable. Staff requested GEH to update the DCD in supplemental RAI 6.3-41. **RAI 6.3-41 is being tracked as an open item.**

During a LOCA, if the passive PCCS heat exchanger inlets are within the zone of influence, debris ingress is expected. Therefore, in RAI 6.3-42, the staff requested that GEH describe the impact of the debris on the heat transfer performance of the heat exchanger. In response, GEH provided this information. Staff requested GEH to update the DCD in supplemental RAI 6.3-42. **RAI 6.3-42 is being tracked as an open item.**

The ESBWR relies on the PCCS to provide water to the GDCS for core cooling and for containment heat removal for 72 hours after a LOCA. Beyond 72 hours, the ESBWR also relies on the FAPCS. DCD Tier 2, Revision 3, Table 19A-2, identifies the FAPCS operating in suppression pool cooling and low-pressure coolant injection modes as a RTNSS.

However, DCD Tier 2, Revision 3, Table 1C-1, states that NRC BL 95-02 is not applicable to the ESBWR because it does not have a safety-related suppression pool cooling system. The same table states that NRC BLs 93-02 and Supplement 1 to BL 93-02, BL 96-03, and BL 98-04 do not apply to the ESBWR because the reactor design provides emergency core cooling via the GDCS and the GDCS pools do not have the debris transport mechanisms to which the suppression pool is subject.

Therefore, in RAI 6.2-173, the staff requested that GEH explain why the debris-plugging issues described in the above BLs should not be applied to the debris plugging of the suppression pool suction strainer for operation of the FAPCS 72 hours after a LOCA. **RAI 6.2-173 is being tracked as an open item**.

6.2.1.7.4 Conclusions

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

6.2.2 Containment Heat Removal System

6.2.2.1 <u>Regulatory Criteria</u>

The staff reviewed DCD Tier 2, Section 6.2.2, "Passive Containment Cooling System," in accordance with SRP Section 6.2.2, "Containment Heat Removal Systems." The applicant's containment heat removal system is acceptable if it meets the requirements of the following Commission regulations:

- GDC 38 as it relates to the following:
  - the ability of the containment heat removal system to rapidly reduce the containment pressure and temperature following a LOCA and to maintain these indicators at acceptably low levels
  - the ability of the containment heat removal system to perform in a manner consistent with the function of other systems
  - the safety-grade design of the containment heat removal system providing suitable redundancy in components and features and suitable interconnections, leak detection, isolation, and containment capability to ensure that, for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available), the system safety function can be accomplished in the event of a single failure
- GDC 39, "Inspection of Containment Heat Removal System," as it relates to the design of the containment heat removal system to permit periodic inspection of components
- GDC 40, "Testing of Containment Heat Removal System," as it relates to (1) the structural and leaktight integrity of its components, (2) the operability and performance of the active components of the system, and (3) the operability of the system as a whole, and under conditions as close to the design as practical the performance of the full operational sequence that brings the system into operation, including operation of applicable portions of the protection system, the transfer between normal and emergency power sources, and the operation of the associated cooling water system.

The regulations governing the evaluation of standard plant designs explicitly recognize the unique characteristics of the ESBWR PCCS. The regulation in 10 CFR 52.47(b)(2)(i)(A) states that, in the absence of a prototype plant that has been tested over an appropriate range of normal, transient, and accident conditions, a plant that "utilizes simplified, inherent, passive, or other innovative means to accomplish its safety functions" must meet the following requirements:

- The performance of each safety feature of the design has been demonstrated through either analysis, appropriate test programs, experience, or a combination thereof.
- Interdependent effects among the safety features of the design have been found acceptable by analysis, appropriate test programs, experience, or a combination thereof.
- Sufficient data exist on the safety features of the design to assess the analytical tools used for safety analyses over a sufficient range of normal operating conditions, transient conditions, and specified accident sequences, including equilibrium core conditions.

6.2.2.2 Summary of Technical Information

Consistent with the applicable requirements, the passive plant vendor, GEH, developed and performed design certification tests of sufficient scope, including both separate effects and integral systems experiments, to provide data with which to assess the computer programs used to analyze plant behavior over the range of conditions described in the third requirement above. To satisfy the requirements of 10 CFR 52.47(c)(2)(i)(A), GEH developed test programs to investigate the PCCS, including both component and phenomenological (separate effects) tests and integral systems tests.

The PCCS removes the core decay heat rejected to the containment after a LOCA. It provides containment cooling for a minimum of 72 hours post-LOCA, with containment pressure never exceeding its DPL, and with the isolation condenser/passive containment cooling (IC/PCC) pool inventory not being replenished.

GEH considers the PCCS condenser as an extension of the containment pressure boundary, and the PCCS condenser is used to mitigate the consequences of an accident. This function classifies it as a safety-related ESF. ASME Code, Section III, Class 2, and Section XI requirements for design and accessibility of welds for inservice inspection apply to meet GDC 16. Quality Group B requirements apply per RG 1.26, "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants." The system is designed to seismic Category I per RG 1.29, "Seismic Design Classification." The common cooling pool that PCCS condensers share with the ICs is a safety-related ESF, and it is designed such that no locally generated force (such as an ICS rupture) can destroy its function. Protection requirements against mechanical damage, fire, and flood apply to the common IC/PCC pool.

The PCCS condenser is sized to maintain the containment pressure within its design limit for DBAs. Section 6.2.2.2.2 of the DCD states, "The system is designed as a passive system with no components that must actively function, and it is also designed for conditions that equal or exceed the upper limits of containment reference severe accident capability." GEH clarified the reference to severe accident capability as those postulated for severe accident conditions as described in DCD Tier 2, Revision 4, Appendix 19B. For the postulated severe accident conditions, the service Level C pressure capacity for the PCCS heat exchangers at the temperature of 260 ^{o}C is 1.33 MPa gauge (193 psig). For comparison, the ESBWR containment design pressure is 0.312 MPa gauge (45 psig.)

The PCCS consists of six, low-pressure, separate loops sharing a common cooling pool. Each loop contains a two-module steam condenser (passive containment cooling (PCC) condenser) designed to reject up to 11 megawatt thermal (MWt) of heat.

Following a postulated accident, after initial energy deposition into the pressure suppression pool, the PCCS keeps the containment pressure below its design limit for at least 72 hours, without water makeup to the IC/PCC pool, and beyond 72 hours with pool makeup.

The PCCS is open to the containment and receives a steam-gas mixture supply directly from the drywell. The condensed steam is drained to a GDCS pool, and the gas is vented through the vent line, which is submerged in the pressure suppression pool.

The PCCS operates in two distinct modes, a condensing mode and a pressure differential mode. Its operation is initiated by the difference in pressure between the drywell and the wetwell. Once a sufficient rate of steam condensation is established, the pressure inside the PCCS tubes is lower than the pressure in the drywell, which causes the flow of the steam-gas mixture into the heat exchange units. The condensate is then drained by gravity to a GDCS pool, and the noncondensable gases are collected in the lower drum of the PCCS units until its pressure exceeds the submergence head of the PCCS vent pipes in the suppression pool.

In the pressure differential mode, a pressure buildup in the drywell, caused by insufficient steam condensation inside the PCC condenser, will force flow through the PCCS which pushes the noncondensable gases and the noncondensed steam into the suppression pool and potentially reestablishes the condensing mode of operation. This pressure buildup has to be greater than the submergence of PCCS vent pipes but not sufficient to clear the main vents. For that reason, the PCC vent line outlet is about 0.9 m higher than the outlet of the upper horizontal main vents. .

Since PCCS operation is completely passive, there is no need for sensing, control, logic, or power-actuated devices to function. GEH considers the PCCS condensers as an extension of the safety-related containment and thus not in need of isolation valves.

6.2.2.3 Staff Evaluation

The staff relied on the guidance provided in SRP Section 6.2.2, “Containment Heat Removal Systems,” Revision 4, to perform its review.

GDC 38 states, in part, “The system safety function shall be to reduce rapidly, consistent with the functioning of other associated systems, the containment pressure and temperature following any loss-of-coolant accident and maintain them at acceptably low levels.” The PCCS does not appear to rapidly reduce containment pressure and temperature as evident from the TRACG results presented in Section 6.2, “Containment Systems,” of Revision 3 to the DCD. The applicant needs to demonstrate how the ESBWR meets the safety function of GDC 38. The TRACG results indicate that containment pressure is still rising after 72 hours. The applicant should demonstrate that containment pressure can be rapidly reduced and maintained beyond 72 hours. **RAIs 6.2-139 and 6.2-140 address this issue and are being tracked as open items.**

The applicant should also address the use of any non-safety systems to reduce uncertainties in the TRACG analyses in accordance with the guidance provided in SECY-94-084, "Policy and Technical Issues Associated with the Regulatory Treatment of Non-safety Systems in Passive Plant Designs." **RAIs 6.2-139 and 6.2-140 address this issue and are being tracked as open items.**

The ESBWR PCCS is a safety-related ESF which does not involve pumps, sprays, or fan coolers. Its design pressure is 758.5 kPa(g) (110 psig) compared to the containment design pressure of 310 kPa(g) (45 psig), and its design temperature is 171℃ (340℉), the same as that for the containment. DCD Table 6.2-1 indicates the containment design parameters.

Since PCCS operation is completely passive, there is no need for sensing, control, logic, or power-actuated devices to function. GEH considers the PCCS condensers as an extension of the safety-related containment and thus not in need of isolation valves. The staff is reviewing the acceptability of this design to meet Appendix J requirements. RAI 6.2-102 addresses this issue. **RAI 6.2-102 is being tracked as an open item.**

The PCCS operates in two distinct modes, a condensing mode and a pressure differential mode. In the pressure differential mode, a pressure buildup in the drywell, caused by insufficient steam condensation inside the PCC condenser, will force flow through the PCCS which pushes the noncondensable gases and the noncondensed steam into the suppression pool and potentially reestablishes the condensing mode of operation. This pressure buildup has to be greater than the submergence of PCCS vent pipes but not sufficient to clear the main vents. For that reason, the PCC vent line outlet is about 0.9 m higher than the outlet of the upper horizontal main vents. This is a critical elevation that should be verified by ITAAC and described in Tier 1 and Tier 2 of the DCD. DCD Tier 2, Section 6.2.2, does not include or describe the elevation of the PCC vent line relative to the upper horizontal main vents. RAI 6.2-169 addresses these issues. **RAI 6.2-169 is being tracked as an open item.**

The PCCS is designed to seismic Category I, per RG 1.29, ASME Code, Section III, Class 2, and Section XI requirements, to meet GDC 16 in Appendix A to 10 CFR Part 50. The material used must be a nuclear-grade stainless steel or equivalent material, which is not susceptible to intergranular SCC.

The six PCCS loops are each designed to remove 11 MWt of latent heat during condensation of pure steam inside the tubes at a pressure of 308 kPa (45 psia) and temperature of 134℃ (273.2℉) with an outside pool water temperature of 102℃ (215.6℉). For the steam-gas mixture and/or at the lower pressure and temperature, the condensing power of the condenser is lower. DCD Table 6.2-10 indicates the PCC design parameters.

In order to demonstrate PCCS performance at various flow rates, steam-gas compositions, and thermal conditions, a comprehensive testing program was developed to provide an experimental database for validation of analytical models. The staff reviewed and approved the PCCS-related test program in Chapter 21 of this report. The following is a brief description of the three major tests (i.e., PANTHERS/PCC, PANDA, and GIRAFFE).

PANTHERS/PCC is an FS, two-module test facility at the SIET laboratory in Piacenza, Italy. Of the 63 tests performed using a prototypical heat exchanger, 13 were steady-state steam-only tests, 42 were air-steam tests, and 8 were noncondensable gas buildup tests with air, helium, and a mixture of both. The test matrix covered the range of expected accident conditions (pressure, temperature, and flow rates) as predicted by TRACG calculations. The tests confirmed the expected performance of the PCC condenser.

PANDA is a 1:25 scale (by volume), full-height integral systems test facility at the Paul Scherrer Institute in Switzerland. The PANDA test facility was configured to represent all major ESBWR containment components. It includes three full-height, scaled (by number of tubes) PCC condensers and one scaled IC unit. Of the 22 tests performed, 10 were steady state, covering a wide range of expected steam and airflow rates, and 12 were transient tests, representative of various post-LOCA conditions. The tests confirmed the expected performance of the PCCS.

GIRAFFE is a full-height, small-scale (1:400 by volume) test facility at the Toshiba laboratories in Japan. The PCC condenser is represented by three full-height tubes. The main purpose of the tests was to demonstrate the effect of lighter-than-steam and heavier-than-steam noncondensable gases. Four tests were performed using nitrogen and helium. The tests confirmed that the PCCS can successfully operate in the presence of noncondensable gases.

The staff visited all of these facilities and performed several reviews of the engineering abilities of the personnel involved, testing equipment, and applied QA programs. The staff audited the QA programs and found them acceptable, as discussed in Section 21.7. Therefore, the staff accepted the use of the test results as a demonstration of PCCS performance and their use in support of verification and validation of the relevant analytical models.

The staff also performed its own independent studies of the PCCS performance at the PUMA facility at Purdue University. PUMA is a scaled (1:400 by volume, 1:4 reduced height) integral representation of the SBWR design similar to the PANDA facility. One of the purposes of these studies was to examine the effect of different scaling approaches. Unlike the PANDA facility, which preserves full height, the PUMA facility preserves the aspect ratio. This feature of PUMA provides additional insights into the multidimensional effects of an SBWR-like design. The PUMA tests qualitatively confirmed the PANDA results.

GEH did not include a description of the ESBWR test program as applied to the safety evaluation of the containment heat removal system. Staff requested this information in RAI 6.2-172. GEH response was acceptable; however, staff needs to verify the response is incorporated into a future revision of the DCD. RAI 6.2.172 is being tracked as a confirmatory item.

GEH did not include an evaluation of GL 96-06, “Assurance of Equipment Operability and Containment Integrity During Design-Basis Accident Conditions,” dated September 30, 1996, as indicated in DCD Appendix 1C, Table 1C-1. Staff requested GEH provide this discussion in RAI 6.2-170.

In response, GEH explained that except for the containment isolation function, the CWS equipment is all nonsafety-related and is not required to function during the response to a design basis accident. It is assumed that the nonsafety-related Seismic Category II coolant boundary of the CWS or Drywell Cooling System heat exchanger may fail, opening to the containment atmosphere. Thus, the concerns of GL 96-06 have been taken into consideration in the design of CWS and do not adversely affect the ESBWR response to a design basis accident.

During design basis accident conditions, the design feature providing cooling of the containment air for ESBWR is the PCCS condensers, which condense steam that has been released to the DW following a LOCA or MSLB to transfer the heat to the IC)/PCC) pools. The IC/PCC pools are designed to boil in order to perform their heat removal function. DCD Tier 2, Revision 3, Subsection 6.2.1 discusses the role of the PCCS condensers in maintaining containment pressure and temperature within design limits during design basis accidents and provides information about the function of the PCCS. DCD Tier 2, Revision 3, Subsection 6.2.2 provides design details on the PCCS. The passive nature of the PCCS design prevents it from being subjected to water hammer effects or thermally-induced overpressurization.

Staff found GEH's response is acceptable; however, staff needs to verify the proposed revision to the DCD is incorporated into a future revision of the DCD. RAI 6.2-170 is being tracked as a confirmatory item.

DCD, Tier 2, Revision 3, Chapter 1.11, Table 1.11-1 states that Task Action Plan Item (TAP) B-12, "Containment Cooling Requirements (Non-LOCA), "evaluation is addressed in DCD Sections 6.2.2, 7.3.2, 9.2.7 and 9.4.8. Staff could not locate where, in Section 6.2.2, this item is discussed. Staff requested in RAI 6.2-171 to address TAB B-12.

In response, GEH stated that DCD Tier 2, Revision 3, Subsections 6.2.2 and 7.3.2 have been referenced because they describe the design of the PCCS, which performs the safety-related containment cooling for ESBWR. DCD Tier 2, Revision 3, Subsections 9.2.7 and 9.4.8 have been referenced because they describe the design of the CWS and DCS, respectively. The CWS and DCS perform containment air cooling during normal operation and are isolated on a LOCA signal. A loss of normal containment cooling does not impact the operability of the safety-related PCCS to perform this function or the ability to place the ESBWR in a safe shutdown condition. The PCCS is a passive system that does not have instrumentation, control logic or power-actuated valves, and does not need or use electrical power for its operation.

Staff found GEH's response is acceptable; however, staff needs to verify the proposed revision to the DCD is incorporated into a future revision of the DCD. RAI 6.2-171 is being tracked as a confirmatory item.

6.2.2.4 Conclusions

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

The review of the ESBWR test program revealed that it correctly established the expected containment thermal conditions and the ranges of relevant parameters included in the experimental matrices. The test data appear to be of good engineering quality and sufficient to provide a basis for validation of TRACG analytical models as well as for verification of the code predictions of containment behavior under various accident conditions. The staff accepts the TRACG prediction that, within 72 hours of the DBA, the ESBWR pressure and temperature during the postulated DBA scenarios are sufficiently within the design values.

6.2.3 Reactor Building Functional Design

The RB structure encloses all penetrations through the containment (except for those of the main steam tunnel and IC/PCC pools). The RB provides an added barrier to fission product released from the containment in case of an accident, contains, dilutes, and holds up any leakage from the containment, and houses safety-related systems.

6.2.3.1 Regulatory Criteria

The staff reviewed the RB in accordance with SRP Section 6.2.3 for secondary containment. The ESBWR design is significantly different from the secondary containment of current operating BWR facilities. The staff discusses these differences as part of its evaluation. Conformance with the following regulatory criteria forms the basis for determining the acceptability of the RB functional design:

- GDC 4, as it relates to safety-related SSCs being designed to accommodate the effects of normal operation, maintenance, testing and postulated accidents, and being protected against dynamic effects (e.g., the effects of missiles, pipe whipping, and discharging fluids) that may result from equipment failures

- GDC 16, as it relates to reactor containment and associated systems being provided to establish an essentially leak-tight barrier against the uncontrolled release of radioactive material to the environment

- GDC 43, "Testing of Containment Atmosphere Cleanup Systems," as it relates to systems having the design capability to permit periodic functional testing to ensure system integrity, the operability of active components, and the operability of the system as a whole and the performance of the operational sequence that brings the system into operation

- Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors," to 10 CFR Part 50, as it relates to the secondary containment being designed to permit preoperational and periodic leakage rate testing so that bypass leakage paths are identified

The staff used the following guidance in its review:

- RG 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," issued July 2000, as it relates to guidance in assumptions concerning mixing in the RB in applying the alternative source term

- SRP (NUREG-0800), Section 6.2.3, as it provides methods acceptable to the staff for the review of secondary containments

6.2.3.2 Summary of Technical Information

The RB under accident conditions is automatically isolated to provide a holdup and plate-out barrier. When isolated, the RB can be serviced by the RB heating, ventilation, and air conditioning (HVAC) system through a high-efficiency particulate air (HEPA) filtration system (see

Section 9.4.6). With low leakage and stagnant conditions, holdup and plate-out mechanisms perform the basic mitigating functions.

The ESBWR design does not include a secondary containment, and minimal credit is taken for the existence of the RB surrounding the primary containment vessel in any radiological analyses. The radiological dose consequences for LOCAs, based on an assumed containment leak rate of 0.5 percent per day and RB bypass leakage equal to 100 percent of the containment leak rate, show that offsite and control room doses after an accident are less than allowable limits, as discussed in Chapter 15. The RB envelope is not intended to provide a leak-tight barrier against radiological releases. Therefore, GEH indicted the design criterion of GDC 16 does not apply.

During normal plant operation, potentially contaminated areas within the RB are kept at a negative pressure with respect to the environment, while clean areas are maintained at positive pressure. The ESBWR does not need and thus has no filter system that performs a safety-related function following a DBA, as discussed in Section 6.5.1. Therefore, GEH indicated the design criterion of GDC 43 is not applicable.

Personnel and equipment entrances to the RB consist of vestibules with interlocked doors and hatches. Large equipment access is by means of a dedicated, external access tower that provides the necessary interlocks.

The RB is a reinforced concrete structure that forms an envelope completely surrounding the containment (except the basemat), and it is designed to meet the following safety design bases:

- The RB maintains its integrity during the environmental conditions postulated for a DBA.

- The RB HVAC system automatically isolates upon detection of high radiation levels in the ventilation exhaust system.

- All openings through the RB boundary, such as personnel and equipment doors, are closed during normal operation and after a DBA by interlocks or administrative control. These doors are provided with position indicators and alarms that are monitored in the control room.

- Detection and isolation capability for high-energy pipe breaks within the RB is provided.

- The compartments within the RB are designed to withstand the maximum pressure resulting from an HELB. Each line break analyzed is a double-ended break. This analysis considers the rupture producing the greatest blowdown of mass and enthalpy in conjunction with the worst-case single active component failure. Blowout panels between compartments provide flowpaths to relieve pressure.

- The RB can be periodically tested to assure that the leakage rates assumed in the radiological analyses are met.

During normal operation, the potentially contaminated areas in the RB are maintained at a slightly negative pressure relative to adjoining areas by the contaminated area HVAC subsystem (CONAVS) portion of the RB HVAC system (see Section 9.4.6). This assures that any leakage from these areas is collected and treated before release. Airflow is from clean to potentially contaminated areas. RB effluents are monitored for radioactivity by stack radiation monitors. If the radioactivity level rises above set levels, the discharge can be routed through the CONAVS purge system for treatment before further release. Penetrations through the RB envelope are designed to minimize leakage. All piping and electrical penetrations are sealed for leakage. Access to the RB is through interlocked doors. The RB HVAC system is designed and tested for isolation under accident conditions.

High-energy pipe breaks in any of the RB compartments do not require the building to be isolated. These breaks are detected, and the broken pipe is isolated by the closure of system isolation valves. No significant release of radioactivity is postulated from these types of accidents because reactor fuel is not damaged

Reactor Water Cleanup Equipment and Valve Rooms

The two independent RWCU divisions are located in the 0–90° and 270–0° quadrants of the RB. The RWCU equipment (pumps, heat exchangers, and filter/demineralizers) is located on floor elevations -11,500 mm and -6400 mm with separate rooms for equipment and valves. The RWCU piping originates at the RPV. High-energy piping leads to the RWCU divisions through a dedicated, enclosed pipe chase. The steam-air mixture resulting from an HELB in any RWCU compartment is directed through adjoining compartments and pipe chase to the RB operating floor. The envelope profile represents the calculated maximum pressure response values for the given room or region resulting from all postulated RWCU/SDC system pipe breaks. These pressure profiles include no margin.

Isolation Condenser System

The ICs are located in the RB at the 27,000-mm elevation. The IC steam supply line is connected directly to the RPV. The supply line leads to a steam distribution header, which feeds four pipes. Each pipe has a flow limiter to mitigate the consequences of an IC line break. The IC design-basis break is a double-ended break in the piping after the steam header and flow restrictors. The IC/PCC pool is vented to atmosphere to remove steam generated in the IC pools by the condenser operation. In the event of an IC break, the steam-air mixture is expected to preferentially exhaust through hatches in the refueling floor and into the RB operating area with portions of the steam directed through the pool compartments to the stack, which is vented to the atmosphere. Because the vent path through the hatches leads to the refueling floor area, which is a large open space with no safety implications, this event was excluded from the pressurization analysis.

Main Steam Tunnel

The RB main steam tunnel is located between the primary containment vessel and the turbine building. The limiting break is an MSL longitudinal break. The MSLs originate at the RPV and are routed through the steam tunnel to the turbine building. The steam-air mixture resulting from an MSLB is directed to the turbine building through the steam tunnel.

DCD Section 3G.1.5.2.1.10 discusses the pressure capability of the steam tunnel compartment. No blowout panels are required in the steam tunnel because the flowpath between the steam tunnel and the turbine building is open. The MSLB is excluded from the pressurization analysis because of the ability of the steam to blow down into the turbine building.

Fission Product Containment

Sufficient water is stored within the containment to cover the core during both the blowdown phase of a LOCA and during the long-term post-blowdown condition. Because of this continuous core cooling, fuel damage and fission product release is a very low probability event. If there is a release from the fuel, most fission products are readily trapped in water. Consequently, the large volume of water in the containment is expected to be an effective fission product scrubbing and retention mechanism. Also, because the containment is located entirely within the RB, multiple structural barriers exist between the containment and the environment, thus mitigating fission product leakage from the RB.

Compartment Pressurization Analysis

RWCU pipe breaks in the RB and outside the containment were postulated and analyzed. For compartment pressurization analyses, HELB accidents are postulated as the result of piping failures in the RWCU system where the location and size of breaks result in maximum pressure values. The analysis considers calculated pressure responses in order to define the peak pressure of the RB compartments for structural design purposes. The calculated peak compartment pressure is 3.26 kPa(g), which is below the RB compartment pressurization design requirements, as discussed in DCD Section 3G.1.5.2.1.11.

Values of the mass and energy releases produced by each break are in accordance with American National Standards Institute/American Nuclear Society (ANSI/ANS)-56.4. The break fluid enthalpy for energy release considerations is equal to the stagnation enthalpy of the fluid in the ruptured pipe. The mass and energy blowdown from the postulated broken pipe terminates when system isolation valves are fully closed after receiving the pertinent isolation closure signal.

Subcompartment pressurization effects resulting from postulated breaks of high-energy piping have been analyzed according to ANSI/ANS-56.10. To calculate the pressure response in the RB and outside the containment resulting from HELB accidents, the CONTAIN 2.0 code was used according to the nodalization schemes shown in DCD Figure 6.2-18. The nodalization contains the rooms where breaks occur and all interconnected rooms/regions through flowpaths such as doors and hatches. DCD Table 6.2-12 gives flowpath and blowout panel characteristics, and DCD Table 6.12-12a contains subcompartment nodal descriptions.

6.2.3.3 Staff Evaluation

The staff review focused on compliance with the GDC listed in Section 6.2.3.1.

GDC 4 states that SSCs important to safety shall be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents including a LOCA.

The applicant did not state in the DCD the method or manner in which the ESBWR achieves compliance with GDC 4 for the RB. The applicant should state which components, systems, and structures are important to safety and the method or features that assure compliance with GDC 4. Open Item RAI 6.2-155 addresses this issue. **RAI 6.2-155 is being tracked as an open item.**

GDC 16 states that reactor containment and associated systems shall be provided to establish an essentially leak tight barrier against the uncontrolled release of radioactivity to the environment and to assure that the containment design conditions important to safety are not exceeded for as long as postulated accident conditions require. The applicant stated that GDC 16 does not apply because the RB is not considered to be a leak tight barrier.

The staff is considering the applicant's statement with respect to the applicability of GDC 16. The applicant makes two assumptions in the design-basis analyses that impact the control of radioactive release. The first assumption is that the primary containment leakage into the RB is diluted by 40 percent of the RB volume. The second assumption is that the reactor RB leakage to the environment is 50-percent volume per day. These two assumptions directly affect the results of the design-basis analyses required by 10 CFR 34(a)(1) and control room operator dose stated in GDC 19, "Control Room." The applicant needs to demonstrate how the ESBWR meets the safety function of controlling radioactive releases to the environment if it is not a leak-tight barrier. RG 1.183 (Appendix A, paragraph 4.4) states that "credit for dilution in the secondary containment may be allowed when adequate means to cause mixing can be demonstrated. Otherwise the leakage from the primary containment should be assumed to be transported directly to the exhaust systems without mixing." The applicant has not demonstrated an adequate means to cause the 40-percent dilution of the source term before its release from the RB and has not addressed the uncertainties. The applicant also needs to identify (1) the maximum leak rate that could occur from the RB under design-basis conditions, including weather conditions, (2) the procedures used to test RB leakage, and (3) the frequency of the test. The applicant should also address the use of any non-safety systems to reduce uncertainties in these two assumptions in accordance with SECY-94-084. RAI 6.2-165 addresses this issue. **RAI 6.2-165 is being tracked as an open item.**

SRP Section 6.2.3 references GDC 43 as applying to secondary containments and states that the containment atmosphere cleanup systems shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leak-tight integrity of its components, (2) the operability and performance of the active components of the systems such as fans, filters, dampers, pumps, and valves, and (3) the operability of the systems as a whole and, under conditions as close to design as practical, the performance of the full operational sequence that brings the systems into operation, including operation of applicable portions of the protection system, the transfer between normal and emergency power sources, and the operation of associated systems.

The applicant stated the following:

> …during normal plant operation, potentially contaminated areas within the RB are kept at a negative pressure with respect to the environment while clean areas are maintained at positive pressure. The ESBWR does not need, and thus has no filter system that performs a safety-related function following a design basis accident, as discussed in Subsection 6.5.1. Therefore the design criterion of GDC 43 is not applicable.

The staff is concerned that the applicant has not fully addressed the need for an atmospheric cleanup system post-accident for the RB. The reference to Subsection 6.5.1 is not relevant since it refers to containment spray systems (drywell containment sprays) that are non-safety-related and not part of the RB. An atmospheric cleanup system for the RB as used in essentially all existing BWR facilities would provide the means to control, process, monitor, and release through an identified path radioactivity entering the building, would prevent the buildup of radioactivity in the RB, would reduce the release to the environment, and would meet the requirements of

GDC 16 or GDC 60, “Control of Releases of Radioactive Materials to the Environment.” Control of radioactivity buildup in the RB should be considered in the light of the guidance provided in NUREG-0737, II.B.2, which provides guidance on radiation levels acceptable for postaccident entry into vital areas should entry into the RB be required and guidance on the need to assure that safety-related equipment is environmentally qualified for postaccident doses. RAI 6.2-166 addresses this issue. **RAI 6.2-166 is being tracked as an open item.**

In Appendix J to 10 CFR Part 50, Option A states in IV.B that other structures of multiple barrier or subatmospheric containments (e.g., secondary containments for BWRs and shield buildings for pressurized-water reactors that enclose the entire primary reactor containment or portions thereof) shall be subject to individual tests in accordance with the procedure established in the TSs or associated bases.

The staff is concerned about the type of test that will be used to bound the RB leakage, the conditions under which the test will be run, the degree to which these conditions will reflect worst-case accident conditions, the frequency of such a test, and the establishment of test criteria. RAI 6.2-167 and RAI 15.4-26 address these issues. **RAI 6.2-167 and RAI 15.4-26 are being tracked as open items.**

With respect to fission product containment, the applicant stated that “because the containment is located entirely within the RB, multiple structural barriers exist between the containment and the environment. Therefore, fission product leakage from the RB is mitigated.”

The staff is reviewing the degree of mitigation provided by the RB in connection with RAIs on mixing assumptions and building leakage. Building leakage is especially important because of its impact on the effectiveness of the building as a barrier to the release of radioactivity to the environment. DCD Tier 1, Section 2.16.5, states that “offsite dose requirements are met assuming a 100 percent volume change out per day in the RB volume outside of the RCCV.” This is inconsistent with the assumption used in the design-basis analyses of 50-percent volume per day in Chapter 15, Table 15.4-5. The staff asked the applicant to explain the difference and correct the appropriate sections of the DCD. RAI 6.2-168 addresses this issue. **RAI 6.2-168 is being tracked as an open item.**

With respect to compartment pressurization analysis, the staff needs additional information to perform confirmatory RB subcompartment analyses as requested in Supplemental RAI 6.2-46 and RAI 6.2-154. **RAI 6.2-46 and 6.2-154 are being tracked as open items.**

6.2.3.4 Conclusions

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

6.2.4 Containment Isolation System

The containment isolation system (CIS) consists of isolation barriers, such as valves, blind flanges, and closed systems, and the associated instrumentation and controls required for the automatic or manual initiation of containment isolation. The purpose of the CIS is to permit the normal or postaccident passage of fluids through the containment boundary, while protecting against release of fission products to the environment that may be present in the containment atmosphere and fluids as a result of postulated accidents.

6.2.4.1 Regulatory Criteria

Acceptance criteria are based on meeting the relevant requirements of the following Commission regulations:

- GDC 1, “Quality Standards and Records,” as it relates to designing, fabricating, erecting, and testing safety-related SSCs to quality standards commensurate with the importance of the safety functions to be performed

- GDC 2, “Design Bases for Protection Against Natural Phenomena,” as it relates to designing safety-related SSCs to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches without loss of capability to perform safety functions

- GDC 4, as it relates to designing safety-related SSCs to accommodate the effects of and to be compatible with environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, and as it relates to the requirement that these SSCs shall be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids

- GDC 16, as it relates to the requirement that reactor containment and associated systems establish an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment

- GDC 54, “Systems Penetrating Containment,” as it relates to the requirement that piping systems penetrating the containment be provided with leak detection, isolation, and containment capabilities having redundancy, reliability, and performance capabilities that reflect their importance to safety and as it relates to designing such piping systems with a capability to periodically test the operability of the isolation valves and associated apparatus and to determine if valve leakage is within acceptable limits

- GDC 55, “Reactor Coolant Pressure Boundary Penetrating Containment,” and GDC 56, “Primary Containment Isolation,” as they relate to isolation valves for lines penetrating the primary containment boundary as parts of the RCPB (GDC 55) or as direct connections to the containment atmosphere (GDC 56) as follows:

  – one locked-closed isolation valve inside and one outside containment
  – one automatic isolation valve inside and one locked-closed isolation valve outside containment

  – one locked-closed isolation valve inside and one automatic isolation valve outside containment
  – one automatic isolation valve inside and one outside containment

- GDC 57, “Closed Systems Isolation Valves,” as it relates to the requirement that lines that penetrate the primary containment boundary and are neither part of the RCPB nor connected directly to the containment atmosphere have at least one locked-closed, remote-manual, or automatic isolation valve outside containment

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

- 10 CFR 52.47(a)(8) and 10 CFR 52.79(a)(17), as they relate to demonstrating compliance with any technically relevant portions of the requirements related to Three Mile Island (TMI) in 10 CFR 50.34(f)(2)(xiv) and 10 CFR 50.34(f)(2)(xv), for design certification and COL reviews, respectively

6.2.4.2 Summary of Technical Information

The proposed CIS for the ESBWR is described in ESBWR DCD Tier 2, Section 6.2.4. This subsection describes the CIS to provide protection against releases of radioactive materials to the environment as a result of an accident.

The containment isolation function is accomplished by valves and control signals, required for the isolation of lines penetrating the containment. The CIS automatically closes fluid penetrations of fluid systems not required for emergency operation. Fluid penetrations supporting ESF systems have remote manual isolation valves that can be closed from the control room, if required.

DCD Table 6.2-13 identifies the RCPB influent lines, and DCD Table 6.2-14 identifies the RCPB effluent lines. DCD Tables 6.2-15 through 6.2-42 show the pertinent data for the containment isolation valves. DCD Section 7.1.2 lists the criteria for the design of the leak detection and isolation system (LD&IS), which provides containment and reactor vessel isolation control. DCD Section 7.3.3 lists and explains the bases for assigning certain signals for containment isolation.

Power-operated containment isolation valves have position-indicating switches in the control room to show whether the valve is open or closed. Power for valves used in series originates from physically independent sources without cross-ties to assure that no single event can interrupt motive power to both closure devices.

Containment isolation valve closure times are established by determining the isolation requirements necessary to keep radiological effects from exceeding the guidelines in 10 CFR 50.67, "Accident Source Term." Chapter 15 discusses valve closure time bases for system lines, which can provide an open path from the containment to the environment. The design values of closure times for power-operated valves are more conservative than the above requirements.

Sensing instrument lines penetrating the containment follow all the recommendations of RG 1.11, "Instrument Lines Penetrating Primary Reactor Containment (Safety Guide 11) Supplement to Safety Guide 11, Backfitting Considerations," issued March 1971. Each line has a 6-mm (1/4-in.) orifice inside the drywell, as close to the beginning of the instrument line as possible, a manually operated isolation valve just outside the containment, followed by an excess flow check valve. The instrument line is designed such that the instrument response time is acceptable with the presence of the orifice and that the flow restriction is not plugged.

In general, all requirements of GDC 54, 55, 56, 57 and RGs 1.11 and 1.141 are met in the design of the CIS. DCD Section 6.2.4.3 gives a case-by-case analysis of all such penetrations. Table 1.9-6 lists exemptions from the GDC.

The PCCS does not have isolation valves, as the heat exchanger modules and piping are designed as extensions of the safety-related containment. The design pressure of the PCCS is greater than twice the containment design pressure, and the design temperature is the same as the drywell design temperature.

Isolation valves, actuators, and controls are protected against damage from missiles. Tornado missile protection is afforded by the location of all containment isolation valves inside the missile-proof RB. The arrangement of containment isolation valves inside and outside the containment affords sufficient physical separation such that a high-energy pipe break would not preclude containment isolation. The CIS piping and valves are designed in accordance with seismic Category I.

Containment isolation valves and associated pipes are designed to withstand the peak calculated temperatures and pressures to which they would be exposed during postulated DBAs. Containment isolation valves are designed in accordance with the requirements of ASME Code, Section III, and meet at least Group B quality standards, as defined in RG 1.26. The power-operated and automatic isolation valves will be cycled during normal operation to assure their operability.

Redundancy is provided in all design aspects to satisfy the requirement that no single active failure of any kind should prevent containment isolation. Mechanical components are redundant, in that isolation valve arrangements provide backup in the event of accident conditions. Electrical redundancy is provided for each set of isolation valves to eliminate dependency on one power source to attain isolation. Electrical cables for isolation valves in the same line are routed separately.

Plant operators will apply administrative controls by using established procedures and checklist for all nonpowered containment isolation valves to ensure that their position is maintained and known. The position of all power-operated isolation valves is indicated in the control room. DCD Section 7.3.3 discusses instrumentation and controls for the isolation valves. DCD Section 6.2.6 discusses leak-rate testing of isolation valves.

6.2.4.3 Staff Evaluation

The staff reviewed the description of the CIS using the review guidance and acceptance criteria of Section 6.2.4 of the SRP. SRP Section 6.2.4 identifies the staff’s review methodology and acceptance criteria for evaluating compliance with GDC related to those piping systems penetrating containment.

The staff’s review encompassed the following areas specified by Section 6.2.4 of the SRP and 10 CFR 50.34(f)(2)(xiv):

- CIS design, including the following:
  - the number and location of isolation valves (e.g., the isolation valve arrangements, location of isolation valves with respect to the containment wall, purge and vent

valve conformance to Branch Technical Position CSB 6-4, and instrument line conformance to RG 1.11)

- the actuation and control features for isolation valves
- the normal positions of valves and the positions valves take in the event of failures
- the initiating variables for isolation signals and the diversity and redundancy of isolation signals
- the basis for selecting closure time limits for isolation valves
- the redundancy of isolation barriers
- the use of closed systems as isolation barrier substitutes for valves

- the protection provided for CISs against loss of function caused by missiles, pipe whip, and natural phenomena
- environmental conditions in the vicinity of CISs and equipment and their potential effect
- the mechanical engineering design criteria applied to isolation barriers and equipment
- the provisions for alerting operators of the need to isolate manually controlled isolation barriers
- the provisions for, and TSs pertaining to, operability and leak-rate testing of isolation barriers
- the calculation of containment atmosphere released before isolation valve closure for lines that provide a direct path to the environs
- containment purging/venting requirements of 10 CFR 50.34(f)(2)(xiv) and (xv)

Based on its review of the CIS described in ESBWR DCD Tier 2, Section 6.2.4, the staff needed the additional information indicated below to resolve the open issues.

RAI 6.2-102

DCD Tier 2, Revision 3, Sections 6.2.4.3.2.1 and 6.2.4.3.2.2, state that the passive containment cooling system (PCCS) has no containment isolation valves (CIVs). The heat exchanger modules and piping of the PCCS, outside containment, form closed systems. As the justification for having no CIVs, the DCD states that the PCCS does not penetrate containment, because the heat exchanger modules and piping are designed as extensions of the safety-related containment, and that the design pressure of the PCCS is greater than twice the containment design pressure and the design temperature is the same as the drywell design temperature. In RAI 6.2-102, the staff stated that the PCCS must have CIVs, and, supported its position with extensive citations from the regulations (10 CFR Part 50, Appendix A, General Design

Criterion 56) and the applicable official NRC guidance (Standard Review Plan 6.2.4, Rev. 2, “Containment Isolation System,” and RG 1.141, “Containment Isolation Provisions for Fluid Systems,” dated April 1978, which endorses national standard ANS-56.2/ANSI N271-1976, “Containment Isolation Provisions for Fluid Systems” (national standard)). Staff provided a quotation from the national standard that stated that even if the closed system outside containment is treated as an extension of containment, at least one CIV per line is still necessary. GE’s response, MFN 06-466, was a reiteration of their position that the system is considered an extension of the containment boundary, meaning that there are no containment penetrations in the PCCS, and therefore GDC 56, the SRP, the RG, and the national standard do not apply. The applicant cites several documents (other SRPs and GDC) which contain design provisions for the containment boundary, and states that the PCCS satisfies these provisions and so is an extension of containment.

Staff’s Review of GE’s Response:

(1) Staff’s review found that the documents cited by the applicant only address design provisions for the containment in general such as for the walls and roof. The documents cited do not address any situation which is like the applicant’s design (that is, a piping system outside of containment) or explain why no CIVs are needed in such a design. On the other hand, the guidance documents cited by the staff do specifically address designs like the PCCS.

(2) Staff understands that there is no explicit definition of “containment penetration” in the documents cited in staff’s original RAI. Perhaps the authors felt that, when a pipe passes through the containment wall or roof (like the PCCS does), that this was obviously a containment piping penetration. However, there is the following definition in the national standard, in section 2, “Definitions and Terminology”: Penetration assembly. An assembly that allows fluid lines or electrical circuits to pass through a single aperture (nozzle or other opening) in the containment.

Also, the national standard begins as follows:

1. Purpose and Scope

The primary purposes of this standard are to specify minimum design, testing and maintenance requirements for the isolation of fluid systems which penetrate the primary containment of light water reactors. These fluid systems include piping systems (including instrumentation and control) for all fluids entering or leaving the containment. (2) When applying the definitions of the national standard, it can reasonable be interpreted that the PCCS design does indeed have containment penetrations thus requiring CIVs. (3) Even within the DCD, there is contradiction as to whether the PCCS has containment penetrations. Revision 3 of the DCD contains a new table, 6.2-47, titled “Containment Penetrations Subject to Type A, B, and C Testing.” This table lists 18 containment penetrations in the PCCS, numbered T15-MPEN-0001 through T15-MPEN-0018. Staff agrees that the portion of the PCCS outside of containment is considered to be an extension of containment. However, the applicant concludes without sufficient justification that this inherently means there are no containment penetrations and thus no requirement for any CIVs. The applicant has not provided precedents, regulations, guidance documents, or any other reference to support this conclusion. Alternatively, staff has cited a national standard endorsed by RG 1.141 which specifically address the case of a closed system outside of containment which is considered to be an extension of containment. This national standard states that there must be at least one CIV in each line. Provide additional justification for the current design of the PCCS, or revise the DCD with a redesign of the system to include CIVs, per the NRC's applicable regulatory position.

RAI 6.2-102 is being tracked as an open item.

RAI 6.2-103

> RAI 6.2-103 asked for DCD, Tier 2, Table 1.9-6, "Summary of Differences from SRP Section 6," to be revised to state that the passive containment cooling system (PCCS) was different from SRP Section 6.2.4 acceptance criteria, in that it had no containment isolation valves (CIVs). The applicant, consistent with their response to RAI 6.2-102, stated that the PCCS do not require CIVs and do not deviate from SRP Section 6.2.4 acceptance criteria. Consistent with the staff's RAI 6.2-102 supplemental question, the staff requests that the applicant add the PCCS to Table 1.9-6 or change its design to bring it into conformance with SRP Section 6.2.4. The staff also asked that the Process Radiation Monitoring System be added to the table, because it has both CIVs outside containment. The applicant responded that these lines conform to the provisions of RG 1.11, "Instrument Lines Penetrating Primary Reactor Containment" (as described in their response to RAI 6.2-127), which would mean that they do conform to SRP Section 6.2.4 acceptance criteria. However, the applicant has not demonstrated that the system does conform with RG 1.11 (see RAI 6.2-127 supplemental question), and so the staff repeats its request that the applicant add the Process Radiation Monitoring System to Table 1.9-6 or change its design to bring it into conformance with SRP Section 6.2.4.

RAI 6.2-103 is being tracked as an open item.

RAI 6.2-104

> One part of RAI 6.2-104 pointed out that DCD, Chapter 6.2.4, 2nd paragraph, stated that the plant meets the relevant requirements of various GDC for containment isolation design. The staff noted that, to the contrary, at least four systems did not meet the specific requirements of GDC 55 and 56. Three of the systems were listed in DCD, Tier 2, Revision 3, Table 1.9-6, and the fourth was the PCCS.

The staff asked the applicant to clarify or correct this apparent discrepancy. The applicant responded by referring to RAI 6.2-102, which addressed the PCCS CIV issue, and concluded that they would make no changes to the DCD. Putting aside the PCCS issue, which is unresolved RAI 6.2-102, the applicant failed to address the other three systems, as requested in the original RAI, for which still states in DCD, Tier 2, Revision 3, Table 1.9-6, in the row titled, "SRP 6.2.4," that: "ESBWR design takes specific exceptions to GDC 55 and GDC 56, while satisfying the intent.

(1) FAPCS suppression pool suction line contains one isolation valve outside containment;

(2) ICS piping contains two isolation valves inside containment; and

(3) Containment Inerting System piping contains two isolation valves outside containment.

Please address the above three systems as they relate to the inconsistency between Chapter 6.2.4, 2nd paragraph, 4th bullet, “the plant meets the relevant requirements of various GDC 55 and 56...” to the statement in Table 1.9-6, “ESBWR design takes specific exceptions to GDC 55 and GDC 56...”

RAI 6.2-104 is being tracked as an open item.

<u>RAI 6.2-106</u>

The applicant’s response to RAI 6.2-106 did not address the intent of the original RAI. A portion of the RAI is restated below:

In DCD, subsection 6.2.4.1, “Design Bases,” under the heading “Safety Design Bases,” the 3rd bullet states: “The design of isolation valves for lines penetrating the containment follows the requirements of General Design Criteria 54 through 57 to the greatest extent practicable consistent with safety and reliability.” [emphasis added]. The staff does not understand the intent of the highlighted phrase. As applicable, remove this statement, request an exemption, or revise the statement to include, “... except as noted below,” and then provide the specific exceptions.

RAI 6.2-106 is being tracked as an open item.

RAI 6.2-107 is being tracked as a confirmatory item.

<u>RAI 6.2-109</u>

RAI 6.2-109 requested information about containment isolation valve (CIV) closure times. In DCD, Revision 3, the applicant made appropriate revisions and included acceptable CIV closure times in Tables 6.2-16 through 6.2-42, except as follows:

Isolation Condenser System—In DCD, Tier 2, Revision 3, Tables 6.2-24, 6.2-26, 6.2-28, and 6.2-30, a number of 20 mm (0.8 inch) CIVs have closure times of 30 seconds or less.

High Pressure Nitrogen Gas Supply System—The CIVs are 50 mm (2 inches) in diameter with closure times for valves F0009 and F0026 listed in Table 6.2-40 as 30 seconds or less.

Because DCD, Tier 2, Revision 3, Subsection 6.2.4.2.1, states that CIVs which are 80 mm (3 inches) or less in diameter “generally close within 15 seconds,” consistent with national standard ANS-56.2/ANSI N271-1976, section 4.4.4, “Valve Closure Time,” staff is unsure if the quoted closure times of “30 seconds or less” for the above two systems are correct. Please verify. Revise and explain any inconsistency in the DCD.

RAI 6.2-109 is being tracked as an open item.

RAI 6.2-110

Original RAI 6.2-110 referred to DCD, Tier 2, Revision 1, Section 6.2.4.2.2, "Instrument Lines Penetrating Containment," and Section 6.2.4.3.2.5, "Evaluation Against Regulatory Guide (RG) 1.11." The original RAI questioned whether the instrument lines in the ESBWR design conformed with the provisions of RG 1.11. In GE's response, DCD, Revision 3 contains additional information in Section 6.2.4.2.2. RAI 6.2-110 also asked the applicant to identify and describe, in the DCD tables, all instrument lines penetrating containment. In GE's response, DCD, Revision 3, Table 6.2-47 lists many instrument lines.

Supplemental Request:

The applicant's response is incomplete. The DCD, Revision 3, text specifically addresses some, but not all, of the provisions of RG 1.11, and appears to mean to address the remaining provisions by stating that the instrument lines "follow all the recommendations of Regulatory Guide 1.11." The staff cannot review or verify a simple assertion of conformance to a RG. Further, the new information in Table 6.2-47 is incomplete. Most or all of the instrument line listings have containment penetration identifying numbers ending in "TBD" (presumably "To Be Determined") and give no information regarding compliance with RG 1.11. Considering that the design of the instrument lines appears to be incomplete, provide in the DCD complete information demonstrating conformance with each of the specific regulatory positions of RG 1.11, for every instrument line.

RAI 6.2-110 is being tracked as an open item.

RAI 6.2-115

RAI 6.2-115(B) stated:

DCD, Tier 2, Revision 1, Section 6.2.4.3.3, "Evaluation of Single Failure," discusses, in general, the principles used to evaluate single failure. It implies that evaluations were performed for the containment isolation system, but does not provide the actual evaluations or even specific conclusions, other than an unsupported statement that "Electrical and mechanical systems are designed to meet the single failure criterion...." It refers to DCD, Section 3.1 for more information, but 3.1 is only a general discussion of the ESBWR's compliance with the GDC. Provide the actual single failure evaluations performed for the containment isolation system, or at least a better discussion of the evaluation. Address particularly the example given in part 1 of this RAI. [The example was of two redundant CIVs on the same emergency power bus, where a single failure of that bus would fail both CIVs.]

The applicant's response was:

Subsection 6.2.4.3.3, as noted on the attached markup, will be revised to include statement that each of the power operated containment isolation valve for any given penetration is powered from a different division in order to meet the single failure criteria.

Supplemental Request:

GEH's response only addresses the one item particularly called out by the staff. The response does not address the request as a whole. It is necessary for the applicant to demonstrate the soundness of their method to evaluate single failure events. Therefore, for each type or class of penetrations, provide a detailed single failure analyses, with charts and tables naming each failure considered for each penetration or class of penetrations, and explanations for why each single failure would not cause loss of safety function.

RAI 6.2-115 is being tracked as an open item.

RAI 6.2-117 is being tracked as a confirmatory item.

RAI 6.2-119

The containment isolation provisions of the isolation condenser condensate, venting, and purge lines consist of one barrier (a closed system) outside containment and two CIVs inside containment. RAI 6.2-119 stated that this design did not comply with the explicit requirements of GDC 55 or GDC 56, and was inconsistent with the guidelines of the appropriate guidance documents (SRP 6.2.4, Rev. 2; RG 1.141; and national standard ANS-56.2/ANSI N271-1976) for alternate means for complying with GDC 55 or GDC 56. These GDCs allow alternate isolation provisions, other than their explicit requirements, if "it can be demonstrated that the containment isolation provisions for a specific class of lines, such as instrument lines, are acceptable on some other defined basis." The guidance documents define other acceptable bases. The applicant's response stated that, effectively, the isolation condenser system (ICS) has three barriers (one outside and two inside containment) and goes "beyond the requirements" in GDC 55 and 56.

Supplemental Request:

The explicit requirements of GDC 55 and 56 are to have one CIV inside and one CIV outside containment. If a containment penetration had, for example, two CIVs inside and one CIV outside containment, or one CIV inside containment, one CIV outside containment, and a closed system outside containment, then it would clearly and simply go beyond the requirements of the GDC. However, one cannot simply add up the number of containment isolation barriers and conclude that three must be better than two. It depends on the configuration. For example, three CIVs inside containment, and none outside, does not satisfy the explicit requirements of the GDC because there is no valve outside containment. It would also not be in accordance with the guidance documents.

Containment isolation design philosophy, as set forth in the regulations and the guidance documents, requires redundant isolation barriers such that no single failure of a pipe or valve can disable the isolation function. Even passive failures are implicitly considered in the design provisions. For example, one locked-closed manual isolation valve on a penetration is not enough, even though no active failure could cause it to fail; a second, redundant barrier is required. Likewise, a closed piping system, inside or outside containment, is not by itself sufficient; a second barrier, typically a valve, is required, and the requirements and guidelines state that it must be outside of containment, presumably to be accessible for manual operator action if it fails to close. Furthermore, when there is a closed system and one CIV outside containment, there must be a special provision to protect against a failure of the pipe segment between the containment wall and the CIV, either by enclosing the pipe segment and valve in a leak-tight or controlled leakage enclosure or by designing them to particular conservative design requirements which are assumed to preclude a breach. This is done because a pipe breach in this location would be unisolable. It is true that standard technical specifications allow, in many circumstances, continued plant operation with only a single isolation barrier in place, but this is with a recognition that the containment isolation system is degraded by this condition and must eventually be restored to the full design capability.

In addition to the explicit GDC 55 and 56 configuration of one CIV inside and one outside containment, the guidance documents allow two other configurations: 1) one CIV and a closed system, both outside containment, or 2) two CIVs outside containment. The ICS design does not conform to either of these. The NRC has the authority to approve additional isolation configurations under the "other defined basis" provision of the GDC, but the applicant must adequately justify their proposed alternative to assure sufficient safety, consistent with the overall containment isolation design philosophy expressed in the GDC and guidance documents. For example, SRP Section 6.2.4 states, "If it is not practical to locate a valve inside containment (for example, the valve may be under water as a result of an accident), both valves may be located outside containment." In the ICS case, locating a CIV outside containment would place it under water all of the time. This is good justification for moving it inside containment, if it can also be shown that a single failure would not disable the containment isolation function.

Provide additional justification for the proposed design, as discussed above, in DCD, Tier 2, Section 6.2.4.3.1.1, or revise the design to conform to the GDC requirements and guidance documents provisions.

RAI 6.2-119 is being tracked as an open item.

RAI 6.2-120

RAI 6.2-120 noted that DCD, Tier 2, Revision 1, Section 6.2.4.3.1.2, "Effluent Lines," under the heading "Main Steam and Drain Lines," described the power-operated main steam isolation valves (MSIVs) as closing under either spring force or gas pressure. The staff questioned this statement, considering that virtually every BWR MSIV in the U.S. needs both gas pressure and spring force to close under accident conditions.

Supplemental Request:

The applicant provided an adequate response to this RAI, MFN 06-436, that explained the operation of the valves, which is like the MSIVs in other BWRs. The response included a proposed DCD, Revision 3, Section 6.2.4.3.1.2, which, if it had been incorporated into the DCD, would have resolved this issue. However, the proposed revision was not incorporated in DCD, Revision 3, Section 6.2.4.3.1.2. The version in Revision 3 contains even less information than it did in Revision 1, which causes this RAI to remain unresolved. On another note, the RAI response and current DCD version refers to DCD, Section 5.4.5, for further information, but that section does not seem to address this particular issue.

Please revise the DCD to include the appropriate information as presented in the proposed DCD Revision 3 and revisit the need to reference Section 5.4.5.

RAI 6.2-120 is being tracked as an open item.

RAI 6.2-121

The resolution of this RAI is a subsidiary of RAI 6.2-119. Depending on the resolution of supplemental RAI 6.2-119, the response to RAI 6.2-121 may need to be revised.

RAI 6.2-121 is being tracked as an open item.

RAI 6.2-122

RAI 6.2-122 requested that information about the containment isolation design for the Fuel and Auxiliary Pools Cooling System, currently located in DCD, Tier 2, Revision 3, Subsection 9.1.3.3, be provided in Subsection 6.2.4.3.2.1. The applicant responded: In order to minimize the risk of errors and inconsistencies in future DCD updates, it is preferable to provide a detailed description in only one location and reference it as needed in other sections. By taking this approach, fewer DCD changes will be required if this information needs to be revised in the future. RG 1.70 supports this approach.

Supplemental Request:

The staff does not disagree that a single location for detailed information is preferable. However, the staff believes that the containment isolation design information in Subsection 9.1.3.3 does not belong there. It should be removed from Subsection 9.1.3.3 and placed in Subsection 6.2.4.3.2.1. The staff reviewed DCD Revision 3, Subsection 9.1.3.3, information and associated tables. Please address the following:

A. Subsection 9.1.3.3 states that the containment isolation valves (CIVs) in the suppression pool supply and return lines fail as-is on loss of electric power or the air supply.

1. DCD Tier 2, Revision 3, Table 6.2-33a agrees with this for the “A” lines. To the contrary, Table 6.2-33b states that the CIVs fail closed in the “B” lines. Correct this discrepancy.

2. The DCD simply states that the CIVs in the suppression pool supply and return lines fail as-is on loss of electric power or the air supply, without any explanation or justification as to why this is acceptable. GDCs 55 and 56 state that, upon loss of actuating power, automatic isolation valves shall be designed to take the position that provides greater safety. SRP Section 6.2.4, Revision 2, Section II.6.j, and ANS-56.2/ANSI N271-

1976, sections 4.4.3 and 4.4.7, provide guidance for this requirement. Normally, a CIV should take the post-accident position upon failure; the post-accident position for these valves is “closed,” per the DCD tables. The two CIVs in each suction line are both nitrogen-motor operated without accumulator (NMO) valves, which, by their nature, fail as-is. The guidance documents allow both CIVs in a line to be motor-operated if independent power sources serve the two valves, so that a single power failure does not fail both valves. Considering that the subject valves will fail on loss of either electric power or the air (or nitrogen) supply, provide in the DCD an explanation of the manner in which the CIVs in the suction line(s) are protected from a single power failure rendering both valves inoperable, and potentially open, during an accident.

B. In the suppression pool suction lines, having both CIVs located outside containment is acceptable per Section II.6.d of SRP Section 6.2.4, Revision 2, as cited in the DCD, except for the following two points:

1. There is no indication as to whether the design provides a capability to detect leakage from the valve shaft and/or bonnet seals and terminate the leakage, which is also a provision of section II.6.d of SRP Section 6.2.4, Revision 2. Provide the missing information in the DCD.

2. Note that the option of having both CIVs in a line outside of containment is available only for engineered safety feature (ESF) or ESF-related systems, or systems needed for safe shutdown of the plant. Tables 6.2-33a and 6.2-33b state that these lines are not ESF. Discuss in the DCD whether the suppression pool suction lines satisfy this SRP criterion (for example, are ESF-related or needed for safe shutdown), and, if not, justify the deviation from the guidelines.

C. DCD, Tier 2, Revision 3, Tables 6.2-33a through 6.2-35, provide containment isolation design information for seven containment penetrations in the Fuel and Auxiliary Pools Cooling System, designated G21-MPEN-0001 through G21-MPEN-0007. However, DCD, Tier 2, Revision 3, contains a new table, Table 6.2-47, “Containment Penetrations Subject to Type A, B, and C Testing.” This table indicates that there is an additional penetration in the system, designated G21-MPEN-TBD (To Be Determined?) and described as the Reactor Well Drain Line. Provide in the DCD containment isolation design information for this penetration.

RAI 6.2-122 is being tracked as an open item.

<u>RAI 6.2-123</u>

RAI 6.2-123 noted that the influent and effluent lines of the containment inerting system, described in DCD, Tier 2, Revision 1, Sections 6.2.4.3.2.1 and 6.2.4.3.2.2, had all of their containment isolation valves (CIVs) outside of containment, but without adequate justification per the guidelines of SRP Section 6.2.4, Revision 2, (Section II.d.), RG 1.141, and national standard ANS-56.2/ANSI N271-1976 (Sections 3.6.5 and 3.7). The applicant’s response provided DCD changes (later made in DCD Revision 3) to address the guidelines.

Supplemental Request:

The DCD revision (Revision 3) satisfies the provisions of the guidance documents, except as described in the following two items:

A. The guidelines state that both CIVs in a line may be located outside of containment if it is not practical to place one inside containment. The DCD does not address this point. If, in fact, it is practical to place one CIV inside containment, justify the deviation from the guidelines.

B. The option of having both CIVs outside containment is available only for engineered safety feature (ESF) or ESF-related systems, or systems needed for safe shutdown of the plant. Tables 6.2-36 through 6.2-38 state that the containment inerting system lines are not ESF. Discuss in the DCD whether the containment inerting system lines satisfy this criterion (for example, are ESF-related or needed for safe shutdown), and, if not, justify the deviation from the guidelines.

RAI 6.2-123 is being tracked as an open item.

RAI 6.2-125 is being tracked as a confirmatory item.

RAI 6.2-127

RAI 6.2-127 questioned the design of the Process Radiation Monitoring System, in that all of the containment isolation valves (CIVs) are outside of containment. The applicant responded that the lines, 1 in (25 mm) in diameter, should be treated as instrument lines and that the design was acceptable because it was in accordance with RG 1.11, “Instrument Lines Penetrating Primary Reactor Containment.”

Supplemental Request:

The staff accepts the classification of these lines as instrument lines. However, there is not enough information to conclude that they comply with RG 1.11. RG 1.11, Section C.2., states that the lines should have one CIV inside and one outside containment (which they do not), or else conform to Sections 1.b. through 1.e. The staff needs more information to determine if the lines satisfy each of these provisions, but notes, for example, that 1.c. states, in part, that the CIVs should fail as-is, whereas DCD, Tier 2, Revision 3, Table 6.2-42 states that they fail closed. Provide in the DCD a discussion showing that these lines conform to RG 1.11, or, if not, the requirements for non-instrument lines.

RAI 6.2-127 is being tracked as a open item.

RAI 6.2-128

RAI 6.2-128 noted that DCD, Tier 2, Revision 1, Tables 6.2-39 through 6.2-42 did not include information covering the Chilled Water, High Pressure Nitrogen Gas Supply, and Process Radiation Monitoring Systems. In DCD, Revision 3, the applicant filled in the tables.

Supplemental Request:

The new information is generally acceptable, but the staff has the following questions.

A. For the Chilled Water and High Pressure Nitrogen Gas Supply Systems, the stated applicable basis is GDC 57. The applicant's revised response to RAI 6.2-129 (ML071030343) recognizes that no ESBWR system credits a closed system inside containment (per GDC 57) as a containment isolation barrier. Please correct the tables in the DCD.

B. For the High Pressure Nitrogen Gas Supply and Process Radiation Monitoring Systems, the tables indicate that DCD, Tier 2 figures for the systems are "N/A." Why are system figures not applicable? When will figures be provided?

C. Closure times for CIVs in the High Pressure Nitrogen Gas Supply System are unacceptable. See Supplemental RAI 6.2-109 for details.

RAI 6.2-128 is being tracked as an open item.

RAI 6.2-131

The responses to parts (A) and (B) of the original RAI 6.2-131 are acceptable. However, the staff has a further request for part (C). Part (C) requested a discussion of reducing the containment setpoint pressure that initiates containment isolation for nonessential penetrations to the minimum compatible with normal operating conditions. In the response to this RAI, GEH proposed a change to DCD Tier 2, Appendix IA to include the following: The alarm and initiation setpoints of the LD&IS are set to the minimum compatible with normal operating conditions to initiate containment isolation for containment penetrations containing process lines that are not required for emergency operation. The values for these setpoints are determined analytically or are based on actual measurements made during startup and preoperational tests. If setpoints are to be determined analytically, provide the actual numerical value of the containment setpoint pressure that initiates containment isolation for nonessential penetrations and justify that it is the minimum compatible with normal operating conditions. If the setpoints are to be based on actual measurements during startup and preoperational tests then revise the DCD to provide more details regarding how and when this setpoint will be determined.

RAI 6.2-131 is being tracked as an open item.

RAI 6.2-157

DCD, Tier 2, Revision 3, contains a new table, Table 6.2-47, "Containment Penetrations Subject to Type A, B, and C Testing." The staff compared this table with Tables 6.2-15 through 6.2-42, which were to provide "pertinent data for the containment isolation valves" (DCD Tier 2, Revision 3, Subsection 6.2.4.2), presumably in a comprehensive way. However, Table 6.2-47 includes many containment piping penetrations (approximately 122) which are not covered in Tables 6.2-15 through 6.2-42 or elsewhere in DCD Tier 2, Revision 3, Section 6.2.4, “Containment Isolation Function.” Further, Table 6.2-47 contains virtually no information on the containment isolation provisions for these lines, other than incomplete information on leakage rate testing.

Most of these penetrations are designated by numbers ending in “TBD,” apparently meaning “To Be Determined.” Many of the lines are instrument lines and many are part of systems whose larger lines are addressed in Tables 6.2-15 through 6.2-42. However, some are systems which are not covered at all in Tables 6.2-15 through 6.2-42:

Control Rod Drive System
Gravity Driven Cooling System
Makeup Water System
Service Air System
Containment Monitoring System
Equipment and Floor Drain System

A. Is the design of the containment isolation provisions for the approximately 122 penetrations to be performed by COL applicants? If so, provide a COL Item in DCD subsection 6.2.8. If not, provide the missing information in the DCD. Also, are there any other containment penetrations which are not listed in Table 6.2-47?

B. Table 6.2-47 also lists the containment air locks and hatches, which are not addressed elsewhere in section 6.2.4. Provide in the DCD containment isolation design information for the containment air locks and hatches.

RAI 6.2-157 is being tracked as an open item.

Generic Issues

The two generic issues included in the staff’s review of the CIS are II.E.4.2, Containment Isolation Dependability, and II.E.4.4, Containment Purging During Reactor Operation.

<u>II.E.4.2, Containment Isolation Dependability (10 CFR 50.34(f)(2)(xiv)</u>)

The governing regulation, 10 CFR 50.34(f)(2)(xiv), states the following:

Provide containment isolation systems that: (II.E.4.2)

(A) Ensure all non-essential systems are isolated automatically by the containment isolation system,

(B) For each non-essential penetration (except instrument lines) have two isolation barriers in series,

(C) Do not result in reopening of the containment isolation valves on resetting of the isolation signal,

(D) Utilize a containment set point pressure for initiating containment isolation as low as is compatible with normal operation,

(E) Include automatic closing on a high radiation signal for all systems that provide a path to the environs.

The staff issued RAI 6.2-131 and RAI 6.2-179 to address these issues. **RAI 6.2-131 and 6.2-179 are being tracked as open items.**

6.2.4.4 Conclusions

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

6.2.5 Combustible Gas Control in Containment

During certain accidents, combustible gases could be generated inside containment and, if not controlled, might burn and threaten the operability of the containment or various systems inside the containment that are important to safety.

6.2.5.1 Regulatory Criteria

The requirements for the control of combustible gas in containment, during accidents, are contained in 10 CFR 50.44, "Combustible Gas Control for Nuclear Power Reactors." The NRC extensively revised 10 CFR 50.44 in 2003, made associated changes to 10 CFR 50.34 and 10 CFR 52.47, and added a new section, 10 CFR 50.46a, on reactor coolant system venting. The revisions consolidate combustible gas control regulations for future power reactor applicants and licensees and also apply to current power reactor licensees. The purpose of the revisions was to risk inform the requirements for combustible gas control. The revised rules eliminate the former requirements for hydrogen recombiners and hydrogen purge systems and relax the former requirements for hydrogen- and oxygen-monitoring equipment to make them commensurate with their risk significance.

For the design certification of the ESBWR design, 10 CFR 50.44 requires the following:

- 10 CFR 50.44(c)(2): The containment must either (1) have an inerted atmosphere, or (2) limit hydrogen concentrations in containment during and following an accident that releases an amount of combustible gas equivalent to that generated by a 100 percent fuel clad-coolant reaction, uniformly distributed, to less than 10 percent (by volume) and maintain containment structural integrity and appropriate accident mitigating features. 10 CFR 50.44(a)(1) defines that "inerted atmosphere" means a containment atmosphere with less than 4 percent oxygen by volume.

- 10 CFR 50.44(c)(1): The containment must be capable of ensuring a mixed atmosphere during design-basis and significant beyond design-basis accidents (BDBA). As a definition, the rule states that "mixed atmosphere" means that the concentration of combustible gases in any part of the containment is below a level that supports combustion or detonation that could cause loss of containment integrity.

- 10 CFR 50.44(c)(4)(i): Equipment must be provided for monitoring oxygen in containments that use an inerted atmosphere for combustible gas control. Equipment for monitoring oxygen must be functional, reliable, and capable of continuously measuring the concentration of oxygen in the containment atmosphere following a significant BDBA for combustible gas control and accident management, including emergency planning.

- 10 CFR 50.44(c)(4)(ii): Equipment must be provided for monitoring hydrogen in the containment. Equipment for monitoring hydrogen must be functional, reliable, and capable of continuously measuring the concentration of hydrogen in the containment atmosphere following a significant BDBA for accident management, including emergency planning.

- 10 CFR 50.44(c)(5): The applicant must perform an analysis that demonstrates containment structural integrity. This demonstration must use an analytical technique that is accepted by the NRC and must include sufficient supporting justification to show that the technique describes the containment response to the structural loads involved. The analysis must address an accident that releases hydrogen generated from a 100 percent fuel clad-coolant reaction accompanied by hydrogen burning. Systems necessary to ensure containment integrity must also be demonstrated to perform their function under these conditions.

The appropriate staff guidance documents for this review are RG 1.7, “Control of Combustible Gas Concentrations in Containment,” and SRP Section 6.2.5, “Combustible Gas Control in Containment.” The staff is using Revision 3 of both documents, even though they were not formally issued until March 2007, which was after the ESBWR DCD was docketed. These revisions were issued to support the 2003 revision to 10 CFR 50.44. Draft versions of the guidance documents have been publicly available since 2003 and were substantially like the final versions. The applicant has cited the draft Revision 3 of RG 1.7 in the DCD.

The following regulations also have a bearing on this review:

- GDC 5, “Sharing of Structures, Systems, and Components,” as it relates to providing assurance that sharing of SSCs important to safety among nuclear power units will not significantly impair their ability to perform their safety functions

- GDC 41, as it relates to systems being provided to control the concentration of hydrogen or oxygen that may be released into the reactor containment following postulated accidents to ensure that containment integrity is maintained; systems being designed to suitable requirements (i.e., that suitable redundancy in components and features exists) and suitable interconnections to ensure that, for either a loss of onsite or offsite power, the system safety function can be accomplished, assuming a single failure; and systems being provided with suitable leak detection, isolation, and containment capability to ensure that system safety function can be accomplished

- GDC 42, “Inspection of Containment Atmosphere Cleanup Systems,” as it relates to the design of the systems to permit appropriate periodic inspection of components to ensure the integrity and capability of the systems

- GDC 43, as it relates to the systems being designed to permit periodic testing to ensure system integrity and the operability of the systems and active components

- 10 CFR 52.47(b)(1), which requires that a design certification application contain the proposed ITAACa that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a facility that incorporates the design certification has been constructed and will be operated in conformity with the design certification, the provisions of the Act, and the Commission's rules and regulations

- 10 CFR 52.80(a), which requires that a COL application contain the proposed inspections, tests, and analyses, including those applicable to emergency planning, that the licensee shall perform and the acceptance criteria that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will operate in conformity with the COL, the provisions of the Atomic Energy Act, and the NRC’s regulations

Because the latter regulation applies to COL applications, it does not apply to the ESBWR, a design certification application.

6.2.5.2 Summary of Technical Information

The design of the ESBWR provides for an inerted containment (with oxygen concentration in the containment maintained at less than 4 percent by volume) during normal operation, according to 10 CFR 50.44 (c)(2), and, as a result, no system to limit hydrogen concentration is required.

DCD Tier 2, Revision 3, states that the ESBWR meets the relevant requirements of the following:

- 10 CFR 50.44 and 10 CFR 50.46 apply as they relate to BWR plants being designed to have containments with an inerted atmosphere.

- GDC 5 does not apply to the inerting function because there is no sharing of SSCs between different units

- GDC 41, as it relates to systems being provided to control the concentration of hydrogen or oxygen that may be released into the reactor containment following postulated accidents to ensure that containment integrity is maintained, does not apply to the ESBWR because the safety function is accomplished by keeping the containment inerted. Thus, no redundancy or single failure criteria shall be considered, as the inerted containment is intrinsically safe and passive

- GDC 42 and 43, as they relate to the design of the systems to permit appropriate periodic inspection and periodic testing of components to ensure the integrity and capability of the systems, do not apply to the inerting function. Periodic monitoring of oxygen concentration is adequate to confirm the safety function

- RG 1.7, Revision 3, applies as it relates to the systems being designed to limit the oxygen gas concentrations within the containment

Containment Inerting System: The containment inerting system is provided to establish and maintain an inert atmosphere within the containment and is discussed in DCD Section 9.4.9. The containment inerting system can be used under postaccident conditions for containment atmosphere dilution to maintain the containment in an inerted condition by a controlled purge of the containment atmosphere with nitrogen to prevent reaching a combustible gas condition.

Containment Atmosphere Monitoring: The containment monitoring system (CMS) provides the function that is necessary to meet or exceed the requirements of 10 CFR 50.44(c)(4) with regard to oxygen and hydrogen monitoring. The CMS is a safety-related, seismic Category 1 system consisting of two redundant, physically and electrically independent postaccident monitoring divisions. Each division is capable of measuring and recording the radiation levels and the oxygen and hydrogen concentration levels in the drywell and suppression chamber. Hydrogen and oxygen monitoring consists of two hydrogen- and two oxygen-monitoring channels containing hydrogen and oxygen sensors, sample lines to bring a sample from the drywell or suppression chamber to the sensor, hydrogen and oxygen monitor electronics assemblies, visual displays, and a calibration gas supply. The data are transmitted to the main control room where they are continuously displayed. High hydrogen and oxygen concentration alarms are provided. The channels are equipped with an inoperative alarm to indicate malfunctions. The channels are divided into two redundant divisions.

Containment Atmosphere Mixing: The ESBWR design provides protection from localized combustible gas deflagrations, including the capability to mix the steam and noncondensable gases throughout the containment atmosphere and minimize the accumulation of high concentrations of combustible gases in local areas. DCD Section 19.3 lists the containment design features that will reduce the likelihood of combustible gas deflagrations resulting from localized buildup of combustible gases during degraded core accidents.

Containment Overpressure Protection: The pressure capability of the ESBWR containment vessel is such that it will not be exceeded by any design-basis or special event. The pressure capability of the containment's limiting component is greater than the pressure that results from assuming a 100-percent fuel clad-coolant reaction. There is sufficient margin to the containment pressure capability such that there is no need for an automatic containment overpressure protection system. In the hypothetical situation, in which containment depressurization is required, this depressurization can be performed by manual operator action.

Containment Structural Integrity: In accordance with RG 1.7, Revision 3, an acceptable method for demonstrating containment structural integrity is to meet the ASME Code Section III acceptance criteria as follows:

- Steel containments meet the requirements of the ASME Boiler and Pressure Vessel Code (edition and addenda as incorporated by reference in 10 CFR 50.55a(b)(1)), Section III, Division 1, Subarticle NE-3220, "Service Level C Limits," considering pressure and dead load alone (evaluation of instability is not required).

- Concrete containments meet the requirements of the ASME Boiler and Pressure Vessel Code, Section III, Division 2, Subarticle CC-3720, "Factored Load Category," considering pressure and dead load alone.

DCD Table 6.2-46 summarizes the Level C pressure capabilities of the steel components of major penetrations. The Level C pressure capability of the ESBWR containment structure is 1.182 MPa (171.4 psi).

Postaccident Radiolytic Oxygen Generation: For a design-basis LOCA in the ESBWR, the ADS would depressurize the reactor vessel and the GDCS would provide gravity-driven flow into the vessel for emergency core cooling. The safety analyses show that the core does not uncover during this event and, as a result, there is no fuel damage or fuel clad-coolant interaction that would result in the release of fission products or hydrogen. Thus, for a design-basis LOCA, the generation of postaccident oxygen would not result in a combustible gas condition, and a design-basis LOCA does not have to be considered in this regard.

For the purposes of postaccident radiolytic oxygen generation for the ESBWR, a severe accident with a significant release of iodine and hydrogen is more appropriate to consider. Because the ESBWR containment is inerted, the prevention of a combustible gas deflagration is assured in the short term following a severe accident. In the longer term, an increase in the oxygen concentration would result from the continued radiolytic decomposition of the water in the containment. Because the possibility of a combustible gas condition is oxygen limited for an inerted containment, it is important to evaluate the containment oxygen concentration versus time following a severe accident to ensure that there will be sufficient time to implement severe accident management (SAM) actions. It is desirable to have at least a 24-hour period following an accident to allow for SAM implementation.

The DCD states that the analysis of the radiolytic oxygen concentration in containment was performed consistent with the methodology of Appendix A to SRP Section 6.2.5 and RG 1.7. Some of the key assumptions are as follows:

- reactor power was 102 percent of rated
- $G(O_2)$ = 0.25 molecules/100 electron volts
- initial containment O_2 concentration = 4 percent
- allowed containment O_2 concentration = 5 percent
- stripping of drywell noncondensable gases to wetwell vapor space
- fuel clad-coolant reaction up to 100 percent
- iodine release up 100 percent

The analysis results show that the time required for the oxygen concentration to increase to the deinerting value of 5 percent is significantly greater than 24 hours for a wide range of fuel clad-coolant interaction and iodine release assumptions up to and including 100 percent. The results support the conclusion that sufficient time will be available to activate the emergency response organization and implement the SAM actions necessary to preclude a combustible gas deflagration.

6.2.5.3 Staff Evaluation

6.2.5.3.1 Combustible Gas Control

The ESBWR design specifies that the containment will be inerted with nitrogen gas during normal operation. This means that the concentration of oxygen in the containment atmosphere will be maintained at less than 4 percent by volume while the reactor is in operation. This satisfies the requirement of 10 CFR 50.44(c)(2) and is therefore acceptable.

There is, however, an open item concerning the placement of a 4 percent by volume limitation on containment oxygen concentration in the TSs. RAI 16.2-110 (ADAMS Accession No. ML062840401) and the applicant's response (ADAMS Accession No. ML070320097) and RAI 16.2-110, Supplement 1, and the applicant's response (ADAMS Accession No. ML071990227) address this issue, which is described as follows:

> Proposed Technical Specification (TS) Section 3.6, Containment Systems, apparently does not have a TS for containment oxygen concentration. GE's response to RAI 16.0-1, dated August 8, 2006, in Enclosure 1, Attachment 2, item 27, asserts that an operating restriction on oxygen concentration (to less than 4% by volume) is not required as an initial condition in the analysis of any design-basis event, so it does not meet Criterion 2 of 10 CFR 50.36 and is not included in the proposed Technical Specifications.
>
> However, both the NRC staff and the nuclear industry's Technical Specification Task Force have stated that such a TS is required.
>
> - When 10 CFR 50.44, "Combustible Gas Control in Containment," was revised in 2003, he staff issued a model safety evaluation (SE) for implementation of the revised rule through the Consolidated Line Item Improvement Process (ADAMS Accession No. ML032600597, September 12, 2003). The model SE states, on page 13, that "...requirements for primary containment oxygen concentration will be retained in TS for plant designs with an inerted containment." Furthermore, the current standard TS for BWR/4 plants (NUREG-1433, Rev. 3.1) includes TS 3.6.3.2, Primary Containment Oxygen Concentration, which states that "The primary containment oxygen concentration shall be < 4.0 volume percent."
>
> - Technical Specification Task Force Traveler TSTF-447, Revision 1, dated July 18, 2003, "Elimination of Hydrogen Recombiners and Change to Hydrogen and Oxygen Monitors," which has been accepted by the staff, states: "For plant designs with an inerted containment, the requirement for primary containment oxygen concentration will be retained in Technical Specifications."
>
> In light of these positions, add a TS limiting containment oxygen concentration to less than 4 percent by volume.

The applicant responded to RAI 16.2-110 as follows:

> As stated in the model safety evaluation for implementation of the revised 10 CFR 50.44, "Combustible Gas Control In Containment," dated September 12, 2003, the basis for retention of this requirement in Technical Specifications (TS) is that it meets Criterion 2 of 10 CFR 50.36(c)(2)(ii) in that it is a process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

This is based on the fact that calculations typically included in Chapter 6 of Updated Final Safety Analysis Reports assume that the primary containment is inerted, that is, oxygen concentration < 4.0 volume percent, when a design basis LOCA occurs.

Design Control Document (DCD), Tier 2, Subsection 6.2.5.5, "Post Accident Radiolytic Oxygen Generation," states that for a design basis loss of coolant accident (LOCA) in the ESBWR, the Automatic Depressurization System (ADS) would depressurize the reactor vessel and the Gravity Driven Cooling System (GDCS) would provide gravity driven flow into the vessel for emergency core cooling. The safety analyses show that the core does not uncover during this event and as a result, there is no fuel damage or fuel clad-coolant interaction that would result in the release of fission products or hydrogen. Thus, for the ESBWR Design Basis Accident (DBA), the generation of post accident oxygen would not result in a combustible gas condition and a design basis LOCA does not have to be considered in this regard. Therefore, GE's response to RAI 16.0-1, dated August 8, 2006, in Enclosure 1, Attachment 2, item 27, concluded that containment oxygen assumptions do not meet Criterion 2 of 10 CFR 50.36 and are not included in the proposed Technical Specifications. This conclusion, that Criterion 2 is not applicable, is also consistent with the existing Industry proposal to revise the Bases for those plants committed to retaining a Specification on oxygen concentration to reflect retention based on Criterion 4 of 10 CFR 50.36 (i.e., TSTF-478, "BWR Technical Specification Changes that Implement the Revised Rule for Combustible Gas Control").

Furthermore, from the Statements of Considerations (SOCs) for the Final Rule adopting the revisions to 10 CFR 50.44 (68FR54123, September 16, 2003) combustible gas control is clearly a beyond design basis accident (i.e., severe accident) issue. Limitations for these beyond design basis accidents have not been applied to evaluations against the criteria of 10 CFR 50.36(c)(2)(ii). Regarding the Technical Specification requirement for inerting, these SOCs acknowledge that for the existing BWR plants: "Retaining the requirement maintains the current level of public protection." This, in effect, mandates applicability of 10 CFR 50.36(c)(2)(ii), Criterion 4, on existing plants.

The ESBWR design certification does not fall under this discussion and reasoning for existing plants (i.e., there is no "current level of public protection" standard to evaluate). Furthermore, 50.36(c)(2)(ii)(D), Criterion 4, does not apply to a process variable or initial condition (e.g., as Criterion 2 does). Criterion 4 is restricted to SSCs. However, because the basis of the ESBWR severe accident analysis assumes containment inerting, GE commits to include an Availability Control, similar to other Regulatory Treatment of Non-Safety Systems (RTNSS) Availability Controls, in an Appendix to DCD Chapter 19.

The Availability Control will be modeled after the BWR4 NUREG-1433, LCO 3.6.3.2, "Primary Containment Oxygen Concentration," and will be incorporated in DCD Chapter 19, Revision 3.

DCD Impact

An Availability Control for containment oxygen concentration will be included in an Appendix to DCD, Tier 2, Chapter 19, Revision 3.

In RAI 16.2-110, Supplement 1, the NRC staff stated the following:

RAI 16.2-110 requested that GE add a Technical Specification (TS) limiting containment oxygen concentration to less than 4 percent.

GE has responded that the four criteria of 10 CFR 50.36(c)(2)(ii) do not require it. Criterion 2 covers process variables and operating restrictions, but only those which are related to design basis accidents. They argue that the requirements of 10 CFR 50.44, combustible gas control, are derived from beyond-design-basis or severe accidents, so Criterion 2 does not apply.

They further argue that Criterion 4 does not apply: "A structure, system, or component [SSC] which operating experience or probabilistic risk assessment has shown to be significant to public health and safety." They point out that Criterion 4 does not apply to process variables or initial conditions, but rather is restricted to SSCs.

The staff asserts that the fundamental basis for ESBWR's compliance with 50.44 depends on the containment being inerted. The Federal Register Notice for the final 10 CFR 50.44 rulemaking stated that combustible gases produced by beyond design-basis accidents involving both fuel-cladding oxidation and core-concrete interaction would be risk-significant for plants with inerted containments, if not for the inerted containment atmosphere. If not inerted, the ESBWR containment will not be protected from combustible gas events and will not be safe enough to allow reactor operation. The public would not have the protection required by the regulation. The staff's position is that there must be a license requirement limiting containment oxygen concentration to less than 4 percent. If necessary, the TS on containment operability could be enhanced by adding an oxygen concentration limit or surveillance requirement as being necessary for containment operability (a system, per Criterion 4). An explicit TS limit would seem to be prudent for a future licensee; if the TS were silent on oxygen concentration, then an uninerted containment could be declared an inoperable containment, and ESBWR proposed LCO 3.6.1.1 ("Containment shall be OPERABLE.") would allow only one hour before requiring initiation of shutdown. Plant operation with an uninerted containment would result in noncompliance with the requirements of 50.44, which could, at the least, lead to violations, citations, enforcement action, and an over-all less stable regulatory environment, without appropriate surveillance requirements, limiting conditions, and associated actions.

One approach could be to create a TS safety limit for oxygen concentration. 10 CFR 50.36(c)(1) says that "Safety limits for nuclear reactors are limits upon important process variables [e.g., oxygen concentration] that are found to be necessary to reasonably protect the integrity of certain of the physical barriers that guard against the uncontrolled release of radioactivity [e.g., containment]. If any safety limit is exceeded, the reactor must be shut down."

Alternately, a license condition could be imposed to prohibit plant operation if oxygen concentration is greater than or equal to 4 percent. This would be outside the purview of 50.36.

These approaches to place a regulatory limit on containment oxygen concentration during operation of ESBWR plants would need to be further developed.

The point the staff wishes to make is that it is essential to have a regulatory limit on containment oxygen concentration in ESBWR licenses. Various mechanisms are available, but a separate TS on oxygen concentration, similar to TS 3.6.3.2 in the BWR/4 STS, would allow 24 hours before requiring initiation of shutdown, as well as leeway on inerting and de-inerting during start-up and shut down.

Please propose a regulatory limit requiring containment oxygen concentration to be less than 4 percent.

The applicant responded to RAI 16.2-110, Supplement 1, in the following manner:

As stated in the Staff's comment, ESBWR compliance with 10 CFR 50.44 requires the containment be inerted. Supporting that regulatory requirement, DCD Tier 2, Revision 3, subsections 6.2.5.1 and 9.4.9.1 state the design basis for the ESBWR is for an inerted containment. Since Tier 2 is incorporated by reference in the Regulations upon design certification, there are ESBWR-specific regulatory limitations imposed to assure the containment is inerted.

GE recognizes the benefit of proposing a regulatory allowance for a limited time to operate with the containment oxygen concentration below the limit. Such a control was proposed in the previous response by way of the inclusion of an Availability Control within the Regulatory Treatment of Non-Safety Systems (RTNSS) Controls to be included in DCD, Tier 2, Chapter 19.

This Availability Control, imposing a limiting condition for containment oxygen concentration, will also provide for the appropriate compensatory actions and restoration timeframes for operation with the containment atmosphere not inerted to within limits. Appropriate surveillance requirements to monitor this condition will also be provided.

The original GE action to include an Availability Control, similar to other RTNSS Availability Controls, modeled after the BWR4 NUREG-1433, LCO 3.6.3.2, "Primary Containment Oxygen Concentration," will provide the limit on containment oxygen concentration as well as the leeway on inerting and de-inerting during start-up and shut down that the Staff discusses above.

DCD Impact

An Availability Control for containment oxygen concentration will be included in an Appendix to DCD, Tier 2, Chapter 19, Revision 4.

The staff finds that the applicant's latest response is unacceptable. The regulatory limit proposed by the applicant, based on the future design certification rulemaking for the ESBWR, will be too far removed from the day-to-day operation of a plant to provide sufficient control of and attention to the containment oxygen concentration limit. It adds little to the requirements already present in 10 CFR 50.44. Furthermore, using the applicant's suggested availability control also lacks sufficient regulatory force. The staff's position is that a TS limiting condition for operation must be established for an inerted containment to meet 10 CFR 50.36(c)(2)(iv). The structure is the inerted containment. The NRC has determined that combustible gases produced by BDBA involving both fuel-cladding oxidation and core-concrete interaction would be risk significant for plants with inerted containments, if not for the inerted containment atmosphere. It is essential to have a regulatory limit on containment oxygen concentration in each ESBWR plant license, meaning a TS limiting condition for operation. The staff's additional information requirements are contained in RAI 16.2-110, Supplement 2**. RAI 16.2-110 is being tracked as an open item**.

6.2.5.3.2 Mixed Atmosphere

The DCD states that the containment will have a mixed atmosphere during accidents, but presents no justification or analysis of the design's capability for ensuring a mixed atmosphere. This was discussed in RAI 6.2-138, to which the applicant has not yet responded. RAI 6.2-138 is reproduced below.

<u>RAI 6.2-138</u>

> Describe and justify capability for ensuring a mixed containment atmosphere.
>
> 10 CFR 50.44(c)(1) states:
>
> Mixed atmosphere. All containments must have a capability for ensuring a mixed atmosphere during design-basis and significant beyond design-basis accidents.
>
> The following is the complete text of DCD, Tier 2, Revision 1, Section 6.2.5.3.4, "Containment Atmosphere Mixing":
>
>> The ESBWR design provides protection from localized combustible gas deflagrations including the capability to mix the steam and non-condensable gases throughout the containment atmosphere and minimize the accumulation of high concentrations of combustible gases in local areas. The containment design features that will reduce the likelihood of combustible gas deflagrations resulting from localized buildup of combustible gases during degraded core accidents are listed in Section 19.3.
>>
>> It appears that Section 19.3.2.1, "Hydrogen Generation and Control," is the only part of Section 19.3 that mentions containment atmosphere mixing. The problem is that the only mention of it is a statement that the analysis of post-accident oxygen concentration assumes "Adequate gas mixing throughout containment."

Insofar as an assumption is not an explanation or justification, add an appropriate discussion to the DCD which explains and justifies ESBWR's capability for ensuring a mixed atmosphere during design-basis and significant beyond design-basis accidents. The discussion should address: passive features of the design, including containment/subcompartment layout, elevations, and openings between compartments that impact mixing; active features of the design, including ventilation systems, cooling systems, and spray systems; and the effectiveness of the passive and active features in providing a mixed atmosphere in the design-basis and significant beyond design-basis events. If non-safety related systems are relied upon for mixing, the availability of these systems in the frequency-dominant beyond design-basis events and any "special treatment" requirements for these systems should also be addressed.

RAI 6.2-138 is being tracked as an open item.

6.2.5.3.3 Oxygen Monitor

RAI 6.2-137 and the applicant's response describe the issues regarding oxygen monitors.

<u>RAI 6.2-137, Supplement 1</u>

The response to this RAI is not specific enough to allow the staff to draw conclusions as to the acceptability of the design of the oxygen monitors. Also, the information contained in the responses to the RAI and its supplement(s) needs to be put into the DCD, Tier 2. Here is a detailed description of the additional requested information:

Item (1) of the RAI response states that the instrument range will be met under "the specified pressure conditions" for the ESBWR design, yet the response did not include any specified pressure conditions. It is not clear if the "specified pressure conditions" means containment design pressure, pressures resulting from significant beyond design-basis accidents, or something else. Provide the "specified pressure conditions."

Item (2) gives numbers for the instrument accuracies, but the numbers are enclosed in square brackets. What does this mean? A conventional meaning of square brackets is that the numbers are suggested or typical values, but that individual plants may choose different numbers based on various design considerations. Provide specific accuracies for the oxygen monitors and justify that they are adequate for their intended function, or develop a COL Action Item to require COL applicants to do so, subject to NRC review and approval during COL reviews. Also in item (2), the staff had asked the applicant to provide the placement of the monitor's sampling points, and to justify that this placement is adequate for their intended function. This information was not provided. Instead, the response stated that sampling points "will be selected" according to certain criteria. Provide the specific information which was requested, or develop a COL Action Item to require COL applicants to do so, subject to NRC review and approval during COL reviews.

For Item (3), the staff had asked whether the monitoring system would remain functional and reliable when exposed internally to the temperature, pressure, humidity, and radioactivity of containment atmosphere during a significant beyond design-basis accident. The response stated that the equipment chosen "will be specified" and "will be evaluated" in accordance with certain general criteria. Provide an evaluation of the system's functionality and reliability against ESBWR-specific containment temperature, pressure, humidity, and radioactivity conditions during significant beyond design-basis accidents, or develop a COL Action Item to require COL applicants to do so, subject to NRC review and approval during COL reviews. The staff cautions the applicant that the recommended design provisions for oxygen monitors in the final issue of RG 1.7, Revision 3, Section 2.2, are significantly different from those in draft Revision 3, at least in form. If the applicant cites RG 1.7 in the future, the applicant should specify which version (draft or final) is being used.

RAI 6.2-137 is being tracked as an open item.

6.2.5.3.4 Hydrogen Monitor

RAI 6.2-136 and the applicant's response describe the issues regarding hydrogen monitors.

<u>RAI 6.2-136, Supplement 1</u>

The response to this RAI is not specific enough to allow the staff to draw conclusions as to the acceptability of the design of the hydrogen monitors. Also, the information contained in the responses to the RAI and its supplement(s) needs to be put into the DCD, Tier 2. Here is a detailed description of the additional requested information:

Item (A) a) of the RAI response states that the instrument range will be met under "the specified pressure conditions" for the ESBWR design, yet the response did not include any specified pressure conditions. It is not clear if the "specified pressure conditions" means containment design pressure, pressures resulting from significant beyond design-basis accidents, or something else. Provide the "specified pressure conditions." Item (A) b) gives numbers for the instrument accuracies, but the numbers are enclosed in square brackets. Staff is not clear on the meaning of the enclosed square brackets. The conventional meaning of square brackets is that the numbers are suggested or typical values, but that individual plants may choose different numbers based on various design considerations. Provide specific accuracies for the hydrogen monitors and justify that they are adequate for their intended function, or develop a COL Action Item to require COL applicants to do so, subject to NRC review and approval during COL reviews. Also in item (A) b), the staff had asked the applicant to provide the placement of the monitor's sampling points, and to justify that this placement is adequate for their intended function. This information was not provided. Instead, the response stated that sampling points "will be selected" according to certain criteria. Provide the specific information that was originally requested, or develop a COL Action Item to require COL applicants to do so, subject to NRC review and approval during COL reviews.

The Item (B) response stated that the equipment warmup time "will be evaluated" during the specification and procurement process to ensure that the warmup time noted in RG 1.7, Revision 3, is not exceeded. Develop a COL Action Item to require COL applicants to do this, subject to NRC review and approval during COL reviews.

For Item (C), the staff had asked whether the monitoring system would remain functional and reliable when exposed internally to the temperature, pressure, humidity, and radioactivity of containment atmosphere during a significant beyond design-basis accident. The response stated that the equipment chosen "will be specified" and "will be evaluated" in accordance with certain general criteria. Provide an evaluation of the system's functionality and reliability against ESBWR-specific containment temperature, pressure, humidity, and radioactivity conditions during significant beyond design-basis accidents, or develop a COL Action Item to require COL applicants to do so, subject to NRC review and approval during COL reviews. The staff cautions the applicant that the recommended design provisions for oxygen monitors in the final issue of RG 1.7, Revision 3, Section 2.2, are significantly different from those in draft Revision 3, at least in form. If the applicant cites RG 1.7 in the future, the applicant should specify which version (draft or final) is being used.

RAI 6.2-136 is being tracked as an open item.

6.2.5.3.5 Structural Analysis

As required by 10 CFR 50.44(c)(5), the applicant must perform an analysis that demonstrates containment structural integrity. This demonstration must use an analytical technique that is accepted by the NRC and includes sufficient supporting justification to show that the technique describes the containment response to the structural loads involved. The analysis must address an accident that releases hydrogen generated from a 100 percent fuel clad-coolant reaction accompanied by hydrogen burning. Systems necessary to ensure containment integrity must also be demonstrated to perform their function under these conditions.

RG 1.7, Revision 3, Section C.5, describes an analytical technique that is accepted by the NRC. The applicant has used this technique in DCD Tier 2, Section 6.2.5.4.2, and concluded that the Level C pressure capability of the containment structure is 1.182 MPa (171.4 psi).

However, the applicant has not included sufficient supporting justification to show that the technique describes the containment response to the structural loads involved, in that the applicant has not addressed the structural loads involved. DCD Tier 2, Section 6.2.5.4.1, does make the following two statements addressing this point:

- "The pressure capability of the ESBWR containment vessel is such that it will not be exceeded by any design basis or special event."

- "The pressure capability of the containment's limiting component is higher than the pressure that results from assuming 100 percent fuel clad-coolant reaction. There is sufficient margin to the containment pressure capability such that there is no need for an automatic containment overpressure protection system."

These statements are not specific enough for the staff to be able to determine whether the structural integrity of the containment design is acceptable. First, it is unclear as to what is meant by a "special event." Second, the DCD does not provide the actual pressure that results from assuming a 100 percent fuel clad-coolant reaction, and most especially does not indicate whether the assumption of a 100-percent fuel clad-coolant reaction includes hydrogen burning, as required by 10 CFR 50.44(c)(5). It may be that the inerted condition of the containment would preclude burning for many or most accidents, but there may be DBA sequences in which sufficient oxygen is generated by radiolysis of water to support combustion.

RAI 6.2-178 includes the staff's additional information requirements.

RAI 6.2-178

DCD, Revision 3, Tier 2, Subsection 6.2.5.4.1, makes these two statements addressing containment structural integrity:

- The pressure capability of the ESBWR containment vessel is such that it will not be exceeded by any design basis or special event.

- The pressure capability of the containment's limiting component is higher than he pressure that results from assuming 100 percent fuel clad-coolant reaction. There is sufficient margin to the containment pressure capability such that there is no need for an automatic containment overpressure protection system.

These statements are not specific enough for the staff to be able to determine whether the structural integrity of the containment design is acceptable. First, it is unclear as to what is meant by a "special event." Second, the DCD does not provide the actual pressure that results from assuming 100 percent fuel clad-coolant reaction, and most especially does not indicate whether the assumption of 100 percent fuel clad-coolant reaction includes hydrogen burning, as required by 10 CFR 50.44(c)(5). It may be that the inerted condition of the containment would preclude burning for many or most accidents, but there may be beyond design-basis accident sequences in which sufficient oxygen is generated by radiolysis of water to support combustion.

Provide in the DCD a description of which "special events" were considered in the analysis. Provide the actual pressure that results from assuming 100 percent fuel clad-coolant reaction, and whether the assumption of 100 percent fuel clad-coolant reaction includes hydrogen burning. If no hydrogen burning was assumed for any accident, justify this assumption, with consideration of beyond design-basis accident information from DCD, Tier 2, Chapter 19.

RAI 6.2-178 is being tracked as an open item.

6.2.5.3.6 Other Regulations

This section addresses regulations, other than 10 CFR 50.44, that relate to combustible gas control in containment. Section 6.2.5.1 above lists these regulations.

The ESBWR design meets the relevant requirements of the following:

- GDC 5 does not apply because there is no sharing of SSCs between different units.

- GDC 41, as it relates to systems being provided to control the concentration of hydrogen or oxygen that may be released into the reactor containment following postulated accidents to ensure that containment integrity is maintained, is satisfied because the safety function is accomplished by keeping the containment inerted. Thus, no redundancy or single failure criteria need be considered, as the inerted containment is intrinsically safe and passive.

- GDC 42 and 43, related to the design of the systems to permit appropriate periodic inspection and periodic testing of components to ensure the integrity and capability of the systems, do not apply to the inerting function; periodic monitoring of oxygen concentration is adequate to confirm the safety function.

- 10 CFR 52.47(b)(1) relates to ITAAC. Chapter 14 of this SER addresses this topic.

- 10 CFR 52.80(a) applies to COL applications, not a design certification application like that submitted for the ESBWR.

6.2.5.4 Conclusions

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

6.2.6 Containment Leakage Testing

DCD Tier 2, Section 6.2.6, describes the proposed containment leakage rate testing program for the ESBWR.

6.2.6.1 Regulatory Criteria

Conformance with the requirements of either Option A or B of Appendix J to 10 CFR Part 50 and the provisions of RG 1.163, "Performance-Based Containment Leak-Test Program," issued September 1995, constitutes an acceptable basis for satisfying the requirements of the following GDC applicable to containment leakage rate testing:

GDC 52, "Capability for Containment Leakage Rate Testing," as it relates to the reactor containment and exposed equipment being designed to accommodate the test conditions for the containment integrated leakage rate test (up to the containment design pressure)

GDC 53, as it relates to the reactor containment being designed to permit appropriate inspection of important areas (such as penetrations), an appropriate surveillance program, and leakage rate testing at the containment design pressure of penetrations having resilient seals and expansion bellows

GDC 54, as it relates to piping systems that penetrate primary reactor containment being designed with a capability to determine whether the valve leakage rate is within acceptable limits

10 CFR 52.47(b)(1), which requires that a design certification application contain the proposed ITAAC that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design certification is built and will operate in

accordance with the design certification, the provisions of the Atomic Energy Act, and the NRC's regulations

6.2.6.2 Summary of Technical Information

This subsection describes the testing program for determining the containment integrated leakage rate (Type A tests), containment penetration leakage rates (Type B tests), and containment isolation valve leakage rates (Type C tests) that complies with Option A or B of Appendix J to 10 CFR Part 50, in accordance with RG 1.163 and GDC 52, 53, and 54. The leakage rate testing capability is consistent with the testing requirements of ANS-56.8.

Type A, B, and C tests are performed before operations and periodically thereafter to ensure that leakage rates through the containment and through systems or components that penetrate containment do not exceed the maximum allowable rates. Maintenance of the containment, including repairs on systems and components penetrating the containment, is performed as necessary to maintain leakage rates at or below acceptable values.

6.2.6.2.1 Containment Integrated Leakage Rate Test (Type A)

Integrated leak rate tests (ILRTs) (Type A tests) are conducted periodically in conformance with Appendix J to 10 CFR Part 50 to ensure that the containment integrity is maintained and to determine whether the leakage rate has increased since the previous ILRT. The tests are performed after major repairs and upon indication of excessive leakage. Verification tests are also performed after each ILRT. After the initial ILRT, periodic ILRTs will be performed at intervals depending on whether Option A or Option B of Appendix J to 10 CFR Part 50 is selected by the COL holder. If Option A is selected, the ILRTs will be performed at least three times during each 10-year service period. If Option B is selected, the test interval will be in accordance with RG 1.163.

In addition, any major modification or replacement of components of the reactor containment performed after the initial ILRT is followed by either a Type A or a Type B test of the area affected by the modification, with the affected area meeting the applicable acceptance criteria.

A standard statistical analysis of the data is conducted by a linear regression analysis using the method of least squares to determine the leakage rate and associated 95-percent upper confidence limit (UCL). ILRT results are satisfactory if the UCL is less than 75 percent of the maximum allowable leakage rate, L_a. As an exemption from the definition of L_a in Appendix J to 10 CFR Part 50, the maximum allowable leakage rate (L_a) is redefined as "containment leakage rate" in DCD Table 6.2-1, which excludes the MSIV leakage rate. The treatment of MSIV leakage pathway separately in radiological dose analysis in Section 15.4.4.5.2 of the DCD justifies this exemption.

After completing the initial ILRT, a verification test is conducted to confirm the ability of the ILRT method and equipment to satisfactorily determine the containment leakage rate. The accuracy of the leakage rate tests is verified by superimposing a calibrated leak on the normal containment leakage rate or by other methods of demonstrated equivalency. The difference between the total leakage and the superimposed known leakage is the actual leakage rate. This method confirms the test accuracy. The measurements are acceptable if the correlation between the verification test data and the ILRT data demonstrates an agreement within ± 0.25

L_a. Appendix C to ANSI/ANS-56.8, “Containment System Leakage Testing Requirements,” includes more descriptive information on verification methods.

During the ILRT (including the verification test), if excessive leakage occurs through locally testable penetrations or isolation valves to the extent that it would interfere with satisfactory completion of the test, these leakage paths may be isolated and the Type A test continued until completion. A local test shall be performed before and after the repair of each isolated path. The test results shall be reported with both pre-and post-repair local leakage rates, as if two Type A tests had been conducted. A record of corrective actions shall be documented as described below:

> For Option A of Appendix J to 10 CFR Part 50, the sum of the local leakage rates and the UCL shall be less than 0.75 L_a. Local leakage rates shall not be subtracted from the Type A test results to determine acceptability of the test.
>
> For Option B of Appendix J to 10 CFR Part 50, the acceptance criteria shall be based on a calculated performance leakage rate that is defined as the sum of the Type A UCL and the as-left minimum pathway leakage rate (MNPLR) for all Type B and Type C pathways that were in service, isolated, or not lined up in their test position (i.e., drained and vented to containment atmosphere) before performing the Type A test. In addition, any leakage pathways that were isolated during performance of the test shall be factored into the performance determination. If the leakage can be determined by a local leak rate test, the as-left MNPLR for that leakage path must also be added to the Type A UCL. If the leakage cannot be determined by local leak rate testing, the performance criteria for the Type A test are not met.

If Option A of Appendix J to 10 CFR Part 50 is followed, and if two consecutive periodic ILRTs fail to meet the acceptance criteria before corrective action, an ILRT is performed at each plant shutdown for major refueling or approximately every 24 months (whichever occurs first), until two consecutive ILRTs meet the acceptance criteria, after which time the previously established periodic retest schedule may be resumed.

If Option B of Appendix J to 10 CFR Part 50 is followed, and if the ILRT results are not acceptable, then a determination should be performed to identify the cause of the unacceptable performance and to determine appropriate corrective actions.

Once the cause determination and corrective actions have been completed, acceptable performance should be reestablished by performing an ILRT within 48 months following the unsuccessful ILRT test. Following a successful ILRT, the surveillance frequency may be returned to once every 10 years.

The following additional criteria will be met for ILRTs if Option A of Appendix J to 10 CFR Part 50 is implemented:

- The following portions of systems are kept open or vented to the containment atmosphere during the ILRT:

- portions of fluid systems that are part of the RCPB that are open directly to the reactor containment atmosphere under postaccident conditions and that become an extension of the boundary of the reactor containment

- portions of closed systems inside containment that penetrate containment and that are not relied upon for containment isolation purposes following a LOCA

- portions of closed systems inside containment that penetrate containment and rupture as a result of a LOCA

Note, however, that the ESBWR does not have any system that penetrates the containment and ruptures as a result of LOCA.

- All systems not designed to remain filled with fluid (e.g., vented) after a LOCA are drained of water to the extent necessary to ensure exposure of the system containment isolation valves to the containment air test pressure.

- Those portions of fluid systems penetrating containment that are external to the containment and that are not designed to provide a containment isolation barrier are vented to the outside atmosphere, as applicable, to ensure that full postaccident differential pressure is maintained across the containment isolation barrier.

- Systems that are required to maintain the plant in a safe condition during the ILRT are operable in their normal mode and are not vented. Also, systems that are normally filled with water and operating under post-LOCA conditions are not vented. Results of local leakage rate tests of penetrations associated with these systems are added to the ILRT results.

The following additional criteria will be met for ILRTs if Option B of Appendix J to 10 CFR Part 50 is implemented. All Appendix J pathways must be properly drained and vented during the performance of ILRT, with the following exceptions:

- pathways in systems that are required for proper conduct of the ILRT or to maintain the plant in a safe-shutdown condition during the ILRT

- pathways in systems that are normally filled with fluid and operable under postaccident conditions

- portions of pathways outside primary containment that are designed to seismic Category I and at least Safety Class 2

- for planning and scheduling purpose, or as low as reasonably achievable (ALARA) considerations, pathways that are Type B or C tested within the previous 24 calendar months need not be vented or drained during the ILRT

6.2.6.2.2 Containment Penetration Leakage Rate Test (Type B)

Containment penetrations designed to incorporate resilient seals, bellows, gaskets, or sealant compounds; air locks and air-lock door seals; equipment and access hatch seals; and electrical penetration canisters receive preoperational and periodic Type B leakage rate tests in accordance with Appendix J to 10 CFR Part 50. The local leak detection tests of Type B and Type C are completed before the preoperational or periodic Type A tests.

Type B tests are performed at containment peak accident pressure, P_a, by local pressurization using either the pressure-decay or flowmeter method. For the pressure-decay method, a test volume is pressurized with air or nitrogen to at least P_a. The rate of decay of pressure of the known test volume is monitored to calculate the leakage rate. For the flowmeter method, the required test pressure is maintained in the test volume by making up air or nitrogen, through a calibrated flowmeter. The flowmeter fluid flow rate is the leakage rate from the test volume. The plant-specific TSs include the acceptance criteria for Type B tests. The combined leakage rate of all components subject to Type B and Type C tests should not exceed 60 percent of L_a.

In accordance with Appendix J to 10 CFR Part 50, Type B tests are performed at intervals depending on whether Option A or Option B is selected on a unit-specific basis. If Option A is selected, Type B tests (except for air locks) will be performed during each reactor shutdown for major fuel reloading, or other convenient intervals, but never at intervals greater than 2 years. Under this option, air locks opened when containment integrity is required are tested in manual mode within 3 days of being opened. If the air lock is to be opened more frequently than once every 3 days, it is tested at least once every 3 days during the period of frequent openings. The acceptance criterion for an air lock is a leakage rate of less than or equal to 0.05 L_a, when tested at a pressure greater than or equal to P_a.

As an exemption from Appendix J to 10 CFR Part 50, Section III.D.2.(b)(ii), can be satisfied by testing at the end of periods when containment integrity is not required by the plant’s TSs at a lower test pressure specified in the TS applied between the door seals with an acceptable maximum measured leakage rate of 0.01 L_a. Air locks are tested at initial fuel loading, and at least once every 6 months thereafter. If Option B is selected, the test interval will be in accordance with RG 1.163.

Air locks that are allowed to be opened during power operation may be tested at power operation so as to avoid shutting down. Personnel air locks through the containment include provisions for testing the door seals and the overall air-lock leakage rates. Each door includes test connections that allow the annulus between the seals to be pressurized and the pressure decay (if the pressure-decay method is used) or flow (if the flowmeter method is used) is monitored to determine the leaktight integrity of the seals.

Test connections are also provided on the outer face of each bulkhead so that the entire lock interior can be pressurized and the pressure decay or flow monitored to determine the overall lock leakage. Clamps or tie-downs are installed to keep the doors sealed during the overall lock test because normal locking mechanisms are not designed for the full differential pressure across the door in the reverse direction.

6.2.6.2.3 Containment Isolation Valve Leakage Rate Test (Type C)

Type C tests are performed on all containment isolation valves required to be tested by either Option A or Option B of Appendix J to 10 CFR Part 50. Type C tests (like Type B tests) are performed by local pressurization using either the pressure decay or flowmeter method. The test pressure is applied in the same direction as when the valve is required to perform its safety function, unless it can be shown that results from tests with pressure applied in a different direction are equivalent or conservative.

Valves that are sealed with a fluid from a seal system or valves not provided with a seal system and that may be justified to be equivalent to valves with a seal system shall be tested in accordance with Option A or Option B of Appendix J to 10 CFR Part 50.

A valid justification for equivalency of such valves is that they are located in lines designed to be, or remain, filled with water for at least 30 days subsequent to a LOCA. All test connections, vent lines, or drainlines consisting of double or multiple barriers (e.g., two valves in series, one valve and a cap, or one valve and a flange) that are connected between isolation valves, form a part of the containment boundary, and are 25.4 mm (1 in.) or less in size may not be Type C-tested because of their infrequent use and because the multiple barrier configurations are maintained using an administrative control program.

Type C testing shall be performed in the correct direction of the leakage path unless it can be demonstrated that testing in the reverse direction is equivalent or more conservative. The correct direction of the leakage path is from inside the containment to outside containment. Instrument lines that penetrate containment conform to RG 1.11 and may not be Type C tested. The lines that connect to the RCPB include a restricting orifice inside containment, are seismic Category I, and terminate in seismic Category I instruments. The instrument lines also include manual isolation valves and excess flow check valves or equivalent.

These valves are normally open and are considered extensions of the containment, the integrity of which is continuously demonstrated during normal operation. In addition, these lines are subject to the periodic Type A test because they are open (up to the pressure boundary instruments) during the ILRT. Leaktight integrity is also verified during functional and surveillance activities as well as visual observations during operator tours. The combined leakage rate of all components subject to Type B and Type C tests shall not exceed 60 percent of L_a. The plant-specific TSs detail the periodic leakage rate test schedule requirements for Types A, B, and C tests. Type B and C tests may be conducted at any time during normal plant operations or during shutdown periods, with test intervals that conform to either Option A or Option B of Appendix J to 10 CFR Part 50. Each time a Type B or Type C test is completed, the overall total leakage rate for all required Type B and Type C tests is updated to reflect the most recent test results.

In addition to the periodic tests, any major modification or replacement of a component that is part of the primary reactor containment boundary performed after the preoperational leakage rate test will be followed by either a Type A, B, or C test (as applicable) for the area affected by the modification. The leakage test summary report will describe the containment inspection method, any repairs necessary to meet the acceptance criteria, and the test results. Following the drywell structural integrity test, a preoperational drywell-to-wetwell leakage rate test is performed at the peak drywell-to-wetwell differential pressure. Also, drywell-to-wetwell leakage rate tests are conducted at a reduced differential pressure corresponding approximately to the submergence of the vents. These tests are performed following the preoperational ILRT and periodically thereafter. They verify that no paths exist for gross leakage from the drywell to the wetwell air space that bypass the pressure suppression pool. The combination of the peak pressure and reduced pressure leakage tests also verifies adequate performance of the drywell over the full range of postulated primary system break sizes.

Drywell-to-wetwell leakage rate tests are performed with the drywell isolated from the wetwell. Valves and system lineups are the same as for the ILRT, except for paths that equalize drywell and wetwell pressure, which are open during the ILRT and are isolated during the drywell leakage test. The drywell atmosphere is allowed to stabilize for a period of 1 hour after attaining the test pressure. Leakage rate test calculations, using the wetwell pressure rise method, commence after the stabilization period.

The pressure rise method is based on containment atmosphere pressure and temperature observations and the known wetwell volume. The leakage rate is calculated from the pressure and temperature data, wetwell free air volume, and elapsed time.

The plant-specific TSs specify the periodic drywell-to-wetwell leakage rate test pressure, duration, frequency, and acceptance criteria.

6.2.6.3 Staff Evaluation

The staff reviewed the information in DCD Tier 2 for conformance with the requirements of Appendix J to 10 CFR Part 50 and GDC 52, 53, and 54. The staff used the guidance, staff positions, and acceptance criteria of SRP Section 6.2.6 and RG 1.163 in conducting its review.

Meeting the requirements of Appendix J to 10 CFR Part 50 provides assurance that the leaktightness of the containment will be within the values specified in the facility TSs and that offsite radiation doses in excess of the reference values specified in 10 CFR Part 100 will not occur. Chapter 14 of this SER addresses both 10 CFR 52.47(b)(1), as it relates to ITAAC, and the ITAAC themselves.

Based on its review, the staff has two open items. The applicant has to respond to RAI 6.2-90, Supplement 1, to resolve that water cannot be a Type C test medium and RAI 6.2-91, Supplement 1, to revise the DCD such that the seal system provisions comply with the Appendix J requirements, as discussed below.

In RAI 6.2-90, the staff stated the following:

> DCD Tier 2, Section 6.2.6.3, “Containment Isolation Valve Leakage Rate Test (Type C),” states that, for the flowmeter method, water may be used as a test medium for Type C tests, “if applicable.”
>
> Option A, section III.C.2.(a), “Test Pressure,” states: “Valves, unless pressurized with fluid (e.g., water, nitrogen) from a seal system, shall be pressurized with air or nitrogen at a pressure of P_a.”
>
> Option B, section III.B., begins: “Type B pneumatic tests...and Type C pneumatic tests….” Applicable guidance is in ANSI/ANS-56.8-1994, section 3.3.5, “Test Medium,” which states, in part, “Type B and Type C tests shall be conducted with air or nitrogen.”
>
> The leakage rate tests for containment isolation valves (CIVs) served by seal systems are not Type C tests per se and are addressed in RAI 6.2-91.
>
> Delete the option for water as a Type C test medium from the DCD.

RAI 6.2-90 and RAI 6.2-91 are being tracked as open items.

6.2.6.4 Generic Issues

One generic issue is included in the staff's review of containment leakage rate testing, which is Item A-23, "Containment Leak Testing."

The staff addressed Item A-23 by revising and clarifying Appendix J to 10 CFR Part 50 and issuing RG 1.163. Thus, no additional review or action is required relative to Item A-23 for the ESBWR.

6.2.6.5 Conclusions

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

6.2.7 Fracture Prevention of Containment Pressure Boundary

6.2.7.1 Regulatory Criteria

The reactor containment system includes the functional capability of enclosing the reactor system and of providing a final barrier against the release of radioactive fission products. Fracture prevention of the containment pressure boundary needs to be assured. The ESBWR must address the following regulations:

- GDC 1 requires that SSCs important to safety shall be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed. Section 6.1.1 addresses the applicant's discussion and the staff's evaluation.

- GDC 16 requires that the reactor containment and associated systems shall be provided to establish an essentially leaktight barrier against the uncontrolled release of radioactivity to the environment and to ensure that the containment design conditions important to safety are not exceeded for as long as postulated accident conditions require. Sections 6.2.3 and 6.2.4 address the applicant's discussion and the staff's evaluation.

- GDC 51, "Fracture Prevention of Containment Pressure Boundary," requires that the reactor containment boundary shall be designed with sufficient margins to ensure that under operating, maintenance, testing, and postulated accident conditions (1) its ferritic materials behave in a nonbrittle manner, and (2) the probability of rapidly propagating fracture is minimized. The design shall reflect consideration of service temperatures and other conditions of the containment boundary material during operation, maintenance, testing, and postulated accident conditions and the uncertainties in determining (1) material properties, (2) residual, steady-state, and transient stresses, and (3) flaw size.

The staff reviewed the ESBWR DCD to ascertain whether containment pressure boundary materials meet the requirements of GDC 51.

6.2.7.2 Summary of Technical Information

The containment vessel of the ESBWR is a reinforced concrete structure with ferritic parts, such as a liner and a removable head. The ferritic parts are made of materials that have a nil ductility transition temperature sufficiently below the minimum service temperature to ensure that under operating, maintenance, testing, and postulated accident conditions the ferritic materials behave in a nonbrittle manner, considering the uncertainties in determining the material properties, stresses, and size of flaws. In Table 6.1-1 of DCD Tier 2, the applicant identified the containment vessel liner materials which are in conformance with the ASME Code, Section III (CC-2520, Fracture Toughness Requirements for Materials). This meets the requirements of GDC 51. GDC 51 is only applicable to those parts of the containment that are to be made of ferritic materials.

6.2.7.3 Staff Evaluation

The staff reviewed the ESBWR measures involving fracture prevention of ferritic materials used in the containment pressure boundary in accordance with SRP Section 6.2.7. Containment pressure boundary ferritic materials are acceptable if they meet the requirements of GDC 51 as it relates to the reactor containment pressure boundary being designed with sufficient margins to ensure that under operating, maintenance, testing, and postulated accident conditions the ferritic materials will behave in a nonbrittle manner and the probability of rapidly propagating fracture is minimized.

6.2.7.4 Conclusions

Based on the review of the information included in the ESBWR, the staff finds that the fracture toughness of the materials of the reactor containment pressure boundary meets the fracture toughness requirements specified in GDC 51. This satisfies the requirements of GDC 51 for fracture prevention of the containment pressure boundary.

The staff, therefore, concludes that the ESBWR will provide reasonable assurance that the materials of the reactor containment pressure boundary, under operating, maintenance, testing, and postulated accident conditions, will not undergo brittle fracture and that the probability of rapidly propagating fracture will be minimized, thereby meeting the requirements of GDC 51.

6.3 Emergency Core Cooling Systems

6.3.1 Emergency Core Cooling Systems Design

6.3.1.1 Regulatory Criteria

The staff reviewed DCD Tier 2, Section 6.3, “Emergency Core Cooling System,” for the ESBWR, in accordance with Sections 6.3 and 15.6.5 of the SRP.

The staff based its acceptance criteria on the following requirements:

- GDC 2, as it relates to the seismic design of SSCs whose failure could cause an unacceptable reduction in the capability of the ECCS to perform its safety function

- GDC 4, as it relates to dynamic effects associated with flow instabilities and loads (e.g., water hammer)

- GDC 5, "Sharing of Structures, Systems, and Components," as it relates to nuclear power units not sharing SSCs important to safety unless the applicant can demonstrate that sharing will not impair the ability of such SSCs to perform their safety function

- GDC 17, "Electric Power Systems," as it relates to the design of the ECCS having sufficient capacity and capability to ensure that the system does not exceed specified acceptable fuel design limits and the design conditions of the RCPB during anticipated operational occurrences and that the core is cooled during accident conditions

- GDC 27, "Combined Reactivity Control Systems Capability," as it relates to the ECCS design having the capability to ensure that, under postulated accident conditions and with appropriate margin for stuck rods, the system will maintain the capability to cool the core

- GDC 35, as it relates to the provision of an abundance of core cooling to transfer heat from the core at a rate so that fuel and clad damage will not interfere with continued effective core cooling

- GDC 36, "Inspection of Emergency Core Cooling System," as it relates to the appropriate periodic inspection of important components

- GDC 37, "Testing of Emergency Core Cooling System," as it relates to periodic pressure and functional testing

- 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors," as it relates to (1) the design of the ECCS, (2) ensuring that ECCS cooling performance is calculated in accordance with an acceptable evaluation model, and (3) demonstrating that the following five major ECCS acceptance criteria are met:

  (1) The calculated maximum fuel element cladding temperature does not exceed 1204℃ (2200℉).

  (2) The calculated total local oxidation of the cladding does not exceed 17 percent of the total cladding thickness before oxidation.

  (3) The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam does not exceed 1 percent of the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react.

  (4) Calculated changes in core geometry are such that the core remains amenable to cooling.

(5) After any calculated successful initial operation of the ECCS, the system maintains the calculated core temperature at an acceptably low value and removes decay heat for the extended period of time required by the long-lived radioactivity.

The staff also evaluated DCD Tier 2, Section 6.3 for conformance with the following sections of the TMI action plan:

- TMI Action Plan Item II.K.3.15 of NUREG-0737, which involves isolation of the high-pressure coolant injection and the reactor core isolation cooling for BWR plants

- TMI Action Plan Item II.K.3.18 of NUREG-0737, which is equivalent to 10 CFR 50.34(f)(1)(vii), with respect to eliminating the need for manual actuation of the BWR ADS to ensure adequate core cooling

- TMI Action Plan Item II.K.3.28 of NUREG-0737, which is equivalent to 10 CFR 50.34(f)(1)(x), with respect to BWR ADS-associated equipment and instrumentation being capable of performing their intended functions during and following an accident, while taking no credit for non-safety-related equipment or instrumentation and accounting for normal expected air (or nitrogen) leakage through valves

- TMI Action Plan Item II.K.3.45 of NUREG-0737, which is equivalent to 10 CFR 50.34(f)(1)(xi), with regard to providing an evaluation of depressurization methods, other than full actuation of the ADS, that would reduce the possibility of exceeding vessel integrity limits during rapid cooldown for BWRs

- TMI Action Plan Item III.D.1.1 of NUREG-0737, which is equivalent to 10 CFR 50.34(f)(2)(xxvi), with respect to the provisions for a leakage detection and control program to minimize the leakage from those portions of the ECCS outside of the containment that contain or may contain radioactive material following an accident

6.3.1.2 Summary of Technical Information

In DCD Tier 2, Section 6.3, GEH described the ECCS and the design criteria that satisfy the NRC regulatory requirements. Following is a brief summary of the GEH description.

Passive Core Cooling System

The Passive Core Cooling System is comprised of the GDCS, ADS, ICS, and the SLCS. The GDCS, in conjunction with the ADS, the ICS, and the SLCS, provides emergency core cooling in case of a LOCA. When the ADS receives an initiation signal, the ADS depressurizes the reactor vessel and the GDCS injects cooling water, in addition to that supplied by the ICS and SLCS, to maintain the peak cladding temperatures below the limits defined in 10 CFR 50.46.

Gravity-Driven Cooling System

The GDCS is a passive makeup water system. Water flows into the vessel by gravity from the GDCS pools. This differs from the ECCS in currently operating BWR/2-6 designs, which rely on active pumps and support systems. The GDCS injects water into the downcomer annulus region of the RPV following a LOCA and reactor vessel depressurization. It provides short-term, gravity-driven water makeup from three separate water pools located within the upper drywell at an elevation above the active core region through eight separate injection nozzles in the RPV. In the long-term, most of the coolant boil-off is returned to the RPV as condensate from the isolation condensers or the PCCS heat exchangers, however, there will be some boil-off loss of inventory to the drywell. The GDCS provides long-term, post-LOCA makeup from the suppression pool to meet long-term core decay heat boil-off requirements through four separate equalizing lines.

The GDCS is completely automatic in actuation and operation. A backup to automatic actuation is the ability to actuate by operator action. The GDCS consists of four identical trains independent of one another, both electrically and mechanically, with the exception of two trains sharing one of the three GDCS pools. Each GDCS injection and equalizing line consists of two normally locked-open manual valves, a check valve, and a squib-actuated valve. A confirmed low RPV water level signal actuates the ADS to reduce RPV pressure. In the GDCS logic, short-term and long-term timers simultaneously start. After timeout and satisfying permissive conditions, squib valves actuate to provide an open flow path from the water sources (GDCS pools in the short term and suppression pool in the long term) to the vessel.

In the event of a core melt sequence that causes failure of the lower vessel head and results in molten fuel reaching the lower drywell cavity floor, the GDCS floods the lower drywell region with water through four separate deluge lines. Logic circuits receiving input signals from an array of temperature sensors in the lower drywell actuate squib valves to initiate the water flow. Actuation occurs when the lower drywell basemat temperature exceeds 537 degrees Celsius (1000 degrees Fahrenheit). Once the squib valves are actuated, the GDCS deluge lines provide a flow path from the GDCS pool to lower drywell cavity.

Squib Valve

The ECCS uses squib-actuated valves for injection to the RPV. Specifically, the function of the squib-actuated valve is to open upon receiving a signal and to remain in its full open position without any continuing external power source and thereby to admit reactor coolant makeup into the RPV in the event of a LOCA. The valves also function in the closed position to prevent RPV backflow and to maintain the RCPB during normal plant operation. The valves are horizontally mounted straight-through, pyrotechnic-actuated, and non-reclosing, with metal diaphragm seals and flanged ends. The valve diaphragms form part of the reactor pressure boundary. The valves actuate when either of the two squib initiators ignite, causing the valves to open. The squib valves can be refurbished once fired. Squib-actuated valves are also used in the equalizing lines and the deluge lines. To minimize the potential for common mode failure, different batches of pyrotechnic charges are used for the equalizing valves and the GDCS injection valves, and a different booster material is used for the deluge line squib valves.

Automatic Depressurization System

The ADS is a part of the ECCS and operates to depressurize the reactor so that the low-pressure GDCS can inject makeup coolant to the reactor. The ADS is composed of 10 SRVs and 8 squib-actuated DPVs and their associated instrumentation and controls. The SRVs are mounted on top of the MSLs in the drywell and discharge through lines routed to quenchers in the suppression pool. Section 5.2.2 of this report describes the SRVs.

The DPVs are straight-through, squib-actuated, non-reclosing valves. The valve size provides about twice the depressurization capacity of the SRV. The DPVs are designed so that there is low leakage throughout the life of the valve. Two initiators (squibs), singly or jointly, actuate a booster, which actuates the shearing plunger. Either one or both of two battery-powered, independent firing circuits initiate the squibs. The firing of one initiator booster is adequate to activate the plunger. All 8 DPVs are horizontally mounted on horizontal stub tubes connected to the RPV at about the elevation of the MSLs. The DPVs discharge into the drywell airspace.

Isolation Condenser System

The ICS provides additional liquid inventory upon the opening of the condensate return valves to initiate the system. The ICS also provides initial depressurization of the reactor before ADS in the event of a loss of feedwater. (See Section 5.4.6 of this report for a detailed description of the ICS.)

Standby Liquid Control System

The SLCS provides additional liquid inventory in the event of DPV actuation. The firing of squib-actuated injection valves initiates the SLCS to accomplish this function. (See Section 9.3.5 of this report for a detailed description of the SLCS.)

Strainers

See Section 6.2.1.7 of this report for a description of the strainers.

6.3.1.3 Staff Evaluation

Emergency Core Cooling Systems

The ECCS is designed to provide coolant inventory to the reactor coolant system in the event of a LOCA. The ECCS is designed with sufficient capacity to make up for loss of coolant from a large spectrum of pipe breaks, up to and including a double-ended rupture of the largest pipe carrying water or steam connected to the RCPB, as well as spurious SRV operation. The passive ECCS is a safety-related system designed to perform the emergency core cooling function. The ECCS consists of the GDCS, the ADS, the ICS, and the SLCS.

The ECCS is passive and its subsystems or components require only a one-time alignment of valves upon actuation. Once the initial actuation alignment is made, they rely solely on natural forces, such as gravity and stored energy, to operate. Once opened, the injection valves remain open and can not be closed or overridden by operators. The use of active equipment or supporting systems, such as pumps, alternating current (ac) power sources, component cooling water, or service water, is not required for the first 72 hours following an accident.

Unlike current operating BWR/2-6 designs, the ICS and SLCS in the ESBWR design are part of the ECCS. The ICS and SLCS provide additional liquid inventory that is credited in the LOCA analysis for ESBWR. The GDCS, ADS, and SLCS are initiated on low RPV Level 1 with a timer delay. The ICS injection is initiated on RPV Level 2 with a timer delay or RPV Level 1 with no timer delay.

Gravity-Driven Cooling System

The GDCS is an ESF system. It is classified as safety related and seismic Category I. The GDCS instrumentation and associated dc power supply are Institute of Electrical and Electronics Engineers (IEEE) Class 1E. The GDCS injection squib valves are opened after a 150-second delay from the ECCS initiation start signal. This time delay allows the reactor to depressurize so that the GDCS can inject into the RPV. In addition, suction from the suppression pool is initiated when the RPV level drops to 0.5 level (1.0 m (3.28 ft)) above the TAF, with a time delay of 30 minutes. In this mode, the GDCS equalizing lines allow coolant from the suppression pool into the RPV to provide long term inventory control.

In RAI 6.3-33, the staff requested additional information regarding the normal and post-accident water level in the GCDS pool. The staff requested this information in order to assess the equilibrium between the reactor decay heat and the condensate flow rate from the PCCS. The applicant provided the information requested, however, this information has not been included in the DCD. RAI 6.3-33 is being tracked as a confirmatory item to ensure that the information is provided in a future revision of the DCD.

A perforated steel plate covers the GDCS pool opening to the drywell airspace to prevent debris from entering the GDCS pool. The holes in the perforated steel plate will be smaller than the orifice holes in the fuel support orifice. In addition, an intake strainer is provided at the suction line from the suppression pool to prevent debris from entering the RPV when the GDCS draws suction from the suppression pool. Section 6.2.1.5.3 of this report provides the staff's evaluation of the strainers.

As noted earlier, the GDCS also provides cooling water to the drywell floor during a hypothetical severe accident. The staff's evaluation of the severe accident mitigation features is provided in Section 19.2 of this report.

All piping in the GDCS is stainless steel and rated for reactor pressure and temperature. The RPV injection line and the equalizing line nozzles all contain integral flow limiters with a venturi shape for pressure recovery. The minimum throat diameters of the nozzles are 7.62 cm (3 in.) and 5.08 cm (2 in.), respectively. GEH states that the nozzle throat length is long enough to ensure that the homogeneous flow model can be used in the LOCA analyses. In RAI 6.3-13, the staff asked GEH to provide additional information on the choked flow model in its LOCA analyses and the nozzle throat lengths for which it is applicable. The staff requested this information in order to address the applicability of the TRACG04 flow choking model to the ESBWR RPV injection line and equalizing line nozzles. R**AI 6.3-13 is being tracked as an open item.**

A squib valve is installed on each GDCS line. The valve is leakproof during normal operation. After opening, the squib valves will remain fully open. This type of squib-actuated valve is smaller than the squib-actuated DPV that has been tested at full size. The test results for the squib-actuated DPV bound the smaller GDCS squib valve and hence is acceptable.

A check valve is installed on each of the GDCS injection lines to the RPV, upstream of the squib-actuated injection valves. The check valve prevent backflow from the RPV to the GDCS, thereby mitigating the consequences of spurious GDCS squib-actuated valve operation. The check valve is classified as Quality Group A, seismic Category I, and ASME Code, Section III, Class 1. Remote check valve position indication is provided in the main control room. The staff noted that the applicant changed the description of the check valves in DCD Revision 3, Section 6.3.2.7.2. The staff requested additional information about the changes to the check valve in RAI 6.3-78 in order to evaluate the changes for acceptability. RAI 6.3-78 is as follows:

In DCD Tier 2, Revision 3, Section 6.3.2.7.2, the "Biased open check valve" name is changed to "GDCS check valve," and Figure 6.3-3, which was showing the "Biased Open Check Valve," is deleted. The Description of Change provided with DCD Revision 3 states: "Revised description of GDC check valve." It does not explain why the valve design was changed, although it seems there was a significant change in the design of the check valve. The check valve will be normally open instead of biased-open. Please address the following:

A. Describe the design differences between the old and the new design.

B. Add the typical check valve figure in the DCD as before.

C. Confirm that the check valves used for injection and equalization are of different types.

D. Provide additional information demonstrating that the core remains covered considering failure of GDCS check valves as the single active failure for design basis LOCA events. Provide this information for the cases where reactor vessel pressure is higher than that of the GDCS and the check valve fails to close.

RAI 6.3-78 is being tracked as an open item.

Automatic Depressurization System

The ADS is part of the ECCS and it depressurizes the reactor so the low-pressure GDCS can provide makeup coolant to the reactor. Depressurization is achieved through sequenced operation of 10 SRVs and 8 DPVs. Initially, five SRVs open upon an ECCS signal to start reducing RPV pressure, followed by five more SRVs after a time delay of 10 seconds. The sequence continues with groups of DPVs opening after successive time delays. Group I (three DPVs) opens after a time delay of 50 seconds, Group II (two DPVs) opens after a time delay of 100 seconds, Group III (two DPVs) opens after a time delay of 150 seconds, and Group IV (one DPV) opens after a time delay of 200 seconds.

Using a combination of SRVs and DPVs to accomplish the ADS function provides diversity in the design. The design of the DPVs reduces components and maintenance, as compared to SRVs. The use of DPVs also reduces the number of SRVs and the need for SRV maintenance and periodic calibration and testing. In Addition, since DPVs discharge into the drywell atmosphere, the use of DPVs reduces the number of SRV discharge lines and quenchers in the suppression pool. The SRVs and DPVs and associated controls and actuation circuits are located or protected so that the consequential effects of an accident cannot impair their function. The ADS is designed to withstand the effects of flooding, pipe whip, and jet impingement. ADS components are also qualified to withstand the harsh environment postulated for DBAs inside containment, including temperature, pressure, and radiation. Section 3.11 of this report provides further details regarding environmental qualification. The SRVs and DPVs are designed with flange connections to allow easy removal for maintenance, testing, or rebuilding. In addition, they are designed such that routine maintenance and inspection can take place at their installed locations. The squib valve is classified as Quality Group A, seismic Category I, and ASME Code, Section III, Class 1.

GEH successfully conducted full-size testing of the DPV to demonstrate its operation.

Each of the 10 ADS SRVs is equipped with a seismically qualified pneumatic accumulator and check valve. These accumulators assure that the valves can be opened following failure of the gas supplying the accumulators. The accumulator capacity is sufficient for one actuation at drywell design pressure. The accumulator capacity is enough to operate the valve at least twice at accident conditions. The containment design pressure is approximately 45 psig. At the beginning of the accident, the containment pressure is much lower then design pressure, and hence the valve can function twice. The DPVs are squib-actuated and are not dependent on accumulators. Thus the applicant has met TMI Action Plan Item II.K.3.28 of NUREG-0737, which is equivalent to 10 CFR 50.34(f)(1)(x), with respect to BWR ADS-associated equipment and instrumentation being capable of performing their intended functions during and following an accident, while taking no credit for non-safety-related equipment or instrumentation and accounting for normal expected air (or nitrogen) leakage through valves.

The SRVs and DPVs are sized such that vessel depressurization and cooldown are slow enough that the system does not exceed vessel integrity limits. GEH performed a thermal analysis that considered the effect of blowdown. Because of the unique design of the ESBWR, depressurization is expected to be slower than in the current BWR operating reactors. The RPV and the containment are designed to withstand structural integrity under the ADS event. Thus the applicant has met TMI Action Plan Item II.K.3.45 of NUREG-0737, which is equivalent to 10 CFR 50.34(f)(1)(xi), with regard to providing an evaluation of depressurization methods, other than full actuation of the ADS, that would reduce the possibility of exceeding vessel integrity limits during rapid cooldown for BWRs.

Isolation Condenser System

The ICS has four passive high-pressure loops, each containing a heat exchanger that condenses steam on the tube side. The steamline connected to the vessel is normally open, and the condensate return line is normally closed. During a LOCA, the condensate return valves open to initiate the ICS operation. The water volume in the condensate return line is credited in the LOCA analysis. Section 5.4.6 of this report provides the staff’s evaluation of the ICS.

Standby Liquid Control System

The SLCS also provides the reactor with additional liquid inventory during a LOCA. The SLCS accomplishes this function by firing squib valves to inject boron solution from the two accumulator tanks pressurized by nitrogen. Section 9.3.5 of this report provides the staff’s evaluation of the SLCS.

Qualification of Emergency Core Cooling System

The ECCS is designed to meet seismic Category I requirements, in accordance with RG 1.29, Revision 3. The ECCS will be housed in structures designed to withstand seismic events, tornadoes, floods, and other phenomena, in accordance with the requirements of GDC 2. The ECCS equipment design complies with the guidance of RG 1.26, Revision 3, regarding the quality group classifications and standards for water, steam, and radioactive waste containing components. The ECCS is protected against pipe whip and discharging fluids, in compliance with the requirements of GDC 4. In addition, the ECCS equipment also meets the environmental qualification requirements of GDC 4 regarding operation under normal and accident conditions. These aspects of the ECCS design are discussed in Chapter 3 of this report.

The ESBWR is proposed as a single unit design and therefore GDC 5, which concern sharing of structures, systems, and components among units is not applicable to the ESBWR design.

The ESBWR core remains covered during all anticipated operational occurrences and accident conditions. Therefore, the ESBWR ECCS meets the requirements of GDC 17, as it relates to the design of the ECCS having sufficient capacity and capability to ensure core cooling.

Sections 4.2, 4.6, and 9.3.5 of this report discuss GDC 27, as it relates to the reactivity control systems having a combined capability, in conjunction with poison addition by the ECCS, of reliably controlling reactivity changes under postulated accident conditions, with appropriate margin for stuck rods.

The GDCS, ICS, and SLCS provide abundant emergency core cooling, thus satisfying the requirements of GDC 35. All the systems within the ECCS are designed to permit appropriate periodic inspection of important components, such as heat exchanger, valves, water injection nozzles, and piping, to ensure the integrity and capability of the systems; thus, GDC 36 is satisfied. Section 3.9.6 of this report discusses compliance with the inservice inspection requirements of GDC 36. The design of the systems within the ECCS permit appropriate periodic pressure and functional testing, thus satisfying GDC 37.

Section 7.3 of this report evaluates the ECCS instrumentation and controls. Section 6.2 of this report discusses the periodic testing and leak-rate criteria for those valves that will isolate the reactor system from the ECCS. Section 5.2.5 of this report discusses the detection of leaks from those portions of the ECCS within the primary coolant pressure boundary.

In response to RAI 5.4-43, GEH stated that even though the ICS uses differential pressure transmitters to detect a possible pipe break, the ICS does not use steam-driven pumps. Thus, TMI Action Plan Item II.K.3.15, "Modify Break Detection Logic to Prevent Spurious Isolation of HPCI and RCIC Systems," is not applicable to the ESBWR.

For transient and accident events that do not directly produce a high drywell pressure signal (e.g., stuck open relief valve or steamline break outside containment), and that are further complicated by the loss of all high-pressure systems, manual activation of the ADS was originally required to provide adequate core cooling. However, TMI Action Plan Item II.K.3.18 required all BWRs to modify their ADS actuation logic to eliminate the need for manual actuation to assure adequate core cooling. RAI 6.3-10 requests additional information on the ADS control logic used to model the Level 1 setpoint in TRACG. **RAI 6.3-10 is being tracked as an open item.**

Preoperational Tests

Preoperational tests will ensure proper functioning of controls, instrumentation, pumps, piping, and valves. The applicant will measure pressure differentials and flow rates for later use in determining acceptable performance in periodic tests. The applicant commits to conform to the guidelines of RG 1.68, "Initial Test Programs for Water-Cooled Nuclear Power Plants," for preoperational and initial startup testing of the ECCS, as noted in Section 14.2 of this report.

Safe Shutdown

Establishing a safe-shutdown condition requires maintenance of the reactor in a subcritical condition and adequate cooling to remove residual heat. One of the functional requirements for the ESBWR is that the plant can be brought to a stable condition using the safety-grade systems for all events. The Commission, in a staff requirements memorandum dated June 30, 1994, approved the position proposed in SECY-94-084. This position accepts temperatures of 215.6 C (420 F) or below, rather than the cold shutdown (less than 93.3 C (200 F)) specified in SRP Branch Technical Position (BTP) RSB 5-1, as the safe, stable condition that the passive decay heat removal system must be capable of achieving and maintaining following non-LOCA events. The SLCS establishes safe shutdown by providing the necessary reactivity control to maintain the core in a subcritical condition and by providing residual heat removal capability to maintain adequate core cooling. DCD Tier 2, Section 7.4, discusses the systems required for safe shutdown.

For all events, the ECCS will use the following systems to keep the reactor in a stable condition:
- ICs
- SLCS
- SRVs
- DPVs
- GDCS
- PCCS

The passive ICS automatically initiates upon closure of the MSIVs to remove decay heat following scram and isolation, and ICS condensate flow provides initial reactor coolant inventory makeup to the RPV. When the water reaches Level 1 in the reactor, the ADS, with operation of SRVs and DPVs, initiates to depressurize the RPV.

Post-72-Hour Actions

The ESBWR passive decay heat removal systems are capable of achieving and maintaining safe, stable conditions for at least 72 hours without operator action following LOCA events. The IC/PCCS expansion pools have an installed capacity that provides at least 72 hours of reactor decay heat removal capability. Replenishing the IC/PCCS expansion pool inventory allows the heat rejection process to continue indefinitely. A safety-related independent FAPCS makeup line provides makeup water into the IC/PCCS expansion pool. A dedicated diesel-driven makeup pump system is connected to the FAPCS. This connection enables the site FPS to fill the upper IC/PCCS pools. This is acceptable because it complies with the guidelines in SECY-94-084.

Mechanical and Electrical Separation

The staff reviewed the ECCS design to confirm that the systems' mechanical and electrical separation criteria are satisfied. Although a common tie exists between the ICS and DPVs on the stub line from the reactor vessel, there is no safety impact resulting from the cross-tie between the ICS and the DPVs. The GDCS Divisions B and C injection lines both connect to a common GDCS pool . This exception is acceptable since there is sufficient redundancy in the GDCS system. In response to RAI 6.3-12 supplement 1, GEH provided a draft paragraph for inclusion in the DCD to clarify the mechanical seperation provided in the design. The response was acceptable. RAI 6.3-12 is being tracked as a confirmatory item.

System Reliability

The ESBWR ECCS is designed to satisfy a variety of requirements to ensure the availability and reliability of its safety functions, including redundancy (e.g., for components, power supplies, actuation signals, and instrumentation), equipment testing to confirm operability, procurement of qualified components, and provisions for periodic maintenance. In addition, the design provides protection against single active and passive component failures; spurious failures; physical damage from fires, flooding, missiles, pipe whip, and accident loads; and environmental conditions, such as high-temperature and containment flood-up. All risk-significant ECCS will be included in the DRAP as described in Section 17.4 of this report.

Generic Issues related to the ECCS

Staff evaluation of TMI Action Plan Item III.D.1.1 is included in Section 20 of this report.

Since the ESBWR design does not include core spray and low-pressure coolant injection systems that can restart after the LOCA, TMI Action Plan Item II.K.21 is not applicable.

Inspections, Tests, Analyses and Acceptance Criteria

The ESBWR ITAAC are provided in DCD Tier 1. The staff's evaluation of the ESBWR ITAAC is provided in Section 14.3 of this report. In RAI 6.3-18, the staff requested GEH to provide pool inventory in the ITAAC. The GEH response submitted in MFN 06-241 dated July 28, 2006, was not satisfactory. The staff requested physical elevation inspection of the GDCS pool level. **RAI 6.3-18 is being tracked as an open item.**

6.3.1.4 Conclusion

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

6.3.2 Emergency Core Cooling System Performance Analysis for Loss-of-Coolant Accident

6.3.2.1 Regulatory Criteria

DCD Tier 2, Section 6.3.3.7, “Emergency Core Cooling Systems,” presents the design bases for the ESBWR ECCS and the LOCA ECCS performance analysis. The staff based its review of the ECCS performance for the LOCA on information in DCD Tier 2, Revision 3, responses to RAIs, and topical reports referenced by the applicant. The staff conducted its evaluation in accordance with the requirements of 10 CFR 50.46 and the guidelines provided by SRP Section 15.6.5, “Loss-of-Coolant Accidents Resulting from Spectrum of Postulated Piping Breaks Within the Reactor Coolant Pressure Boundary.”

6.3.2.2 Summary of Technical Information

6.3.2.2.1 Evaluation Model

GEH used the TRACG code, as accepted in NEDC-33O83P-A “TRACG Application for ESBWR” with Open Items, to evaluate the ESBWR system response during a LOCA. Section 21.6 of this report summarizes the staff’s evaluation of the TRACG code as applied to the ESBWR.

6.3.2.2.2 Uncertainty Analysis

In MFN 05-096, GEH stated that since there is no core heatup, an uncertainty analysis on the peak cladding temperature would not provide useful results. GEH stated that a bounding evaluation for the minimum water level in the chimney during a LOCA event demonstrates that there is margin to core uncovery and heatup.

6.3.2.2.3 Failure Mode Analysis

As discussed in Section 6.3.1.2 of this report, in case of a LOCA, the GDCS, in conjunction with the ADS, the ICS, and the SLCS, provides the emergency core cooling. In DCD Tier 2, Revision 3, and in response to RAI 6.3-46, GEH analyzed eight LOCA events using the failure of one GDCS valve, one SRV, or one DPV. DCD, Tier 2, Table 6.3-1, shown below, shows the most limiting combinations.

Table 6.3-1 Single-Failure Evaluation

Assumed Failure	Systems Remaining
1 DPV	10 SRVs, 7 DPVs, 3 ICS, 2 SLCS accumulators, and 4 GDCS with 8 injection lines
1 SRV	9 SRVs, 8 DPVs, 3 ICS, 2 SLCS accumulators, and 4 GDCS with 8 injection lines
1 GDCS injection valve	10 SRVs, 8 DPVs, 4 ICS, 2 SLCS accumulators, and 4 GDCS with 7 injection lines

6.3.2.2.4 Loss of Offsite Power

GEH analyzed the LOCA events with a loss of offsite power occurring at the same time as the initiation of the break.

6.3.2.2.5 Break Spectrum

Table 6.3-2 of this report shows all of the connections to the ESBWR RPV.

Table 6.3-2 ESBWR RPV Penetrations

Piping Connection	**Number of Lines**	**Elevation** (relative to bottom of the vessel)	**Break Area**	**Notes**
Main Steamlines	4	22.84 m (74.93 ft)	0.09832 m^2 (1.058 ft^2)	limited by venturi throat area
DPV/IC (DPV Stub Tube)	4	21.91 m (71.88 ft)	0.08320 m^2 (0.8956 ft^2)	limited by venturi throat area (16-in. Schedule 160 pipe)
Feedwater Nozzle	6	18.915 m (62.06 ft)	0.07420 m^2 (0.7986 ft^2)	limited by feedwater nozzle area
RWCU/SDC Suction Line	2	17.215 m (56.48 ft)	0.06558 m^2 (0.7059 ft^2)	12-in. Schedule 80 pipe
IC Return Line	4	13.025 m (42.73 ft)	0.01824 m^2 (0.1963 ft^2)	limited by venturi throat area (6-in. diameter)
GDCS Injection Lines	8	10.453 m (32.29 ft)	0.004561 m^2 (0.04910 ft^2)	limited by venturi throat area (3-in. diameter)
SLCS Injection Line	2	9.709 m (31.85 ft)	0.000453 m^2	limited by the nozzle area at the shroud penetration
GDCS Equalizing Line	4	8.453 m (27.73 ft)	0.002026 m^2 (0.02181 ft^2)	limited by venturi throat area
RWCU/SDC Drainline (bottom head drainline)	4	0.0 m	0.004052 m^2 (0.04361 ft^2)	area of 2 nozzles (2 in. diameter)

The values in the above table are from Table 6.3-47-1 in the applicant's response to RAI 6.3-47. In a supplement request to RAI 6.3-47, staff requested the information in the above table to be incorporated into the DCD. **RAI 6.3-47 is being tracked as an open item.**

Two bottom drainlines join a common header. In response to RAI 6.3-58, GEH stated that the penetration of the bottom drainline to the vessel is 50.8 mm. Although the break area for the common header is larger than that of the two drainline nozzles, the flow is choked at the vessel penetrations, and GEH therefore assumes the area of the break to be the size of two of the nozzles.

GEH selected a representative set of cases to evaluate the spectrum of postulated break locations to demonstrate the ECCS performance. Specifically, GEH analyzed the following break locations, each with various single failures:

- steamline inside containment
- FWL
- GDCS injection line
- bottom head drainline

The largest possible line breaks for the ESBWR are the DPV stub tube break, MSLB, FWLB, and RWCU/shutdown cooling (SDC) suction line break. The DPV stub tube break will also include backflow through the IC return line; similarly, the total RWCU/SDC suction line breakflow includes flow through the bottom head drainline. GEH analyzed the maximum inside steamline break and the maximum FWLB as representative cases for these four break locations. After an IC return line break, the ESBWR will rapidly depressurize through the ADS valves. Therefore, the results for this case are similar to the large steamline break case. For small line breaks, GEH analyzed the GDCS injection line break and the bottom head drainline breaks.

In response to RAI 6.3-46, GEH submitted the minimum water level results for the following additional break locations:

- GDCS equalizing line
- DPV stub tube (DPV/IC steamline)
- RWCU/SDC return line
- IC return line

GEH stated that the limiting cases are the MSL and gravity drainline (GDL) breaks. The applicant’s results do not show heatup or core uncovery for any of the analyzed LOCAs. Since the acceptance criteria for 10 CFR 50.46 are not challenged for this event, GEH uses minimum static head in the chimney as a metric to determine the most limiting break. Table 6.3-3 of this report shows the various break scenarios analyzed by GEH and the minimum static head in the chimney during each event.

Table 6.3-3 Nominal ESBWR LOCA Calculations

Break Location	Break Size m^2 (ft^2)	Single Failure	Minimum Chimney Static Head m (ft)
Steamline Inside Containment	0.09832 (1.058)	1 SRV	9.00 (29.53)
Steamline Inside Containment	0.09832 (1.058)	1 GDCS Valve	9.15 (30.02)
Steamline Inside Containment	0.09832 (1.058)	1 DPV	8.90 (29.20)
Feedwater Line	0.07420 (0.7986)	1 SRV	9.48 (31.10)
Feedwater Line	0.07420 (0.7986)	1 GDCS Valve	9.48 (31.10)
Feedwater Line	0.07420 (0.7986)	1 DPV	9.34 (30.64)
GDCS Injection Line	0.004561 (0.04910)	1 SRV	9.24 (30.31)
GDCS Injection Line	0.004561 (0.04910)	1 GDCS Valve	9.08 (29.77)
GDCS Injection Line	0.004561 (0.04910)	1 DPV	9.34 (30.64)
Bottom Head Drainline	0.004053 (0.04361)	1 SRV	9.45 (31.00)
Bottom Head Drainline	0.004053 (0.04361)	1 GDCS Valve	9.30 (30.52)
Bottom Head Drainline	0.004053 (0.04361)	1 DPV	9.46 (31.02)

The values in the above table are from DCD Tier 2, Revision 3, Table 6.3-5.

GEH used nominal plant calculations to obtain the minimum chimney static head measurements set forth in Table 6.3-3 of this report. GEH did not perform an uncertainty analysis on the minimum chimney static head. Instead, it performed a bounding calculation on the two most limiting break locations—the steamline inside containment break and the GDCS injection line break. The staff previously reviewed and approved the bounding assumptions, as documented in Section 2.7.2.1 of NEDC-33083P-A. Table 6.3-4 below presents the results of the applicant's calculations.

Table 6.3-4 Bounding ESBWR LOCA Calculations

Break Location	Break Size m^2 (ft^2)	Single Failure	Minimum Chimney Static Head m (ft)
Steamline Inside Containment	0.09832 (1.058)	1 SRV	9.49 (31.14)
Steamline Inside Containment	0.09832 (1.058)	1 GDCS Valve	9.37 (30.74)
Steamline Inside Containment	0.09832 (1.058)	1 DPV	9.31 (30.54)
GDCS Injection Line	0.004561 (0.04910)	1 SRV	8.81 (28.89)
GDCS Injection Line	0.004561 (0.04910)	1 GDCS Valve	8.61 (28.25)
GDCS Injection Line	0.004561 (0.04910)	1 DPV	8.94 (29.34)

The values in the above table are from DCD Tier 2, Revision 3, Table 6.3-5.

6.3.2.2.6 Evaluation Model Parameters and Assumptions

GEH chose the evaluation model parameters and assumptions discussed in the following sections.

<u>Initial Power Level</u>. DCD Tier 2, Table 6.3-11 states that GEH is using a core power of rated +2 percent for its bounding LOCA analysis.

<u>Maximum Linear Heat Generation Rate</u>. DCD Tier 2, Table 6.3-11 states that GEH is using a peak linear heat generation rate (LHGR) of 44.8 kilowatts per meter (kW/m) (13.7 kW/foot (ft)) for its bounding LOCA analysis.

<u>Axial Power Shapes</u>. The applicant's TRACG model uses 35 axial nodes, with 32 representing the heated section of the channel. In response to RAI 408 in NEDC-33083P-A , GEH stated that it is using a bottom peaked axial power shape. GEH does not perform the analysis with other power shapes.

<u>Initial Stored Energy</u>. GEH assumes constant gap conductance throughout the LOCA event. GEH uses these gap conductances as inputs into the TRACG code and calculates them through the GSTRM fuel mechanical code. Section 4.2 of this report discusses applicability of the GSTRM code to the ESBWR. The applicant's fuel thermal conductivity model is based on that used in the PRIME03 code. Applicability of the PRIME03 code to ESBWR is an Open Item. **RAI 6.3-55 is being tracked as an open item.**

<u>Control Rod Insertion</u>. GEH uses a scram time delay with each LOCA case analyzed. DCD Tier 2, Table 6.3-1 states that the events are analyzed with a 2-second scram delay time. DCD Tier 2, Table 6.3-11 states that GEH is using the 1994 ANS decay heat standard.

Boric Acid Precipitation. Boric acid will be injected into the RPV bypass during a LOCA event as part of the SLCS initiation. GEH does not consider boric acid precipitation as part of short-term or long-term core cooling.

Containment Pressure Response. Section 6.2 of this report discusses containment pressure response.

ECCS Strainer Performance Evaluation. Section 6.2.1.5.3 of this report discusses ECCS strainer performance evaluation.

6.3.2.2.7 Reactor Protection and ECCS Actions

The following sections give a narrative description of the sequence of events for the break locations presented in DCD Tier 2, Revision 3, and the ESBWR ECCS and reactor protection system (RPS) response.

Gravity-Driven Cooling System Line Break. In DCD Tier 2, Revision 3, the GDCS line break with the failure of one injection valve is the most limiting. GEH showed the results of the TRACG analysis of this break in DCD Tier 2, Revision 3, Figures 6.3-31a to 6.3-38b . The operational sequence of the RPS and ECCS actions is in DCD Tier 2, Revision 3, Table 6.3-9.

DCD Tier 2, Figures 6.3-32a and 6.3-32b show the static head in the chimney and two-phase level in the chimney. During the first 30 seconds after the break, water is pushed into the vessel from the downcomer, causing the collapsed chimney level to increase. DCD Tier 2, Figure 6.3-33b shows the decrease in downcomer level.

The system reaches the Level 2 setpoint approximately 15 seconds after the break. The MSIV will either close after the 30-second delay or on a low MSL pressure signal plus delay time. For the GDCS line break, the system first reaches the low MSL pressure setpoint at around 17 seconds, and the MSIVs close about 1 second later. DCD Tier 2, Figure 6.3-35b shows that, at this point, flow in the steamline goes to zero, and DCD Tier 2, Figure 6.3-34b shows that the RPV pressure decrease slows.

DCD Tier 2, Figure 6.3-35b shows that the break flow decreases at about 8 seconds. This is a result of the swell in the downcomer and the break flow reaching saturation temperature, when it begins voiding. GEH showed a plot of the void fraction of the break flow in Figure 6.3-69-4a of the applicant's response to RAI 6.3-69. GEH stated in its response to RAI 6.3-69 that it would add these figures to the DCD. RAI 6.3-69 is resolved and is being tracked as a confirmatory item. This plot shows that the void fraction increases until about 20 seconds when the MSIVs close. At this time, the voids begin to collapse. The break flow void fraction is reduced to zero at about 30 seconds. At this time, the downcomer two-phase level begins to nearly equal that of the collapsed level in the downcomer.

The IC drain valves open on the loss of offsite power. The drain valves open after a 15-second delay. DCD Tier 2, Figure 6.3-37b shows the IC drain flow. The IC drain flow peaks early because of the additional water inventory in the IC drain tanks. There is a high flow rate until about 55 seconds, which is the time it takes for the IC drain tanks to empty. IC flow then decreases and begins to show oscillating behavior. This flow is from the IC steam condensation. In response to RAI 6.3-74, GEH explained that 180 seconds into the transient there is a drop in the IC drainline water level, which causes an abrupt increase in the IC steamflow rate to fill the voided volume.

The system reaches the Level 1 setpoint approximately 170 seconds after the break. Level 1 must persist for 10 seconds to be confirmed. ADS is confirmed at approximately 180 seconds after the break, and SRV actuation begins. DCD Tier 2, Figure 6.3-35a shows the increase in steamflow at this time resulting from the opening of the ADS valves. DCD Tier 2, Figure 6.3-36a shows the steamflow contribution separately from the SRVs and DPVs and also the IC. The downcomer flashes when ADS actuation begins. DCD Tier 2, Figure 6.3-33a shows this as an increase in the two-phase level. DCD Tier 2, Figure 6.3-32a shows that the collapsed chimney level decrease at this time. The collapsed level oscillates with each SRV and DPV actuation then steadily decreases to its minimum at around 500 seconds into the transient.

The SLCS timer begins when the system reaches the Level 1 setpoint. The SLCS timer times out in 50 seconds, at the same time as DPV actuation, and the SLCS actuates at about 228 seconds, as shown in DCD Tier 2, Figure 6.3-37a.

The GDCS timer is also initiated with the Level 1 setpoint. The GDCS timer times out in 150 seconds, and the GDCS injection valves open at 328 seconds after the initiation of the break. DCD Tier 2, Figure 6.3-35a shows the GDCS pool in the broken line starting to empty into the drywell at this time. However, the RPV pressure is still too high for the other GDCS trains to inject into the RPV.

DCD Tier 2, Figure 6.3-37a shows the GDCS beginning to inject at about 450 seconds into the event when the RPV pressure decreases to that of the GDCS. There is a spike in flow at the onset of GDCS initiation. This shows the steam from the RPV colliding with subcooled GDCS flow. The collapsed level in the chimney begins to rise from its minimum value after the GDCS injection begins.

The collapsed chimney level continues to recover as a result of the GDCS injection. At about 1100 seconds into the transient, the level starts to experience large oscillations. GEH states in NEDC-33083P-A that these are manometric type of oscillations. The collapsed and two-phase chimney levels continue to oscillate. However, on average, the level continues to recover until it reaches the top of the chimney partitions.

The staff also requested that GEH provide reactor power as a function of time in RAI 6.3-68 and state if the cases presented in the DCD are run with nominal or bounding conditions in RAI 6.3-70. **RAI 6.3-68 and 6.3-70 are being tracked as open items.**

Main Steamline Break. DCD Tier 2, Revision 3, Figures 6.3-15a through 6.3-22b describe the system response of an MSLB inside containment with one GDCS valve failure. DCD Tier 2, Revision 3, Table 6.3-8, sets forth the operational sequence of the RPS and ECCS actions. The RPS and ECCS response is similar to the GDCS line break discussed above. One of the main differences between the responses of the MSLB and the GDCS line break is that the MSLB depressurizes much faster than the GDCS line break because of the larger break size. In addition, the Level 1 setpoint actuates later because of the level swell.

Feedwater Line Break. DCD Tier 2, Revision 3, Figures 6.3-7a through 6.3-14b describe the systems response of an FWLB with one GDCS valve failure. DCD Tier 2, Revision 3, Table 6.3-7, shows the operational sequence of the RPS and ECCS actions. The RPS and ECCS response is similar to the GDCS line break discussed above. One of the main differences between the FWLB and the GDCS line break is that the FWLB depressurizes faster than the GDCS line break because of the larger break area. Also, the higher elevation causes the FWLB to respond more like the steamline break once the two-phase level drops below the elevation of the feedwater sparger. Similar to the MSLB, the Level 1 setpoint actuates later than the GDCS line break because of the level swell, but sooner than the MSLB.

Bottom Drainline Break. DCD Tier 2, Revision 3, Figures 6.3-23a through 6.3-30b describe the systems response of a bottom drainline break with one GDCS valve failure. DCD Tier 2, Revision 3, Table 6.3-10 , shows the operational sequence of the RPS and ECCS actions. The RPS and ECCS response is similar to the GDCS line break discussed above. The bottom drainline break has a lower elevation and a smaller break area than the GDCS line break. Hence, the vessel depressurizes slower and the Level 1 setpoint actuates later than in the GDCS line break.

6.3.2.2.8 Long-Term Core Cooling

In MFN 05-105, GEH submitted details on long-term core cooling. This letter included a discussion on long-term inventory distribution for four break locations—(1) MSLB, (2) FWLB, (3) bottom drainline break, and (4) GDCS line break. GEH based these analyses on Revision 0 of the ESBWR DCD and will update them when it submits responses to RAIs 6.3-64 and 21.6-98. In response to RAI 6.3-64, GEH submitted a long-term core cooling analysis for the GDCS line break with one DPV failure. The staff asked GEH for additional information on this analysis. **RAI 6.3-64 and RAI 21.6-98 are being tracked as open items.**

Long-Term Core Cooling for Main Steamline Break. In the long-term MSLB, the GDCS will drain to the level of the break (i.e., the DPVs), which will leave about two-thirds of the GDCS inventory in the pools. The PCCS will condense the steam generated by decay heat and return it to the vessel via the GDCS. Some steam will condense on the drywell surfaces and not return to the RPV, leaving a small amount in the lower drywell.

Long-Term Core Cooling for Feedwater Line Break. The long-term core cooling for the FWLB is similar to the MSLB because it is a higher elevation break. The GDCS pools will drain down to the level of the FWL sparger, which is close to the bottom of the elevation of the GDCS pools. There is a period of time when the GDCS pools are drained and the PCCS does not condense the steam at the same rate as the decay heat power generated by the core and so the level in the downcomer decreases at a faster rate. DCD Tier 2, Revision 3, Figure 6.2-13c1 , shows that PCCS power begins to equal that of decay heat at 6 hours into the event. This is different from the time shown in DCD Revision 2, where PCCS power becomes equal to that of decay heat at about 8 hours into the event. Since the DCD Revision 3 plots no longer apply to the long-term core cooling analyses that GEH submitted MFN 05-105, the staff will wait until GEH submits its responses to RAIs 6.3-64 and 21.6-98 to address concerns related to long-term core cooling. This is an open item in Section 6.3.2.2.8 of this report. In MFN 05-105, Figure 4, GEH shows the levels at 12 hours above the top of the chimney partitions. Some steam will condense on the drywell surfaces and not return to the RPV, leaving a small amount in the lower drywell. The level in the drywell gets high enough to return to the suppression pool via the spillover holes in the vertical vent pipes. However, the drywell level remains well below the RPV break location in the FWL sparger.

Long-Term Core Cooling for Bottom Drainline Break. Since this break is on the bottom of the vessel, the inventory in the lower drywell becomes important. The GDCS pool empties approximately 5.5 hours into the event, and the level in the downcomer begins to decrease at a faster rate. The drywell fills up to the elevation of the spillover hole at about 5 hours. The level in the downcomer and the RPV goes below the top of the chimney partitions about 6.5 hours into the event and continues to drop until the level reaches that of the spillover hole. The elevation of the spillover hole is 17.5 m (57.4 ft) above the bottom of the reactor vessel (DCD Tier 2, Figures 3G.1-57 and 3G.1-58), which is approximately 10 m (32.8 ft) above the TAF. At about 8 hours into the event, the levels in the drywell, RPV, and downcomer remain nearly constant at the spillover hole level. The PCCS maintains the levels by condensing the steam from decay heat and returning it to the vessel via the GDCS.

Long-Term Core Cooling for Gravity-Driven Line Break. The long-term behavior of this break is similar to that of the bottom drainline break described above in that once the GDCS pool drains and the levels in the downcomer and the RPV start to fall, they will level out at the spillover hole elevation because the GDCS injection line is below that of the spillover hole. Since the GDCS line is broken, more inventory enters the drywell earlier in the event, and the level in the drywell reaches that of the spillover holes at about 3 hours into the event. Also, since the GDCS pools lose inventory faster because of the broken line, the GDCS pools empty at about 4 hours into the event.

6.3.2.3 Staff Evaluation

6.3.2.3.1 Evaluation Model

The staff reviewed and approved the GEH evaluation model (TRACG) for the 4000 MWt preapplication ESBWR design, NEDC-33083-A. Section 21.6 of this report provides an evaluation of its applicability to the current ESBWR 4500 MWt design. Due to the open items that remain to be resolved for this section the staff was unable to finalize its conclusions regarding acceptability for the use of this evaluation model. The ESBWR LOCA analyses shows that the core does not heatup or uncover. Therefore, the staff did not review or approve the use of TRACG for core heatup or uncovery. The staff's acceptance of the ECCS performance for the ESBWR is based on it maintaining a static head of water above the TAF.

6.3.2.3.2 Uncertainty Analysis

Section 50.46(a)(1)(i) of 10 CFR states in part that "comparisons to applicable experimental data must be made and uncertainties in the analysis method and inputs must be identified and assessed so that the uncertainty in the calculated results can be estimated. This uncertainty must be accounted for." Section 50.46(a)(1)(ii) of 10 CFR states, "Alternately, an ECCS evaluation model may be developed in conformance with the required and acceptable features of appendix K ECCS Evaluation Models." GEH has not done either of these options and, therefore, does not comply with the requirements of 10 CFR 50.46. The staff issued RAI 6.3-81 requesting that GEH demonstrate how the LOCA analyses comply with this requirement. **RAI 6.3-81 is being tracked as an open item.**

6.3.2.3.3 Failure Mode Analysis

GDC 35 requires that the ECCS be able to accomplish its function in the event of a single failure. In DCD Tier 2, Section 6.3.3, GEH provided an analysis to demonstrate the most limiting break size, break location, and single failure for the ESBWR. The staff reviewed the system description, piping and instrumentation drawings, and the applicant's ECCS performance analysis to ensure that the applicant considered all credible single active failures. The following sections describe the staff's evaluation of the single failures assumed in the analyses for each of the credited ECCS.

Section 6.3 of the SRP states that the long term cooling capacity is adequate in the event of failure of any single active or passive component of the ECCS. The staff expects GEH to confirm that the ESBWR design satisfies this feature. Please state if the ESBWR design takes credit for any passive component during long term post LOCA (i.e., beyond 72 hours) cooling **RAI 6.3-79 is being tracked as an Open Item.**

GDCS Single Failure. The GDCS consists of three pools and eight injection lines. In response to RAI 6.3-43, GEH provided additional information on the modeling of the GDCS. Because there are three GDCS pools and eight injection lines, the choice of valve failure and line break may give different results. The staff confirmed that GEH chose the most conservative combination of valve failure and line break.

The GDCS check valve must be closed upon initiation of the squib valves since the RPV pressure is higher than that of the GDCS. The staff requested that GEH evaluate the possibility of this failure in RAI 6.3-78, because this may result in additional coolant loss. **RAI 6.3-78 is being tracked as an open item.**

GEH performed analyses of all design-basis LOCA events assuming one GDCS squib valve fails to open. DCD Tier 2, Revision 3, Table 6.3-5, provides the results of the analyses, which show that the core remains covered for the GDCS line break, the MSLB, the bottom drainline break, and the FWLB with one GDCS injection line failure. Table 6.3-46-1 in the applicant's response to RAI 6.3-46, shows that the core remains covered for the GDCS equalizing line, DPV stub tube (DPV/IC steamline), RWCU/SDC return line, and the IC return line breaks with one GDCS injection line failure. GEH did not consider the effect of the failure of the GDCS injection valve on the SLCS line break because the break area was too small to initiate ECCS in the 2000-second phase of the event. The staff requested as a supplement to RAI 6.3-65 that GEH discuss the long-term results of the SLCS line break. **RAI 6.3-65 is being tracked as an open item.** See Section 6.3.2.3.5 of this report for additional discussion of this RAI.

ADS Single Failure. The ADS consists of DPVs and SRVs as described in Section 6.3.2 of this report. GEH performed analyses of all design-basis LOCA events assuming failure of either a DPV or an SRV. DCD Tier 2, Revision 3, Table 6.3-5 presents the results of the analyses, which show that the core remains covered for the GDCS line break, the MSLB, the bottom drainline break, and the FWLB with one SRV or one DPV failure. Table 6.3-46-1 in the applicant's response to RAI 6.3-46, shows that the core remains covered for the GDCS equalizing line, DPV stub tube (DPV/IC steamline), RWCU/SDC return line, and the IC return line breaks with a failure of either one DPV or SRV. GEH did not consider the effect of the failure of the ADS injection valve on the SLCS line break because the break area was too small to initiate ECCS in the 2000-second phase of the event. The staff requested as a supplement to RAI 6.3-65 that GEH discuss the long-term results of the SLCS line break. **RAI 6.3-65 is being tracked as an open item.** See Section 6.3.2.3.5 of this report for additional discussion of this RAI.

SLCS Single Failure. Section 9.3.5 of this report provides the SLCS evaluation. This section shows that no single active failure of the SLCS can prevent either of the SLCS trains from injecting. Therefore, the staff finds that the applicant's assumption that the SLCS does not fail during any LOCA event is acceptable and that the design of the SLCS complies with GDC 35 as it relates to ECCS performance.

One train of the SLCS will fail if there is a break in an SLCS line because of inventory lost out the break. In response to RAI 6.3-65, GEH showed that the collapsed liquid level in the downcomer does not drop to the Level 1 elevation and therefore does not initiate any ECCS during the first 2000 seconds of the event and does not require SLCS injection. The staff requested as a supplement to RAI 6.3-65 that GEH discuss the long-term results of the SLCS line break. In addition, there are several SLCS nozzles penetrating the ESBWR shroud that join to a common pipe header. It is not clear whether GEH included the combined area of all of the nozzles or just one nozzle. **RAI 6.3-65 is being tracked as an open item.** See Section 6.3.2.3.5 of this report for additional discussion of this RAI.

ICS Single Failure. GEH credits the heat removal capability of the ICS and the inventory in the ICS drain tanks during a LOCA event. The condensate return valve for the ICS is single-failure proof. There is a bypass valve that may be actuated in the event that the condensate return valve fails to open. Section 5.4.6.2.2 of this report describes this, and DCD Tier 2, Revision 3, Figure 5.1-3 depicts it. For all design-basis LOCA analyses, GEH always assumed that only three out of the four ICs are available during a LOCA because one may be out of service. DCD Tier 2, Revision 3, Table 6.3-5, presents the results of the analyses, which show that the core remains covered for the GDCS line break, the MSLB, the bottom drainline break, and the FWLB with one inoperable IC. Table 6.3-46-1 in the applicant's response to RAI 6.3-46, shows that the core remains covered for the GDCS equalizing line, DPV stub tube (DPV/IC steamline), and RWCU/SDC return line breaks with one inoperable IC. For the IC return line break, the IC out of service may be a different IC than the one attached to the broken line, so GEH assumed only two ICs available for this event. The results in Table 6.3-46-1 show that the core remains covered for this event. The staff asked GEH to verify how many ICs were operating during the SLCS break evaluation presented in the applicant's response to RAI 6.3-43 Supplement 1. The staff sent this request to GEH as a supplement to RAI 6.3-65. **RAI 6.3-65 is being tracked as an open item.** See Section 6.3.2.3.5 of this report for additional discussion of this RAI.

In RAI 6.3-66, staff requested GEH include a statement that the loss of coolant accident reactor pressure vessel level analyses takes credit for isolation condenser heat removal capacity and the CRD hydraulic control unit injection. In response, GEH stated that it will revise Table 6.3-1B.3 to include the drainline water inventory. Staff finds this approach acceptable. RAI 6.3-66 is being tracked as a confirmatory item to ensure that the information is included in a future revision of the DCD.

Vacuum Breaker Failure. There is a vacuum breaker between the drywell and the suppression pool that opens if the suppression pool pressure exceeds that of the drywell. Failure of this valve to close after opening would render the PCCS incapable of condensing the steam because a pressure differential would not be maintained across the PCCS. This would result in an inability of the system to maintain long-term core cooling. In DCD Tier 2, Revision 3, Section 6.2.1.1.2, GEH added an alternate means to close this opening by adding a block valve that would allow the vacuum breaker to remain operable with a single active failure. The staff requested additional information about the block valve operation, control and its impact on containment and RPV analysis in RAI 6.3-63. **RAI 6.3-63 is being tracked as an open item.**

Bottom Drainline Isolation. The bottom drainline is open during normal operations for RWCU operations. In the event of a LOCA, it is possible that if this line fails to isolate, additional loss of inventory may occur. In response to RAI 6.3-59, GEH confirmed that there are two isolation valves in series; therefore, failure of one valve to close would not result in a failure of the system to isolate. In addition, GEH provided the signals that would isolate the bottom drainline in the event of a LOCA. These signals include the following:

- reactor vessel low water Level 2
- reactor vessel low water Level 1
- MSL tunnel high ambient temperature
- high flow in the RWCU/SDC loop
- SLCS in operation

The staff finds that this system will be isolated upon a LOCA and that there will be no additional inventory lost.

CRD Hydraulic Control Unit. In its analyses, GEH assumes that HCU inventory is injected into the vessel during the scram. GEH does not fail this injection source as part of its LOCA analyses. The volume of water injected into the vessel for one HCU unit is negligible as compared to the other ECCS sources, and its failure will not provide limiting results. In response to RAI 6.3-66, GEH stated that it would include the HCU injection as Table 6.3-1 B.6 in the DCD. This is acceptable. RAI 6.3-66 is being tracked as a confirmatory item to ensure that this information is included in a future revision of the DCD.

Conclusion of Single-Failure Evaluation. Due to the open items that remain to be resolved for this section the staff was unable to finalize its conclusions regarding acceptability.

6.3.2.3.4 Loss of Offsite Power

GDC 35 requires that the ECCS be able to accomplish its function in the event of a loss of offsite power. To demonstrate that the ECCS performance meets the design requirements, GEH assumed a loss of offsite power occurs coincident with the break for each of the design-basis LOCA events analyzed. This causes the reactor to scram and the ICS to initiate upon the loss-of-power signal. If there was not a loss of offsite power at the initiation of the break, there would be a delay in the actuation of these systems, as they would actuate on their own trip setpoints. GEH states that there is a loss of feedwater upon loss of offsite power and assuming a loss of feedwater is more conservative than incorporating the delays. In DCD Tier 2, Revision 3, GEH changed the feedwater isolation to be a safety grade and it is isolated upon a sensed differential pressure between the FWLs coincident with high drywell pressure. The staff agrees with GEH that the loss of offsite power coincident with the break is a conservative assumption because of the feedwater isolation. For the FWLB, the high drywell pressure signal occurs at 1 second into the transient, meaning that the assumption of the loss of power at the break gives virtually the same scram and ICS response. GEH did not evaluate the effects of allowing the reactor and ICS to initiate on their own trip setpoints for a small break LOCA. However, the staff agrees with GEH that the loss of feedwater during this time is a more conservative analysis. Also, since all ESBWR LOCA analyses show large safety margins, the staff does not believe that this possible nonconservatism would have any measurable impact on safety. Therefore, the staff finds the applicant’s loss-of-offsite-power assumption to be appropriate for ESBWR LOCA analyses.

6.3.2.3.5 Break Spectrum

GEH showed the results of a LOCA event at eight different break locations with three single failures. In each of these 24 cases, the core remains covered throughout the entire blowdown phase and through the reflood phase (until 2000 seconds after the break). GEH uses minimum static head in the chimney as a metric to determine the most liming break. The staff finds the labeling of DCD Tier 2, Revision 3, Table 6.3-5, in DCD Revision 3 misleading because GEH labels these values as "minimum chimney static head level above vessel zero," and calculates these values by collapsing the level in the chimney and does not consider the void fraction in the core. The staff requested in RAI 6.3-77 that GEH change this label in the next revision of the DCD or else justify that it is the same (i.e., show that when considering the void fraction in the core, the collapsed level remains the same). In its response to this RAI, GEH stated that it would update the language in the next revision of the DCD. RAI 6.3-77 is being tracked as a confirmatory item.

GEH performed each of the 24 calculations using nominal conditions. GEH stated that the GDL break and the MSLB are the most limiting and performed calculations for these two break locations using bounding conditions. For these two cases, the core still remains covered. Upon review of this information, the staff discovered that the IC return line break appears to be more limiting than the GDCS line break. The staff requested supplemental information to RAI 6.3-46. In the applicant's response to this RAI, GEH stated that the IC return line gives a lower minimum static head in the chimney than the GDL break for the case where there is one GDCS injection valve failure. The staff requested that GEH justify its selection of the GDL break as the limiting case, considering that the IC return line break gives a lower minimum static head in the chimney. **RAI 6.3-46 is being tracked as an open item.**

For the 24 breaks analyzed by GEH using the nominal conditions, the two cases giving the smallest static head in the chimney are the MSLB and the DPV stub tube line break with the failure of one DPV. GEH chose to only run the bounding cases for the MSLB and the GDL break because the DPV stub tube line break is similar to the MSLB. The staff agrees that these two breaks would exhibit similar behavior and that the MSLB is bounding for the DPV stub tube line break.

GEH performed a sensitivity study of the GDL break size in response to RAI 6.3-46. The break sizes for this study ranged from the full DEG break, and then 80, 60, 40, and 20 percent of this size. GEH ran these cases using nominal conditions and failure of one GDCS injection valve. The 80-percent case gave the most limiting results. GEH then ran the cases for the 100-percent and 80-percent break sizes using bounding assumptions. For these two cases, the 100-percent size is still the most limiting break for this location. In supplemental RAI 6.3-46, the staff asked GEH to explain why this is so. In addition, GEH did not evaluate the 60-, 40-, or 20-percent break sizes using bounding assumptions. The staff asked GEH why it did not do so. GEH also stated during a teleconference on this RAI that it would provide a qualitative argument as to why very small breaks (i.e., smaller than 20 percent) are not limiting. The staff requested that GEH provide the additional information to address these questions in the context of RAI 6.3-46. **RAI 6.3-46 is being tracked as an open item.**

GEH did not analyze a break in the SLCS injection line. The staff is concerned about this break since this break would also cause the loss of an SLCS injection train. One train of the SLCS will fail if there is a break in an SLCS line because of inventory lost out the break. In response to RAI 6.3-65, GEH showed that the collapsed liquid level in the downcomer does not drop to the Level 1 elevation and therefore does not initiate any ECCS during the first 2000 seconds of the event. The staff requested additional information from GEH as a supplement to RAI 6.3-65. The staff asked GEH to provide information on the event after 2000 seconds. **RAI 6.3-65 is being tracked as an open item.**

In response to a supplemental request for RAIs 6.3-10 and 6.3-16, GEH submitted the results of a spectrum of break sizes for the MSLB with a failure of one DPV. GEH provided results for break sizes that are 100, 80, 60, 40, 20, and 10 percent of the DEG-sized break. GEH demonstrated that for each of these break sizes, the "minimum chimney static head level above vessel zero" remains above the top of fuel and that the most limiting size break for the MSL is the 100-percent DEG size. The staff needs additional information to the applicant's response to RAI 6.3-10. The staff requested that GEH clarify the language "minimum chimney static head level above vessel zero." In the DCD, GEH calculates this as the static head in the chimney added to the elevation of the top of the core. In the DCD, GEH also uses "minimum chimney static above vessel zero" but does not use the qualifying statement that "DCD Chimney static head is calculated by adding the static head in the chimney to the elevation of bottom of chimney." The staff requested that GEH clarify whether the level calculation accounts for the void fraction of the core. The staff also noticed that although the core remains covered for all the break sizes, there is a decreasing trend from 40 percent and 20 percent down to 10 percent. The staff requested that GEH address the break sizes below 10 percent and provide the maximum break size that does not exceed the makeup system. **RAI 6.3-10 is being tracked as an open item.**

In RAI 6.3-76, the staff requested GEH to explain why the bounding steamline break gives a higher collapsed liquid level in the chimney than the nominal case. In response to this RAI, GEH showed plots comparing the downcomer and collapsed chimney level for the nominal and bounding cases. The collapsed level in the chimney is directly related to the level in the downcomer because of the manometer effect. For the bounding case, the downcomer reaches a lower collapsed level at a later time than the nominal case. This causes the GDCS injection phase to begin later in the bounding case transient. At this time, the core void fraction will be lower and the decay heat reduced from the nominal case. During the injection phase, the collapsed level in the chimney will experience oscillations resulting from the interaction of the core void with incoming subcooled water from the lower plenum. The lower core void fraction and decay heat will reduce the magnitude of the oscillations. For the nominal case, the minimum static head in the chimney occurs during these oscillations, whereas for the bounding it occurs before.

Although the bounding steamline break inside containment gives a higher collapsed liquid level in the chimney than the nominal, the staff finds that the analysis is conservative and still shows that the ESBWR design has a safety margin with respect to this event. The staff agrees with the explanation given by GEH in its response to RAI 6.3-76. The minimum collapsed chimney level for the nominal case happens during the GDCS injection phase when the core is experiencing oscillations in level (Section 6.3.4.2.7.6 of this report provides further discussion on the oscillations). The minimum collapsed chimney level for the bounding case happens just before the injection and before the core oscillations. The staff's review of the plots submitted by GEH in response to RAI 6.3-76. The plots show that for the nominal and bounding conditions, the bounding case is qualitatively a more conservative analysis. Both analyses demonstrate that the ESBWR design has margin to core uncovery for this event, and therefore the staff finds the results of the analyses acceptable.

The staff did not request that GEH provide an analysis of the MSLB outside containment. This event is bounded by the MSLB inside containment.

6.3.2.3.6 Evaluation Model Parameters and Assumptions

The following sections discuss the staff's review of the evaluation model parameters and assumptions to ensure that the applicant chose them conservatively.

Initial Power Level. DCD Tier 2, Revision 3, Table 6.3-11 states that GEH is using a core power of rated +2 percent for its bounding LOCA analysis. This is consistent with the requirements in SRP Section 15.6.5. The staff finds this value acceptable.

Maximum Linear Heat Generation Rate. DCD Tier 2, Revision 3, Table 6.3-11 states that GEH is using a peak LHGR of 13.7 kW/ft for its bounding LOCA analysis. This value is consistent with the limit in NEDO-33242, Revision 1, "GE14 for ESBWR Fuel Rod Thermal-Mechanical Design Report," which gives a thermal-mechanical limit of 13.4 kW/ft. For the LOCA event, GEH has shown that the ESBWR will experience no core uncovery. Because of the high margin to safety limits for this set of events, the staff finds that the maximum linear heat generation rate (MLHGR) for the ESBWR will be limited by the fuel rod thermal-mechanical design or minimum critical power ratio (MCPR), and therefore the staff finds the value used for the LOCA analysis acceptable. The staff understands that TS 3.2.1 sets limits on the LHGR and that the MLHGR will be specified in the core operating limits report. In accordance with the requirements of 10 CFR 50.46, GEH will update the LOCA evaluations in the event that the MLHGR specified in the core operating limits report is allowed to exceed that used in the current LOCA analyses of record.

Axial Power Shapes. The staff finds that the applicant's axial resolution of about 0.04m (1.5 in.) at the inlet and 0.015m (6 in.) for the rest of the channel is fine enough to discern the location of the peak cladding temperature and adequately represent the axial power shape. GEH submitted the power shape used for the LOCA analyses in response to RAI 6.3-50. The staff finds that the power shape submitted may not be the most conservative for LOCA applications where the core experiences heatup; however, since the core remains covered during all analyzed LOCA transients and the limiting bundle does not heat up, other power shapes would not produce appreciably different results.

Initial Stored Energy. For the ESBWR LOCA analyses, GEH used a constant gap conductance. The gap conductance values come from the GEH GSTRM fuel mechanical code. Section 4.2 of this report describes the applicability of the GSTRM code to the ESBWR. The staff has questions about the gas gap conductance values from the GSTRM code. The staff does not have any information about the assumptions GEH used in the GSTRM code to generate these values. However, since the LOCA event for the ESBWR does not cause any core heatup and the core remains covered throughout the entire transient, the staff finds that these values will not have any effect on the calculated figure of merit (i.e., collapsed chimney level) for the LOCA transient and therefore finds their use acceptable.

The TRACG04 code uses fuel thermal conductivity values based on the PRIME03 code. The NRC staff is currently evaluating the GSTRM code for use in the ESBWR, and its use for LOCA analysis will depend upon resolution of RAI 4.8-16 as discussed in Section 4.2.3.2 of this report. The NRC has not reviewed and approved PRIME03; RAI 6.3-54 asks GEH to justify using this model. RAI 6.3-55 also asks GEH to justify using gap conductance and fuel thermal conductivity from different models. **RAI 6.3-54 and 6.3-55 are being tracked as open items.**

Control Rod Insertion. RAI 6.3-52 asks GEH to provide the scram time delay and justify the delay time selected. During an audit of TRACG as applied to the ESBWR LOCA analyses, GEH stated that the travel time of the rods into the core is factored into the decay heat curve. GEH will explain how this is done in response to this RAI. **RAI 6.3-52 is being tracked as an open item.**

In DCD Revision 3, GEH stated that it is using the 1994 ANS decay heat standard. The 1994 decay heat standard is different from that used in DCD Revision 2. GEH used the 1979 ANS decay heat standard in DCD Revision 2. GEH updated DCD Tier 2, Revision 3, Figure 6.3-39 to show the new decay heat curve. The figures in the DCD demonstrating ECCS performance are unchanged. In RAI 6.3-80, the staff asked GEH to explain why the analysis plots did not change. In addition, the staff performed a detailed review of the applicant's decay heat model during an audit of TRACG as applied to the ESBWR LOCA . RAI 6.3-80 also requests that GEH note any differences in the methodology that the staff reviewed during the audit and that which was used to generate the decay heat curve in DCD Revision 3. GEH received the questions that resulted from the staff audit in RAI 6.3-62. **RAI 6.3-62 and 6.3-80 are being tracked as open items.**

Boric Acid Precipitation. DCD Tier 2, Revision 3, Figure 5.1-1 gives a core volume of 96 cubic meters (m^3) (which does not include volume in the chimney, separator, and lower plenum). In response to RAI 6.3-60, GEH gave the maximum volume of SLCS inventory that can be injected into the core. This volume is based on the closure of the SLCS injection line shutoff valves at the accumulator low-level setpoint. The volume of each of the two SLCS tanks is 8.31 m^3, giving a total possible SLCS injection inventory of 16.62 m^3. The SLCS tanks are at ambient temperature with a 12.5 weight-percent sodium pentaborate solution. The volume of the core is more than five times that of the SLCS tanks. The average void fraction of the core would have to exceed 80 percent to create a mixing volume comparable to that of the SLCS tanks. The staff finds that it is unlikely that boron will precipitate during a LOCA event in the ESBWR and therefore finds that GEH not analyzing this possibility is acceptable.

Containment Pressure Response. Section 6.2 of this report discusses containment pressure response.

ECCS Strainer Performance Evaluation. Section 6.2.1.5.3 of this report addresses ECCS strainer performance.

6.3.2.3.7 RPS and ECCS Actions

The staff reviewed the timing, sequencing, and capacity of the RPS and ECCS in relation to the design-basis LOCA event analyses. In DCD Revision 3, GEH stated that the GDCS line break with failure of one injection valve is the limiting break for the minimum collapsed chimney level for the ESBWR. Section 6.3.2.2.7 of this report describes the sequence of the RPS actions. The following sections discuss the evaluation of the RPS and ECCS functions for the design-basis events presented in DCD Tier 2, Revision 3.

The staff requested in RAI 6.3-56, GEH provide more details on the sequence of events than the information that is provided in DCD Tier 2, Tables 6.3-7 through 6.3-10. Staff requested GEH to include trip signals and set-points for all reactor protection system (RPS) actions and to include actions necessary for long-term core cooling. **RAI 6.3-56 is being tracked as an open item.**

Reactor Scram. For a LOCA event, the mitigation function of the reactor scram is to shut down the nuclear chain reaction and reduce power to decay heat levels. For the design analyses, the reactor scram signal is from the loss of power generation buses (i.e., loss of offsite power that results in a loss of all feedwater). DCD Tier 2, Revision 3, Section 7.2.1.2.4.2, gives a complete list of reactor scram signals. The reactor scram signals that would likely cause the reactor to scram during a LOCA event include the following:

- high drywell pressure
- loss of power generation buses
- reactor water level reaching Level 3 and indicating it is decreasing
- MSIV closure indication
- manual

The staff finds that the timing and function of the reactor scram are adequate for it to perform its safety function.

Isolation Condenser System. The LOCA mitigation function of the ICS is to provide injection at high-pressure conditions from the drainlines. In addition, the IC will be used to condense the RPV steam. The IC drainline valves open on the same signal that scrams the reactor. This occurs upon the loss of power generation buses (i.e., loss of offsite power that results in a loss of all feedwater). DCD Tier 2, Revision 3, Section 7.4.4.3 , gives a complete list of IC actuation signals. The following signals would likely cause the IC to actuate during a LOCA event:

- loss of power generation buses
- reactor water level reaching Level 2 with a time delay
- reactor water level reaching Level 1
- loss of feedwater
- MSIV closure indication
- manual

The staff finds that the timing and function of the ICS are adequate for it to perform its safety function.

MSIV Closure. The MSIV closure helps mitigate the depressurization and loss of inventory during a LOCA event. The MSIV closure in the limiting LOCA analysis (GDCS line break with failure of one injection valve) is initiated on a low MSL pressure (plus a delay). The MSIV will also close based on a Level 2 signal plus a 30-second delay. The staff finds that the MSIV closure is adequate to perform its mitigation function during a LOCA.

ADS Actuation. The purpose of the ADS is to depressurize the reactor vessel so that the GDCS may inject cooling water. The ADS is initiated upon confirmation of the Level 1 setpoint. Confirmation of Level 1 occurs when Level 1 persists for 10 seconds. Section 6.3.2 of this report discusses the ADS, including sequencing of the valves. The results of the ECCS performance analyses show that the ADS initiation, sequencing, and capacity enable it to perform its ECCS safety function.

SLCS Actuation. The LOCA mitigation function of the SLCS is to provide additional injection inventory at high-pressure conditions. The SLCS timer is initiated upon confirmation of the Level 1 setpoint. The SLCS will actuate after a 50-second delay. The results of the ECCS performance analyses show that the SLCS initiation and capacity make it capable of performing its ECCS safety function.

GDCS Actuation. The main function of the GDCS is to provide low-pressure coolant inventory in the event of a LOCA once the RPV is depressurized. The GDCS timer is initiated upon confirmation of the Level 1 setpoint. The GDCS squib valves will then actuate after a 150-second delay. GDCS pools will drain once the RPV depressurizes below that of the GDCS. During the later stages of the GDCS injection phase of the LOCA event, the collapsed chimney level experiences large oscillations. In NEDC-33083P-A, GH stated that these are manometric types of oscillations. The manometric oscillations occur as the voids in the core are quenched and a larger static head is created inside the shroud and reduces the flow from the downcomer; this leads to an increase in void fraction. The increase in void fraction will cause a decrease in static head inside the shroud, and the downcomer flow will increase and quench the voids to start the cycle all over again. This is also why the downcomer shows oscillations. Since the channel represented in the ECCS performance plots of collapsed chimney level is the hot channel, the oscillations shown in the chimney are much larger. The staff believes that this may also be a result of geysering. In either case, the staff does not find these observed oscillations to be a safety concern. The mechanism for these oscillations requires that there be recirculation flow and water above the TAF. In addition, at decay heat levels, the core would need to experience a sustained uncovery to heat up to levels that would cause fuel damage. These oscillations currently do not show that the level goes below the TAF. In the case that TRACG is not adequately calculating the magnitude of these oscillations and the level reaches below the TAF, then fuel would only be exposed for a few seconds at a time and be requenched a few seconds later. Structurally, the core would be experiencing some loads from these oscillations; however, these would not be more than those seen during normal operations. Overall, the results of the ECCS performance analyses show that the GDCS is capable of performing its ECCS safety function.

6.3.2.3.8 Long-Term Core Cooling

The staff reviewed the long-term core cooling calculations presented in MFN 05-105. These calculations show that the core remains covered for up to 12 hours. The calculations do not show the levels up to 72 hours. The staff requested this information in RAIs 6.3-64 and 21.6-98. The applicant will also update these calculations to reflect the most recent design. **RAI 6.3-64 and 21.6-98 are being tracked as open items.** See Section 6.3.2.2.8 of this report for a discussion of Open Item RAI 6.3-64.

In RAI 6.3-45 staff requested GEH to provide the differences between the TRACG input decks used to calculate minimum water levels and perform containment peak pressure analyses. In a supplemental RAI, staff requested GEH provide justification that even though the input deck for calculating minimum water level lacks the modifications applied to the containment input deck, that the results are still accurate and conservative for the long-term core cooling analysis. **RAI 6.3-45 is being tracked as an open item.**

6.3.2.3.9 Independent Staff Calculations

The staff is using the TRACE thermal hydraulics code to independently verify the ESBWR system response in the event of a LOCA. The staff is currently performing confirmatory calculations for the following LOCA events:

- steamline inside containment
- FWL
- GDCS injection line

- bottom head drainline

6.3.4.2.10 ECCS Performance Criteria

Section 6.3.1.1 of this report discusses the five performance criteria of the ECCS. For the ESBWR, GEH demonstrated that there is no core uncovery or heatup for any design-basis LOCA. The fuel does not heat up during a LOCA event; therefore, the PCT is about the same as it is during normal operations. There is no additional oxidation of the cladding as a result of the LOCA. There is no additional hydrogen generated from the chemical reaction of the cladding with water or steam because the temperatures are not high enough to create this chemical reaction. There are no changes in core geometry resulting from the LOCA event that would prevent the core from being amenable to cooling. GEH has not yet demonstrated that the ECCS maintains the calculated temperature at an acceptably low value and removes that decay heat for the extended period of time as discussed in Section 6.3.2.2.8 of this report. **RAI 6.3-64 and 21.6-98 are being tracked as open items.**

6.3.2.4 Conclusions

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

6.4 Control Room Habitability Systems

The control room habitability area (CRHA) is served by a combination of individual systems that collectively provide the habitability functions. The systems that make up the habitability systems are the CRHA HVAC subsystem (CRHAVS), radiation monitoring subsystem (RMS), lighting system, and the FPS. ESBWR design features are provided to ensure that the control room operators can remain in the control room and take actions to safely operate the plant under normal conditions and to maintain it in a safe condition under accident conditions. These habitability features include missile protection, radiation shielding, radiation monitoring, air filtration and ventilation systems, lighting, personnel and administrative support, and fire protection.

6.4.1 Regulatory Criteria

The staff reviewed the CRHA in accordance with SRP Section 6.4, which discusses the control room habitability system. Conformance with the SRP acceptance criteria forms the basis for the staff's evaluation of the CRHA systems. The following regulations and NRC guidance documents apply to this system:

- GDC 4, as it relates to structures, systems, and components important to safety being designed to accommodate the effects of and to be compatible with the environmental conditions associated with postulated accidents
- GDC 5, as it relates to ensuring that sharing among nuclear power units of structures, systems, and components 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, as it relates to maintaining the nuclear power unit in a safe condition under accident conditions and providing adequate radiation protection

- 10 CFR 50.34(f)(2)(xxviii), as it relates to evaluations and design provisions to preclude certain control room habitability problems

- TMI Action Plan Item III.D.3.4 (see NUREG-0737), regarding protection against the effects of toxic substance releases, either on or off the site.

- RG 1.78, "Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release"

- RG 1.196, "Control Room Habitability at Light-Water Nuclear Power Reactors"

- RG 1.197, "Demonstrating Control Room Envelope Integrity at Nuclear Power Reactors," issued May 2003

- Generic Safety Issue, Item B-36, "Develop Design, Testing and Maintenance Criteria for Atmosphere Cleanup System Air Filtration and Adsorption Units for Engineered Safety Features Systems and for Normal Ventilation Systems"

- Generic Safety Issue, Item B-66, "Control Room Infiltration Measurements"

- Generic Safety Issue, Issue 83, "Control Room Habitability (Rev. 3)"

The generic safety issues can be found in NUREG-0933, "A Prioritization of Generic Safety Issues," issued September 2007.

6.4.2 Summary of Technical Information

The CRHA is served by a combination of individual systems that collectively provide the habitability functions. The systems that make up the habitability systems include the following:

- CRHAVS
- RMS
- lighting system
- FPS

When ac power is available, the CRHAVS provides normal and abnormal HVAC service to the CRHA as described in DCD Tier 2, Revision 3, Section 9.4.1. When ac power is unavailable for an extended time, or if high radioactivity is detected in the CRHA outside air supply duct, the RMS automatically isolates the CRHA normal air supply. When this happens, the habitability requirements are then met by the operation of an emergency filter unit (EFU). The EFUs provide emergency ventilation and pressurization for the CRHA. When ac power is unavailable, the CRHA is passively cooled by the CRHA passive heat sink.

The RMS provides radiation monitoring of the CRHA environment and outside air intake.

The FPS provides smoke detection and fire damper isolation.

The lighting system provides emergency lighting.

Storage capacity is provided in the main control room for personnel support equipment. Manual hose stations outside the CRHA and portable fire extinguishers provide fire suppression in the CRHA.

The CRHA contains the following features:

- main control consoles and associated equipment
- shielding and area radiation monitoring
- provisions for emergency food, water, storage and air supply systems
- kitchen and sanitary facilities
- provision for protection from airborne radioactive contaminants

The CRHA is contained inside a seismic Category I structure (the Control Building) and is protected from wind and tornado effects, external floods and internal flooding, external and internal missiles, and the dynamic effects associated with the postulated rupture of piping.

The habitability systems maintain the main control room environment suitable for prolonged occupancy throughout the duration of the postulated accidents. In particular, the systems ensure the following:

- The main control room is designed to withstand the effects of an SSE and a design-basis tornado.

- The radiation exposure of main control room personnel throughout the duration of the postulated limiting faults discussed in Chapter 15 does not exceed the limits set by GDC 19.

- The emergency habitability system maintains the American Society of Heating, Refrigeration and Air Conditioning Engineers (ASHRAE) Standard 62, “Ventilation for Acceptable Air Quality,” fresh air requirements for up to 21 main control room occupants.

- The habitability systems detect and protect main control room personnel from external fire, smoke, and airborne radioactivity.

- Automatic actuation of the individual systems that perform a habitability system function is provided. Smoke detectors, radiation detectors, and associated control equipment are installed at various plant locations as necessary to provide the appropriate operation of the systems.

- The CRHA includes all instrumentation and controls necessary during safe shutdown of the plant and is limited to those areas requiring operator access during and after a DBA.

- CRHA habitability requirements are satisfied without the need for individual breathing apparatus and/or special protective clothing.

- The CRHA EFUs and associated fans and ductwork; the CRHA envelope structures; and the CRHA heat sink, doors, isolation dampers and/or valves, including supporting ductwork/piping, and associated controls are safety related and seismic Category I.

- Non-safety-related pipe, ductwork, or other components located in the control room are designed as necessary to ensure that they do not adversely affect safety-related components or the plant operators during an SSE.

- The EFU trains are designed with sufficient redundancy to ensure operation under emergency conditions.

- The EFUs are operable during loss of preferred power, loss of onsite ac power or station blackout (SBO).

- The EFUs operate during an emergency to ensure the safety of the control room operators and the integrity of the control room by maintaining a minimum positive differential pressure inside the CRHA.

- The CRHA envelope is sufficiently leaktight to maintain positive differential pressure with one EFU in operation.

The EFUs are redundant safety-related components that supply filtered air to the CRHA for breathing and pressurization to minimize in-leakage. The EFUs and their related components form a safety-related subset of the CRHAVS. The EFU portion of the system and the associated components are designed, constructed, and tested as a safety-related nuclear air filtration system in accordance with ASME AG-1, "Code on Nuclear Air and Gas Treatment," requirements. An EFU is automatically initiated. There are two redundant EFU trains to provide protection against a single failure. Each train consists of an air intake, fan filtration housing, ductwork, and dampers. The EFUs have been sized to provide sufficient breathing quality air and to maintain a positive pressure in the CRHA.

Control Room Habitability Area

The CRHA boundary is located in the Control Building at the 2000 mm elevation. DCD Tier 2, Revision 3, Figure 1.2-3 depicts the layout of the CRHA, which includes the main control room.

The CRHA envelope includes the following areas:

- main control room (Room 3275)
- shift supervisor office (Room 3272)
- shift supervisor conference room (Room 3273)
- operator's area (Room 3270)
- shift technical advisor office (Room 3271)
- main control room storage room (Room 3204)
- electrical panel board room (Room 3205)
- restroom A (Room 3201)
- restroom B (Room 3201)

These areas constitute the operation control area, which can be isolated and remain habitable for 72 hours without ac power if high radiation conditions exist. Potential sources of danger, such as steamlines, pressurized piping, pressure vessels, carbon dioxide firefighting containers, and the like, are located outside of the CRHA.

Heat Sink

The function of providing a passive heat sink for the CRHA is part of the CRHA emergency habitability system. The heat sink for each room is designed to limit the temperature rise inside the room during the 72-hour period following a loss of CRHAVS operation. The heat sink consists of the thermal mass of the concrete that makes up the ceilings and walls of these rooms.

Shielding Design

The design-basis LOCA dictates the shielding requirements for the CRHA. Section 12.3 of DCD Tier 2, Revision 3 discusses the main control room shielding design bases. Section 15.4 of DCD Tier 2, Revision 3 describes the design-basis LOCA source terms, main control room shielding parameters, and evaluation of doses to main control room personnel. DCD Tier 2, Revision 3, Figure 12.3-3 shows the main control room location in the plant with respect to designated radiation zones.

Component Descriptions

The EFU outside air supply portion of the CRHAVS is safety related and seismic Category I. Single active failure protection is provided by the use of two trains, which are physically and electrically redundant and separated. If one train fails, the failed train is isolated and the alternate train is automatically initiated. Both trains are 100-percent capacity and capable of supplying 99-percent credited efficiency filtered air to the CRHA pressure boundary at the required flow rate. The EFU design uses a HEPA filter, carbon filter, and postfilter to provide radiological protection of the CRHA outside air supply.

The CRHA pressure boundary includes penetrations, dampers and/or valves, interconnecting duct or piping, and related test connections and manual valves. The isolation dampers and/or valves are classified as Safety Class 3 and seismic Category I. The dampers and/or valves have spring return actuators that fail close on a loss of electrical power. Isolation valves are qualified to provide a leaktight barrier for the CRHA envelope pressure. The boundary isolation function of isolation dampers and/or valves will be demonstrated by pressure testing of the CRHA and in-leakage testing.

Tornado protection dampers are a split wing or equivalent type designed to close automatically. The tornado protection dampers are designed to mitigate the effect of a design-basis tornado.

All shutoff, balancing, and backdraft dampers in the EFU outside air delivery path are constructed, qualified, and tested in accordance with ANSI/Air Moving and Conditioning Association (AMCA) 500-D, "Laboratory Methods of Testing Dampers for Rating," or ASME AG-1, Section DA. Backdraft dampers meet the Leakage Class II requirements of ASME AG-1. Remotely operated, two-position-type shutoff dampers are designed for the maximum fan static pressure.

Ductwork, duct supports, and accessories are constructed of galvanized or stainless steel, or, of carbon or stainless steel if standard pipe is used. Ductwork subject to fan shutoff pressures is structurally designed to accommodate such pressures. The EFU-related ductwork, including the EFUs and the related ductwork outside the CRHA boundary, is designed in accordance with ASME AG-1, Article SA-4500, to provide low-leakage components necessary to maintain the CRHA habitability.

Two sets of doors, with a vestibule between them that acts as an air lock, are provided at each access to the main control room.

Leaktightness

The CRHA boundary envelope structures are designed with low-leakage construction. The CRHA is located in an underground portion of the Control Building. The boundary walls are adjacent to underground fill or underground internal areas of the Control Building. The construction consists of cast-in-place reinforced concrete walls and slabs and is constructed to minimize leakage through joints and penetrations. The following features are applied as required to achieve the leaktightness objective:

- The EFU filter train is located downstream of the EFU fan. This maintains the filter train and delivery ductwork to the CRHA at a positive pressure, precluding any unfiltered in-leakage into the system.
- The access doors are designed with self-closing devices, which close and latch the doors automatically. Double-door air locks allow access and egress during emergencies.
- The outside surface of penetration sleeves in contact with concrete are sealed with epoxy or an equivalent sealant. Piping and electrical cable penetrations are sealed with a qualified pressure-resistant material compatible with penetration materials and/or cable jacketing.
- Inside surfaces of penetrations and sleeves in contact with commodities are sealed.
- Penetration sealing materials are designed to withstand at least a 1/4-inch water gauge pressure differential. The bulk penetration sealing material is gypsum cement or equivalent, with epoxy or equivalent sealants applied to complement penetration sealing.
- The CRHA uses internal air handling units (AHUs) that preclude any AHU ductwork external to the CRHA envelope.

During normal operation, the CRHA is heated, cooled, ventilated, and pressurized by either of a redundant set of recirculating AHUs and either of a redundant set of outside air intake fans for ventilation and pressurization purposes. During a radiological event or an SBO, an EFU maintains a positive pressure in the CRHA to minimize infiltration of airborne contamination.

Interlocked, double-vestibule-type doors maintain the positive pressure, thereby minimizing infiltration when a door is opened.

Emergency Habitability

The CRHA emergency habitability portion of the CRHAVS is not required to operate during normal conditions. The normal operation of the CRHAVS maintains the air temperature of the CRHA within a predetermined temperature range. This maintains the CRHA emergency habitability system's passive heat sink at or below a predetermined temperature. The normal operation portion of the CRHAVS operates during all modes of normal power plant operation, including startup and shutdown.

Operation of the emergency habitability portion of the CRHAVS is automatically initiated by either of the following conditions—high radioactivity in the main control room supply air duct and/or extended loss of ac power.

Operation can also be initiated by manual actuation. If radiation levels in the main control room supply air duct exceed the high setpoint, the normal outside air intake and restroom exhaust are isolated from the CRHA pressure boundary by automatic closure of the isolation devices in the system ductwork. At the same time, an EFU begins to deliver filtered air from one of the two unique safety-related outside air intake locations. A constant airflow rate is maintained, and this flow rate is sufficient to pressurize the CRHA boundary to at least 31 Pa (1/8-inch water gauge) positive differential pressure with respect to the surroundings. The EFU system airflow rate is also sufficient to supply the ASHRAE Standard 62 fresh air requirement of 9.5 liters per second L/s (20 cubic feet per minute (cfm)) per person for up to 21 occupants.

With a source of ac power available, an EFU can operate indefinitely. In the event that ac power is not available, the safety-related 1E battery power supply is sized to provide the required power to an EFU fan for 72 hours of operation. The temperature and humidity in the CRHA pressure boundary following a loss of the normal portion of the CRAHVS remain within the limits for reliable human performance over a 72-hour period. The CRHA isolation dampers fail closed upon a loss of ac power or instrument air. A portable, dedicated RTNSS generator will provide backup power to the safety-related control room EFU fans (after 72 hours).

Upon a loss of preferred power or SBO, the initial ranges of temperature/relative humidity in the CRHA are 22.8–25.6℃ (73–78℉) and 25–60 percent relative humidity. During the first 2 hours of an SBO, most of the equipment in the main control room remains powered by the non-1E battery supply. After 2 hours, the non-1E batteries are exhausted, and only a small amount of safety-related equipment remains powered. During the first 2 hours, the environmental conditions are maintained within the normal ranges listed above. This is accomplished by the continued operation of a CRHA AHU and chilled water pump powered from the same non-1E battery supply that powers the non-safety main control room equipment. Chilled water from a chilled water thermal storage tank is used as the heat sink. The cooling function for this 2-hour period is not a safety function; if this cooling function is lost, the non-safety-related equipment and associated heat loads are automatically de-energized.

If power remains unavailable beyond 2 hours, the remaining CRHA safety-related equipment heat loads are dissipated passively to the CRHA heat sink. The CRHA heat sink limits the temperature rise. The CRHA is passively cooled by conduction into the walls and ceiling. Sufficient thermal mass is provided in the walls and ceiling of the main control room to absorb the heat generated by the equipment, lights, and occupants.

Doses to main control room personnel are calculated for the accident scenario in which an EFU provides filtered air to pressurize the CRHA. The dose analyses are performed in accordance with the requirements of RG 1.194, “Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments for Nuclear Power Plants,” issued June 2003, and RG 1.196. For all events, the dose is within the dose acceptance limit of 5 rem total effective dose equivalent. Chapter 15 of DCD Tier 2, Revision 3, discusses the analytical assumptions for modeling the doses to the main control room personnel. No radioactive materials are stored or transported near the main control room pressure boundary. The use of noncombustible construction and heat-and flame-resistant materials throughout the plant reduces the likelihood of fire and its consequential impact on the main control room atmosphere.

Section 9.4.1 of DCD Tier 2, Revision 3, discusses operation of the CRHAVS in the event of a fire. The exhaust stacks of the onsite standby power diesel generators are located in excess of 48 m (157 ft) away from the fresh air intakes of the main control room. The onsite standby power system fuel oil storage tanks are located in excess of 55 m (180 ft) from the main control room fresh air intakes. These separation distances reduce the possibility that combustion fumes or smoke from an oil fire would be drawn into the main control room.

DCD Tier 2, Revision 3, Table 6.4-2 lists the sources of onsite chemicals, and DCD Tier 2, Revision 3, Figure 1.1-1 provides their locations. Analysis of these sources in accordance with RG 1.78 and the methodology in NUREG-0570, “Toxic Vapor Concentrations in the Control Room Following a Postulated Accidental Release,” (see DCD Tier 2, Rev. 3, Table 1.9-23) is to be performed on a site-specific basis (see DCD Tier 2, Rev. 3, Section 6.4.9).
During emergency operation, the CRHA emergency habitability system passive heat sink is designed to limit the temperature rise inside the CRHA to no more than 8.3℃ (15℉). This maintains the CRHA within the limits for reliable human performance over 72 hours. The walls and ceiling that act as the passive heat sink contain sufficient thermal mass to accommodate the heat sources from equipment, personnel, and lighting for 72 hours. The EFU portion of the CRHAVS nominally provides 200 L/s (424 cfm) of ventilation air to the main control room and is sufficient to pressurize the control room to at least a positive 1/8-inch water gauge differential pressure with respect to the surrounding areas. This flow rate also supplies the ASHRAE Standard 62 (Table 1.9-22) recommended fresh air supply of 9.5 L/s (20 cfm) per person for a maximum occupancy of 21 persons. Automatic isolation of the normal air intake and transfer of outside air supply to an EFU is initiated by either of the following conditions—high radioactivity in the CRHA normal air supply duct or extended loss of ac power.

The airborne fission product source term in the reactor containment following the postulated LOCA is assumed to leak from the containment. The concentration of radioactivity is evaluated as a function of the fission product decay constants, the containment leak rate, and the meteorological conditions assumed. The assessment of the amount of radioactivity within the CRHA considers the radiological decay of fission products and the infiltration/exfiltration rates to and from the CRHA pressure boundary. Section 15.4 of DCD Tier 2, Revision 3, discusses specific radiological protection assumptions used in the generation of post-LOCA radiation source terms.

Preoperational Inspection and Testing

Preoperational testing of the CRHAVS is performed to verify that the minimum airflow rate of 200 L/s (424 cfm) is sufficient to maintain pressurization of the main control room envelope of at least 31 Pa (1/8" wg) with respect to the adjacent areas. The positive pressure within the main control room is confirmed by means of the differential pressure transmitters within the control room. The installed flow meters are used to verify the system flow rates.

The pressurization of the control room limits the ingress of radioactivity to maintain operator dose limits below regulatory limits.

Air quality within the CRHA environment is confirmed to be within the guidelines of ASHRAE Standard 62 requirements for continued occupancy because it meets the fresh air supply requirement of 9.5 L/s (20 cfm) per person for the type of occupancy expected in the CRHA.

The capacity of the safety-related batteries is verified to be capable of powering an EFU fan for a minimum of 72 hours. An inspection will verify that the heat loads within the CRHA are less than the specified values.

Preoperational testing of the CRHAVS isolation dampers is performed to verify the leaktightness of the dampers. Preoperational testing for CRHA in-leakage during EFU operation will be conducted in accordance with ASTM E741, "Standard Test Method for Determining Air Change in a Single Zone by Means of a Tracer Gas Dilution."

Section 11.5 of DCD Tier 2, Revision 3, discusses the testing and inspection of the radiation monitors. Chapter 14 of DCD Tier 2, Revision 3, discusses the other tests noted above.

Inservice Testing

Inservice testing of the CRHAVS is conducted in accordance with the surveillance requirements specified in the TSs provided in DCD Tier 2, Revision 3, Chapter 16. Leaktightness testing of the CRHA pressure boundary is conducted at the frequency specified in the TSs.

Nuclear Air Filtration Unit Testing

The EFU filtration components are periodically tested in accordance with ASME AG-1 to meet the requirements of RG 1.52, "Design, Inspection, and Testing Criteria for Air Filtration and Adsorption Units fo Post-Accident Engineered-Safety-Feature Atmosphere Cleanup Systems in Light-Water-Cooled Nuclear Power Plants." Periodic surveillance testing of safety-related CRHA isolation dampers and the EFU components are carried out in accordance with IEEE-338, "Standard Criteria for the Periodic Surveillance Testing of Nuclear Power Generating Station Safety Systems." Safety-related CRHA isolation dampers and the EFU are operational during plant normal and abnormal operating modes.

Air In-Leakage Testing

Testing to demonstrate the integrity of the CRHA envelope is performed in accordance with RG 1.197 and ASTM E741.

Instrumentation Requirements

Section 7.3 of DCD Tier 2, Revision 3, provides a list of the instrumentation required for actuation of the CRHAVS emergency habitability system, a description of initiating circuits, logic, periodic testing requirements, and a discussion of the redundancy of instrumentation relating to the habitability systems. Section 11.5 of DCD Tier 2, Revision 3, details the radiation monitors used to provide the main control room indication of actuation of a CRHA isolation and EFU initiation. Alarms for the following CRHA/CRHAVS conditions are provided in the main control room:

- low airflow (each EFU fan, AHU, and outside air intake fan)
- high filter pressure drop (each EFU and normal outside air intake filters)
- high space temperatures
- low space temperatures
- low AHU entering air temperature
- low CRHA differential pressure
- smoke detection
- high and low humidity in the CRHA
- CRHA airlock doors open during an SBO
- area high radiation in the CRHA
- high radiation in the outside air intake duct

COL Information

COL applicants referencing the ESBWR certified design are responsible for the amount and location of possible sources of toxic chemicals in or near the plant and for seismic Category I, Class 1E toxic gas monitoring, as required. 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 and GDC 19. COL applicants referencing the ESBWR certified design are responsible for verifying that procedures and training for control room habitability are consistent with the intent of Generic Safety Issue 83. The COL applicant will provide the testing frequency for the main control room in-leakage test. This is identified as COL Information item in DCD Tier 2, Revision 3, Section 6.4.9.

6.4.3 Staff Evaluation

The staff reviewed the information in DCD Tier 2, Revision 3, Section 6.4, and referenced sections to determine compliance with the GDC, TMI action items, and other appropriate regulatory criteria and guidance documents.

GDC 4 requires that structures, systems, and components 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 LOCAs. These structures, systems, and components shall be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids that may result from equipment failures and from events and conditions outside the nuclear power unit. However, dynamic effects associated with postulated pipe ruptures in nuclear power plant units may be excluded from the design basis when analyses reviewed and approved by the Commission demonstrate that the probability of fluid system piping rupture is extremely low under conditions consistent with the design basis for the piping.

The CRHAVS and components are located in a seismic Category I structure that is protected from tornado, missile, pressure, and flood damage. The EFU portion of the CRHAVS is safety related and designed to seismic Category I standards.

Tornado dampers are provided at some locations for the ESBWR intakes. The staff requested additional information in RAI 6.4-13 to identify which intakes are protected and to assess the impact of a sudden pressure drop resulting from a tornado. **RAI 6.4-13 is being tracked as an open item.**

Nonseismic pipe, ductwork for kitchen and sanitary facilities, and other nonessential components in the control room habitability area are designed to ensure that their failure during an SSE will not adversely affect essential components.

Potential sources of danger, such as pressure vessels and carbon dioxide firefighting containers, are located outside the CRHA.

There are no high-energy lines near the control room; therefore, the habitability systems are protected against the dynamic effects that may result from possible failures of such lines.

The staff finds that the ESBWR CRHA design complies with GDC 4 in that the essentially underground structure is contained within a seismically qualified Class 1 building and is protected from the effects of external environmental conditions, such as wind, flooding, pipe whip, and discharging fluids.

GDC 5 requires that structures, systems, and components important to safety not be shared among nuclear power units unless it can be shown that such sharing will not significantly impair their ability to perform their safety functions, including, in the event of an accident in one unit, an orderly shutdown and cooldown of the remaining units. The staff review determined that the CRHAVS meets the acceptance criteria of GDC 5. The ESBWR control room habitability design supports a single unit. Structures, systems, and components important to safety are not shared among nuclear power units. As such, the requirements of GDC 5 are satisfied.

GDC 19 requires that a control room be provided from which the nuclear power unit can be safely operated under normal conditions and maintained in a safe condition under accident conditions, including LOCAs. It also requires that adequate radiation protection be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident. Implicit in GDC 19 is that the environmental conditions (such as temperature, humidity, and oxygenation) will be acceptable for personnel and equipment to function.

The CRHA includes the main control room, shift supervisor’s office, shift supervisor’s conference room, operator area, shift technical advisor’s office, electrical panel board room, storage areas, and restrooms. The CRHA is required to be habitable during (1) normal operations, (2) accident conditions, including a LOCA, for 72 hours without reliance on active or non-safety-related systems, (3) accident conditions for the period from 72 hours to 30 days during which some active or RTNSS-qualified systems may be used, and (4) toxic gas or smoke purging events.

The principal means for maintaining control room habitability is the operation of the CRHAVS and associated cooling and heating. The system provides control of unfiltered in-leakage through pressurization of the control room, as well as smoke purge capability. Section 9.4.1 of this report discusses the system features.

For normal operation, maintenance, and testing, the non-safety CRHAVS supply fan provides fresh air and pressurization. A recirculation AHU provides for control room air circulation. Cooling coils and heating coils are included in the recirculation AHU for temperature control. The recirculation AHU also provides humidity control. The system is powered from the station ac system.

For the first 72 hours after an accident, including an SBO with a LOCA, the CRHA zone is isolated. The unfiltered supply air system is shut down and isolated by safety-related dampers. The bathroom and kitchen exhaust and the smoke purge inlet and outlet are also isolated. One of two EFU fans starts and supplies filtered air to the CRHA at 200 L/s (424 cfm). The operating recirculation AHU is shut down, and the entire system operates on the safety-related battery system.

With the isolation of the recirculation AHU, temperature control is lost. The applicant states that the passive heat sink, consisting of the walls, ceilings, and floors of the CRHA, is sufficient to prevent a temperature rise greater than 15°F in the CRHA.

The ability of the passive heat sink to maintain the control room atmosphere temperature at a habitable level to permit prolonged personnel occupancy throughout a postulated design-basis event has not been established. The staff needs additional information on the impact of heat sink operation at both the summer design temperature and the winter design temperature. The staff issued RAI 6.4-7 to address this issue. **RAI 6.4-7 is being tracked as an open item.**

The applicant also stated that for the first 2 hours after an accident, including an SBO with a LOCA, non-safety-related heat loads are removed by the operation of a recirculation AHU and chilled water pump by non-safety-related battery systems. The applicant claims that if the non-safety-related batteries are not available, these non-safety-related heat loads can be isolated. The staff issued an RAI relating to this issue in connection with its review of Section 9.4.1 of this report (see open item 9.4-32). The staff issued an additional RAI, RAI 6.4-16, requesting additional clarification. **RAI 6.4-16 is being tracked as an open item.**

In addition, with the shutdown of the isolation unit during the first 72 hours, there is no air mixing in the CRHA. Although the air is being refreshed by the EFU units and the flow rate meets the ASHRAE 62 Standard in principle, it is not clear whether the ASHRAE standard is based on a volume that does not include recirculation or mixing of air. For the ESBWR, it appears that the 200 L/s (424 cfm), which could be at the summer design temperature, is entering in the return plenum above the false ceiling. Without forced mixing, this could lead to stratification and leave the breathable zone of operators essentially unrefreshed. Also, carbon dioxide levels near or above toxicity limits may develop without mixing. This issue is being addressed in Section 9.4.1 of this report (see open Item RAI 9.4-30 associated with RAI 9.4-30).

Section 9.4.1 of this report discusses the staff review of the CRHAVS, including the safety-related EFU.

The evaluation above speaks to the first 72 hours after an accident. The CRHA needs to be habitable for the 30-day duration of the DBA. If this requires the restoration of ac power after 72 hours, the DCD needs to identify the source of this power. This issue is being addressed by the Open Item associated with RAI 6.4-7, discussed above.

The staff developed several RAIs with respect to CRHA pressurization, temperature control, adequacy of breathable air, power sources, and operation of components. These need to be addressed to complete the CRHA review. The staff issued RAI 6.4-8 to address the issue of heat sink initial conditions and surveillances. **RAI 6.4-8 is being tracked as an open item.**

CRHA pressurization is maintained during an emergency mode by the EFU system flow rate of 200 L/s (424 cfm). Based on the acceptance criteria presented in SRP Section 6.4, the staff needs additional information about the adequacy of this makeup flow to maintain pressurization. The staff issued RAI 6.4-9 to address this issue. **RAI 6.4-9 is being tracked as an open item.**

The applicant states that the EFUs are being designed, constructed, and tested in accordance with ASME AG-1 and RG 1.52. This is consistent with the closure of Generic Safety Issue B-36, which addresses the testing criteria for atmosphere cleanup system air filtration and adsorption units, and thus the generic issue is resolved. SRP Section 6.4 acceptance criteria also include the AG-1a-92 addenda. This addenda should be added to the DCD. The staff issued RAI 6.4-10 to address this issue. **RAI 6.4-10 is being tracked as an open item.**

The staff considered the relative location of a source with respect to the Control Building in which the CRHA is located. It appears from Figures 1.2-4 and 1.2-7 that two louvers are identified, one on the Reactor Building side of the Control Building and one on the opposite side of the Control Building. Both louvers appear to be 3–4 meters above grade. However, it is not clear as to which of these louvers services the EFU system and which services the normal HVAC supply to the CRHA. Also, staff is not clear if these control room ventilation inlets are consistent with the values suggested in acceptance criterion no. 5A in Standard Review Plan (SRP) Section 6.4, Revision 3, March 2007. Specifically, staff would like to know if they are separated from potential release points by 100 feet laterally and 50 feet vertically, and, are the actual minimum distances based on the dose analyses. The staff issued RAI 6.4-11 to address these issues.. **RAI 6.4-11 is being tracked as an open item.**

One of the essential requirements of the CRHA is to protect the control room operators from the effects of radiation after an accident. Chapter 15 of this report provides the assessment of dose on control room operators for DBAs. Physical aspects of the CRHA that support the dose assessment are shielding design, unfiltered in-leakage based on leaktightness, pressurization of the CRHA, and location of the control room air intakes with respect to potential releases. The staff issued several RAIs addressing leaktightness, external ductwork, and EFU air intake location(s). In addition to RAIs 6.4-10 and 6.4-11, discussed above, staff issued RAI 6.4-14 requesting GE to include a discussion in the DCD of any ductwork external to the CRHA associated with EFU supply, normal outside air supply, and smoke purge paths. The staff also issued RAI 14.3-152 requesting GEH to provide an ITAAC to verify that the leak tightness of the CRHA has been achieved by testing in accordance with the guidance in Regulatory Guide 1.197. This testing is important in that it serves as the basis for unfiltered in-leakage assumed in the design basis analyses of Chapter 15. **RAIs 6.4-14 and 14.3-152 are being tracked as open items.**

In DCD, Tier 2, Revision 3, Section 9.4.1, the applicant stated that the CRHAVS would maintain the control room at a positive pressure of 1/8 inches e.g., to minimize air in leakage. DCD, Tier 2, Revision 3, Chapter 15, Table 15.4-5 states that the assumption on control room unfiltered in leakage is 1.13E-02 cubic meters per minute (0.3 cfm). Typically, a value of 10 cfm as a minimum is assumed for access and egress. In addition, results of tracer gas testing on positive pressure control rooms have shown additional leakage in many cases. Inefficiency of the EFU filters and bypass penetration could add to unfiltered in leakage. Thus, staff requested in RAI 14.3-153, for GEH to confirm the unfiltered in leakage value that will be used in the design basis analyses and be maintained by control room pressurization in the accident mode and to provide an ITAAC to verify that the unfiltered inleakage is no greater than the value assumed in the DBA analysis. The staff's ITAAC review is discussed in section 14.3 of this report. **RAI14.3-153 is being tracked as an open item.**

TMI Action Plan Item III.D.3.4 (see NUREG-0737) requires that control room operators be adequately protected against the effects of accidental release of toxic and radioactive gases and that the nuclear power plant can safely be operated or shut down under DBA conditions. The staff agrees that COL applicants referencing the ESBWR certified design are responsible for the amount and location of possible sources of toxic chemicals in or near the plant and for seismic Category I, Class 1E toxic gas monitoring, as required. Furthermore, COL applicants referencing the ESBWR certified design are responsible for verifying that procedures and training for control room habitability are consistent with the intent of Generic Safety Issue 83. However, the staff requests GEH to state in the DCD: (1) that the test requirements and the testing frequency will be consistent with the guidance of Regulatory Guide 1.197 which establishes an in service test program and (2) that the test requirements are presented in Chapter 16, Technical Specifications, Section 5.5.12, Control Room Habitability Area (CRHA) Boundary Program. The staff issued RAI 6.4-17 to address this issue. **RAI 6.4-17 is being tracked as an open item.**

The DCD does not identify the design features of the ESBWR that provide the capability to respond to a toxic gas event. Concern exists over EFU operation, automatic isolation, and the need for self-contained breathing apparatus equipment for such events. The staff issued RAIs 6.4-12 and RAI 6.4-15 to address this issue. **RAI 6.4-12 and 6.4-15 are being tracked as open items.**

RAIs 6.4-5, 6.4-6, and 6.4-18 are being tracked as open items to correct discrepancies in the DCD.

6.4.4 Conclusions

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

6.5 Atmosphere Cleanup System

6.5.1 Regulatory Criteria

The atmosphere cleanup system is needed to mitigate the radiological consequences of postulated DBAs by removing fission products from the containment atmosphere that may be released from the reactor primary coolant system in the event of an accident and to meet the radiological consequence evaluation factors specified in 10 CFR 50.34(a)(1) and GDC 19.

The staff's acceptance criteria for the atmosphere cleanup systems are based on meeting the relevant requirements of the following regulations:

- GDC 19, as it relates to systems being designed for habitability of the control room during and following postulated DBAs

- GDC 41, as it relates to the design of systems to be used for containment atmosphere cleanup during and following postulated DBAs

- GDC 42 and 43, as they relate to the inspection and testing of the systems

- GDC 61, “Fuel Storage and Handling and Radioactive Material,” as it relates to the design of systems for radioactivity control

- 10 CFR 50.34(a)(1), as it relates to the radiological consequence evaluation factors specified for the exclusion area boundary and the low-population zone

The staff reviewed DCD Tier 2, Section 6.5, “Atmosphere Cleanup System,” in accordance with the following SRP sections:

- Section 6.5.1, “ESF Atmosphere Cleanup System”
- Section 6.5.3, “Fission Product Control Systems and Structures”
- Section 6.5.5, “Pressure Suppression Pool as a Fission Product Cleanup System”

6.5.2 Summary of Technical Information

Containment

The ESBWR design does not provide an active containment atmosphere cleanup system. Instead, the design relies on natural aerosol removal processes, such as gravitational settling and plateout on containment internal structure surfaces through diffusiophoresis and thermophoresis. The containment structure is a reinforced concrete cylindrical structure which encloses the RPV and its related systems and components and has an internal steel liner providing the leaktight containment boundary. The ESBWR containment is designed to a maximum allowable design leak rate of 0.5 weight-percent (wt%) per day. The applicant stated that 2 percent of this 0.5 wt% per day containment leak rate (0.01 wt% per day overall containment leak) is assumed to leak through the PCCS into the air space directly above the PCCS and subsequently leak directly to the environment without mixing with the RB atmosphere.

Passive Containment Cooling System

The PCCS is designed to remove decay heat and fission products from the containment atmosphere following a postulated DBA. The PCC heat exchangers receive a steam-gas mixture and airborne fission products from the drywell atmosphere, condense the steam, and return the condensate with condensed fission products to the RPV though the GDCS pools. The noncondensables, including noble gases and volatile fission products, are drawn to the suppression pool through a submerged ventline driven by the differential pressure between the drywell and wetwell. The noncondensables will become airborne into the wetwell air space and flow back into the drywell during vacuum breaker openings.

Reactor Building

The RB is a reinforced concrete structure, which forms an envelope completely surrounding the containment, and is designed to seismic Category 1 criteria. The RB does not have an atmospheric cleanup system. The reactor building ventilation system (RBVS) isolation dampers will be tested as described in the ESBWR DCD, Tier 1, Section 2.16.2 to support the radiological consequence analysis performed in Chapter 15 of this report. During normal plant operation, the potentially contaminated areas of the RB are maintained at a slightly negative pressure relative to adjoining areas by a non-safety-related RB HVAC system. Following a postulated DBA, the RB HVAC system is automatically isolated. The RB has a design maximum leakage of 50 percent of air mass per day to the environment. The applicant assumed that the releases from the primary containment to the RB would be mixed with the RB air at a mixing efficiency of 40 percent. The applicant stated that the RB envelope is not intended to provide a leaktight barrier against radiological fission product release. The applicant further stated that the RB is capable of periodic testing to ensure that the leakage rates assumed in the radiological consequence analyses are met.

Suppression Pool

The ESBWR design provides, among other things, a suppression pool to condense steam and remove fission products following a postulated DBA. The applicant evaluated the radiological consequences for the DBA in Section 15.4.4, "Loss-of-Coolant Accident Inside Containment Radiological Analysis," postulating the following three LOCA scenarios:

(1) RPV bottom drainline break with ADS operating and degraded low-pressure makeup system

(2) RPV bottom drainline break with ADS failure and with degraded high-pressure makeup system

(3) loss of preferred power with ADS operating and with degraded low-pressure makeup system

The sequences of these three accident scenarios include, among other things, operation and availability of the suppression pool as a passive fission product control and removal system. Two of the accident scenarios evaluated (accident scenarios 1 and 2) involve the reactor bottom drainline breaks. The pipe breaks result in a blowdown of the RPV liquid and steam to the drywell via the severed pipe. The resulting pressure buildup drives the mixture of steam, water, and other gases down through vents to the downcomers and into the suppression pool water, thereby condensing the steam and reducing the pressure. Because of the postulated loss of core cooling, the fuel heats up and melts, resulting in the release of fission products. The fission product release occurs in phases over a 2-hour period. Significant quantities of fission products would not be part of the initial blowdown to the suppression pool. Subsequent fission product releases from reactor safety valves to the suppression pool would remove some fission products by the suppression pool water. The applicant assumed a decontamination factor of 10 for any particulate fission products and for iodine in elemental form.

Control Room Emergency Filter Unit

In DCD Tier 2, Revision 3, the applicant described the control room emergency filter unit (CREFU), which is an ESF atmosphere cleanup system to prevent the intrusion of fission products into the main CRHA and to pressurize the control room with nonradioactive outside air following postulated DBAs. The CREFU, a subsystem of the control building HVAC system, is a safety-related system and is located in the control building. The control building is designed to seismic Category 1 criteria. The CREFU replaces the passive control room emergency air breathing system provided in previous revisions of the DCD.

The CREFU consists of two redundant trains, and each train has a prefilter, HEPA filter, 4-in-deep charcoal adsorber, and postfilter to remove fission products and to pressurize the control room to prevent any in-leakage of radioactive material into the control room following postulated DBAs. Two redundant trains, which are physically and electrically redundant and separated, provide single active failure protection for the CREFU. The CREFU equipment and components are designed to seismic Category 1 and are located in a seismic Category 1 structure. The CREFU trains are operable during loss of preferred power, loss of onsite ac power, or SBO, and they are designed, constructed, and tested to meet the requirements of RG 1.52. The system will be automatically activated by high radioactivity in the main control room air supply duct or can be activated manually from the main control room.

Drywell Spray System

The ESBWR design includes a non-safety-related drywell spray system for severe accident management to aid in postaccident recovery or to mitigate the effects of a severe accident. The non-safety-related drywell spray system is not credited for removal of fission products in the radiological consequence evaluation.

Section 6.2.1 of this report addresses the staff's evaluation of containment performance.

Section 6.2.3 of this report addresses the staff's evaluation of the applicants assumptions related to RB leakage and mixing and the RB functional design.

Section 15.4.3 of this report presents the staff's evaluation of the removal of fission products by the PCCS as a means for meeting the radiological consequence evaluation factors in 10 CFR 50.34 (a)(1) and GDC 19. Section 6.2.2 of this report provides the staff's evaluation of the PCCS.

In performing its independent confirmatory radiological consequence analysis, the staff used the MELCOR computer code, along with the ESBWR design specifics, to estimate fission product transport and removal by these passive systems. Section 15.4.3 of this report presents the staff's evaluation on the removal of fission products by these passive systems and structures as a means for meeting the radiological consequence evaluation factors in 10 CFR 50.34(a)(1) and GDC 19.

Sections 6.4 and 9.4.1 of this report provide the staff's evaluation of whether the CREFU meets the requirements of GDC 40, 41, and 61. Section 15.4.3 of this report summarizes the radiological consequence analysis using the CREFU for the control room habitability following postulated DBAs as a means for meeting the radiological consequence evaluation factors in GDC 19.

6.5.4 Conclusions

Because of the open items that remain to be resolved in Sections 6.2.1, 6.2.2, 6.2.3, 6.4, and 15.4.3 of this report, the staff was unable to finalize its conclusions regarding acceptability for this section.

6.6 Preservice and Inservice Inspection and Testing of Class 2 and 3 Components and Piping

6.6.1 Regulatory Criteria

The staff reviewed DCD Tier 2, Revision 3, Section 6.6, in accordance with Section 6.6, “Inservice Inspection of Class 2 and 3 Components,” of the SRP. This SRP section states that the requirements for periodic inspection and testing of Class 2 and 3 systems in GDC 36, 37, 39, 40, 42, 43, 45, “Inspection of Cooling Water System,” and 46, “Testing of Cooling Water System,” are specified in 10 CFR 50.55a and detailed in Section XI of the ASME Code as described below.

- Compliance with 10 CFR 50.55a requires preservice and periodic inspection and testing requirements of the ASME Code for Class 2 and 3 systems and components.
- Compliance with GDC 36 requires that the design of the ECCS permits appropriate periodic inspection of important safety components, such as spray rings in the RPV.
- Compliance with GDC 37 requires that the design of the ECCS permits appropriate testing to ensure structural integrity, leaktightness, and the operability of the system.
- Compliance with GDC 39 requires that the design of the containment heat removal system permits inspection of important components, such as the torus and spray nozzles, to assure the integrity and capability of the system.
- Compliance with GDC 40 requires that the design of the containment heat removal system permits appropriate periodic pressure and functional testing to assure the structural and leaktight integrity of its components, the operability and performance of the active components of the system, and the operability of the system as a whole.
- Compliance with GDC 42 requires that the design of the containment atmospheric cleanup system permits appropriate periodic inspection of components such as filter frames and ducts to assure integrity and capability of the system.
- Compliance with GDC 43 requires that the design of the containment atmospheric cleanup system permits appropriate periodic pressure and functional testing to assure the structural integrity of components and the operability and performance of active components of the system, such as fans, filters, and dampers.
- Compliance with GDC 45 requires that the design of the cooling water system permits appropriate periodic inspection of important components such as heat exchangers, to assure the integrity and capability of the system.
- Compliance with GDC 46 requires that the design of the cooling water system permits appropriate pressure and functional testing to assure the structural and leaktight integrity of it components, the operability and the performance of the active components of the system, and the operability of the system as a whole.

The inservice inspection (ISI) program for ASME Class 2 and 3 components relies upon these design provisions to allow performance of an ISI. Compliance with these GDC ensures that the design of the safety systems will allow accessibility of important components so that periodic inspections can be performed to detect degradation, leakage, signs of mechanical or structural distress caused by aging, and fatigue or corrosion before the ability of these systems to perform their intended safety functions is jeopardized.

6.6.2 Summary of Technical Information

DCD Tier 2, Section 6.6, states that the ESBWR meets the requirements for periodic inspection and testing of Class 2 and 3 systems in GDC 36, 37, 39, 40, 42, 43, 45 and 46, as specified, in part, in 10 CFR 50.55a and as detailed in Section XI of the ASME Code. The ESBWR meets SRP Section 6.6, Revision 1, acceptance criteria by conforming to the ISI requirements of the aforementioned GDC and 10 CFR 50.55a for the areas of review described in Subsection I of the SRP.

The applicant stated that all items within the Class 2 and 3 boundaries are designed to provide access for the examinations required by ASME Code, Section XI, Subarticles IWC-2500 and IWD-2500. The COL holder bears the responsibility for designing components for accessibility for preservice and ISI.

The physical arrangement of piping, pumps, and valves provides personnel access to each weld location for performance of ultrasonic and surface (magnetic particle or liquid penetrant) examinations and sufficient access to supports for performance of visual (VT-3) examination. Working platforms are provided in some areas to facilitate the servicing of pumps and valves. Removable thermal insulation is provided on welds and components that require frequent access for examination or are located in high radiation areas. Weld locations are designed to permit ultrasonic examination from at least one side, but where component geometry permits, access from both sides is provided.

The personnel performing examinations shall be qualified in accordance with ASME Code, Section XI, Appendix VII. Ultrasonic examination systems shall be qualified in accordance with an industry-accepted program for implementation of ASME Code, Section XI, Appendix VIII. Circumferential welds in high-energy piping between the containment isolation valves shall be 100-percent volumetrically examined each inspection interval.

Piping systems that are ASME Code, Section III, Code Class 1, 2, 3, as well as non-safety-related piping, and components described in NRC GL 89-08, “Erosion/Corrosion-Induced Pipe Wall Thinning,” dated May 2, 1989, that are determined to be susceptible to erosion/corrosion shall be subject to a program of nondestructive examination (NDE) to verify a system’s structural integrity. The examination schedule and examination methods shall be determined in accordance with the Electric Power Research Institute (EPRI) guidelines in NSAC-202L-R2, which satisfy NRC GL 89-08, or the latest revision approved by the NRC (or an equally effective program) and the applicable rules of ASME Code, Section XI.

The COL holder will be responsible for the development of the site-specific preservice and ISI program plans, which will be based on the ASME Code, Section XI, edition and addenda specified in accordance with 10 CFR 50.55a. The COL holder shall specify the edition of the ASME Code to be used, based on the date of issuance of the construction permit or license, per 10 CFR 50.55a. The requirements presented in DCD Tier 2, Section 6.6, are provided for information and are based on the 2001 Edition of ASME Code, Section XI, with 2003 Addenda. This is identified as a COL action item in DCD Tier 2, Section 6.6.11.

6.6.3 Staff Evaluation

The staff's evaluation of the ISI program description of ASME Code Class 2 and 3 components is contained in the following six sections—(1) components subject to inspection, (2) accessibility, (3) examination categories and methods, (4) evaluation of examination results, (5) system pressure tests, and (6) augmented ISI to protect against postulated piping failure.

6.6.3.1 Components Subject to Inspection

The definitions of ASME Code Class 2 and 3 components and systems subject to an ISI program are acceptable if they agree with the NRC quality group classification system (RG 1.26) or the definitions in Article NCA-2000 of Section III of the ASME Code. Section 3.2.2 of this report contains the staff's evaluation of the applicant's classification of components.

6.6.3.2 Accessibility

The applicant indicated that, in the ESBWR design, all items within the Class 2 and 3 boundaries are designed to provide access for the examinations required by ASME Code, Section XI, Subarticles IWC-2500 and IWD-2500. The applicant also indicated that the COL holder is responsible for ensuring accessibility for preservice and ISI. ASME Code, Section XI, Subarticle IWA-1500 provides several considerations regarding accessibility of components for inspection.

The staff issued several RAIs (6.6-1, 6.6-2, 6.6-3, 6.6-4, 5.2-51, 5.2-53, 5.2-54, 5.2-57, and 5.2-58) regarding the accessibility of components to perform inspections required by ASME Code, Section XI, and 10 CFR 50.55a. The staff developed RAI 5.2-62, which supercedes the aforementioned RAIs, regarding the accessibility and inspectibility of welds and components. In RAI 5.2-62, the staff requested that the applicant modify the DCD to (1) specify the inspection methods that are practical to use for ISI of welds in ASME Code Class 1 and 2 austenitic and dissimilar metal welds and (2) add COL action items to Sections 5.2.4 and 6.6 for COL applicants to ensure that a COL applicant referencing the ESBWR will provide a detailed description of its plans to incorporate, during design and construction, access to piping systems to enable NDE of such welds during ISI.

By way of background, the staff understands that materials selected for use in the ESBWR ASME Code, Class 1 and 2, austenitic and dissimilar metal welds are not expected to encounter SCC or an appreciable amount of other forms of degradation based on currently available information. However, the staff notes that SCC was not expected in previously built pressurized-water reactors and BWRs based on information that was available at the time of their licensing and construction. Accordingly, the staff considers that the design of components should include provisions to enable NDE to detect future component degradation, such as SCC. This is a critical attribute of any new reactor design.

ASME Code, Section XI, as incorporated into 10 CFR 50.55a(g), currently allows for either ultrasonic or radiographic examination of welds in ASME Code Class 1 and 2 piping systems. The staff requested that the applicant modify the DCD in Tier 1 to state that one or both of these types of examination are practical for ISI of austenitic and dissimilar metal welds. The staff notes that ultrasonic examination has advantages with respect to ALARA considerations and, with this change to the DCD any design certification rule that might be issued for the ESBWR will preclude the granting of relief under 10 CFR 50.55a(g)(6) for ISI of such welds. The staff requested that the applicant confirm that austenitic or dissimilar metal welds in ASME Code Class 1 and 2 piping systems will be accessible for examination by either ultrasonic or radiographic examination in accordance with the requirements of 10 CFR 50.55a(g)(3).

In support of these DCD changes, a COL applicant referencing the ESBWR design certification application should inform the staff of how it plans to meet all access requirements during construction and operation as required by 10 CFR 50.55a(g)(3)(i) and (ii). The staff notes that the preservice inspection requirements are known at the time a component is ordered, and 10 CFR 50.55a(g) does not contain provisions for consideration of relief requests for impractical examination during the construction phases of the component. The COL action items requested above should reflect these considerations. The staff identified this issue in RAI 5.2-62. **RAI 5.2-62 is being tracked as an open item.**

The staff approval of the applicant's design for accessibility of ASME Code Class 2 and 3 systems is pending the resolution of Open Items 5.2-62

6.6.3.3 Examination Categories and Methods

The ISI program will follow ASME Code, Section XI, as required by 10 CFR 50.55a. Thus, the examination categories and methods specified in the DCD are acceptable if they are in agreement with the requirements in Subarticles IWA-2000, IWC-2000, and IWD-2000 of Section XI of the ASME Code. The staff will review the COL applicant's description of its ISI program at the time of COL application review.

DCD Tier 2, Section 6.6.3.1 indicates that all of the items selected for inservice examination will receive a preservice examination in accordance with ASME Code, Section XI, Subarticles IWC-2200 and IWD-2200 with the exception of the preservice examinations specifically excluded by the ASME Code. For the aforementioned exception to preservice examination, the applicant provides examples such as the visual VT-2 examinations for Category C-H and D-A.

DCD Tier 2, Section 5.2.4 indicates that the design to perform preservice inspection is based on the requirements of ASME Code, Section XI, as specified in DCD Tier 2, Table 1.9-22. Table 1.9-22 indicates that the above-referenced code is the ASME Code, Section XI, 2001 Edition through the 2003 Addenda. The 2001 edition through the 2003 addenda of ASME Code, Section XI, Subarticle IWD-2200 states that all examinations required by this article (with the exception of Examination Category D-B of Table IWD-2500-1) shall be performed completely, once, as a preservice examination requirement before initial plant startup.

It appears that the applicant has made references to the 1989 edition of ASME Code, Section XI, regarding examination Category D-A. The staff notes that, in other instances, the applicant referenced examination categories from the 1989 ASME Code. In RAI 5.2-56, the staff requested that the applicant update references to examination categories that were apparently referenced from the 1989 ASME Code. Given that GEH has indicated that the information it supplied is based on the 2001 edition through the 2003 addenda of ASME Code, Section XI, the staff requested, in RAI 6.6-8, that GEH modify DCD Tier 2, Section 6.6, to reference the appropriate examination categories for the 2001 edition through 2003 addenda. The staff also requested that the applicant verify that it has reviewed DCD Sections 5.2.4 and 6.6 to ensure that all references to ASME Code, Section XI, are consistent with the 2001 edition through the

2003 addenda. The staff identified this issue in RAI 6.6-8. **RAI 6.6-8 is being tracked as an open item.**

DCD Section 6.6.3.2.6 indicates that personnel performing ultrasonic examinations will be qualified in accordance with ASME Code, Section XI, Appendix VII. Ultrasonic examination systems will be qualified in accordance with an industry-accepted program for implementation of ASME Code, Section XI, Division 1, Appendix VIII. The staff finds this acceptable given that any industry-accepted program is required to meet Appendix VIII requirements in accordance with the implementation requirements of 10 CFR 50.55a.

The staff requested information regarding the ISI requirements for ICs and PCCS heat exchangers (condensers) because it is not clear whether ASME Code requirements are sufficient to ensure that these components will be inspected in a manner that will provide reasonable assurance that degradation that may occur will be detected in a timely fashion and thus prevent component failure. The ICs and PCCS heat exchangers are ASME Code, Section III, Class 2.

In response to RAI 5.4-58, the applicant indicated that modified Alloy 600 has been in service for a number of years, but that it has not currently been inspected as part of a formal ISI program. In RAI 5.4-56, the staff requested that the applicant confirm that the method/technique for inspecting IC tubes is capable of detecting general wall thinning, pit-like defects, and SCC along the entire length of the tube. In response to RAI 5.4-56, the applicant indicated that, because of the size of the IC tubes (nominal pipe size (NPS) of 2), the IC tubes are exempt from volumetric and surface inservice examinations by ASME Code, Section XI, Subarticle IWC-1220 which exempts sizes NPS 4 and smaller. The applicant indicated that the ICs are subject to leakage (VT-2) examination under ASME Code, Section XI. However, visual examination will only indicate whether the degradation has penetrated through wall (which would normally be detected through radiation monitoring techniques). There is a lack of long-term service experience (with inspection results) and the limitations of accelerated corrosion testing to fully simulate the range of variables that may exist in the field (and pertinent to corrosion). Therefore, in supplemental RAI 5.4-58, the staff requested additional information concerning the inspection and acceptance criteria for the IC tubes or justification for why no inspection requirements are needed. Supplemental RAI 5.4-58 also requested that the applicant provide a response that addresses the original RAI-5.4-56 since visual inspections will not indicate whether the IC tubes have degraded by corrosion or mechanical mechanisms unless the degradation has penetrated through wall (at this point, the IC tubes may no longer have adequate integrity). In summary, the staff requested that the applicant provide the inspection and acceptance criteria for the IC tubes and confirm that volumetric inservice examination techniques exist for finding the forms of degradation that may affect the IC tubes. The staff identified these issues in RAI 5.4-56 and RAI 5.4-58. **RAI 5.4-56 and 5.4-58 are being tracked as open items.**

Since the limitations of accelerated corrosion testing also apply to the PCCS heat exchanger tubes, the staff requested similar information for the PCCS heat exchanger. In addition, the staff requested clarification to determine whether the cracking that occurred in earlier ICs could occur in the PCCS heat exchanger. The staff identified this issue in RAI 5.4-57. **RAI 5.4-57 is being tracked as an open item.**

Due to the open items that remain to be resolved for this section the staff was unable to finalize its conclusions regarding acceptability of the applicant's examination categories and methods for ASME Code Class 2 and 3 systems.

6.6.3.4 Evaluation of Examination Results

The ISI program will follow ASME Code, Section XI, as required by 10 CFR 50.55a. GEH indicated that examination results are evaluated in accordance with ASME Code, Section XI, Subarticle IWC-3000 for Class 2 components, with repairs based on the requirements of Subarticle IWA-4000. Examination results are evaluated in accordance with ASME Code, Section XI, Subarticle IWD-3000 for Class 3 components, with repairs based on the requirements of Subarticle IWA-4000. The GEH description of the evaluation of examination results is consistent with ASME Code, Section XI, and meets the acceptance criteria in SRP Section 6.6, Subsection II.5, and is therefore acceptable.

6.6.3.5 System Pressure Tests

DCD Tier 2, Sections 5.2.4.6 and 6.6.6, reference certain portions of ASME Code, Section XI, Subarticles IWA-5000, IWB-5000, IWC-5000, and IWD-5000 in the description of system leakage and hydrostatic pressure tests for ASME Code Class 1, 2, and 3 systems. In RAI 5.2-65, the staff requested that the applicant modify DCD Sections 5.2.4.6 and 6.6 to clarify that system leakage and hydrostatic pressure tests will meet all requirements of ASME Code, Section XI, Subarticles IWA-5000, IWB-5000, IWC-5000, and IWD-5000. The staff identified this issue in RAI 5.2-65. **RAI 5.2-65 is being tracked as an open item.**

6.6.3.6 Augmented ISI to Protect against Postulated Piping Failure

The augmented ISI program for high-energy fluid systems piping between containment isolation valves is acceptable if the extent of ISI examinations completed during each inspection interval provides 100-percent volumetric examination of circumferential and longitudinal pipe welds with the boundary of these portions of piping. DCD Tier 2, Section 6.6.7, indicates that high-energy piping (as defined in Subsection 3.6.2) between containment isolation valves is subject to additional inspection requirements. Circumferential welds shall be 100-percent volumetrically examined each inspection interval. The piping in these areas is seamless, thereby eliminating longitudinal welds. The applicants augmented ISI program to protect against postulated pipe failure is consistent with SRP Section 6.6 and therefore acceptable.

6.6.3.7 Augmented Erosion/Corrosion Inspection Program

As described in GL 89-08, an appropriate long-term monitoring program for potential wall thinning of high-energy piping by erosion/corrosion must be implemented. The applicant has indicated that all piping systems that are ASME Code, Section III, Code Class 1, 2, 3, as well as non-safety-related piping, and components described in GL 89-08 that are determined to be susceptible to erosion/corrosion shall be subject to a program of NDE to verify system integrity. The applicant further stated that examination schedule and methods shall be determined in accordance with EPRI guidelines in NSAC-202L-R2 or the latest revision approved by the NRC (or an equally effective program). The staff finds this acceptable because it meets current NRC guidance. To verify that COL applicants will develop an appropriate long-term monitoring program for potential wall-thinning of high-energy piping by erosion/corrosion before plant startup, the staff requested, in RAI 5.2-64, that the applicant revise DCD Sections 5.2.4 and 6.6 to include a COL applicant action item to provide a detailed description of the preservice/ISI and augmented inspection programs and to provide milestones for their implementation. The staff identified this issue in RAI 5.2-64. **RAI 5.2-64 is being tracked as an open item.**

6.6.3.8 COL Information

DCD Tier 2, Section 6.6.11 states that "The unit specific PSI/ISI Plan includes detailed plant information and is the responsibility of the COL Holder as per Subsection 6.6.10." In RAI 5.2-64, the staff requested that the applicant revise DCD Sections 5.2.4 and 6.6 to include a COL applicant action item to provide a description of the preservice/ISI and augmented inspection programs and to provide milestones for their implementation. The staff is concerned that the GEH reference to the COL holder does not make it clear that the COL applicant must provide a description of its preservice/inservice and augmented inspection programs with commitments for scheduled implementation of those programs identified in the COL application. It is understood that the COL holder will fully develop and implement the actual programs. However, the COL applicant must fully describe the preservice/ISI and augmented inspection programs to allow the staff to make a reasonable assurance finding of acceptability.

6.6.4 Conclusions

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

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