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, 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 in Title 10 of the *Code of Federal Regulations* (10 CFR) 50.55a, "Codes and Standards"; 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, 4, 14, 31, 35, and 41; and Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," to 10 CFR Part 50.

- GDC 1, "Quality Standards and Records," 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, "Environmental and Dynamic Effects Design Bases," 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, "Reactor Coolant Pressure Boundary," 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, "Fracture Prevention of Reactor Coolant Pressure Boundary," requires that the design of the RCPB include sufficient margin to ensure 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 will be minimized.

- GDC 35, "Emergency Core Cooling," 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, "Containment Atmosphere Cleanup," 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 meet 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 prevention or mitigation of 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. DCD, Tier 2, Chapter 6, identifies the ESF systems, which include (1) fission product containment and containment cooling systems, (2) emergency core cooling systems (ECCSs), and (3) control room habitability systems.

The applicant has provided a Tier 2 description of the ESF systems materials in Section 6.1.1, summarized here in part as follows:

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

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 for the 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.

DCD Section 6.1.1.2 states that all materials of construction used in essential portions of ESF systems are resistant to corrosion, both in the medium contained and the external environment.

DCD Section 6.1.1.2 also states that general corrosion of all materials, except carbon and low-alloy steel, is negligible and conservative corrosion allowances are provided for all exposed surfaces of carbon and low-alloy steel.

ESBWR core cooling water and containment sprays employ demineralized water with no additives, as stated in DCD, Tier 2, Section 6.1.1.2. DCD, Tier 2, Section 9.2.3, describes the water quality requirements. The applicant contends that leaching of chlorides from concrete and other substances is not significant and no detrimental effects occur on any of the ESF construction materials from allowable containment levels in the high-purity water. Thus, the applicant concludes that materials are compatible with the post-LOCA environment.

As described in DCD, Tier 2, Section 6.1.1 the ESBWR design conforms to 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”
- Generic Letter (GL) 88-01, “NRC Position on IGSCC in BWR Austenitic Stainless Steel Piping,” dated 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*

6.1.1.3.1 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 the applicable provisions of Section III, Divisions 1 or 2, 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 ASME Code, Section III, and therefore will only be materials that appear in ASME Code, Section III, 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. Insert concluding sentence on RAI resolution

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 staff considered this RAI response to be incomplete. In Supplement 1 of 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, asking that the applicant modify DCD, Tier 2, Tables 5.2-4 and 6.1-1 to include the correct weld filler material classifications.

The applicant's proposed revision to DCD, Tier 2, 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 DCD, Tier 2, 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 that the applicant 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 DCD, Tier 2, Table 5.2-4 if it does not apply.

The applicant's proposed revision to DCD, Tier 2, 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, asking the applicant to provide the complete GE-Hitachi Nuclear Energy Americas LLC (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 identified the above issues regarding weld filler metal specifications and P numbers as RAI 6.1-2. RAI 6.1-2 was being tracked as an open item in the safety evaluation report (SER) with open items.

In its response, the applicant indicated that it would modify Tables 6.1-1 and 5.2-4 to delete obsolete filler material classifications, delete references to P5C Group 1 materials, and delete E8018-G filler material classifications. The staff reviewed the ESBWR DCD, Tier 2, Revision 5, and verified that the appropriate modifications were made. Based on the applicant's response, RAI 6.1-2 was resolved.

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. The applicant responded that the material of construction for the IC heat exchanger tubes will be modified SB-167 in accordance with Code Case N-580-1. The staff confirmed that the applicant had appropriately modified DCD, Tier 2, Table 6.1-1. RG 1.84 endorses Code Case N-580-1 for use, without conditions. The staff therefore finds this acceptable. RAI 5.4-20(D) regarding IC materials specifications was resolved.

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. The applicant responded by letter dated February 15, 2008, and indicated that although the hardware has not yet been fabricated, GEH will perform a qualification of induction bent tubing. For tubes that will be subjected to induction bending after solution annealing, a qualification sample of the material will be subjected to mechanical testing (including yield, ultimate strength, and percent elongation). The acceptance criteria for this testing will be the mechanical properties listed in the material specification. Verification that testing is performed will be completed as part of DCD, Revision 7, Tier 1, "ITAAC for The Isolation Condenser System," ITAAC 2a3, Table 2.4.1-3. The staff finds this acceptable because the applicant will provide a testing program for induction-bending operations that will ensure that the mechanical properties of the IC tubes required by the ASME Code will be acceptable following bending operations.

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) information item to require submittal of this information. The applicant responded by letter dated February 15, 2008, and stated that an ASME Code design specification, as well as a design report, will be available at the plant site for review. In addition, the applicant stated that crevices have been eliminated to the extent possible in the IC design. The applicant therefore believes that no COL information item is needed. The actual IC system operation will be less than 1,000 hours. The staff notes that the applicant indicated, by letter dated January 16, 2008, that the normal operating temperature of the IC pool is less than 65 degrees Celsius (°C). Given that the normal operating temperature of the IC pool is relatively low, the amount of

operating time is less than 1,000 hours, crevices have been eliminated to the extent possible in the IC design, and the IC pool is demineralized water with controlled impurity limits, the staff considers the likelihood of any significant degradation to be minimal. The staff therefore finds the applicant's decision not to include the aforementioned COL information item acceptable. RAI 5.4-20 was being tracked as an open item in the SER with open items. Based on the applicant's response, RAI 5.4-20 was resolved.

In RAI 6.1-17, the staff requested that the applicant modify the containment liner materials listed in Table 6.1-1 to be consistent with the liner materials listed in DCD, Tier 2, Section 3.8. The applicant responded and modified Table 6.1-1 to reference DCD, Tier 2, Section 3.8, for materials used for the containment vessel liner plate, penetrations, gravity-driven cooling system (GDCS) pool liner, and suppression pool liner. The staff reviewed the materials for the above components and verified that they are permitted for use in accordance with ASME Code, Section III, with the exception of American Society for Testing and Materials (ASTM) A709 HPS 70W, which is not listed as a permitted material specification in accordance with ASME Code, Section III, Division II, Article CC-2000. The applicant indicated that it intends to use this material in accordance with ASME Code Case N-763 for the containment liner and structural attachments welded to the containment liner. Code Case N-763 has gone through the ASME Committee approval process and has been found acceptable. ASTM A709 HPS 70W is a high-performance quenched and tempered weathering steel that is widely used in the fabrication of steel bridges. This material has high toughness in the as-welded condition and exhibits good resistance to corrosion when exposed to atmospheric conditions. The staff notes that ASTM A709 HPS 70W steel is currently permitted for use by American National Standards Institute/American Institute of Steel Construction (ANSI/AISC) N690, "Specification for the Design, Fabrication, and Erection of Steel Safety-Related Structures for Nuclear Facilities." Based on the above-listed considerations, the staff finds that the use of A709 HPS 70W is acceptable for its intended use. Based on the applicant's response, RAI 6.1-17 was resolved.

The staff finds that the ESF materials 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.

6.1.1.3.2 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, 1.37, and 1.44. Designs must also provide 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 fabrication and processing of austenitic stainless steels, as well as conformance to 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 to 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 because cold work will be controlled by the applicant during fabrication by applying limits in

hardness, bend radii and the surface finish on ground surfaces which will reduce the susceptibility of components to stress corrosion cracking.

6.1.1.3.3 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," 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.

6.1.1.3.4 Dissimilar Metal Welds

The applicant described 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.

6.1.1.3.5 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 it 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.

6.1.1.3.6 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 and 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 2. 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 2.

To meet the requirements of GDC 4, 14, and 41, the plant design should control the water used in the ESF to ensure 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 used for reactor shroud support and stub tubes (see the 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 S01, the staff requested that the applicant discuss whether there have been any other "incidents" associated with the use of these materials in these applications. The applicant responded by letter dated January 7, 2008, and indicated that a review of IC industry experience did not identify any incidents associated with the use of Alloy 600 material. RAI 5.4-55 was being tracked as an open item in the SER with open items. Based on the applicant's response, RAI 5.4-55 was resolved.

6.1.1.3.7 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, Revision 8, "GE Nuclear Energy Quality Assurance Program Description," Class I (nonproprietary), dated March 31, 1989, documents the alternative that the applicant may use. The alternative involves using methods, other than mechanical ones, 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.

6.1.1.3.8 Thermal Insulation

The type of thermal insulation used in the ESBWR containment will be primarily metallic and metal-encapsulated insulation, except for the 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 levels practical. 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. The applicant has stated that it will follow the guidance in RG 1.36, and the staff finds this acceptable as it will meet the requirements of GDC 1, 14, and 31.

6.1.1.4 *Conclusions*

Based on its review of the information provided by GEH, the staff concludes that 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,. 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

DCD, Tier 2, Revision 6, states that all field-applied epoxy coatings inside containment will meet the requirements of RG 1.54 and are qualified using the standard ASTM tests, as applicable to procurement, installation, and maintenance.

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 D3842, "Selection of Test Methods for Coatings for Use in Light Water Nuclear Power Plants"; ASTM D3911, "Evaluating Coatings Used in Light Water Nuclear Power Plants at Simulated Design Basis Accident (DBA) Conditions"; and ASTM D5144-00, "Standard Guide for Use of Protective Coating Standards in Nuclear Power Plants."

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 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 D5144-00, as applicable.

The applicant also stated that not all coatings inside containment will meet the criteria of RG 1.54, Revision 1, and ASTM D5144-00. The exceptions are for small 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 D5144-00 and RG 1.54, Revision 1, as per SRP 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 COL Information Item 6.1-1-A in DCD, Tier 2, Revision 3, Section 6.1.3.1.

Because the amount of organic materials does not meet the requirements of RG 1.54 and will not be available before the procurement of the components, the staff requested, in RAI 6.1-16, that the applicant revise the DCD (including addressing a COL information item) to ensure that the COL applicant provides a bounding value for the amount of unqualified coatings and the assumptions used to determine this bounding value. In Revision 5 of DCD Tier 1, the applicant deleted COL Information Item 6.1-1-A and revised the DCD to specify that all field-applied epoxy coatings inside containment will meet the requirements of RG 1.54 and that the coatings are qualified using the standard ASTM tests. In addition, consistent with the rationale of RG 1.54, the wetwell and attendant vertical vents are designated as a Service Level I area. All surfaces and equipment in this area are either uncoated, corrosion-resistant stainless steel, or coated in accordance with RG 1.54 and referenced ASTM standards, as applicable. The staff finds Revision 5 of the DCD acceptable because all field-applied epoxy coatings inside containment will meet the requirements of RG 1.54 and the coatings are qualified using the standard ASTM tests. Based on the applicant's response, RAI 6.1-16 was resolved.

6.1.2.4 *Conclusions*

The staff concludes that the protective coating systems and their applications are acceptable and meet the requirements of Appendix B to 10 CFR Part 50. This conclusion is based on the applicant having met the QA requirements of Appendix B to 10 CFR Part 50, as the coating systems and their applications will meet the requirements of RG 1.54, Revision 1. By meeting the recommendations in RG 1.54, Revision 1, the COL applicant will have evaluated the suitability of the coatings to withstand a postulated DBA environment, in accordance with NRC accepted practices and procedures.

6.2 <u>Containment Systems</u>

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, in accordance with SRP Section 6.2.1, Revision 3, issued March 2007; SRP Section 6.2.1.1.C, Revision 7, issued March 2007; and SRP Section 6.2.1.3, Revision 3, issued March 2007.

In accordance with SRP Section 6.2.1.1.C, Revision 7, 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, “Environmental and Dynamic Effects Design Bases,” 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 post accident 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 Code, Section III, Subsection NE, to ensure 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 technical specification (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 0.35 percent by weight per day at the calculated peak containment pressure related to the 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, Tier 2, 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, GDCS

pools and piping, passive containment cooling system (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 leaktight connections.

The drywell, which has a net free volume of 7,206 cubic meters (m^3) (254,500 cubic feet (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 kilopascals gauge (kPa(g)) (45 pounds per square inch gauge (psig)) and 171 degrees C (340 degrees F), respectively. The design drywell-wetwell pressure differences (i.e., drywell pressure being higher or lower than wetwell pressure) are +241 kilopascals 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 5,350 m^3 (188,900 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 degrees C (250 degrees F).

- 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 leaktight 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 1.2 meters (m) (3.9 feet (ft)), each containing 3 horizontal vents of a nominal diameter of 0.70 m (2.3 ft), 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-millimeter (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 backflooding 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 each vacuum breaker, pneumatically operated fail-as-is safety-related isolation valves are provided to isolate a leaking (not fully closed) or stuck open vacuum breaker. 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.

Vacuum Breakers. 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 backflooding of the suppression pool water into the drywell. The vacuum breaker is provided with redundant proximity sensors to detect its closed position. One out of the three vacuum breakers is required to perform the vacuum relief function. The third vacuum breaker provides redundancy, while the second vacuum breaker provides single-failure protection for opening. On the upstream side of each vacuum breaker, a pneumatically operated fail-as-is safety-related isolation valve is provided to isolate a leaking or stuck-open vacuum breaker. 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, Tier 2, 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 2 square centimeters (cm^2) ($A/\sqrt{K}$), as specified in TS Surveillance Requirement (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 staff based its containment functional evaluation 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, as described in NEDC-33083P-A, "TRACG Application for ESBWR," issued March 2005, and NEDE-32176P, "TRACG Model Description," issued January 2008. The staff's safety evaluation in Section 4 of NEDC-33083P-A contains items needing confirmation during the ESBWR design certification stage. The staff addresses these confirmatory items in the "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."

DCD, Tier 2, 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. DCD, Tier 2, 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 DCD, Tier 2, Table 6.2-1, for containment performance evaluation. DCD, Tier 2, Table 6.2-6, provides the nominal and

bounding values for the plant initial and operating conditions for evaluating the containment performance.

Using the nominal initial and operating conditions listed in Table 6.2-1, GEH evaluated four cases, the three liquid line break cases and the steamline break case. 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 show 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. DCD, Tier 2, 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 to within design values.

The following events may cause containment depressurization:

- Post-LOCA drywell depressurization is caused by the ECCS (e.g., GDCS, control rod drive (CRD) system) 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 show 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 9.67E-2 square meters (m^2) (1.041 square feet (ft^2)). The calculation also assumed a drywell spray flow rate of 127 m^3/hour (h) (560 gallons per minute (gpm)) at a temperature of 293 K (67.7 °F) which is conservatively initiated when the drywell pressure has peaked just before opening of the vacuum breakers.

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 SRVs, 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 SRV 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 SRV in-plant confirmatory test program
- the evaluation of analytical models used for containment analysis

DCD, Tier 2, Revision 4, 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. In RAI 6.2-175, the staff requested that GEH add this information to the DCD. RAI 6.2-175 was being tracked as an open item in the SER with open items. In response to RAI 6.2-175, GEH added Appendix E to DCD, Tier 2, Revision 5, to provide the chronology of progression of a LOCA as predicted by TRACG containment analysis. This addressed the staff's concern. The staff's evaluation of TRACG LOCA containment analysis and staff's confirmatory analysis are described later in this section. RAI 6.2-175 was resolved.

Table 6.2-1 of this report reproduces DCD, Tier 2, Table 6.2-6. DCD, Tier 2, Revision 4, 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. In RAI 6.2-174, the staff asked GEH to define NWL and provide its value. In its response, GEH defined NWL as "normal water level" and added a footnote to DCD, Tier 2, Revision 5, Table 6.2-6, stating that the NWL value is provided in DCD, Tier 2, Revision 5, Table 15.2-1. The staff confirmed that this information was incorporated in DCD, Tier 2, Revision 5.

RAI 6.2-174 was being tracked as an open item. Based on the applicant's response, RAI 6.2-174 was resolved.

Table 6.2-1 of this document shows the major plant initial and operational parameters used in the containment analysis.

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 (15.8 ft)	4.8 m (15.8 ft)
4	PCC pool temperature	43.3 °C (110 °F)	43.3 °C (110 °F)
5	Drywell pressure	101.3 kPa (14.7 psia)	106.9 kPa (15.5 psia)
6	Drywell temperature	46.1 °C (115 °F)	46.1 °C (115 °F)
7	Wetwell pressure	101.3 kPa (14.7 psia)	106.9 kPa (15.5 psia)
8	Wetwell temperature	43.3 °C (110 °F)	43.3 °C (110 °F)
9	Suppression pool temperature	43.3 °C (110 °F)	43.3 °C(110 °F)
10	GDCS pool temperature	46.1 °C(115 °F)	46.1 °C(115 °F)
11	Suppression pool level	5.45 m (17.9 ft)	5.50 m (18.1 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 (1 ft)
16	RPV Dome Vapor and Saturation Temperature	287.4°C (549.3°F)	288.4°C (551.0°F)
17	RPV Lower Plenum Liquid Temperature	272.3°C (522.2°F)	272.2°C (522.0°F)

* NWL—Normal Water Level, 20.72 m (815.7 in.)

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 ensure 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 according to 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 each vacuum breaker, a pneumatically operated fail-as-is safety-related isolation valve 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, but it was also necessary that the information be added to the DCD. RAI 6.2-99 was being tracked as an open item in the SER with open items. The staff confirmed that the information was incorporated in DCD, Tier 2, Revision 4, Table 6.2-1, which addressed the staff's concern. Based on the applicant's response, RAI 6.2-99 was resolved.

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 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 its response by 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, having two versus three vacuum breakers open was expected to have a minimal impact on the PCCS performance in the long term and thus on the maximum containment pressure. The applicant's response addresses the staff's concern and is acceptable because the applicant correctly described the effect of two versus three vacuum breakers opening. RAI 6.2-142 was resolved.

<u>Steam Bypass of the Suppression Pool</u>. The potential exists for steam to bypass the suppression pool by various leak paths, primarily through the vacuum breakers. 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 hours 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 asked GEH to add this information to the DCD. In its response by 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/√K). Therefore, the bypass leakage of 100 cm^2 (0.107 ft^2) (A/√K) is no longer limiting, and a DCD update is not needed. The applicant's response addresses the staff's concern and is acceptable because the staff's confirmatory analysis confirms the applicant's conclusions in Appendix E to DCD Tier 2. RAI 6.2-147 was resolved.

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/√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/√K), respectively (SRP Section 6.2.1.1.C, Revision 6, issued 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/√K) (i.e., 0.1 cm^2 (A/√K)). The staff was concerned that this may be a low value for bypass leakage, which plants may find difficult to confirm. Therefore, in RAI 6.2-145, the staff asked GEH to verify that plants will be able to measure such a low bypass leakage value.

In response to RAI 6.2-145, GEH proposed an alternative 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 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 was unacceptable and requested that GEH propose an acceptable bypass leakage acceptance criterion. RAI 6.2-145 was being tracked as an open item in the SER with open items.

In its response to RAI 6.2-145 by letter dated April 18, 2008, GEH proposed (1) to increase the acceptance criterion for the suppression pool bypass leakage test to a value less than or equal to 1 cm^2 (1.08×10^{-3} ft^2) (A/√K), which amounts to 50 percent of the design-basis bypass leakage value, and (2) to increase the frequency of the overall suppression pool bypass leakage test to be the same as the integrated leak rate test (ILRT) frequency. GEH stated that General Electric established 10 percent of the containment capacity as the acceptance criterion for the suppression pool bypass leakage test during licensing of the initial pressure suppression containments in the early 1970s for BWRs with active ECCSs. GEH stated that the value of 10 percent of containment capability was intended to leave sufficient margin for increases in bypass leakage between outages, and it was chosen, in part, because of the limited amount of field-testing experience and data and the large number of penetrations through the diaphragm floor of the Mark II containment. In support of its position, GEH provided bypass leakage test data for Mark II containments.

These data show that, for each plant, the measured bypass leakages are significantly less than the surveillance test acceptance criteria. These data also show that plants have measured significantly lower bypass leakages than the leakage proposed for the ESBWR. In addition, each ESBWR vacuum breaker consists of an upstream isolation valve, which can isolate a leaking vacuum breaker during a LOCA upon detecting the leakage. Vacuum breakers are equipped with temperature gauges for detecting a leakage. Therefore, the staff determined that the bypass leakage surveillance criterion of 50 percent of the design value proposed is acceptable for the ESBWR.

When proposing in its letter dated April 18, 2008, to increase the overall suppression pool bypass leakage test frequency to the same frequency as the ILRT, GEH stated that this frequency was similar to that employed at the following operating BWRs with Mark II containments: Columbia Generating Station, Nine Mile Point Unit 2, Susquehanna Units 1 and 2, and Limerick Units 1 and 2. Since the extensions to test frequency for the above plants were approved based on plant-specific data, the staff requested that GEH provide additional justification for the proposed change for the ESBWR. Instead, in a letter dated April 27, 2009, GEH changed the overall suppression pool bypass leakage test frequency to once every 24 months and made appropriate changes to the DCD.

RAI 6.2-145 was being tracked as an open item. The applicant's response is acceptable because the staff agrees with the applicant's rationale for the 24-month bypass leakage test frequency. Based on the applicant's response, RAI 6.2-145 was resolved.

DCD, Tier 2, Revision 2, Section 6.2.1.1.2 states that "[o]n the upstream side of the vacuum breaker, a DC solenoid operated isolation valve designed to fail-close is provided." The vacuum breaker isolation valve (VBIV) provides a safety function of closing a leaking vacuum breaker. A vacuum breaker leaking at a rate higher than its design leakage value would cause steam to leak from the drywell to the wetwell bypassing the suppression pool at a rate higher than the design steam leakage value. Steam that enters the wetwell bypassing the suppression does not get condensed by the suppression pool and raises the wetwell pressure and eventually the drywell pressure. In RAI 6.2-148 staff asked GEH to state the type of isolation valve and how the fail-close function is provided.

In response GEH stated the following. VBIV is a pneumatically operated *fail-as-is* safety-related valve that isolates a leaking or stuck open vacuum breaker. Both the vacuum breaker and VBIV are located in the drywell side of the diaphragm floor. The VBIV valve type will be of similar design to a triple offset metal-seated butterfly valve. Automatic actuation logic will close the VBIV based upon an open indication provided by the vacuum breaker proximity sensors with temperature confirmation or indication of bypass leakage provided by temperature sensors. These temperature sensors are located within the cavity of the vacuum breaker/VBIV assembly. Additional temperature sensors are located in close proximity to the vacuum breaker outlets screens and in the drywell and wetwell.

GEH stated that during a LOCA, if a vacuum breaker leaks, these same temperature sensors will detect a decrease in temperature differential between the hot drywell gas leaking past the vacuum breaker seat and the wetwell gas. This will generate a signal to close the VBIV. Proximity sensors located on the vacuum breaker seat can also generate a close signal if they detect a stuck-open VB coincident with a separate temperature confirmation.

GEH's response did not provide information on the limit of bypass leakage that activates the sensors to close the VBIV and the value of temperature differential that activates the sensors. Therefore, in RAI 6.2-148, Supplement 1, the staff asked GEH to provide this information.

In response GEH stated that a vacuum breaker not fully closing, which is considered a single failure, is defined as a bypass leakage area greater than 0.6 cm^2 (0.093 in^2) ($A/\sqrt{K}$). GEH stated that "DCD, Tier 1, Table 2.15.1-2, ITAAC 16b will be changed to a type test to detect bypass leakage from 0.3 cm^2 to 0.6 cm^2 ($A/\sqrt{K}$) using temperature sensors. Detecting leakage starting from 0.3 cm^2 ($A/\sqrt{K}$) assures the setpoint calculation will have margin to the 0.6 cm^2 ($A/\sqrt{K}$) analytical limit to close a VBIV." GEH stated that "[t]he temperature difference value that will activate the sensors will be dependent on the final location of the temperature sensors, the

instrument accuracy of the temperature sensors, and the height of the vacuum breaker seat from the diaphragm floor, which is dependent on the end-to-end dimension of the VBIV."

In RAI 6.2-148, Supplement 2 and 3, staff asked GEH to provide details of the type test and how the setpoint will be determined. In response GEH submitted licensing topical report, NEDE-33564P, "Leakage Detection Instrumentation Confirmatory Test for the ESBWR Wetwell-Drywell Vacuum Breakers," dated March 2010, providing details of the type test and the method of determining the setpoint and agreed to incorporate this report by reference in ESBWR DCD, Tier 2, Revision 8. After reviewing GEH's responses including NEDE-33564P, staff determined that GEH responses addressed staff's concerns and were acceptable. **RAI 6.2-148 is being tracked as a confirmatory item.**

Loss-of-Coolant Accidents. The staff reviewed the information provided in DCD, Tier 2, Section 6.2.1.1 and performed an audit of the GEH containment analysis on December 11 through December 15, 2006. In addition, the staff performed confirmatory containment analyses using the MELCOR computer code that produced qualitative agreement with those of GEH.

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 that 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 addresses 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 needed 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 changes; the impact on containment performance from the modeling changes was minimal. GEH described the behavior of noncondensable gases in the containment adequately. However, GEH did not provide justification for modeling changes. RAI 6.2-52 was being tracked as an open item in the SER with open items.

In a supplemental request to RAI 6.2-52, the staff asked GEH to justify modeling changes and provide the justification and the results of the tieback calculations in the DCD or in a supplement to NEDC-33083P-A. In response, GEH added Appendix B to DCD, Tier 2, Revision 5, justifying modeling changes and providing results of the tie-back calculations. GEH stated that the analysis for the ESBWR containment evaluation followed the application methodology outlines in NEDC-33083P-A and that TRACG nodalization approach in the licensing analysis was similar

to that used in NEDC-33083P-A. GEH stated that this licensing nodalization includes additional features and details. Some of these features were to address the confirmatory items listed in the safety evaluation report of NEDC-33083P-A and others were implemented due to design changes. GEH added Table 6.2-6a to DCD Revision 4 summarizing the list of these changes in the TRACG nodalization. GEH addressed ESBWR design changes which were made after staff evaluated NEDC-33083P-A as described in the corresponding SER. Therefore, the staff determined that RAI 6.2-52 was resolved.

DCD, Tier 2, Revision 1, 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 for the review of the containment performance in response to the limiting DBA. Therefore, in RAI 6.2-53, the staff requested 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 by performing bounding analysis.

Also, because the limiting DBA changed from the FWLB to the MSLB as discussed in RAI 6.2-59 (above), the staff requested in a supplement to RAI 6.2-53 that GEH reanalyze the containment response to MSLB as the limiting DBA. In response, GEH added the results of containment response to the limiting DBA, with respect to the movement of noncondensable gases and mixing and stratification in the containment for the FWLB and MSLB scenarios.

RAIs 6.2-53 and 6.2-98 were being tracked as open items in the SER with open items. The applicant's response addresses the staff's concern and is acceptable because the applicant's treatment of noncondensable gases is bounding. Based on the applicant's responses, RAIs 6.2-59 and 6.2-98 were resolved.

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 hookup 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.

DCD, Tier 2, Revision 1, was not clear as to whether the containment analysis takes credit for the non-safety systems. 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 the 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 CRD system, reactor water cleanup/shutdown cooling system (RWCU/SDC), FAPCS in suppression pool cooling mode, FAPCS in drywell spray mode, and FAPCS in low-pressure coolant injection (LPCI) mode. GEH updated the DCD to include this information. The staff confirmed that the information was incorporated in DCD, Tier 2, Revision 5.

RAI 6.2-57 was being tracked as an open item in the SER with open items. The applicant's response addresses the staff's concern and is acceptable because the applicant's explanation of the treatment of non-safety systems is satisfactory. Based on the applicant's response, RAI 6.2-57 was resolved.

Maximum Containment Pressure

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. The section below titled "Post-72-Hour Containment Pressure Control," discusses this issue.

DCD, Tier 2, Revision 3, Section 6.2.1.1.3.5, states that "the peak drywell pressure for the bounding case is below the containment design pressure." DCD, Tier 2, Revision 3, Table 6.2-5, lists peak drywell pressure and peak wetwell pressure. However, the TRACG analysis results provided in the DCD show no peak drywell or wetwell pressures for the limiting FWLB and MSLB DBAs. Instead, the pressure continues to rise and reaches its maximum value for the duration of analysis at 72 hours as stated above. In RAI 6.2-177, the staff requested that GEH correct this discrepancy. In response to this RAI, GEH changed references to "peak pressure" to "maximum pressure" in DCD, Tier 2, Revision 4. The staff confirmed that the change was incorporated in DCD, Tier 2, Revision 4.

RAI 6.2-177 was being tracked as an open item in the SER with open items. The applicant's response addresses the staff's concern and is acceptable because the applicant revised DCD Tier 2 as requested. Based on the applicant's response, RAI 6.2-177 was resolved.

Single Failures Considered

DCD, Tier 2, Revision 1, 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 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 the ECCS analysis. However, it was not clear whether the single failures considered would bound the single failures affecting the maximum 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.

In response to RAI 6.2-58, GEH stated that to avoid the injection of nitrogen into the reactor vessel, four divisional, safety-related level sensors per SLC accumulator are used to provide automatic isolation of the associated accumulator shutoff valves (two in series) on a low accumulator level signal, using a two-out-of-four voting logic as stated in DCD, Tier 2, Section 7.4.1.2. Therefore, the staff determined that a failure of a shutoff valve in one of the SLCS trains will not cause continuous injection of nitrogen in the pressure vessel and need not be considered as a credible single failure for containment analysis.

RAI 6.2-58 was being tracked as an open item in the SER with open items. The applicant's response is acceptable because the staff determined that the single active failures considered by GEH produced the highest maximum containment pressure. Based on the applicant's response, RAI 6.2-58 was resolved.

Initial Containment Conditions

DCD, Tier 2, Table 6.2-2, lists the average drywell temperature during normal operation as 57.2 degrees C (135 degrees F). However, DCD, Tier 2, Table 6.2-6, lists the initial temperature used in analyzing the containment DBA cases as 46.1 degrees C (115 degrees F). 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 degrees C (115 degrees F) to 57.2 degrees C (135 degrees F). Results from a previous sensitivity study of the simplified boiling-water reactor (SBWR) design (Figure 4.3-2, NEDE-32178, Revision 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 degrees C (115 degrees F) to

ensure conservative (i.e., maximum) calculated peak drywell pressure. GEH agreed to update the DCD to include this response.

RAI 6.2-64 was being tracked as a confirmatory item in the SER with open items. The staff confirmed that the change was incorporated in DCD, Tier 2, Revision 5. Based on the applicant's response, RAI 6.2-64 was resolved.

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.

RAI 6.2-65 was being tracked as a confirmatory item in the SER with open items. The staff confirmed that the change was incorporated in DCD, Tier 2, Revision 5. Based on the applicant's response, RAI 6.2-65 was resolved.

DCD, Tier 2, Table 6.2-2, lists the suppression pool temperature in hot standby as 54.4 degrees C (130 degrees F), while DCD, Tier 2, Table 6.2-6, lists the initial suppression pool temperature used for the DBA analyses as 43.3 degrees C (110 degrees F), which is lower than the hot standby temperature. In RAI 6.2-67, the staff asked 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.

Regarding initial pool temperature, GEH stated that the suppression pool average temperature during normal operation was less than 43.3 degrees C (110 degrees F), and the maximum pool temperature of 43.3 degrees C (110 degrees F) was used in the safety analyses. According to the TS (DCD, Tier 2, Chapter 16), 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 43.3 degrees C (110 degrees F), and the reactor will be switched to shutdown mode immediately when the suppression pool temperature is greater than or equal to 48.9 degrees C (120 degrees F).

Regarding the second concern, the mass and energy releases in the case of a reactor operating at less than 100-percent power are bounded by those for 100-percent power scenarios, and, therefore, are less severe than the limiting DBA case.

RAI 6.2-67 was being tracked as an open item. The applicant's response is acceptable because the applicant's choice of initial suppression pool temperature is consistent with relevant TS. Based on the applicant's response, RAI 6.2-67 was resolved.

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 continues in the long-term heatup, so the flow area restriction is not applied. The staff determined that because of the long-term energy addition to the pool by spillover flow following an FWLB, the exception for FWLB is acceptable. However, the applicant modified the design by removing the spillover pipes and accomplishing the spillover function by spillover horizontal holes, which is reflected in DCD, Tier 2, Revision 3, Section 6.2.1.1.2, thus invalidating the above concern. Therefore, RAI 6.2-55 was resolved.

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. This information is needed for proper review of the TRACG analyses, as well as for the staff's performance of confirmatory containment analysis using the MELCOR computer code. GEH provided the requested information in the revised DCD, Tier 2, Revision 5, Section 6.2.1.3, and added DCD, Tier 2, Table 6.2-12d and Figures 6.2-9e1, 6.2-9e2, 6.2-10e1, and 6.2-10e2.

RAI 6.2-63 was being tracked as an open item in the SER with open items. The applicant's response is acceptable because the applicant revised DCD Tier 2 as requested. Based on the applicant's response, RAI 6.2-63 was resolved.

Previous versions of the DCD did not contain 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 (RSW), GDCS pool structures, RPV support brackets, fine motion CRDs, 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 is 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. In its response to RAI 6.2-69, GEH revised DCD, Tier 2, Table 6.2-6, and DCD, Tier 1, Table 2.15.1-2. The revision specified the maximum and minimum analytical values for drywell and wetwell volumes used in the licensing analyses, and the inspection, test, analysis, and acceptance criteria (ITAAC) ensure that the as-built volumes match or are conservative with respect to the containment performance analysis.

RAI 6.2-69 was being tracked as a confirmatory item in the SER with open items. The staff confirmed that the change was incorporated in DCD Tier 1 and Tier 2, Revision 6. Based on the applicant's response, RAI 6.2-69 was resolved.

Previous versions of the DCD did not include information on how GEH evaluated the various primary system volumes and heat structures (piping, RPV, and others). 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. In RAI 6.2-70, the staff requested that GEH provide this information. In its 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. RAI 6.2-70 was being tracked as a confirmatory item in the SER with open items. The staff confirmed that the changes were incorporated in DCD, Tier 2, Section 6A and Table 6.2-6. Based on the applicant's response, RAI 6.2-70 was resolved.

GDCS Airspace

DCD, Tier 2, Section 6.2.1.1.10.2, states that the GDCS pools are placed above the RPV with their airspace 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 its 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 deleted the statement from the DCD in a later revision, and the staff confirmed the change.

RAI 6.2-152 was being tracked as a confirmatory item in the SER with open items. Based on the applicant's response, RAI 6.2-152 was resolved.

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 asked 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. GEH agreed to add this information in a later revision of DCD Tier 2.

RAI 6.2-54 was being tracked as a confirmatory item in the SER with open items. The staff confirmed that the information was incorporated in DCD, Tier 2, Revision 5, Appendix 6B. Based on the applicant's response, RAI 6.2-54 was resolved.

The original DCD did not provide information on passive heat sinks used in the containment analysis. The staff needed 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. RAI 6.2-62 was being tracked as an open item in the SER with open items. The applicant provided the requested information in Appendix 6D to DCD, Tier 2, Revision 3. Based on the applicant's response, RAI 6.2-62 was resolved.

The applicant 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 needed 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.

RAI 6.2-72 was being tracked as a confirmatory item. The staff confirmed that GEH added the modeling information for the SLCS in Appendix B to DCD, Tier 2, Revision 5. Based on the applicant's response, RAI 6.2-72 was resolved.

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) which critical flow models were used for choked flowpaths. Because the staff needed this information for its review, in RAI 6.2-73, the staff asked GEH to provide this information. In response to part (1) of the request, GEH stated that the upper limit (+2σ) is applied to the bounding short-term peak pressure calculations, and the lower bound (-2σ) is applied in the long-term peak pressure calculations. This response is acceptable because the chosen uncertainty values for the choked flow provide conservative results for accident scenarios, which have bounding short-term or long-term peak pressure. However, as stated above under resolution of RAI 6.2-59, after error corrections in TRACG calculations, no accident scenario showed bounding short-term peak pressures. In RAI 6.2-141 the staff requested that GEH revise all previous responses to the containment-related RAIs, which includes RAI 6.2-73.

In response to part (2) of RAI 6.2-73, 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 side), and drywell main vents. The staff determined that applying the choked flow model to all flow paths was reasonable and acceptable. In its response, GEH agreed to update the DCD to provide information submitted in response to RAI 6.2-73. RAIs 6.2-73 and 6.2-141 were being tracked as open items in the DCD with open items.

In response to RAI 6.2-141, the applicant stated that all of the most recent containment analyses confirmed the MSLB scenario as the bounding case, as documented in DCD, Tier 2, Section 6.2, Revision 3. The staff determined that this response was acceptable because it addressed the staff's concern. RAI 6.2-141 is closed. GEH updated DCD, Tier 2, Revision 5, Section 6.2 to provide information submitted in response to RAIs 6.2-73. RAI 6.2-73 is closed.

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 needed 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 provided graphical results of FWLB, GDCS line break, vessel bottom line break, and MSLB. Each of these cases considered a single failure and nominal conditions given on Table 6.2-6 of DCD, Tier 2, Revision 1, and assumed 100 percent double-ended guillotine break. In its response GEH agreed to include above results in the DCD. After reviewing the results the staff determined that they were acceptable.

However, in its response to RAI 6.2-59, GEH also stated that it had discovered an erroneous result for FWLB (i.e., an early peak in drywell pressure), because the FWLB analysis was sensitive to the time step selection. 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 steps 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 revised the analysis presented in NEDC-33083P-A, reflecting the correction of the error and model enhancement applied to FWLB, GDCS line break, vessel bottom line break, and MSLB, and updated the DCD. The staff determined that GEH's error corrections as described in its response were acceptable. In a supplement to RAI 6.2-59, the staff requested GEH include the input error corrections information in a licensing document. RAI 6.2-59 was being tracked as an open item in the SER with open items.

In response, GEH added the input error corrections information to Appendix B to ESBWR DCD, Tier 2, Revision 4. This addressed the staff's concern. RAI 6.2-59 was resolved.

DCD Tier 2, states that only DEG breaks were analyzed. However, the DCD also states that a spectrum of break 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. 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. 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. After reviewing GEH's response the staff determined that it was acceptable because it addressed the staff's concerns. In a supplement to RAI 6.2-60, the staff requested GEH to incorporate the response into the DCD. RAI 6.2-60 was being tracked as an open item in the SER with open items.

In response GEH added a discussion of spectrum of break sizes and break elevations as DCD, Tier 2, Appendix 6F. This addressed the staff's concerns. RAI 6.2-60 was resolved.

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 balance-of-plant 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.

During an NRC audit conducted December 11–20, 2006, GEH stated that it had made several changes to the TRACG containment model. GEH identified these changes in DCD, Tier 2, Revision 3, Appendix 6A. 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 to perform confirmatory containment performance analysis with the MELCOR computer code. The response is acceptable because the applicant provided the revised results for FWL and MSL cases and modified DCD Tier 2 accordingly. Based on the applicant's response, RAI 6.2-61 was 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 updated the graphs in the DCD to identify cells for which temperatures plotted. The applicant provided the requested information, in support of the staff's confirmatory calculations, and revised the DCD accordingly. Based on the applicant's response, RAI 6.2-71 was resolved.

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. GEH made the requested editorial changes replacing all "GDCS Air Pressures" captions with "GDCS NC Gas Pressures." The staff confirmed that these changes were incorporated in DCD, Tier 2, Revision 7.

RAI 6.2-176 was being tracked as an open item in the SER with open items. Based on the applicant's response, RAI 6.2-176 was resolved.

Post-72-Hour Containment Pressure Control

ESBWR DCD Tier 2 provides 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 pounds per square inch absolute (psia)), which is 29.0 kPa (4.2 psi) below the containment design pressure of 411.7 kPa (59.7 psia) (310 kPa(g) (45 psig)). 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 the containment and its associated systems to accommodate, without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions resulting from any LOCA.

The staff's concern about the long-term cooling capability was the subject of RAI 6.2-140 and its subsequent Supplements 1–6. RAI 6.2-140 was being tracked as an open item in the SER with open items.

GEH, in a series of responses, proposed the following after 72 of initiation of a LOCA to reduce the long-term containment pressure:

- continuous refilling of the PCCS pools at rate commensurate with decay heat rate,

- taking credit for the passive radiolytical recombiners and removing from the system hydrogen at the rate of its generation, and

- implementing a design modification by installing vent fans, teed off of each PCCS vent line, thus establishing a sufficient gas flow from the DW atmosphere to the exhausts submerged in the GDCS pool. This fan system is to be designed to satisfy minimum requirements such as to assure the long term removal of noncondensable gas from the PCCS for continued condenser efficiency.

With these modifications, the calculated containment pressure drops rapidly shortly after 72 hours of the postulated limiting DBA from the maximum pressure to about 330 kPa, and continues to decrease over the period of 30 days to about 290 kPa. Thus, during the whole 30-day period following a LOCA the predicted containment pressure remained below the containment design pressure of 411.7 kPa (59.7 psia). After reviewing the proposed design modifications the staff found them acceptable.

The results of staff's confirmatory calculations using MELCOR computer code showed similar results as GEH's TRACG calculation. These addressed the staff's concerns. RAI 6.2-140 was resolved.

Staff Audit of TRACG Containment Analysis

The staff audited the GEH TRACG containment analysis on December 11–20, 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 by considering 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, assumes 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 the 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, Revision 3, issued March 2007, and the associated Branch Technical Position (BTP) 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 (PWR), 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 asked GEH to justify the containment back pressure used for determining the minimum RPV water level considering BTP CSB 6-1. The staff requested this information in RAI 6.2-144. In its response, GEH evaluated the impact of containment back pressure on the ECCS performance and presented this evaluation in ESBWR DCD, Tier 2, Revision 4, Appendix 6C. The staff reviewed the applicant'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. Based on the applicant's response, RAI 6.2-144 was 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/passive containment cooling (PCC) tank reflood would occur 34,376 seconds (9.55 hours) earlier than predicted by TRACG. However, the reflood timing has a small impact on containment pressure responses since the PCCS 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 TRACG and MELCOR event timings agree reasonably well.

Table 6.2-3 summarizes the 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
Drywell 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 its response, GEH provided results of the inadvertent spray actuation analysis. The conclusion was that FWLB and MLSB scenarios are bounding possible containment conditions, with FWLB producing the highest peak drywell pressure and MSLB producing the lowest one, during the initial 2,000 seconds after the break. GEH modified DCD Section 6.2.1.1.4 accordingly.

RAI 6.2-11 was being tracked as a confirmatory item in the SER with open items. The staff confirmed that the changes were incorporated in DCD, Tier 2, Revision 5. Based on the applicant's response, RAI 6.2-11 was resolved.

6.2.1.1.4 Conclusions

Based on the staff's review of the submitted containment analysis, as presented in DCD, Tier 2, Revision 7, and the staff's independent confirmatory calculations of containment responses to the postulated DBA LOCAs, the staff finds the GEH containment analysis acceptable.

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, in accordance with SRP Section 6.2.1.2, Revision 3, issued March 2007.

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

- GDC 4, "Environmental and Dynamic Effects Design Bases," as it relates to the environmental and missile protection provided to ensure 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, "Containment Design Basis," 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" (see Moody, F.J., "Maximum Flow Rate of a Single Component, Two-Phase Mixture," Journal of Heat Transfer, Trans. ASME, Series C, Volume 87, page 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. No high-energy lines are in the drywell head region.

Reactor Shield Annulus

The RSA exists between the RSW and the RPV. The RSW is a steel cylinder surrounding the RPV and extending 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 DCD, Tier 2, Section 6.2.1.2, and performed independent confirmatory analyses of the containment subcompartment by using alternative methodology (TRAC/RELAP Advanced Computational Engine (TRACE) computer code). The confirmatory calculations were based on additional information the staff requested in RAI 6.2-13, including synopsis of the piping break analyses, justification for the selection of the DBA (break size and location), and whether the leak-before-break was assumed to limit the pipe break area. In its 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 (the MSL, FWL, and GDCS injection line and the 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 released 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.

RAI 6.2-13 was being tracked as a confirmatory item in the SER with open items. The staff reviewed the applicant's subcompartment analysis and, based on the staff's confirmatory calculations, accepted the results. The staff confirmed that the changes were incorporated in DCD, Tier 2, Revision 5. Based on the applicant's response, RAI 6.2-13 was resolved.

DCD Tier 2 was unclear as to whether pipe restraints are used to limit the break area of the pipe ruptures. Therefore, in RAI 6.2-14, the staff asked GEH to clarify. In response, GEH stated that it took no credit in the analysis to limit the break area because of the presence of pipe restraints. The staff agreed with the information provided in the response. RAI 6.2-14 was being tracked as an open item in the SER with open items. Based on the applicant's response, RAI 6.2-14 was resolved.

DCD, Tier 2, Revision 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, Revision 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. RAI 6.2-15 was being tracked as an open item in the SER with open items.

In response, GEH clarified that it is "at least 15 percent margin" applied for design-basis structural analysis. Also, in DCD, Tier 2, Revision 5, the factor of 1.4 was changed to 1.2. The staff determined that the clarification and modifications are consistent with SRP Section 6.2.1.2 and acceptable. This addressed the staff's concern. RAI 6.2-15 was resolved.

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. Therefore, in RAI 6.2-17, the staff requested that GEH clarify. In response, GEH stated that the RSA subcompartment vent areas in ESBWR containment are always open, and no insulation collapse would occur in this subcompartment. The staff determined that this response was acceptable as it provides design basis. In a supplement to RAI 6.2-17, the staff asked GEH to provide this information in the DCD. RAI 6.2-17 was being tracked as an open

item in the SER with open items. In response, GEH incorporated the above in DCD, Tier 2, Revision 5. This addressed the staff's concern. RAI 6.2-17 was resolved.

DCD, Tier 2, Section 6.2.1.2.3, stated 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.

DCD, Tier 2, Section 6.2.1.2.3, stated that the TRACG computer code was used for the ESBWR containment subcompartment analysis. However, ESBWR DCD Tier 2 did 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 that GEH provide this information.

In its response, in a letter dated June 5, 2006, GEH stated that TRACG was qualified for analysis of the 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. After reviewing GEH's response, the staff determined that the GEH response addressed its concerns. GEH included its response in the revised licensing topical report (LTR), NEDE-33440P, Revision 1, "ESBWR Safety Analysis—Additional Information," issued June 2009. RAI 6.2-19 is resolved.

ESBWR DCD, Tier 2, Revision 1, did not provide the assumed initial operating conditions of the plant such as reactor power level and subcompartment pressure, temperature, and humidity which were assumed for the RSA subcompartment evaluation. Therefore, in RAI 6.2-20, the staff asked GEH to provide this information in the DCD. In response, GEH updated the DCD, Tier 2, Section 6.2.1.2.3 in Revision 2 of the DCD to state 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 degrees C when the postulated pipe break occurs. However, 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 clarify. In response, GEH stated that uncertainties associated with either "100% vs 102% power" or "air vs nitrogen" are bounded by the use of a 1.2 multiplier applied to the peak pressures calculated for annulus pressurization before being applied to the structural analyses. Based on its own independent analysis, the staff agrees with this information. RAI 6.2-20 was being tracked as an open item in the SER with open items. Based on the applicant's response, RAI 6.2-20 was resolved.

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 needed 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. GEH stated that the degree of entrainment was not a TRACG input and 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). GEH revised DCD, Tier 2, Revision 5, Section 6.2.1.2.3, accordingly. . The staff determined that GEH's modeling of vent flow and entrainment was acceptable because it is consistent with SRP Section 6.2.1.2. This addressed staff's concerns. RAI 6.2-21 was resolved.

ESBWR DCD Tier 2 does not provide information on the containment subcompartment nodalization. Therefore, in RAI 6.2-23, the staff asked GEH to provide this information. In response, 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 needed the details of nodalization to perform its confirmatory analysis, and staff requested this information in supplements to RAI 6.2-23. RAI 6.2-23 was being tracked as an open item in the SER with open items.

In its response to RAI 6.2-23, Supplements 1–3, GEH provided the requested information. The staff confirmed that the discussion addressing these concerns is included in LTR NEDE-33440P, Revision 1.

RAI 6.2-23 was being tracked as an open item in the SER with open items. The staff reviewed and accepted the applicant's response as it is consistent with previously approved Mark III methodology and also is supported by the insights gained from subcompartment analysis performed independently by the staff using alternate methodology (TRACE code). Based on the applicant's response, RAI 6.2-23 was resolved.

ESBWR DCD Tier 2 does not provide graphs of the pressure responses of subnodes within a subcompartment as functions of time. This information is needed for evaluations of the effect on structures and component supports. Therefore, in RAI 6.2-24, the staff asked GEH to provide this information. RAI 6.2-24 was being tracked as an open item in the SER with open items. In response, GEH provided graphs of the pressure responses of subnodes within a subcompartment as functions of time, which were acceptable because they addressed the staff's concern. In RAI 6.2-24, Supplement 1, the staff requested that GEH add this information to the DCD. In its response, GEH provided the requested information in LTR NEDE-33440P, Revision 1, referenced in the DCD. After reviewing GEH's responses the staff determined them acceptable. RAI 6.2-24 was being tracked as an open item in the SER with open items. Based on the applicant's response, RAI 6.2-24 was resolved.

ESBWR DCD Tier 2 does not provide the mass and energy release data for the postulated pipe breaks. Therefore, in RAI 6.2-25, the staff asked GEH to provide this information. In response, GEH provided the method used to calculate mass and energy release data but not the actual data. Therefore, in RAI 6.2-25, Supplement 1, the staff asked GEH to provide this information and update DCD Tier 2. In response, GEH provided the requested information in LTR NEDE-33440P, Revision 1, referenced in the DCD.. RAI 6.2-25 was resolved.

ESBWR DCD Tier 2 does not state the flow conditions (subsonic or sonic) for vent flowpaths up to the time of peak pressure. The staff needs this information 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 asked GEH to provide this information. In response, GEH stated that before the time of peak pressure, the vent flow is subsonic. GEH agreed to update the DCD to include this information. RAI 6.2-26 was being tracked as a confirmatory item in the SER with open items. The staff confirmed that the information was incorporated in DCD, Tier 2, Revision 5, Section 6.2.1.2.3. Based on the applicant's response, RAI 6.2-26 was resolved.

In RAI 6.2-29, the staff expressed concern about the GEH methodology, specifically, with applying the TRACG computer program to the containment subcompartment analysis without providing information on the time-step and nodalization study, code validation, and comparison to approved methods. In its response to subsequent Supplements 1, 2, and 3 to RAI 6.2-29, GEH provided a comparison of the TRACG and CONTAIN analyses.

RAI 6.2-29 was being tracked as an open item in the SER with open items. Based on the submitted additional comparison and the staff's own confirmatory analysis performed with a subcompartment code TRACE, the staff accepts the results of the GEH subcompartment analysis. Based on the applicant's response, RAI 6.2-29 was resolved.

6.2.1.2.4 Conclusions

The staff reviewed the application of the TRACG computer program to the subcompartment analysis and its comparison to alternative methodology (the CONTAIN code). Based on the review, and the staff's own independent analysis (with the TRACE code), the staff finds the GEH subcompartment analysis to be sufficiently conservative and, therefore, acceptable.

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

Section 6.2.1.1 of this report presents the staff's review of the DCD to determine if it meets the criteria of SRP Section 6.2.1.3.

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

SRP Section 6.2.1.4, 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 Section 6.2.1.5, applies to PWRs and thus is not generally applicable to the ESBWR. However, during a December 2006 audit, the staff raised the issue of possible implications of the minimum containment pressure on the initiation timing of GDCS injection and thus on 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 SRVs.

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 the ESBWR is given in a proprietary report, NEDE-33261P, "ESBWR Containment Load Definition," issued May 2006.

NEDE-33261P 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. Table 6.2.1.6-1 lists the key differences between the two containment designs.

Table 6.2.1.6-1: Geometries of the pressure suppression system

Parameter	ESBWR	ABWR
Number of vertical vents	12	10
Suppression pool angular sector per vertical vent (degrees)	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 the 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 NUREG-1503, "Final Safety Evaluation Report Related to the Certification of the Advanced Boiling Water Reactor Design," issued July 1994, and NEDE-33261P.

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 vessel 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. The 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 the 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, Revision 0, “Containment Horizontal Vent Confirmatory Test, Part I” (proprietary), issued March 1987) indicate that the wall, liner, and submerged structures within two vent diameters of each horizontal vent also experience local effects. The methodology, as presented in NEDE-33261P, addresses this phenomenon.

One of the unique design features of the ESBWR 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 concerning the SRV loads that would result from a second opening of the SRV while the SRV tailpipe is still hot from the initial discharge; the staff referred to this 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 GEH 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: Suppression pool configuration

Design Feature	ESBWR	ABWR
Reactor power, MWt	4,500	4,000
Drywell volume, m^3 (ft^3)	7,206 (~254,520)	7,350 (~259,500)
Wetwell gas space volume, m^3 (ft^3)	5,350 (~188,900)	5,960 (~210,000)
Vertical vents (total), m^2 (ft^2)	13.6 (146)	11.6 (125)
Pool surface only, m^2 (ft^2)	799 (~8,600)	507 (~5,450)

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 to 5 seconds), 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 ventline 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 NEDE-33261P, issued 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 MSLB loads bound the FWLB PS 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 a reduction in the pool hydrodynamic loads. NUREG-0808, “Mark II Containment Program Load 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 considered the differences between the Mark II, Mark III, and ABWR databases in determining whether the ESBWR suppression pool wall pressures do not exhibit any unusual characteristics when compared to the Mark III wall pressures. Because the ABWR and ESBWR suppression pool designs are so similar, the staff reviewed a concern (described in 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, Tier 2, Revision 3, does not 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 staff expressed concerns about 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. In RAI 6.2-158, the staff asked GEH to address the above issues. In its response of November 16, 2007, GEH referred to the revised "ESBWR Containment Load Definition" report (NEDE-33261P, Revision 1,) which addressed the staff's concerns. The report demonstrated that the ABWR CO wall load definition was based on SS tests, and the ABWR CH load definition was based on FS tests. The staff accepted these load definitions during the ABWR certification process. Since the similarity between the ABWR and ESBWR containment systems was established, the staff finds the response acceptable. RAI 6.2-158 was being tracked as an open item in the SER with open items. Based on the applicant's response, RAI 6.2-158 was resolved.

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 actions such 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 degrees C (95 degrees F). Operation can continue until the suppression pool reaches 43 degrees C (110 degrees F). At this temperature, an automatic scram on high suppression pool temperature occurs. Finally, if the pool reaches 49 degrees C (120 degrees F), the TS require depressurization of the reactor coolant system and initiation of cold shutdown conditions. The ESBWR TS 3.6.2.1, "Suppression Pool Average Temperature," specifies temperature thresholds for reactor scram, shutdown, and vessel depressurization of 110 degrees F, 120 degrees F, and 130 degrees F, respectively. These limits do not follow the guidance provided in NUREG-0783, "Suppression Pool Temperature Limits for BWR Containment," issued November 1981. In RAI 6.2-159 the staff asked for explanation why the NUREG-0782 guidance was followed including a description of the effect of pool temperature on the SRV load evaluation. In its response, GEH stated that additional test date with X-Quencher, used in the ESBWR, collected after NUREG-0783 was issued, justified elimination of the local pool temperature limit. The staff approved this conclusion in a letter from G. Holahan (NRC) to R. Pinelli (Boiling Water Reactor Owners Group), dated August 29, 1994. The separate but related issue of potential steam ingestion into ECCS pump suction does not apply to the passive ESBWR design. In addition, the TS pool temperature limit requirement is consistent with the assumptions used for the ESBWR safety analyses. RAI 6.2-159 was being tracked as an open item in the SER with open items. Based on the applicant's response, RAI 6.2-159 was resolved.

NEDE-33261P (May 2006) Revision 0, implies that GE used the PICSM computer code to compare Mark III suppression PS test data from the pressure suppression test facility with analytical predictions. GE technical report NEDE-21544P, Revision 0, “Mark II Pressure Suppression Containment Systems: An Analytical Model of the Pool Swell Phenomenon” (proprietary), issued December 1976, describes the code. GE validated the test data generated for the Mark II design; however, the staff did not review and approve the code. GEH addressed the staff’s concern with potential liquid and froth impacts on the vacuum breaker valves in its response to RAI 6.2-160. DCD, Tier 2, Revision 5, requires the diaphragm floor slab to be greater than 9,600 mm above the wetwell floor. This requirement ensures that the maximum PS of 4,100 mm will not reach the vacuum breaker valves assuming the maximum allowable pool depth of 5,500 mm, as specified by TS SR 3.6.2.2.1. The froth impacts are predicted by using the same methodology as previously approved for the ABWR certification. Based on the established similarity between the ABWR and ESBWR containments, the staff accepts the response. RAI 6.2-160 was being tracked as an open item in the SER with open items. Based on the applicant’s response, RAI 6.2-160 was resolved.

GEH applied 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 (described in NEDO-21061, Revision 0, “Mark II Containment Dynamic Forcing Functions Information Report,” issued September 1975) 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. In RAI 6.2-161, the staff requested that GEH address the above issue. In its response, dated November 16, 2007, GEH explained that both the Mark II program and the ABWR certification used a different computer program, PICSM, for the pool hydrodynamic loads, as presented and approved by the staff in NUREG-1503. Based on design similarities between the ABWR and ESBWR designs, as discussed in NEDE-33261P, Revision 1, GEH claimed that this methodology can be applied to the ESBWR hydrodynamic loads definition. Based on the use of methodology previously accepted during the ABWR certification process and the established similarity between the ABWR and ESBWR containments, the staff accepts the applicant’s response. RAI 6.2-161 was being tracked as an open item in the SER with open items. Based on the applicant’s response, RAI 6.2-161 was resolved.

In RAI 6.2-164 the staff request details of analysis of the suppression pool and its associated structure, systems and components (SSCs) subjected to hydrodynamic loads described in DCD, Tier 2, Revision 4, Appendix 3B. In its response the applicant added Appendices 3F and 3G to DCD, Tier 2, Revision 4 providing qualification for the suppression pool and its associated SSCs to withstand imposed hydrodynamic loads. GEH addressed an additional concern regarding the integrity of the PCCS vent pipe (described by the staff in RAI 6.2-164, Supplement 1), in its response to RAI 14.3-131, Supplement 1, where GEH indicated that the PCCS piping is included in the ITAAC in DCD, Tier 1, Section 3.1. Based on the review of the Appendices 3F, 3G, and the audits performed on the applicant's suppression pool analyses, the staff found the GEH responses regarding the above concerns to be acceptable. RAI 6.2-164 was being tracked as an open item in the SER with open items. Based on the applicant’s response, RAI 6.2-164 was resolved.

6.2.1.6.4 Conclusions

The staff reviewed the methodology presented in NEDE-33261P, including Revision 1, 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 (General Electric Nuclear Energy (GENE)), "Safety Evaluation Report Regarding the Application of GENE's TRACG Code to ESBWR LOCA Analyses," dated 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.

6.2.1.7 *Containment Debris Protection for Emergency Core Cooling System Strainers*

6.2.1.7.1 Regulatory Criteria

GDC 35, "Emergency Core Cooling," GDC 38, "Containment Heat Removal," 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, Revision 3, "Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident," issued November 2003, contains guidance on the sizing criteria for ECCS strainers.

The following NRC bulletins (BLs) 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. A perforated steel plate will cover the GDCS pool airspace opening to the drywell 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 and beyond 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.

DCD, Tier 2, Revision 1, Section 6.2.1.1.2, states the following:

> 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 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. Water inventory, including the GDCS, is sufficient to flood the RPV to at least 1 m above the top of active fuel.

The DCD was not clear as to how water is removed from the suppression pool during the post-LOCA period. Therefore, in RAI 6.2-6, the staff asked GEH for clarification. In response, 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.

If the ESBWR design relies on the suppression pool equalization line to maintain a depth of 1 meter of water above active fuel in the RPV, the suppression pool equalization line should be designed as such. To review the functioning of the suppression pool equalization line during DBA LOCA scenario, in RAI 6.3-40, the staff requested GEH to provide the value of differential pressure across the equalization line check valves for each of the DBA LOCA scenario analyzed.

In response, 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." The ESBWR DCD was not clear as to how the intake strainer is designed to prevent the entry of debris material into the system. Therefore, in RAI 6.2-6, Supplement 1, the staff asked GEH to explain. RAIs 6.2-6 and 6.3-40 were being tracked as open items in the SER with open items.

In response to a related RAI, RAI 6.2-173, Supplement 1, which is described below in this section, GEH stated the following:

> As stated in the response to RAI 6.3-40 (MFN 06-488, dated December 22, 2006), and confirmed in the response to RAI 6.3-40 S01 (MFN 06-488, Supplement 1, dated December 21, 2007), reactor pressure

vessel (RPV) water levels stay above Level 0.5 setpoint for 72 hours and beyond for all loss-of-coolant accident (LOCA) scenarios. In addition, a 30 day analysis confirms RPV water level stays above Level 0.5 setpoint, as discussed in the response to RAI 6.2-140 S02 (MFN 08-633, dated August 18, 2008). Therefore, the squib valves in the GDCS equalization lines never open, and the GDCS equalization lines are not required to function in response to a LOCA and do not perform a safety-related function. Therefore, the application of Regulatory Guide (RG) 1.82, Revision 3, is not required.

After reviewing GEH's response to RAI 6.2-173, Supplement 1, the staff determined that an intake strainer for suppression pool equalization line is not required for 30 days following a LOCA. This addresses the staff's concern raised in RAI 6.2-6, Supplement 1, and RAI 6.3-40. RAIs 6.2-6 and 6.3-40 were resolved.

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, and agreed to add this information to the DCD.

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 staff's concern was that debris could clog the temporary strainers and consequently 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 its concerns and was acceptable. GEH needed to update the DCD to include the remaining information as described above. RAI 6.3-41 was being tracked as an open item in the SER with open items.

GEH updated DCD, Tier 2, Revision 4, Section 6.3.2.7.2, to provide the dimensions of the holes in the perforated plate and to state that the temporary strainer will be removed after initial flushing of GDCS injection lines. This addresses the staff's concerns raised in RAI 6.3-41. RAI 6.3-41 was resolved.

During a LOCA, if the PCCS heat exchanger inlets are within the zone of influence, debris ingress is expected. However, DCD, Tier 2, Revision 2, did not address the impact of possible debris ingress into the PCCS. Therefore, in RAI 6.3-42, the staff requested that GEH describe the impact of the debris on the performance of the heat exchanger. In response, GEH stated that the PCCS heat exchanger inlet pipe is provided with a debris filter with holes no greater

than 25 mm (1 in.) to prevent entrance of missiles into the pipe and protection from fluid jets during a LOCA. These holes are smaller than the size of the heat exchanger tubes (50-mm (2-in.) nominal diameter), which have the smallest diameter of the piping components in the PCCS. GEH stated that if there is any debris that enters the PCCS it cannot become lodged in the vertical heat exchanger tubes where the heat transfer function is performed, and thus, debris will not impact the PCCS performance. The staff determined that the PCC inlet pipe debris filter would limit debris entering the PCCS during a LOCA and that the PCCS heat transfer function would not be impacted. This addressed the staff's concern. In a supplement to RAI 6.3-42, the staff requested that the dimension of the holes of the debris filter should be added to the DCD. RAI 6.3-42 was being tracked as an open item in the SER with open items. GEH revised DCD, Tier 2, Revision 5, Section 6.2.2.2.2, to include this dimension. RAI 6.3-42 was resolved.

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 LPCI modes as an 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 BL 93-02 and its Supplement 1, 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 was being tracked as an open item in the SER with open items.

In its response to RAI 6.2-173, GEH stated the following:

> Long-term decay heat removal from the containment is provided by the Passive Containment Cooling System (PCCS), and after 72 hours the PCCS vent fans are available to increase the efficiency of the PCCS condensers. The PCCS along with the vent fans are capable of maintaining containment pressure below the design pressure for 30 days as described in the response to RAI 6.2-140 S02. In addition, the FAPCS lines associated with the suppression pool are not considered to be operational during a LOCA event and would not be considered available for operation until the seventh day after the start of a LOCA event. Therefore, only when determined to be appropriate and available, the FAPCS may be actuated in the low pressure coolant injection (LPCI), suppression pool cooling, or drywell (DW) spray modes to provide additional cooling to bring the plant to cold shutdown. Since the long term operation of the PCCS vent fans is sufficient to protect the integrity of containment, function of the FAPCS suppression pool line is not safety-related and the operation of the FAPCS cooling function is not required. Therefore RG 1.82, Revision 3 is not applicable to this application.

After reviewing GEH's response, the staff determined that RG 1.82, Revision 3, is not applicable to the ESBWR because the FAPCS cooling function is not required and the PCCS and the vent

fans are capable of maintaining containment pressure below the design pressure for 30 days. This addresses the staff's concerns raised in RAI 6.2-173. RAI 6.2-173 was resolved.

6.2.1.7.4 Conclusions

The staff finds that the ESBWR design includes features to limit debris affecting the performance of the decay heat removal function following a LOCA. The staff determined that RG 1.82, Revision 3, is not applicable to the ESBWR because the FAPCS cooling function is not required and the PCCS and the vent fans are capable of maintaining containment pressure below the design pressure for 30 days. The staff finds the ESBWR design acceptable because LOCA-generated or latent debris will not affect the ESBWR's ability to meet GDC 35, 38, and 41.

6.2.2 Containment Heat Removal System

6.2.2.1 *Regulatory Criteria*

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

- GDC 38, "Containment Heat Removal," 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 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 (ISI) apply to meet GDC 16. Quality Group B requirements apply as described in 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 shared by the PCCS condensers and 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. DCD, Tier 2, Section 6.2.2.2.2 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 degrees C is 1.33 megapascals (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 (PCC condenser) designed to reject up to 7.8 megawatts 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 ventline, 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 ventline outlet is 0.85 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 in SRP Section 6.2.2, Revision 5, issued March 2007, 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 TRACG results indicate that containment pressure is still rising at 72 hours, and the PCCS does not appear to rapidly reduce containment pressure and temperature as evident from the TRACG results presented in DCD, Tier 2, Revision 3, Section 6.2. The applicant needs to demonstrate how the ESBWR meets the safety function of GDC 38. The staff issued RAI 6.2-139 to address this issue.

In response to RAI 6.2-139 (and RAI 6.2-140, discussed in Section 6.2.1.1.3 of this report), GEH made design modifications by adding a passive autocatalytic recombiner system (PARS) and PCCS vent fans, including power supplies. The following describes the staff’s evaluation of these design modifications with respect to GDC 38.

The ESBWR pressure suppression concept employs a drywell that houses the nuclear system and a large volume of water outside the drywell called the suppression pool. If a LOCA occurs within the drywell, the pressure suppression system rapidly condenses the steam that is released through the break or that is generated by flashing of water which is released through the break to prevent overpressurization. Pressurization of the drywell results in venting of steam to the suppression pool where it is condensed, thus relieving pressure in the drywell. Decay heat in the core continues to generate steam, which is released into containment through the break. The PCCS removes heat from containment by condensing steam in the drywell. Condensate from the PCCS drains into the GDCS tanks, which provide water to the RPV for cooling the reactor core.

The balance between the rates of decay heat generation and PCCS heat removal causes ESBWR containment pressurization. The design-basis analysis assumes both steam bypass of the suppression pool and radiolytic generation of noncondensable gases. Containment pressure, calculated using a conservative rate of decay heat generation, a bounding value of the steam bypass, and a conservative rate of radiolysis, continues to rise for 72 hours after a LOCA. However, containment pressure remains below the containment design pressure during this time. During the first 72 hours, the PCCS operates without need for active systems, electric power, or operator actions. The staff determined that the PCCS offers potential advantages over current active containment cooling systems and can provide sufficient containment heat removal to maintain containment pressure below its design value during this time.

Beyond 72 hours after a LOCA, the following additional systems supplement the PCCS to continue containment heat removal:

- systems, structures, and components required for IC/PCC pool refill, including power supplies
- the PARS, which is conservatively assumed not to function until 72 hours, and then is assumed to function only to recombine hydrogen from radiolysis from 72 hours on (i.e., hydrogen content at 72 hours is assumed to remain constant for the duration of the LOCA recovery period)
- PCCS vent fans, including power supplies

Note that the PARS would remove hydrogen by initiating its chemical reaction with oxygen to produce steam, which can be condensed by the PCCS, helping to reduce the containment pressure. This reaction generates heat, countering the benefit of removing hydrogen and oxygen in the containment atmosphere. However, the net result is a drop in containment pressure because the PCCS can remove heat thus generated by condensing steam in the containment atmosphere. The PCCS and the additional systems can continue to remove heat from containment, maintaining its pressure below the design value up to 30 days and beyond. Two systems, (1) suppression pool cooling with a crosstie of the FAPCS and the RWCU/SDC heat exchanger and (2) the FAPCS in LPCI mode, will be available after 8 days following a LOCA, if needed to further reduce containment pressure.

In response to RAI 6.2-139, GEH stated the following in a letter dated April 19, 2008:

> The analysis results indicate that the [drywell] pressure remains below the design pressure of 413.7 KPa (60 psia) for the first 72 hours after the [main steamline break accident], and then rapidly reduces and maintains the reduction with the

refill of the [isolation condenser]/PCC pool and operation of the PCCS Vent Fans, achieving even lower pressures when the PARS were credited.

ESBWR containment pressure after a LOCA differs from that of operating BWR plants in all of the following ways:

- ESBWR pressure has a maximum value at 3 days, while operating BWR pressures peak within a few hours.
- The magnitude of ESBWR pressure drop at 3 days is lower than that for the operating BWRs.
- ESBWR pressure remains at elevated values in the long term compared to operating BWRs.

The staff concludes that the ESBWR does not reduce the containment pressure to as low a level as operating BWRs, but the ESBWR does provide adequate containment heat removal and meets the intent of GDC 38 because of the following:

- The PCCS can remove heat from containment and can maintain containment pressure below its design value without operator action or using active systems or electric power for 72 hours after a LOCA.
- The PCCS and additional systems can continue removing heat from containment from 3 days to beyond 30 days after a LOCA to maintain containment pressure below its design value.
- Systems are available after 8 days following a LOCA to further reduce containment pressure and to take the reactor coolant system to cold shutdown conditions, if needed.

The staff interpretation of GDC 38 applies specifically to the ESBWR passive design but, potentially, also to other similar passive safety systems.

The applicant's response, including design changes, addresses the staff's concern and is acceptable because the ESBWR does provide adequate containment heat removal and meets the intent of GDC 38. RAI 6.2-139 was therefore resolved.

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 degrees C (340 degrees F), the same as that for the containment. DCD Table 6.2-1 provides 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 drywell pressure boundary and thus not needing isolation valves. The staff evaluated GEH's position in Section 6.2.4.3 of this report under RAI 6.2-102 and found it acceptable.

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 ventline outlet is 0.85 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, Revision 4, Section 6.2.2, did not include or describe the elevation of the PCC ventline relative to the upper horizontal main vents. Therefore, the staff issued RAI 6.2-169 to request this information. RAI 6.2-169 was being tracked as an open item in the SER with open items.

In response, GEH updated DCD, Tier 2, Revision 5, Section 6.2.2.2.2, to state that "the vent line discharge point is set at an elevation submerged below low water level and at least 0.85 m (33.5 in) and no greater than 0.900 m (35.4 in) above the top of the uppermost horizontal vent." GEH also added an ITAAC to verify the PCC ventline outlet elevation. These modifications address the staff's concerns. Based on the applicant's response, RAI 6.2-169 was resolved.

The PCCS is designed to seismic Category I, as described in RG 1.29 and 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 7.8 MWt of latent heat during condensation of pure steam inside the tubes at a pressure of 308 kPa (45 psia) and a temperature of 134 degrees C (273.2 degrees F), with an outside pool water temperature of 102 degrees C (215.6 degrees F). 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.

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 briefly describes 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 steamflow 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 of this report. Therefore, the staff accepted the use of the test results to demonstrate PCCS performance and to support the verification and validation of the relevant analytical models.

The staff also performed its own independent studies of the PCCS performance at the Purdue University Meteorological Association (PUMA) facility. 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.

In the DCD, GEH did not describe the ESBWR test program as applied to the safety evaluation of the containment heat removal system. The staff requested this information in RAI 6.2-172. The GEH response was acceptable; however, the staff needed to verify that the response is incorporated in a future revision of the DCD. RAI 6.2-172 was being tracked as a confirmatory item. The staff confirmed that the description is included in the TRACG qualification report, NEDC-32725P, Revision 1, August 2002, “TRACG Qualification for SBWR,” which was reviewed and approved separately by the staff (see Conclusions 6.2.1.6.4). Therefore, RAI 6.2-172 was resolved.

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. In RAI 6.2-170, the staff requested that GEH provide this discussion.

In its response, dated July 27, 2007, GEH explained that except for the containment isolation function, the chilled water system (CWS) equipment is all non-safety-related and is not required to function during the response to a DBA. It is assumed that the non-safety-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 considered in the design of the CWS and do not adversely affect the ESBWR response to a DBA.

During DBA conditions, the design feature providing cooling of the containment air for the ESBWR is the PCCS condensers, which condense steam that has been released to the drywell 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, Section 6.2.1, discusses the role of the PCCS condensers in maintaining containment pressure and temperature within design limits during DBAs and provides information about the function of the PCCS. DCD, Tier 2, Revision 3, Section 6.2.2, gives the design details for the PCCS. The passive nature of the PCCS design prevents it from being subject to water hammer effects or thermally induced overpressurization.

Based on the GEH's response to GL 96-06 and the passive nature of the PCCS design, the staff found GEH's response acceptable; however, the staff needed to verify that the proposed revision to the DCD is incorporated in a future revision of the DCD. RAI 6.2-170 was being tracked as a confirmatory item in the SER with open items. The staff confirmed that the changes were incorporated in DCD, Tier 2, Revision 4, Table 1.1C-1. Based on the applicant's response, RAI 6.2-170 was resolved.

DCD, Tier 2, Revision 3, Section 1.11, Table 1.11-1, states that DCD Sections 6.2.2, 7.3.2, 9.2.7, and 9.4.8 address the evaluation of Task Action Plan Item B-12, "Containment Cooling Requirements (Non-LOCA)." The staff could not locate this discussion in Section 6.2.2 and requested, in RAI 6.2-171, that the applicant address Task Action Plan B-12.

In its response, GEH stated that it referenced DCD, Tier 2, Revision 3, Sections 6.2.2 and 7.3.2, because they describe the design of the PCCS, which performs the safety-related containment cooling for the ESBWR. In DCD, Tier 2, Revision 3, Sections 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 affect 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 it does not need or use electrical power for its operation.

The staff found GEH's response acceptable; however, it needed to verify that the proposed revision to the DCD is incorporated in a future revision of the DCD. RAI 6.2-171 was being tracked as a confirmatory item in the SER with open items. The staff confirmed that the changes were incorporated in DCD, Tier 2, Revision 4, Table 1.11-1. Based on the applicant's response, RAI 6.2-171 was resolved.

In RAI 6.2-202, the staff requested that GEH address the possible accumulation of high concentrations of hydrogen and oxygen in the PCCS and ICS to meet 10 CFR 50.46(b)(5).

PCCS

During the blowdown period of a LOCA, most of the nitrogen in the drywell of the ESBWR would relocate into the wetwell airspace. Radiolysis in the core generates hydrogen and oxygen at the stoichiometric ratio, which would be released into the drywell with steam. A mixture of steam, hydrogen, oxygen, and any nitrogen remaining in the drywell would be drawn into the PCCS where steam is condensed, leaving mainly hydrogen and oxygen in the PCCS. Although a part of the hydrogen and oxygen that accumulates in the PCCS would relocate to the wetwell airspace through the PCC vent line, it is possible for the remaining hydrogen and oxygen to reach concentration levels that supports detonation.

In response, GEH agreed with the staff on the possibility of radiolytically generated hydrogen and oxygen accumulating in the PCCS at detonable levels following a LOCA and designed the PCCS to be able to perform its safety function after undergoing multiple hydrogen detonations. GEH's licensing topical report, NEDE-33572P, describes PCCS design changes and the methodology by which the detonation loads were calculated. GEH modified the design of PCCS tubes, lower drum, and vent and drain lines. GEH did not evaluate the steam supply line and upper drums for detonation loading because the hydrogen and oxygen concentrations in those components would be low and would not support combustion as they are constantly being

flushed by steam coming from the drywell. The staff determined that GEH's design of PCCS intake pipe and the upper drum were acceptable because the dilution by high steam concentration would prohibit detonation of hydrogen.

The following is a summary of PCCS design changes:

- Changed condenser tubing material from SA-213 Gr TP304L to SA-312 Gr XM-19 and increased the tube thickness to withstand detonation loading

- Increased the number of tubes per module (each PCCS condenser consists of two modules) to compensate for the reduction of heat transfer due to increased tube thickness and reduced thermal conductivity of the new material

- Increased thickness of the lower drum and changed the material to SA-182 Gr XM-19 to withstand detonation loading

- Added a safety-related catalyst module with platinum or palladium coated plates to the vent lines in the lower drum of the condenser to limit hydrogen and oxygen concentrations in the vent lines to below a detonable level

- Increased the thickness of the vent lines to withstand (1) pressure loading on the exterior of the vent line from a detonation occurring in a drain line and (2) high pressure generated by expansion of the post combustion gas mixture from a detonation postulated to occur in the lower drum

- Increased the thickness of the drain lines to withstand detonation loading

In calculating detonation loading for the PCCS tubes, lower drum, and drain line, GEH assumed a theoretical maximum concentration of hydrogen and oxygen at a stochiometric ratio of 2:1. The staff determined that this treatment is conservative for mixtures in which the flame accelerates from deflagration to detonation (DDT) in a short period of time. However, with the introduction of inert gasses or vapors, the acceleration of the flame front may be delayed causing delayed DDT that can generate higher detonation pressures. During delayed DDT, the deflagration front undergoes a substantial acceleration period before transitioning to a detonation, or when the un-burnt mixture is compressed due to obstructions or closed ends in the structure. This compression at the onset of detonation has the potential to cause much higher localized pressure loads. To address the staff's concern, GEH noted that the detonation cell size for a hydrogen oxygen mixture is too small to support delayed DDT. After reviewing GEH's response, the staff determined that delayed DDT would not be a concern for PCCS components. Therefore, the staff determined that the hydrogen and oxygen concentrations used by GEH to calculate detonation pressure loading were acceptable.

GEH calculated a bounding detonation pressure for a stoichiometric mixture of hydrogen and oxygen using the highest peak pressure that occurs during a loss of coolant accident (LOCA). GEH then applied the detonation pressure statically using dynamic load factors (DLF) in a finite element (FE) model for the PCCS condenser using the ANSYS computer code. GEH determined the resultant pressure following the passage of a detonation wave, which is called the Chapman-Jouguet (CJ) pressure, using a correlation between the CJ pressure and the initial pressure prior to detonation (i.e., CJ pressure ratio). This pressure ratio is based on a 2006 publication by J. E. Shepherd. The correlation is dependent on the composition of the fuel-

oxidizer mixture, the initial conditions (pressure and temperature), and the geometry of the system. GEH used a CJ pressure ratio of 19. The staff determined that GEH's using of a stoichiometric mixture of hydrogen and oxygen, the peak LOCA pressure, and a temperature which is lower than that is expected in the PCCS during a LOCA would conservatively give high CJ pressures. Therefore, the staff determined that GEH's using a CJ pressure ratio of 19 was acceptable.

The presence of bends, constrictions, and closed ends creates opportunities for reflections that can create localized peak pressures in excess of the CJ pressure. Based on a 1991 publication by J. E. Shepherd, et al., GEH assumed a peak pressure for a closed volume as a maximum of 2.5 times the CJ pressure. The staff determined that using a factor of 2.5 corresponding to a reflection by a closed end was conservative because bends and constrictions would generate lower pressure peaks.

GEH used a CJ pressure ratio of 19, as described above, combined with a DLF of 2. GEH determined DLF based on the 2006 publication by J. E. Shepherd. According to this publication, DLF of 2 can be used when the detonation velocity is not near the structural resonance velocity. Diluents, such as steam, cause the detonation velocity to drop, affording the possibility that the detonation velocity would come close to the resonance velocity of the component, in which case a DLF of 4 should be used. However, with addition of diluents the CJ pressure ratio also drops. GEH showed in Section 2.2.2.2 of NEDE-33572P that for a DLF of 4 the steam concentration has to be increased to a value above 65 percent. At a steam concentration of 65 percent, a CJ pressure ratio would be 9.3. Thus, the product of CJ pressure ratio and DLF will be lower than that assumed in the design. Based on GEH's determination of DLF and the staff's confirmatory calculations, the staff determined that a DLF of 2 is acceptable.

As described above, based on its review and confirmatory calculations, the staff determined that GEH's calculation of detonation pressure loading as used in the PCCS design was acceptable.

GEH proposed to revise the DCD, to include the following:

> To prevent the accumulation of combustible gas in the PCCS vent lines, catalyst modules containing metal parallel plates coated with platinum/palladium catalyst are placed at the entrance to the vent line, within each lower drum. These safety-related vent line catalyst modules are seismic category I and are environmentally qualified for the harsh post-accident environment in combination with the operating conditions of catalytic recombination, given their 60 year design life. The vent line catalyst modules are designed and built to withstand detonation loading in combination with other applicable dynamic loads, without losing their catalytic recombination functionality or negatively impacting the venting capability of the condenser.

After reviewing the proposed revision to the DCD and NEDE-33572P, the staff determined that the catalyst module added to the vent line in the lower drum of the condenser would limit hydrogen concentration in the vent line to below detonable level. Therefore, the staff determined that GEH's ignoring of detonations in the PCCS vent line was acceptable.

With regard to PCCS performance, NEDE-33572P, Revision 1, states that the increase in PCCS tube thickness and change in the material will increase conduction resistance through the tube wall, which will have a negative effect on the overall heat transfer coefficient of the PCCS.

To compensate for this effect, based on TRACG evaluations, GEH increased the number of tubes per PCCS module in order to keep the containment pressure response bounded by the values described in DCD Revision 7. To evaluate the effect of PCCS design changes on its heat transfer capability, in RAI 6.2-202 S01, the staff requested GEH to (1) confirm that TRACG validation for calculating PCCS heat transfer is applicable to the new design; (2) provide the results of TRACG analysis confirming that the containment pressure is bounded by values presented in DCD Revision 7; and (3) confirm that scaling groups used in ESBWR Scaling Report, Revision 2, NEDC-33082P, April 2008, are still applicable to the new design.

In response to RAI 6.2-202 S01, in letter MFN 10-044 Supplement 1, Revision 1, dated August 2, 2010, GEH stated that (1) the PANTHERS and PANDA qualification tests as documented in NEDC 32725P, Revision 1, for TRACG validation are still applicable to the new PCCS design considering the different tube material, tube thickness, tube internal diameter, and different number of tubes; (2) the overall changes to the PCCS design have a relatively small impact on the overall heat transfer and the PCCS performance and; (3) GEH provided an evaluation showing the scaling groups remain applicable.

After reviewing the response, the staff determined that TRACG validation for calculating PCCS heat transfer is applicable to the new design.

In its August 2, 2010, response, GEH provided the results of the TRACG analysis that includes the change in tube material, number of tubes, tube thickness, and tube inner diameter. Containment pressure and PCCS heat removal rate as predicted by TRACG for the MSLB bounding case as provided in DCD Revision 7, did not show any appreciable differences. Therefore, the staff determined that after PCCS design changes, the containment pressure is bounded by values presented in DCD Revision 7.

The results of the GEH's calculations verify that the change in PCCS condenser response time due to the design differences is insignificant for the very slow long-term containment pressure response, which is on the order of several hours (100,000 seconds), as discussed in ESBWR Scaling Report, Revision 2, NEDC-33082P, April 2008. The results further demonstrate that the overall changes to the PCCS design have a relatively small impact on the overall heat transfer. Calculations show that the differences between overall heat transfer coefficient, total thermal resistance, fluid transport time and thermal time constant of the tube wall for the two designs, are not significant. From a scaling perspective, these changes are within the same order of magnitude (i.e., within the acceptable range) as those for the ESBWR test program, discussed in Chapter 21 of the staff SER.

Thus, the local or "bottom-up" scaling shows that PANTHERS tests for PCCS are still applicable to the new design since the PCCS overall heat transfer has not changed. Therefore, the Pi-groups for the "top-down" scaling groups are expected to remain the same, and no change is necessary to the scaling groups. As a result, the analysis confirms that the modified PCCS design satisfies the scaling criterion that was used for the ESBWR test program. In addition, the staff believes that the changes in the PCCS design are not expected to create any new and different phenomena that were not observed in the test.

On the basis of the discussion made above, the staff finds the GEH response acceptable. The staff, therefore, concludes that there is reasonable assurance that the PANTHERS/PCCS test data continue to be relevant and sufficient to apply TRACG for the modified PCCS Condenser design.

Structural Analysis of PCCS

The PCCS condensers were designed as part of the containment pressure boundary according to ASME Code, Section III, Subsection NE. Therefore, under Section 3.8.2 of this report, the staff evaluated the structural integrity of the PCCS within the jurisdictional boundary of ASME Code, Subsection NE; in particular, the staff evaluated the capability of the PCCS to withstand the effects of deflagration or detonation of non-condensable gases during the 72 hour-period associated with a LOCA. All issues identified in the evaluation performed in Section 3.8.2 were not completely resolved.

On September 22, 2010, staff conducted an audit of supporting calculations and the basis for GEH's licensing topical report NEDE-33572P, "ESBWR ICS and PCCS Condenser Combustible Gas Mitigation and Structural Evaluation," at the Nuclear Energy Institute's (NEI's) office in Rockville, Maryland. During this audit, the NRC team reviewed calculations associated with the structural analysis of the PCCS to withstand detonation loads, to obtain reasonable assurance that the design is in conformance with the ASME Code, Subsection NE, and the guidance in SRP 3.8.2 – See "Summary of Audit for Review of License Topical Report NEDE-33572P, Revision 2, Appendix C and Supporting Analyses," September 22, 2010.

To resolve the remaining issues, the applicant responded to RAI 6.2-202 S01, in letter MFN 10-044 Supplement 3, by providing details of its structural evaluation in Appendix B and Appendix C of LTR NEDE-33572P, Revision 3, "ESBWR ICS and PCCS Condenser Combustible Gas Mitigation and Structural Evaluation." The information included in these Appendices addresses the staff's concerns as described below:

The applicant determined, and the staff agreed, that the appropriate acceptance criterion to be used in the PCCS structural design, for load combinations including detonation loads, was Service Level C per the ASME Code, Section III, Subsection NE. It was not clear to the staff if all PCCS components within the jurisdictional boundary of ASME Code, Subsection NE, were designed to this criterion. The staff requested that the applicant confirm that all PCCS components within the containment boundary were designed using acceptance criteria for Service Level C. In response, the applicant confirmed that the design of all critical PCCS components within the jurisdictional boundary of ASME Code, Subsection NE, were modified to satisfy the corresponding allowable stress limits for Service Level C. Therefore, this item was resolved.

The structural analysis of critical PCCS components, for detonation loads, followed an equivalent-static approach in which detonation pressures were statically applied to FE submodels. All dynamic effects, including the effects of pressure wave reflections, were accounted for by using appropriate amplification factors. However, this equivalent-static approach did not address the dynamic effects of detonation loads on the entire PCCS assembly, including its supporting structure and anchorage. In RAI 6.2-202 S01, the staff requested that the applicant assess and include in its analysis and design the effects of detonations on the entire PCCS assembly, including its support structure and anchorage.

In response, the applicant performed an additional dynamic FE analysis to evaluate the effects of detonation loads on the entire PCCS assembly, including its supporting structure and anchorage. The dynamic analysis was performed by applying a spatially varying pressure time-history to the interior of the lower drum. This time-history represents the effect of a one-dimensional detonation pressure wave front initiating at one end of the lower drum, propagating along the length of the lower drum, and eventually reaching an internal equilibrium state. An

appropriate factor was considered to account for reflections of the pressure wave-front inside the lower drum. The applicant included the analysis method and results in Appendix B and Appendix C of the LTR. The staff reviewed the analysis method and the results presented by the applicant and considered them acceptable. The analysis appropriately considered the dynamic effects of detonations on the entire PCCS assembly by applying the dynamic pressure loads to the most critical area of the PCCS and evaluating its effects by a FE time-history analysis. For the design of the various components and supports of the PCCS, the applicant also appropriately considered the stresses and reaction loads from the aforementioned analysis. Therefore this item is resolved.

It was not clear to the staff that thermal effects following a detonation were accounted for in the structural design; particularly the thermal effects on the condenser tubes, which are slender elements restrained against longitudinal expansion. In response to RAI 6.2-202 S01, the applicant performed additional calculations to demonstrate that post-detonation thermal stresses induced in the condenser tubes are bounded by stresses due to detonation loads. The applicant added Section 2.2.7 to the LTR to document the results of this evaluation. Since the stresses due to the post-detonation thermal effects are bounded by the stresses due to detonation loads, this item is resolved.

Since the number of detonations expected to occur during the 72 hour-period associated with a LOCA could be high, the applicant was also asked in RAI 6.2-202 S01, to perform a fatigue evaluation for the total number of expected stress cycles. In response, the applicant performed a simplified fatigue evaluation of all critical PCCS components. The applicant demonstrated, and the staff agrees that the corresponding usage factors were sufficiently lower than 1.0 in all cases. Therefore, this item was resolved.

ICS

Similar to hydrogen accumulation in PCCS, there is a potential for hydrogen accumulation in the ICS tubes during post-LOCA conditions. In LTR NEDE-33572P, Section 4.2, GEH stated that during a LOCA event, the ICS injection is credited using the condensate stored in its drain piping. The heat removal through the ICS condenser is not credited for LOCA. However, there is potential for condensation to occur, and given enough time it is possible for combustible gases to accumulate in the ICS condenser to a detonable level following a LOCA. In order to prevent this buildup from occurring, a logic change was implemented for the ICS steam admission isolation valves in which the valves now automatically close after receiving an indication that the DPV have opened. The staff agrees with the applicant that closing the ICS steam admission isolation valves when the RPV is depressurized mitigates the accumulation of hydrogen and oxygen.

The applicant states that a TRACG evaluation shows that once it is isolated from the vessel, the ICS condenser pressure will drop below 15 psia from the reactor operating pressure within 2,000 seconds, and noncondensable gas partial pressure will not exceed 0.63 following isolation. The applicant also stated that detonation under these conditions is highly unlikely; however, if one were to occur the resulting loads would be within the original design pressure (1250 psig) of the ICS. The methodology by which the PCCS CJ pressures were calculated is also applied to the ICS; however, credit is taken for the detonation properties of the mixture, which contains no less than 37 percent steam (based on the TRACG evaluation). As a result, GEH used a CJ pressure ratio of 13.3 corresponding to 20 percent steam present in the noncondensable gas mixture per Table 4-1 of NEDE-33572P and calculated the maximum detonation pressure to be 1207 psia at 75 seconds after isolation, which is below the 1250 psia.

In addition, a fatigue evaluation will be conducted as part of the detailed design of the ICS and will be addressed in the design report for this component in accordance to Subsection NC of the ASME Code. This is acceptable to the staff because the loads from a potential detonation do not exceed the original design pressure of the ICS.

For non-LOCA events such as station blackout (SBO), GEH proposed modifications to the condenser vent function in order to keep the unit continuously purged of noncondensable gas. A logic change was implemented in which the lower head vent valves automatically open six hours after the ICS is initiated regardless of the system pressure. Once open, the vent will bleed steam and noncondensables from the condenser to the suppression pool, keeping the steam fraction at high levels (beyond the detonation range) throughout the event. Also, the vent valves are designed to fail open on a loss of power to provide additional reliability for this function. In addition, a flow restrictor is included in the vent line to keep the condenser purged and maintain the RPV water above Level 1 for 72 hours.

In RAI 6.2-202 S01, the staff requested justification that the six-hour time delay would be short enough to preclude the accumulation of a detonable concentration of hydrogen and oxygen in the ICS. In response, found in letter MFN 10-044 Supplement 3, GEH revised LTR NEDE-33572P Section 4, ICS Methodology, to provide the technical basis for the six-hour delay, and stated that ESBWR radiolytic hydrogen production calculation is consistent with the methodology of Appendix A to SRP Section 6.2.5 and RG 1.7. The staff reviewed the radiolytic gas production calculation results summary included in Section 4.1.2 of NEDE-33572P, Revision 3. The calculation results show very low gas production at six hours after SBO and consequently, Hydrogen and Oxygen concentrations are expected to be below the deflagration limits and hence acceptable and therefore, the issue was resolved.

Based on the above evaluation the staff finds that the applicant has addressed the possible accumulation of high concentrations of hydrogen and oxygen in the PCCS and ICS. The applicant has used an acceptable methodology to calculate concentrations of hydrogen and oxygen, to calculate loads and load combinations, to calculate stresses which meet applicable ASME code requirements. Based on the above, RAI 6.2-202 is resolved. **The applicant's proposed revision to DCD Revision 8, provided in response to RAI 6.2-202 S01, in letter MFN 10-044 Supplement 3, is a confirmatory item.**

6.2.2.4 *Conclusions*

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 staff realized that the ESBWR design has significant differences from the secondary containment of currently operating BWR facilities. The staff evaluation discusses these differences. Conformance with these regulatory criteria forms the basis for determining the acceptability of the RB functional design. The staff also reviewed the subcompartment analyses in accordance with SRP Section 6.2.1.2 for containment integrity. The staff also reviewed the design with respect to the associated regulatory guidance and criteria.

- GDC 4, "Environmental and Dynamic Effects Design Bases," 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, "Containment Design," as it relates to reactor containment and associated systems being provided to establish essentially leaktight barriers against the uncontrolled release of radioactive material to the environment

- GDC 43, "Testing of Containment Atmosphere Cleanup Systems," as it relates to atmosphere cleanup 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

- GDC 50, "Containment Design Basis," as it relates to the design of the containment internal compartments to ensure that the reactor containment structure, including access openings, penetrations, and the containment heat removal system are designed so that the containment structure and its internal compartments can accommodate, without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions resulting from any loss-of-coolant accident

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

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

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

- SRP Section 6.2.1.2, as it provides methods acceptable to the staff for the review of subcompartment analysis

- NUREG-1242, “NRC Review of Electric Power Research Institute’s Advanced Light Water Reactor Utility Requirements Document,” issued August 1992, with specific references to passive plant designs

6.2.3.2 *Summary of Technical Information*

The RB structure encloses penetrations through the containment (except for those of the main steam tunnel and IC/PCC pools). The RB has the following functions:

- 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
- houses safety-related systems.

The RB consists of rooms and compartments, which are served by one of the three ventilation subsystems: the contaminated area ventilation subsystem (CONAVS), refuel and pool area ventilation subsystem (REPAVS), and clean area ventilation subsystem (CLAVS). None of these compartmentalized areas communicates with any other.

Under accident conditions, the CONAVS and REPAVS areas of the RB automatically isolate on high radiation to provide a holdup volume for fission products. When isolated, the RB (CONAVS and REPAVS areas) can be serviced by the RB heating, ventilation, and air conditioning (HVAC) purge exhaust filter units. No credit is taken for the filters in dose consequence analyses. With low leakage and stagnant conditions, the basic mitigating function is the holdup of fission products in the RB CONAVS area itself. The ESBWR design does not include a secondary containment; however, in radiological analyses, credit is taken for the existence of the RB CONAVS area surrounding the primary containment vessel. RB CONAVS areas envelop all containment penetrations except penetration for main steam and feedwater lines located in the main steam tunnel. The radiological dose consequences for LOCAs, based on an assumed containment leak rate of 0.35 percent per day and RB CONAVS area leakage rate of 141.6 liters per second (L/s) (300 cubic feet per minute (cfm)), show that offsite and control room doses after an accident are less than allowable limits, as discussed in Chapter 15.

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. Therefore, GEH indicated that GDC 43 is not applicable.

RB leakage less than the maximum leak rate used in the accident dose calculations has the potential to increase the radiation dose inside the RB following a DBA. The environmental qualification program addresses the evaluation of the effect of increased radiation levels on equipment, and the emergency planning program, through emergency operating procedures, addresses any increased hazards during postaccident RB reentry.

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.

Design Bases

The RB 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 (RBVS) subsystems (CONAVS and REPAVS) automatically isolate upon detection of high radiation levels in their respective ventilation exhaust system.
- 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 due to 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 design allows for periodic testing to ensure that the leakage rates assumed in the radiological analyses are met. The radiological analyses assume that areas served by the RB CONAVS form this boundary.

Design Description

The RB is a reinforced concrete structure that forms an envelope completely surrounding the containment (except the basemat). During normal operation, the potentially contaminated areas in the RB are maintained at a slightly negative pressure relative to adjoining areas by the CONAVS portion of the RBVS. This ensures that any leakage from these areas is collected and treated before release. Airflow is from clean to potentially contaminated areas. Stack radiation monitors check RB effluents for radioactivity. If the radioactivity level rises above set levels, the discharge can be routed through the RB HVAC online purge exhaust filter unit 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. The RBVS is designed with safety-related isolation dampers and tested for isolation under various accident conditions.

HELBs 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.

The following paragraphs briefly describe the major compartments in the ESBWR design.

RWCU Equipment and Valve Rooms

The two independent RWCU divisions are located in the RB. 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 the pipe chase to the RB operating floor. The design-basis break for the RWCU system compartment network is a double-ended break. The applicant provided pressure profiles for all postulated RWCU/SDC system break cases for each individual room or region. The envelope profile represents the calculated maximum pressure response values for the given room or region due to all postulated RWCU/SDC system pipe breaks. These pressure profiles include no margin.

Isolation Condenser System

The ICs are located in the RB. 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, the pressurization analysis excluded this event.

Main Steam Tunnel

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

No blowout panels are required in the steam tunnel because the flowpath between the steam tunnel and the TB is open.

Design Evaluation

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 resulting in 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. Therefore, fission product leakage from the RB is mitigated.

Compartment Pressurization Analysis

RWCU pipe breaks in the RB and outside the containment were postulated and analyzed at 102 percent power and 187.8 degrees C (370 degrees F) feedwater temperature. For compartment pressurization analyses, HELB accidents are postulated as the result of piping failures in the RWCU system, where locations and size of breaks result in maximum pressure values. Calculated pressure responses have been considered in order to define the peak pressure of the RB compartments for structural design purposes. The calculated peak compartment pressures include a 10-percent margin. The maximum is 35.2 kPa(g) (5.11 psig), which is below the RB compartment pressurization design requirement. 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, "Pressure and Temperature Transient Analysis for Light Water Reactor Containments." 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.

A conservative RWCU model based on RELAP5/Mod3.3 has been developed to evaluate the mass and energy release for five break locations. Total blowdown duration is based on the assumption that the isolation valve starts to close at 46 seconds (1 second instrument time plus 45 seconds built-in time delay in blowdown differential flow detection logic) after the break and the isolation valve is fully closed in 15 seconds.

After the initial inventory depletion period, the steady RPV blowdown is choked at the venturi located upstream of the isolation valve since the venturi flow area is smaller than the isolation valve flow area. After the isolation valve starts closing, as soon as the valve area becomes equal to the venturi flow area, the break flow is choked at the isolation valve. The break flow stops when the isolation valve is fully closed.

The narrative of the event described above applies to all five cases analyzed since the breaks are all located downstream of the isolation valve and the dynamics of the break responses are similar.

Subcompartment pressurization effects resulting from the postulated breaks of high-energy piping have been analyzed according to ANSI/ANS-56.10, "Subcompartment Pressure and Temperature Transient Analysis in LWRs." To calculate the pressure response in the RB and outside the containment resulting from HELB accidents, the analysis used the CONTAIN 2.0 code. The nodalization contains the rooms where breaks occur, and all interconnected rooms or regions through flowpaths such as doors and hatches. The selected nodalization maximizes differential pressure. Owing to the geometry of the regions, each room or region was assigned to a node of the model. No simple or artificial divisions of rooms were considered to evaluate the sensitivity of the model to nodalization. A sensitivity study of pressure response was performed to select the time step. Additional sensitivity studies were performed to evaluate the impact of the heat sinks, dropout, and inertia term. Modeling follows the recommendations given by SMSAB-02-04, "CONTAIN Code Qualification Report/User Guide for Auditing Subcompartment Analysis Calculations."

Tests and Inspections

Position status indication and alarms for doors, which are part of the RB envelope, are tested periodically. Leakage testing and inspection of other architectural openings are also performed regularly.

The RB (CONAVS area) can be periodically tested to ensure that the leakage rates assumed in the radiological analysis are met, as required by TS 3.6.3.1. RB exfiltration testing is a positive pressure test of the CONAVS volume to confirm that the test leak criteria bound the analytical limit derived in the dose modeling. A nominal ¼-inch water gauge (WG) differential pressure bounds the effects of worst-case wind loading applied across a face of the RB. Many pressure measurements are taken at designated areas, and interconnecting doors and dampers are opened to ensure that uniform pressure is established within the contaminated areas of the RB (CONAVS area). The RB exfiltration test pressure is maintained for sufficient time to ensure that steady-state conditions are established (approximately ½ hour to 1 hour). These RB exfiltration test leak rate acceptance criteria are adjusted based on the actual CONAVS area test differential pressure applied to ensure minimal impact of test parameter uncertainties (flow instrument uncertainty, CONAVS area temperature, and pressure gradients).

Instrumentation Requirements

DCD, Tier 2, Section 7.3.3 gives details of the initiating signals for isolation. Doors that form part of the RB boundary are fitted with position status indication and alarms.

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 staff issued RAI 6.2-155, dated May 10, 2007, to obtain information on how the ESBWR complies with GDC 4. In response to this RAI, the applicant included in the DCD a description of analyses, such as pressurization due to high-pressure line break, and identified and stated that ITAAC in DCD, Tier 1, Table 2.16.5-2, will verify compliance with GDC 4. The staff concluded that the design complies with the requirements of GDC 4 in that the applicant has shown by analysis that the plant is designed to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including a LOCA. RAI 6.2-155 was being tracked as an open item. Based on the applicant's response, which included information linked to DCD changes and ITAAC, the staff determined that this open item was resolved.

GDC 16 states that 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. In the ESBWR, the RB CONAVS serves as the barrier against uncontrolled release of radioactivity to the environment from primary containment leakage through penetrations. In accordance with the staff position stated in NUREG-1242, the RB CONAVS is considered to be a safety envelope which is a concrete and reinforced steel structure (secondary containment) within the RB that forms an envelope completely surrounding the primary containment. NUREG-1242 allows appropriate credit for fission product holdup without requiring that a negative pressure be maintained in the secondary containment if the secondary containment leakage and mixing performance are consistent with the values used by the staff in its radiological assessment.

The applicant stated that the ESBWR does not include a secondary containment; however, the applicant takes credit for the existence of the RB CONAVS area surrounding the primary containment vessel in radiological analyses. The staff determined that the RB CONAVS functions as the secondary containment by providing tight controls on leakage through concrete and steel construction, a periodic leakage test program, and holdup volumes, as the principal means of controlling radioactive release.

The staff considered the applicant's statement with respect to the applicability of GDC 16, particularly with respect to the control of leakage from the RB CONAVS to the environment, because of its significant impact on the design-basis analysis dose results. The staff's method for calculating dose results is the RADTRAD software that models releases from the facility and determines an integrated dose over 30 days at control room, exclusion area boundary, and low-population zone receptors. Compliance with GDC 16 requires the applicant to show that the secondary containment leakage and mixing performance are consistent with the values used by the staff in its radiological assessment. The secondary containment leakage is the exfiltration rate. The mixing performance is the percent of the secondary containment volume credited for dilution in the RADTRAD design-basis analysis.

The applicant established two parameters based on the RB CONAVS design that are used as direct inputs to the design-basis analysis: an exfiltration rate from the RB CONAVS to the environment of 141.6 L/s (300 cfm), and an effective mixing volume that is 50 percent of the RB CONAVS volume, which is used to determine the dilution of the source term that is being released. The applicant also stated that the source term entering the RB CONAVS would be 0.35-percent mass of the primary containment per day. DCD, Tier 2, Table 15.4-5, documents these three parameters.

The applicant's basis for 141.6 L/s (300 cfm) exfiltration is a pressure test of the RB CONAVS volume, in which makeup airflow from a fan pressurizing the RB CONAVS is measured to be less than or equal to 141.6 L/s (300 cfm) as the RB CONAVS area is raised and maintained at ¼-inch WG positive pressure. In NUREG-1242, the staff agreed to consider holdup as a means to reduce releases to the environment, on the condition that the exfiltration rate be limited to 25-percent volume per day of the safety envelope volume. The RB CONAVS volume is the safety envelope volume. An exfiltration flow of 141.6 L/s (300 cfm) represents approximately 50 percent volume per day. Thus, the applicant is deviating from the staff position stated in NUREG-1242. The applicant's basis for the deviation is that it would be very difficult to conduct an accurate pressure test of a volume the size of the RB CONAVS with a maximum criterion of 25-percent volume per day.

The staff reviewed the deviation and acknowledges that it would be a difficult test situation. The staff is concerned that the quantity of holdup has not been explicitly established and would have a high degree of uncertainty. Keeping the exfiltration rate small lessens the impact of RB CONAVS releases to the environment due to the uncertainty in the holdup. The staff agreed to consider the increase in exfiltration rate, provided that the requirements of the design-basis dose analysis are met and the uncertainty in holdup is appropriately addressed.

The applicant's basis for an effective mixing volume of 50 percent of RB CONAVS volume is twofold: (1) a reference to RG 1.183 in which 50 percent mixing is permitted if adequate means can be shown to cause the environment to mix, and (2) a GOTHIC analysis which demonstrates that the actual release which occurs considering holdup is less than the release that results in the design-basis RADTRAD analysis using the 50-percent mixing volume, thus showing that the RADTRAD analysis is conservative.

The staff reviewed the reference to RG 1.183 and determined that it provides no justification for a 50-percent mixing rate for a passive design. 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." RG 1.183 clearly requires a means of mixing normally provided by the standby gas treatment system to take credit for 50-percent mixing. The applicant states in the DCD that RB CONAVS has low leakage and stagnant conditions, the exact opposite of a well-mixed environment.

The staff reviewed the arrangement and operation of the RB with respect to holdup and determined that the potential leakage from penetrations took place in penetration rooms that were concrete structures and were maintained closed by administrative controls and door alarms in the control room. Thus, if leakage occurred, it would build significantly in these penetration rooms before entering other parts of the RB CONAVS safety envelope. Before leakage from the RB CONAVS safety envelope would occur, levels of primary leakage would have to concentrate in order to be a significant contributor to the dose consequence analysis. The holdup time resulting from passing through multiple barriers provides for some decay of short-lived isotopes and ensures that a degree of mixing does in fact occur before release from the RB CONAVS safety envelope. Based on the robust concrete structures, closed penetrations rooms under administrative controls, and multiple barriers to release, the staff concluded that a 50-percent mixing assumption in the dose consequence analysis was reasonable.

The staff issued RAI 6.2-165, dated December 8, 2008, to obtain information on how the applicant established the assumption on mixing in the RB which is used in the DCD, Tier 2, Chapter 15 dose consequence analysis. In the interim, design changes occurred that changed the safety envelope from the entire RB to the contamination portion only, the mixing assumption to 50 percent per day of the contaminated volume, the primary containment leakage to 0.35 percent per day of the containment volume and added administrative controls on contaminated area doors and other related changes documented in DCD Revisions 5 and 6. In response to RAI 6.2-165 and the supplements that followed, the applicant submitted a GOTHIC analysis of the RB CONAVS volume to demonstrate that the releases from the RB CONAVS were significantly less than those determined by the RADTRAD dose consequence analysis using the 141.6 L/s (300-cfm) exfiltration and 50-percent mixing assumptions. The response included sensitivity studies and addressed uncertainties. The result of the analytical studies added credence to the determination that the 50-percent mixing assumption was acceptable.

RAI 6.2-165 was being tracked as an open item. Based on the applicant's response, which provided information and insight into the holdup capabilities, and in consideration of other staff confirmatory evaluations, this open item was resolved.

Although GOTHIC is a powerful tool for analyzing conditions throughout a building, many parameters require assumptions or careful measurements to obtain the results and would require revalidation over time. The applicant also adjusted some of the parameters, such as door gaps and leakage points, and showed that the sensitivity of most of the parameters had only a small effect. The staff has not previously accepted the use of GOTHIC as an analysis tool for this application. The application of GOTHIC to this safety evaluation is accepted as collaborating information.

The staff accepted the 50-percent mixing volume for use in RADTRAD on the following bases:

- The staff's determination that significant holdup would occur because of the robust concrete building room structures that form multiple barriers to release to the environment.

- A test program that ensures that the RB CONAVS safety envelope leakage would not exceed the 141.6 L/s (300 cfm) criterion that is part of the dose consequence analysis assumptions.

- The 50-percent mixing volume for the RADTRAD analysis adds substantial conservatism and accounts for holdup distribution changes in the RB CONAVS as the result of infiltration/exfiltration flow.

- Analytical evaluations and sensitivity studies provided by the applicant are consistent with the staff's evaluation and indicate that changes in temperature, resistance factors, and penetration leakage points have minimal impact on results.

- Appropriate ITAAC and administrative controls have been established to ensure that the RB CONAVS is constructed and maintained in accordance with the evaluated design.

The staff finds that the applicant has complied with GDC 16 by providing in the design the means to prevent uncontrolled release to the environment of radioactive effluents through holdup and limited leakage. As such, the applicant has ensured that the guidance values and limits of the radiological consequence analyses are not exceeded.

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 ensure (1) the structural and leaktight 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 DCD states that 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 does not have, a filter system that performs a safety-related function following a DBA. Therefore, the design criterion of GDC 43 is not applicable.

The staff issued RAI 6.2-166, dated May 30, 2007, to obtain information on how buildup of postaccident radiation in the RB is controlled and how it impacts access. In response to RAI 6.2-166 and its supplements, the applicant acknowledged that the absence of a standby gas treatment system allowed radiation levels to build in the contaminated portion of the RB after an accident and that these radiation levels could preclude entry for the purpose of making a cross-tie between the RWCU/SDC and the FAPCS to facilitate achieving cold shutdown. The applicant added a 472 L/s (1,000 cfm) RTNSS E filter system that could be used to clean up the contaminated portion of the RB after 72 hours. This system, the RB HVAC accident exhaust

filter system, exhausts to the environment through the RB vent. The applicant evaluated the impact on the dose consequence analysis and determined that the results of the dose consequence analyses presented in DCD, Tier 2, Chapter 15 bound the results of operation of this system on a parametric basis for all times greater than 8 hours into the accident. The applicant assigned a charcoal adsorber efficiency of 95 percent, based on compliance with RG 1.140, "Design, Inspection, and Testing Criteria for Air Filtration and Adsorption Units of Normal Atmosphere Cleanup Systems in Light-Water-Cooled Nuclear Power Plants." The staff determined that if this system were to operate in the 30-day accident recovery period, it would impact the dose analysis which is safety-related, that it is acceptable for the system to be classified as RTNSS since its operation is not required in the timeframe of 0–72 hours, but that the filter testing should be done in accordance with RG 1.52, "Design, Inspection, and Testing Criteria for Air Filtration and Adsorption Units of Post-Accident Engineered-Safety-Feature Atmosphere Cleanup Systems in Light-Water-Cooled Nuclear Power Plants," since it provides filter efficiency parameters to the dose consequence analysis.

The applicant responded that the system is not required after the accident and that it provides defense in depth. Emergency operating procedures would control the operation of the system. These procedures would confirm that there is no adverse impact on the dose consequence analyses before their operation. In addition, the filters would be tested to the same test requirements specified in RG 1.52, but the system would retain its classification as a non-safety system designed in accordance with RG 1.140. The staff concluded that the system facilitates the cleanup of the contaminated portion of the RB, does not impose any additional impact on release of radiation to the environment, and meets the requirements of GDC 43.

RAI 6.2-166 was being tracked as an open item. Based on the applicant's response, which included design changes to add an RTNSS qualified filter system that could be used after an accident and providing additional assurance that dose levels defined in the radiological consequences analyses documented in Chapter 15 would not be exceeded, the staff determined that this open item was resolved.

GDC 50 states that the containment internal compartments will be designed to ensure that the reactor containment structure, including access openings, penetrations, and the containment heat removal system are designed so that the containment structure and its internal compartments can accommodate, without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions resulting from any LOCA. The staff issued RAI 6.2-46 and RAI 6.2-154, dated April 12, 2007, to obtain additional information for the purpose of conducting confirmatory evaluations. In response to these RAIs, the applicant presented analyses using NRC-approved codes to demonstrate that the containment internal compartments are designed to meet GDC 50. The staff conducted confirmatory calculations for HELBs caused by pipe failures in the RWCU system, which show that the applicant's peak pressure is conservative and is below the design value for peak pressurization. In the applicant's conservatively chosen HELB cases, the maximum pressure observed is 35.2 kPa(g), which is less than the applicant's design limit of 36 kPa(g).

RAIs 6.2-46 and 6.2-154 were being tracked as open items in the SER with open items. Based on the applicant's response, which included analyses using NRC-approved codes, the staff evaluated internal compartment pressures and temperatures and determined that this open item was resolved.

In Appendix J to 10 CFR Part 50, Option A states in Section IV.B that other structures of multiple barrier or subatmospheric containments (e.g., secondary containments for BWRs and

shield buildings for PWRs that enclose the entire primary reactor containment or portions thereof) shall be subject to individual tests in accordance with the procedure established in the TS or associated bases.

The staff issued RAI 6.2-167, dated May 30, 2007, and RAI 15.4-26, dated January 29, 2007, to obtain information on leakage from the RB, test methods, and frequency of testing. In response to these RAIs, the applicant provided information on the test program and updated the DCD. The RB contaminated area, which serves as the safety envelope or, effectively, the secondary containment for release to the environment, is tested periodically under a positive pressure test as described in DCD, Tier 2, Revision 6, Section 6.2.3, and ensures that the exfiltration will be less than the value assumed in the dose consequence analyses. The staff concludes that the test program meets the intent of 10 CFR Part 50, Appendix J, Option A.

RAIs 6.2-167 and 15.4-26 were being tracked as an open item. Based on the applicant's response, which included information on testing, RB leakage, and releases to the environment tied to DCD Revision 6 changes, the staff determined that this open item was resolved.

The staff issued RAI 6.2-168 on May 30, 2007, to request clarification of issues concerning leakage from the RB. The RAI was based on DCD Revision 3. In response to RAI 6.2-168, the applicant provided information to address leakage rates from the RB. This information has been superseded by design changes and is no longer relevant. RAI 6.2-168 was being tracked as an open item, and it is now considered resolved.

6.2.3.4 *Conclusions*

The staff finds that the RB functional design, which provides for holdup in the contaminated portion (CONAVS) after an accident, and the subcompartment pressurization analysis are consistent with the guidance and criteria provided in SRP Sections 6.2.3 and 6.2.1.2 and other regulatory documents identified above. Thus, the design is acceptable.

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 to the environment of fission products 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, "Environmental and Dynamic Effects Design Bases," 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, "Containment Design," as it relates to the requirement that reactor containment and associated systems establish an essentially leaktight 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(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

- 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*

ESBWR DCD, Tier 2, Section 6.2.4, describes the proposed CIS for the ESBWR. The CIS protects 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 (CIVs). 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 CIVs 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 ensure that no single event can interrupt motive power to both closure devices.

CIV 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, and 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 such that the flow restriction is not plugged.

The applicant stated that in general, the design of the CIS meets all requirements of GDC 54, 55, 56, and 57 and follows the guidance of RGs 1.11 and 1.141, “Containment Isolation Provisions for Fluid Systems (for Comment).” DCD Section 6.2.4.3 gives a case-by-case analysis of all such penetrations. DCD, Tier 2, 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 CIVs inside the missile-proof RB. The arrangement of CIVs 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 standards.

CIVs and associated pipes are designed to withstand the peak calculated temperatures and pressures to which they would be exposed during postulated DBAs. They 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 ensure 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 the checklist for all nonpowered CIVs 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. During the review period, the applicant issued Revision 7 to DCD Tier 2. The staff finds that DCD, Tier 2, Revision 7, Section 6.2.4, satisfies the guidance and acceptance criteria of Section 6.2.4 of the SRP.

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 BTP 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
- locating as close as practical
- isolating at appropriate pressure
- exceptions listed in Table 1.9-6
- the provisions for, and TS 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 and venting requirements of 10 CFR 50.34(f)(2)(xiv) and (xv)

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

In RAI 6.2-102, the staff requested additional information concerning the need for CIVs for the PCCS in accordance with the guidance in ANS-56.2/ANSI N271-1976, "Containment Isolation Provisions for Fluid Systems." GEH responded that the design of the ESBWR containment cooling function does have precedent. In the Mark I style containment, the "light-bulb" shaped drywell is connected through a reinforced-concrete barrier by a series of metal ducts to the wetwell metal torus. This wetwell design is a contiguous part of the containment (not an extension or closed system outside of containment). This design contains features that are similar to those of the ESBWR, including the vent duct connections between the drywell and torus, which is a structural containment barrier that is not reinforced by concrete. The ESBWR containment is specifically designed to incorporate the safety-related function of containment cooling directly into the containment structure. Accordingly, GEH has pursued the development of a design that satisfies the applicable ASME Code requirements for Class MC containment vessel design and construction.

According to Section 6.2.2.4 of the DCD, the PCCS structural and leaktight integrity can be checked periodically by pressure testing. If additional ISI becomes necessary, ultrasonic testing

(UT) could be performed during refueling outages. The scope and frequency of the inspections will be determined as part of the ISI program as stated in the ASME Code, Section XI.

GEH also considered the need for CIVs for the PCCS from a risk assessment perspective. GEH stated that the question of whether to install CIVs is a classic tradeoff between the following:

- The CIVs are automatically or manually closed before or during accidents involving fuel damage if one or more PCCS tubes and/or heat exchanger modules exhibit significant leakage.

- Inadvertent automatic (or manual) closure of multiple CIVs during any accident requiring successful operation of the PCCS condensers could result in inadequate containment heat removal and an increase in the core damage frequency and/or large release frequency (LRF).

For the first bullet above, it is not evident that instrumentation could be designed with sufficient reliability to correctly identify a significant radiological release from one or more tubes and to automatically close the associated CIVs to and from the PCCS heat exchanger module(s) without isolating intact modules. Depending on operators to manually close the CIVs would be an even less reliable approach.

For the second bullet above, the probabilistic risk assessment uses a containment heat removal success criterion of four of six PCCS loops. Thus, inadvertent isolation of three or more PCCS loops would defeat the function.

The staff performed a confirmatory calculation to assess the existing risk of the PCCS design without CIVs. GEH used a 72-hour mission time for the heat exchanger leakage rate, which is not conservative as it assumes that the only degradation mechanisms that could occur happen during the accident. While the staff acknowledges that the tubes are fabricated from corrosion-resistant material, they are not immune to all degradation mechanisms, and 2 years or more could elapse between test and inspection, depending on the final ISI program. The staff felt that the conservative 1×10^{-6}/h heat exchanger leakage rate compensated for this nonconservative assumption. Finally, the staff used six PCCS heat exchanger modules in its analysis.

The staff used the following inputs when it repeated the risk assessment:

- a total core damage frequency (internal and external events at power) of 2.3×10^{-8}/year (yr) rather than the GEH value of 5.81×10^{-9}/yr

- a standby failure rate for large heat exchanger leaks (greater than 50 gpm) from NUREG/CR-6928, “Industry-Average Performance for Components and Initiating Events at U.S. Commercial Nuclear Power Plants,” issued February 2007 (the most recent and generally accepted operating experience data source) of 3×10^{-8}/h

- a fault exposure time of T/2 of 8,760 hours, assuming 2 years between testing/inspection

- a total of six PCCS heat exchange modules

The existing LRF from the proposed design (without CIVs) was recalculated as

$$(2.3\times10^{-8}/\text{yr}) * (3\times10^{-8}/\text{h}) * (8{,}760\ \text{h}) * 6 = 3.6\times10^{-11}/\text{yr for LRF.}$$

This value is 2 orders of magnitude greater than the GEH estimate of 4×10^{-13}/yr for the existing level of risk from large release due to PCCS leakage during severe accidents. However, the value of 3.6×10^{-11}/yr remains very low compared to the existing LRF from all other at-power severe accidents of about 1.7×10^{-9}/yr. More importantly, it remains lower by 4 or more orders of magnitude than the potential LRF increase in the alternate design due to inadvertent isolation of three or more PCCS heat exchanger modules during accidents requiring containment heat removal.

The staff's evaluation confirms the applicant's risk assessment conclusions and provides reasonable assurance that the proposed PCCS design without isolation valves represents lower risk than the alternative design with isolation valves.

The staff finds that the PCCS provides a functional feature of the ESBWR primary containment that ensures cooling in the event of a DBA. In addition, the PCCS is an inherent capability designed into the containment structure, not a separate fluid process system. This is a specific departure from past BWR plant designs. All previous BWR containment designs have relied on an external, pressurized, active fluid heat exchange system to provide containment cooling in response to a DBA. The PCCS negates the need for a separate, active safety-related cooling system and thus eliminates the need for fluid piping penetrations.

RAI 6.2-102 was being tracked as an open item in the SER with open items. Based on the above review and the precedent of the Mark I containment example, the staff finds the proposed design of the PCCS without isolation valves acceptable. RAI 6.2-102 was resolved.

In RAI 6.2-103, the staff asked that DCD, Tier 2, Table 1.9-6, be revised to state that the PCCS differs from SRP Section 6.2.4 acceptance criteria, in that it has no CIVs. The applicant indicated that it described its position on PCCS isolation in response to RAI 6.2-102, Supplement 1, and the issue will be resolved under that RAI. This staff concern was closed with the resolution of RAI 6.2-102, which concluded that the proposed design of the PCCS does not require CIVs and does not deviate from SRP Section 6.2.4 acceptance criteria.

The staff also asked that the process radiation monitoring system be added to Table 1.9-6, because it has both CIVs outside containment. The applicant responded that these lines conform to the provisions of RG 1.11 (as described in its response to RAI 6.2-127), which would mean that the lines do conform to SRP Section 6.2.4 acceptance criteria.

However, the applicant had not demonstrated that the system does conform to RG 1.11 (see the supplemental question to RAI 6.2-127), and so the staff requested 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. The applicant responded that it would address its position on containment isolation provisions of the process radiation monitoring system as part of its response to RAI 6.2-127, Supplement 1. This staff concern was resolved with the closure of RAI 6.2-127, as the applicant revised the DCD to include both inboard and outboard CIVs.

RAI 6.2-103 was being tracked as an open item in the SER with open items. Based on the applicant's response, RAI 6.2-103 was resolved.

In RAI 6.2-104, the staff pointed out that four systems did not meet the specific requirements of GDC 55 and 56. DCD, Tier 2, Revision 3, Table 1.9-6, listed three of the systems, and the fourth was the PCCS. The staff asked the applicant to clarify or correct this apparent discrepancy. To correct the inconsistency, the applicant responded that in DCD, Tier 2, Section 6.2.4, Revision 5, it had added a statement that there are exceptions to the explicit requirements of GDC 55 and 56 and that these exceptions are listed in Table 1.9-6 and are qualified on a case-by-case basis.

RAI 6.2-104 was being tracked as an open item in the SER with open items. Based on the applicant's response, RAI 6.2-104 was resolved.

In RAI 6.2-106, the staff requested that the third bullet in DCD Section 6.2.4.1 be revised to remove the statement "to the greatest extent practicable consistent with safety and reliability." As applicable, the applicant should request an exemption, or revise the statement to include "except as noted below" and then provide the specific exceptions. In response, the applicant revised DCD Revision 5, Section 6.2.4.1, third bullet, to remove the statement identified above, added a reference to identify the exemptions to the explicit requirements of GDC 55 through 57, and identified these exemptions in DCD Table 1.9-6.

RAI 6.2-106 was being tracked as an open item in the SER with open items. Based on the applicant's response, RAI 6.2-106 was resolved.

RAI 6.2-107 requested that the applicant clarify the following statement in DCD Section 6.2.4.1, seventh bullet: "Containment isolation valves and associated piping and penetrations meet the requirements of the ASME Boiler and Pressure Vessel Code, Section III, Class 1, 2, or MC, in accordance with their quality group classification." Class MC does not appear to meet the guidelines for a CIS. In response, the applicant stated that the seventh bullet refers to the code for the piping (ASME Section III, Class 1 or 2), as well as the steel components (ASME Section III, Class MC) of other than piping penetrations. In response to a supplement request, GEH revised DCD, Tier 2, Revision 5, Section 6.2.4.1, seventh bullet, to clarify that CIVs and associated piping meet the requirements of ASME Code Section III, Class 1 or 2, in accordance with their quality group classifications and added another bullet stating that piping penetrations (that is, penetrations themselves and not the pipes) are designed to the requirements of Subsection NE (MC components) of Section III of the ASME Code.

RAI 6.2-107 was being tracked as a confirmatory item in the SER with open items. Based on the applicant's response, RAI 6.2-107 was resolved. The staff confirmed that this change was included in DCD, Tier 2, Revision 5.

In RAI 6.2-109, the staff requested information about 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, 20-mm (0.8–in.) CIVs have closure times of 30 seconds or less.

- High-Pressure Nitrogen Gas Supply System—In Table 6.2-40, 50-mm (2–in.) CIVs F0009 and F0026 have closure times of 30 seconds or less.

Because DCD, Tier 2, Revision 3, Section 6.2.4.2.1, states that CIVs that are 80 mm (3 in.) 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," the staff was unsure if the quoted closure times of "30 seconds or less" for the above two systems are correct.

The applicant responded that it changed the closure times for the CIVs for the isolation condenser and high-pressure gas supply systems as listed in DCD, Tier 2, Tables 6.2-24, 6.2-26, 6.2-28, 6.2-30, and 6.2-40, in Revision 5 to indicate that the valves close within 15 seconds.

RAI 6.2-109 was being tracked as an open item in the SER with open items. Based on the applicant's response, RAI 6.2-109 was resolved.

In RAI 6.2-110, the staff questioned whether the instrument lines in the ESBWR design conform to the provisions of RG 1.11. GEH stated that it had revised the first paragraph of DCD, Tier 2, Revision 5, Section 6.2.4.2.2, to include sufficient information demonstrating conformance to each of the specific regulatory positions of RG 1.11, for every instrument line.

RAI 6.2-110 was being tracked as an open item in the SER with open items. Based on the applicant's response, RAI 6.2-110 was resolved.

In RAI 6.2-115(B), the staff asked for a more complete discussion of the single-failure evaluations performed for the CIS. The applicant stated that it will revise DCD, Tier 1, Section 2.15.1 and Table 2.15.1-2, and DCD, Tier 2, Section 6.2.4.3.3, as shown in the attached markups in Revision 5. GEH stated that the single-failure evaluation method for containment penetration isolation designs is based on the commitment to standards ANSI/ANS 58.9, "Single Failure Criteria for LWR Safety-Related Fluid Systems," and Institute of Electrical and Electronic Engineers (IEEE) 379-2000, "IEEE Standard Application of the Single-Failure Criterion to Nuclear Power Generating Station Safety Systems—Description" (see DCD, Tier 2, Table 1.9-22), and RG 1.53, "Application of the Single-Failure Criterion to Nuclear Power Plant Protection Systems" (see DCD, Tier 2, Tables 1.9-21 and 7.1-1, and Sections 7.13.3 and 7.5.2). DCD, Tier 2, Section 6.2.4.3.3, clarifies the method by which single failure is evaluated for containment isolation. Those commitments will be demonstrated under DCD, Tier 1, ITAAC Table 2.15.1-2.

RAI 6.2-115 was being tracked as an open item in the SER with open items. The staff has reviewed the applicant's response regarding the single-failure evaluations for the CIS and concluded it meets the requirements of RG1.53 and national standards ANSI/ANS 58.9 and is therefore acceptable. Based on the applicant's response, RAI 6.2-115 was resolved.

In RAI 6.2-117, the staff requested that more detailed information be added to DCD, Tier 2, Section 6.2.4.2.5, to describe the administrative controls to the extent that they are required by the regulations. In response, the applicant revised DCD, Revision 5, Section 6.2.4.2.5, to describe the manual valves that can be configured only to permit administrative control. Compliance with GDC 55 through 57 requires that the manual CIVs be locked closed. The staff has reviewed the applicant's response and finds it acceptable as these administrative controls meet the requirements of RG 1.141 and satisfy the national standards of ANS-56.2/ANSI N271-1976.

RAI 6.2-117 was being tracked as a confirmatory item in the SER with open items. Based on the applicant's response, RAI 6.2-117 was resolved. The staff confirmed that this change was included in DCD, Tier 2, Revision 5.

The containment isolation provisions of the IC condensate, venting, and purge lines consist of one barrier (a closed system) outside containment and two CIVs inside containment. In RAI 6.2-119 S01, the staff stated that this design does not comply with the explicit requirements of GDC 55 or GDC 56 and is inconsistent with the appropriate guidance documents (SRP Section 6.2.4, Revision 2; RG 1.141; and national standard ANS-56.2/ANSI N271-1976) concerning alternate means for complying with GDC 55 or GDC 56. These GDC 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 applicant stated in response, that because of the physical arrangement of the ICS condensate, venting, and purge line piping, it is impractical to locate an isolation valve outside the containment boundary. Such a valve would be under water and therefore inaccessible and less reliable than a valve located inside the containment boundary. As an alternative, two CIVs in series are located inside containment as close as possible to the containment boundary. The piping between the valves and containment boundary is designed to meet conservative requirements, precluding the occurrence of breaks in these areas. The ICS piping and components outside containment form a closed system designed to withstand the full reactor pressure.

The staff finds that 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 its proposed alternative to ensure 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 sufficient justification for moving it inside containment.

Based on the above evaluation, the staff finds that the containment isolation design for the ICS is considered an adequate alternative to the requirements of GDC 55 because a single failure would not disable the containment isolation function. Therefore, RAI 6.2-119 is considered resolved.

RAI 6.2-119 S01 was being tracked as open items in the SER with open items. Based on the applicant's response, RAIs 6.2-119 S01 was resolved.

RAI 6.2-121 is subsidiary to RAI 6.2-119. In RAIs 6.2-119 and 6.2-121, the staff made similar requests regarding the containment isolation design for the ICS. 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. The first RAI concerned the influent lines and the second RAI concerned the effluent lines. The staff's

supplemental RAI 6.2-119 addressed both the influent and effluent lines of the system. Based on the applicant's acceptable response to RAI 6.2-119 S01, RAI 6.2-121 was resolved.

In RAI 6.2-120, the staff noted that DCD, Tier 2, Revision 1, Section 6.2.4.3.1.2, under the heading describes 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 United States needs both gas pressure and spring force to close under accident conditions.

The applicant's response to RAI 6.2-120 explained the operation of the valves, which is similar to the operation of the MSIVs in other BWRs. The response included a proposed DCD Revision 3, Section 6.2.4.3.1.2. However, the applicant did not incorporate the proposed revision in DCD Revision 3, Section 6.2.4.3.1.2. On another note, the RAI response and DCD version refer to DCD Section 5.4.5 for further information, but that section does not address this particular issue. In a supplement, the staff requested the revision of the DCD to include the appropriate information as presented in the proposed DCD Revision 3 and to revisit the reference to Section 5.4.5.

The applicant's response to the supplement stated that DCD, Tier 2, Section 5.4.5, is the correct location for information regarding the design requirements and functional evaluation of the MSIVs, including the description of all relevant forces to which the actuation mechanism must respond during normal or abnormal operating conditions. The applicant provided the revised markup of DCD, Tier 2, Section 5.4.5, instead of revising Section 6.2.4.3.1.2. The staff's review of this information can be found in Section 5.4.5 of this report. Based on the above review, the staff finds this acceptable. Therefore, RAI 6.2-120 was resolved.

RAI 6.2-120 was being tracked as an open item in the SER with open items. Based on the applicant's response, RAI 6.2-120 was resolved.

In RAI 6.2-122, the staff requested that information about the containment isolation design for the FAPCS be provided in Section 6.2.4.3.2 to support the deviation from GDC 56. The staff also indicated that Table 6.2-33b should be corrected to be consistent with Table 6.2-33a for the CIV position on loss of electric or air supply. In response to RAI 6.2-122, GEH corrected DCD, Tier 2, Table 6.2-33b, to be consistent with DCD, Tier 2,Table 6.2-33a, for CIV position in case of power failure.

GEH also revised DCD Section 6.2.4.3.2 for the FAPCS to provide the following information.

The lines from the FAPCS penetrate the containment separately and are connected to the drywell spray, the suppression pool, the GDCS pools, and the reactor well drain.

The reactor well drainline contains two manual valves inside the containment that are locked closed during normal operation. This arrangement is an exception to GDC 56, which requires that such lines contain one isolation valve outside and one isolation valve inside the containment. The alternative arrangement with both valves inside containment is necessary because a valve outside containment would be submerged in the reactor well, making it inaccessible and less reliable. The isolation valves are located as close as possible to the containment, and the piping between the outermost valve and the containment boundary is designed to conservative requirements to preclude breaks in this area.

In each of the remaining influent lines, there is one pneumatic-operated or equivalent-shutoff valve outside and one check valve inside the containment. Only the GDCS pool return line pneumatic-operated or equivalent-shutoff valve is automatically closed on a containment isolation signal.

Before it exits containment, the FAPCS suction line from the suppression pool branches into two parallel lines, each of which penetrates the containment boundary. Once outside, each parallel flowpath contains two pneumatic isolation valves in series, after which the lines converge into a single flowpath. The CIVs are normally closed and fail as-is for improved reliability. "Fail-as-is" valves are acceptable because the valves are normally closed, will only be open when it is necessary to provide cooling to the suppression pool, and do not communicate with the drywell atmosphere. This arrangement is an exception to GDC 56, which requires that such lines contain one isolation valve outside and one isolation valve inside the containment. Such an alternative arrangement is necessary because the inboard valve could potentially be under water under certain accident conditions. Leak detection is provided for CIVs on the suppression pool suction line, and valves are located as close as possible to the containment.

The CIVs on the FAPCS suppression pool suction and return lines are considered to fail in the position of greatest safety. The CIVs in the suppression pool supply and return lines are closed for all normal operating conditions, except for temporary usage when suppression pool cooling or cleaning is needed. However, if the suppression pool cooling mode has been initiated before an accident, then it is more desirable to continue removing decay heat than to terminate the mode and isolate the system. This is clarified in DCD, Tier 2, Revision 5, Section 6.2.4.3.2. Therefore, the fail-as-is feature allows these valves to remain in an open position, which provides additional reliability for the RTNSS functions of suppression pool cooling and LPCI. Furthermore, the CIVs are designed to accommodate a single failure such that the line can still be isolated with the loss of a single division of power.

While the functions of suppression pool cooling and LPCI are not considered ESFs, they are considered RTNSS backups to ESFs, including the PCCS and GDCS. Therefore, the regulatory treatment that has been assigned to these functions, which utilize the FAPCS suppression pool flowpath, is justification for using the provisions of SRP Section 6.2.4, Revision 2, Section II.6.d.

The staff reviewed the information provided by GEH in response to RAI 6.2-122 as indicated above. The staff found that GEH provided the required information about the containment isolation design for the FAPCS in DCD, Tier 2, Section 6.2.4.3.2 to support the deviation from GDC 56 as per guidelines of SRP Section 6.2.4. RAI 6.2-122 was being tracked as an open item in the SER with open items. Based on the applicant's response, RAI 6.2-122 was resolved.

In RAI 6.2-123, the staff noted that, for 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, all of the CIVs were outside of containment, but without adequate justification as described in 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 changes to the DCD that address the guidelines.

The DCD states that the penetration of the containment inerting system consists of two tandem quarter-turn or equivalent shutoff valves (normally closed), in parallel with two tandem stop or

shutoff valves. All isolation valves on these lines are outside of the containment so that they are not exposed to the harsh environment of the wetwell and drywell and are accessible for maintenance, inspection, and testing during reactor operation. Both CIVs are located as close as practical to the containment. The valve nearest to the containment has the capability to detect and terminate a leak. The piping between the containment and the first isolation valve and the piping between the two isolation valves are designed to meet the requirements of SRP Section 3.6.2. The piping is designed to meet Safety Class 2 and seismic Category I design requirements and to withstand the containment design temperature, design pressure, and LOCA transient environment and is protected against an HELB outside containment when needed for containment isolation.

The staff has reviewed the applicant's response and redundant CIV arrangement. Because (1) the containment inerting isolation valves are normally closed during reactor operation, (2) piping between the containment and the CIVs is conservatively designed to preclude a breach of piping integrity, and the design of the valve and/or piping compartment provides the capability to detect leakage from the valve shaft and or bonnet seals and terminate the leakage according to the requirements of SRP Sections 3.6.2 and 6.2.4, and (3) locating both CIVs outside containment protects the valves from the harsh environment of the wetwell and drywell and allows accessibility for inspection and testing, the staff finds acceptable the proposed location of both inerting system CIVs outside the containment.

RAI 6.2-123 was being tracked as an open item in the SER with open items. Based on the applicant's response, RAI 6.2-123 was resolved.

RAI 6.2-125 is subsidiary to RAI 6.2-122. In RAIs 6.2-122 and 6.2-125, the staff made similar requests regarding the containment isolation design for the FAPCS. The first RAI concerned the influent lines and the second RAI concerned the effluent lines. The staff's supplemental RAI 6.2-122 addressed both the influent and effluent lines of the system. Based on the applicant's acceptable response to RAI 6.2-122, RAI 6.2-125 was resolved.

In RAI 6.2-127, the staff questioned the design of the process radiation monitoring system, particularly the placement of all CIVs 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 is acceptable because it follows the guidance in RG 1.11, Revision 1. The staff asked the applicant to provide a discussion showing that these lines conform to RG 1.11, or, if not, to identify the requirements for non-instrument lines.

In a supplement, the applicant stated that the design has been changed to include an inboard and outboard CIV on penetrations for the fission products monitor sampling line and return line. These two isolation valves are designed to a fail-as-is condition. In DCD, Tier 2, Revision 5, the applicant added a new Figure 6.2-30 to show these isolation valves and revised DCD Tables 3.9-8 and 6.2-42 to include both inboard and outboard CIVs.

RAI 6.2-127 was being tracked as an open item in the SER with open items. Based on the acceptable applicant's response, RAI 6.2-127 was resolved.

In RAI 6.2-128, the staff noted that DCD, Tier 2, Revision 1, Tables 6.2-39 through 6.2-42, does not include information covering the chilled water, high-pressure nitrogen gas supply, and process radiation monitoring systems.

In DCD, Tier 2, Revision 3, the applicant filled in the tables for the above systems. Based on its review, the new information was generally acceptable, but the staff had 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.

In response, the applicant stated the following:

> (A) These tables for the Chilled Water System (CWS) and High Pressure Nitrogen Gas Supply System (HPNSS) were corrected in DCD, Tier 2, Revision 4, to indicate GDC 56 as the applicable basis; (B) For the HPNSS, Table 6.2-40 will be revised to reference the appropriate DCD Tier 2 figures. For the Process Radiation Monitoring System, the response to RAI 6.2-127 S01 provides the appropriate DCD Tier 2 changes in Revision 5. For CIVs in the High Pressure Nitrogen Gas Supply System, response to supplement RAI 6.2-109 provides acceptable closure times.

The staff finds the applicant has provided the required information for the CWS, HPNSS and process radiation monitoring system CIVs in the DCD as per GDC 56.

RAI 6.2-128 was being tracked as an open item in the SER with open items. Based on the applicant's response, RAI 6.2-128 was resolved.

In RAI 6.2-131, the staff requested that the applicant discuss the following in the DCD:

A. The automatic isolation signals for CIVs and their diversity of parameters sensed, per item II.l of SRP Section 6.2.4, Revision 2.

B. Classification of systems as essential or non-essential and automatic isolation of non-essential systems during an accident per item II.h of SRP Section 6.2.4, Revision 2, and item II.E.4.2 of NUREG-0737.

C. Reducing the containment setpoint pressure that initiates containment isolation for non-essential penetrations to the minimum compatible with normal operating conditions, per item II.k of SRP Section 6.2.4, Revision 2, and item II.E.4.2 of NUREG-0737.

The GEH responses to parts A and B of RAI 6.2-131 were acceptable. In response to part A, GEH stated that DCD, Tier 2, Subsections 5.2.5 and 7.3.3.2 provides a discussion of the automatic isolation signals for CIVs and their diversity of parameters sensed as per item II.l of SRP Section 6.2.4, Revision 2. DCD Subsection 6.2.4 was revised to include a reference to the discussions in Subsection 5.2.5 and 7.3.3.2. The staff evaluation found the response to part A, acceptable.

In response to part B, GEH stated that instead of terms 'essential' or 'nonessential' for the classification of systems, GEH used the terms 'safety-related' and 'non-safety-related' for clarity when describing the importance of the functions of a system with regard to safety, similar to the terminology in NUREG-0737, item II.E.4.2, Table 1A-1. DCD, Tier 2, Subsection 6.2.4.1, provides the criteria for categorizing the fluid penetrations that require automatic isolation verses remote manual containment isolation based on the same basic criteria further described in SRP Section 6.2.4 Revision 3, Item II.8. Subsection 6.2.4.1 states, "The containment isolation function 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 Subsection 6.2.4.2 describes the systems containing penetrations that support or provide a flow path for emergency operation of ESF systems not automatically isolated. The staff evaluation found the GEH response to part B acceptable.

However, the staff had a further request for part C. In response to a supplement to RAI 6.2-131, part C, 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. In a supplement [to] this RAI, the staff requested that if setpoints are to be determined analytically, provide the actual numerical value and justify that it is 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.

GEH also stated that the ESBWR is in compliance with NUREG-0737, "Clarification of TMI Action Plan Requirements," issued November 1980. As currently stated in DCD, Tier 2, Appendix 1A, Table 1A-1, Item II.E.4.2, the alarm and initiation setpoints for a high-drywell-pressure condition are reduced to the minimum values compatible with normal operating conditions for containment penetrations containing process lines that are not required for emergency operation. However, the primary concern is to ensure that the high-drywell-pressure setpoint is set conservatively to the analytical limit used in the safety analyses. To clarify the basis of the high-drywell-pressure initiation signal, DCD, Tier 2, Appendix 1A, Table 1A-1, Item II.E.4.2, will be revised to state that the high-drywell-pressure setpoint is based on the analytical limit used in the safety analyses, and the reference to startup and preoperational test measurements will be deleted. The staff reviewed the proposed changes in DCD, Tier 2, Revision 6, and finds them acceptable.

The value for the high–drywell-pressure setpoint is the same for both the reactor protection system (RPS) scram signal and the containment isolation signal. DCD, Tier 2, Table 6.2-2, shows the analytical limit for the high-drywell-pressure signal as 13.8 kPa(g) (2 psig). This value is an upper analytical limit and is the basis for a setpoint calculation that will be performed to determine the actual instrument setting. This setpoint calculation will be based on the GEH "ABWR/ESBWR Setpoint Methodology" (see NEDE-33304P). A setpoint based on this analytical limit is compatible with the maximum normal operating drywell pressure of 8.96 kPa(g) (1.3 psig) identified in DCD, Tier 2, Chapter 16, "Technical Specification Limiting Condition for Operation (LCO) 3.6.1.4." The analytical limit is sufficiently low to ensure the performance of the necessary safety actions and, at the same time, high enough not to cause spurious reactor trips.

The alarm and initiation setpoints of the LD&IS are set as low as compatible with normal operation.

The actual setpoint will be based on instrument sensitivity and tolerance relating to actual installed instrument type, instrument range, setpoint drift, postevent function time, and environmental and process conditions and will ensure that the analytical limit is met. DCD, Tier 2, Sections 5.2.5 and 7.3.3, discuss the LD&IS parameters used to initiate these signals.

Based on the above evaluation, the staff finds the GEH response to RAI 6.2-131 acceptable. RAI 6.2-131 was being tracked as an open item in the SER with open items. Based on the applicant's response, RAI 6.2-131 was resolved.

DCD, Tier 2, Revision 3, contained 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, Section 6.2.4.2), presumably in a comprehensive way. However, Table 6.2-47 included many containment piping penetrations (approximately 122) that are not covered in Tables 6.2-15 through 6.2-42 or elsewhere in DCD, Tier 2, Revision 3, Section 6.2.4. Further, Table 6.2-47 contained virtually no information on the containment isolation provisions for these lines, other than incomplete information on leakage rate testing. Some systems were not covered in Tables 6.2-15 through 6.2-42.

In RAI 6.2-157 the staff requested that GEH address this issue. RAI 6.2-157 was being tracked as an open item in the SER with open items.

In response, GEH revised DCD Table 6.2-47 was to contain the required information for containment penetrations subject to Type A, B, and C testing and satisfies SRP Section 6.2.4 criteria. The CIV information DCD Tables 6.2-15 through 6.2-45, was also revised and information was added on the isolation valves in the makeup water system, service air system, containment monitoring system, and equipment and floor drain system.

In Supplement 1 to RAI 6.2-157, the staff stated that COL Item 6.2-1-H in Section 6.2.8 requires the COL holder to provide the missing information in Tables 6.2-16 through 6.2-45. This is the length of pipe between the containment and the isolation valve(s). Although it is understood that this information is not available until detailed design, GEH should provide acceptance criteria such that this information can be validated in ITAAC.

In response, GEH committed to the following design requirements:

> The containment isolation valves shall be located as close to the containment as practical. Sufficient space shall be provided between the valves and containment boundary to permit the following:
>
> - In-service inspection of non-isolable welds
> - Appendix J of 10CFR50 leak testing
> - Cutout and replacement of isolation valves using standard pipe fitting tools and equipment
> - Local control.
> - Valve seat resurfacing in place.

In Supplement 2 to RAI 6.2-157, the staff stated that the proposed design criteria for locating the pipes is reasonable. However, the GEH response did not allow a safety conclusion that the ESBWR complies with GDC 55, 56, and 57. Therefore, GEH must include the appropriate design in the DCD to demonstrate compliance with GDC 55, 56, and 57, and an ITAAC item must also be added to ensure that the detailed design complies with the guidance in the DCD.

In response, GEH stated that the design considerations for locating CIVs as close to the containment as practical, which were provided in the response to Supplement 1 of this RAI (MFN 08-475, dated May 13, 2008), would be added to DCD, Tier 2, Section 6.2.4.2. An ITAAC item would be added to DCD, Tier 1, Table 2.15.1-2, to document the locations of CIVs relative to containment and to review these locations relative to the design considerations. COL Holder Item 6.2-1-H, which was to provide the pipe lengths between the CIVs and containment, would be deleted from DCD, Tier 2, Section 6.2.8. The piping lengths in DCD, Tier 2, Tables 6.2-16 through 6.2-45, would also be deleted.

DCD, Tier 1, Section 2.15.1 and Table 2.15.1-2, and DCD, Tier 2, Sections 6.2.4.2 and 6.2.8 and Tables 6.2-16 through 6.2-45, were to be be revised accordingly. The staff confirmed that these changes were incorporated in DCD Tier 1 and 2, Revision 6.

Based on the applicant's response, RAI 6.2-157 and the associated open items were resolved.

In DCD, Tier 2, Revision 5, Tables 6.2-36, 6.2-37, and 6.2-38, "Containment Isolation Valve Information for the Containment Inerting System," refer to Figure 9.4-14 for valve location. However, in Revision 5, Figure 9.4-14 has been moved to Chapter 6. The staff requested that the applicant update the above tables to reflect the proper reference and update Figure 6.2-29 to include the containment inerting system. In addition, Figure 6.2-29 should include isolation valve F023 and penetration numbers.

GEH agreed to make the necessary changes in DCD, Tier 2, Tables 6.2-36, 6.2-37, and 6.2-38 and Figure 6.2-29. The staff confirmed that these changes were incorporated in DCD, Tier 2, Revision 6. Based on the applicant's response, RAI 6.2-199 was resolved.

In DCD, Tier 2, Revision 5, Tables 6.2-16 to 6.2-40 present CIV design information. These tables typically refer to other Tier 2 figures for information such as isolation valve(s) and containment penetration. However, many of the referenced figures do not show such information. In tables that refer to other figures for design details, the referenced figures should be updated to show the isolation valve(s) and penetration numbers.

In Tables 6.2-41, 6.2-43, 6.2-44, and 6.2-45, the entries that typically give design information show "N/A" for Tier 2 figures. Thus, there is no design figure (e.g., piping and instrumentation diagram, process diagram). In RAI 6.2-200, the staff requested that these tables be revised to include figure(s) showing the isolation valve(s) and penetration numbers.

In response to RAI 6.2-200, GEH stated that it would revise the DCD to ensure that there are figures showing all CIVs, and that all CIVs and penetrations are labeled with their component numbers on the figures. GEH also agreed to make additional changes to the DCD to correct information associated with CIVs.

GEH provided a markup of the revised tables and figures. The staff confirmed that these changes were incorporated in DCD, Tier 2, Revision 6. Based on the applicant's response, RAI 6.2-200 was resolved.

Generic Issues

The two generic issues included in the staff's review of the CIS are TMI Action Plan Items II.E.4.2, "Containment Isolation Dependability," and II.E.4.4, "Containment Purging During Reactor Operation" of NUREG-0737.

II.E.4.2, "Containment Isolation Dependability" (10 CFR 50.34(f)(2)(xiv))

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 CIVs 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.

DCD, Tier 2, Table 1A-1, states that the ESBWR CIS meets the NRC requirements, including the post-TMI requirements. In general, this means that two barriers are provided, as discussed in DCD Section 6.2.4.3.

Redundancy and physical separation are required in the electrical and mechanical design of the CIS to ensure that no single failure in the system prevents it from performing its intended functions. Electrical redundancy is provided for each set of isolation valves, such that the unavailability of any two safety-related electrical divisions will not prevent isolation from occurring. Electrical cables for isolation valves in the same line are routed separately. Cables are selected and based on the specific environment to which they may be subjected (e.g., magnetic fields, high radiation, high temperature, and high humidity).

Safety-related or non-safety-related (essential or nonessential) classification of SSCs for the ESBWR design is addressed in DCD, Tier 2, Section 3.2 and identified in DCD, Tier 2, Table 3.2-1. Section 3.2 also presents the basis for classification.

The CIS, in general, closes fluid penetrations for support systems that are not safety-related. The design of the control systems for automatic CIVs ensures that resetting the isolation signal does not result in the automatic reopening of CIVs.

Actuation of the CIS is automatically initiated by the LD&IS, at specific limits (described in DCD Sections 5.2.5 and 7.3.3) defined for reactor plant operation. The LD&IS is designed to detect, monitor, and alarm leakage inside and outside the containment and automatically initiates the appropriate protective action to isolate the source of the leak. Various plant variables are monitored, including pressure, and these are used in the logic to isolate the containment. The drywell pressure is monitored by four divisional channels, using pressure transmitters to sense the drywell atmospheric pressure from four separate locations. A pressure rise above the nominal level indicates a possible leak or loss of reactor coolant within the drywell. A high-pressure indication is alarmed in the main control room (MCR) and initiates reactor scram and, with the exception of the MSIVs, closure of the CIVs in certain designated process lines.

All ESBWR containment purge valves meet the criteria provided in BTP CSB 6-4. The main purge valves are fail-closed and are verified to be closed at a frequency interval of 31 days as defined in the plant TS (SR 3.6.13.1). All purge and vent valves are pneumatically operated, fail closed, and receive containment isolation signals. Bleed valves and makeup valves can be manually opened remotely in the presence of an isolation signal, by utilizing override control if continued inerting is necessary.

In the ESBWR design, redundant primary CIVs (purge and vent) close automatically upon receipt of an isolation signal from the LD&IS. The LD&IS is a four-division system designed to detect and monitor leakage from the RCPB and, in certain cases, isolates the source of the leak by initiating closure of the appropriate CIVs. Various plant variables are monitored, including radiation level, and these are used in the logic to initiate alarms and the required control signals for containment isolation. High-radiation levels detected in the RB HVAC air exhaust or in the refueling area air exhaust automatically isolate the containment purge and vent isolation valves.

Based on the above review of the information in the DCD, the staff found that the ESBWR CIS design meets the requirements of post-TMI Generic Issue Item II.E.4.2, “Containment Isolation Dependability” as per 10 CFR 50.34(f)(2)(xiv) and follows the guidance provided in SRP Section 6.2.4 and therefore, is acceptable.

II.E.4.4, Containment Purging During Reactor Operation (10 CFR 50.34(f)(2)(xv)

The governing regulation for TMI Action Plan Item II.E.4.4, Containment Purging During Reactor Operation, 10 CFR 50.34(f)(2)(xv), states :

> Provide a capability for containment purging/venting designed to minimize the purging time consistent with ALARA principles for occupational exposure. Provide and demonstrate high assurance that the purge system will reliably isolate under accident conditions. (II.E.4.4)

The DCD entry for this generic issue, in Table 1A-I, "TMI Action Plan Items," simply asserts that the ESBWR design complies with these requirements, without explanation or justification.

The first requirement of the regulation refers to a situation that generally does not occur in a plant with an inerted containment atmosphere, which is unwarranted or excessive containment purging. The NRC established this generic issue because it had found that some (noninerted) plants were purging/venting their containments for sizable fractions of the plant's operating time, or even continuously. The NRC recognized that an open purge/ventline constitutes a sizable hole in the containment boundary, which is intrinsically a less safe condition than having all purge/vent valves closed, in case an accident occurs.

One legitimate reason for purging while the reactor is operating is to reduce the concentration of airborne radioactive material in the containment atmosphere, which would reduce the occupational exposure of personnel who enter containment. The regulation, then, calls for minimized purging time, consistent with as low as reasonably achievable (ALARA) principles for occupational exposure. However, personnel do not enter containments while they are inerted, so there is no need to purge for this reason. In general, plants with inerted containment will naturally minimize purge/vent time (except when inerting or de-inerting) because of the cost of the nitrogen gas needed to replace that which is expelled from containment. Also, as mentioned before, personnel exposure during containment entries is not a factor.

Despite these facts, the applicant must provide a discussion in the DCD that presents these or similar arguments to demonstrate compliance with the requirement of 10 CFR 50.34(f)(2)(xv).

The second requirement of the regulation, to provide and demonstrate high assurance that the purge system will reliably isolate under accident conditions, is explained in more detail in NUREG-0737, Item II.E.4.2, subpart (6) and Attachment 1. The staff had found that some purge/vent valves (typically butterfly valves) in operating plants were not capable of closing if a design-basis LOCA occurred while the valves were open.

In a design-basis LOCA, containment pressure increases so rapidly that the containment atmosphere rushes out through open purge/vent valves before they can begin to close. Some valves were found to be incapable of closing against the aerodynamic forces induced by the rapidly moving gas; in fact, some valves would even be damaged by the transient so that they would be stuck open and incapable of closing again until repaired. The regulation, therefore, requires the applicant to demonstrate, by analysis and/or testing, that the purge/vent valves would be capable of closing under these conditions. An alternative to such demonstration is to ensure that purge/vent valves will never be open while the plant is operating, by including a requirement in the TS that the valves must be locked or sealed closed in Modes 1 through 4, with no exception for even momentary opening of a purge/ventline while in Modes 1 through 4.

In RAI 6.2-179, the staff requested that the applicant provide the following information in the DCD to demonstrate compliance with the requirements of 10 CFR 50.34(f)(2)(xv):

- Containment purging/venting capability is designed to minimize the purging time consistent with ALARA principles for occupational exposure.

- There is high assurance that the purge system will reliably isolate under accident conditions, or the applicant should provide TS which require purge/vent valves to be sealed closed in Modes 1 through 4.

- The applicant should identify all purge/vent valves. This includes all CIVs in lines that perform a purging or venting function.

In response to RAI 6.2-179, the applicant revised DCD, Tier 2, Chapter 16, TS SR 3.6.1.3.1, to eliminate the specific sizes of the purge/vent valves, and DCD, Tier 2, Chapter 16B, TS SR 3.6.1.3.1, "Bases," to include the 25-mm, 350-mm, and 400-mm purge/vent valves, as well as the 500-mm purge/vent valves. These other purge/vent valves exist within the same system (described below) as the 500-mm valves. The other systems that penetrate containment and have direct contact with the containment atmosphere (the process radiation monitoring system and the containment monitoring system) do not have a purge/vent capability. The following information is the response that GEH provided to RAI 6.2-179:

- The containment purging/venting is performed using the containment inerting system. DCD, Tier 2, Section 6.2.5.2A, describes this system. The containment inerting system is used to establish and maintain an inert atmosphere within the containment during all plant operating modes, except during plant shutdown for refueling or maintenance and during limited periods of time to permit access for inspection and maintenance during reactor low-power operation. The system is designed to permit de-inerting the containment for safe operator access and minimizing personnel exposure. DCD, Tier 2, Chapter 16, TS SR 3.6.1.8, sets out the conditions for inerting and de-inerting containment (see the response to RAI 16.2-110, Supplement 2 in MFN 07-025, Supplement 2). The applicant revised DCD Section 6.2.5.1.1 to describe the function of the containment inerting system in relation to minimizing personnel exposure.

- As discussed in DCD, Tier 2, Section 3.9.3.5, valves that perform an active safety-related function will be functionally qualified to perform their required functions, using ASME QME-1-2007, "Qualification of Active Mechanical Equipment Used in Nuclear Power Plants," as guidance. A qualification specification (i.e., purchase specification), consistent with Appendices QV-I and QV-A to QME-1, will be prepared for the containment purge valves to ensure that the operating conditions and safety functions for which the valves are to be qualified are communicated to the manufacturer or qualification facility. In addition, as discussed in the DCD markup of Tier 2, Revision 4, Section 3.9.6.8 (MFN 08-131), active safety-related valves, including the containment purge valves, will be preoperationally tested to verify that they are properly set to perform their required functions. Finally, the containment purge valves will be periodically tested as shown in DCD, Tier 2, Revision 4, Table 3.9-8, as part of the inservice testing program. This testing includes periodic valve exercise testing (including stroke time measurement), verification of fail-safe performance, local leakage rate testing, and remote position indicator tests.

- Containment purging/venting is performed using the containment inerting system. A complete list of CIVs for this system appears in DCD Tables 6.2-36, 6.2-37, and 6.2-38.

The applicant revised the DCD to show the specific design information of purge valves which staff found acceptable. Based on the above review, the staff finds GEH's response to RAI 6.2-179 demonstrates compliance with the requirements of 10 CFR 50.34(f)(2)(xv) and follows guidance provided in SRP Section 6.2.4 and therefore, is acceptable. RAI 6.2-179 was being tracked as an open item in the SER with open items. Based on the applicant response, RAI 6.2-179 was resolved, and the CIS meets the requirements of post-TMI Generic Issue Item II.E.4.4.

6.2.4.4 *Conclusions*

On the basis of its review, the staff concludes that the proposed ESBWR CIS, described in the DCD, complies with the acceptance criteria of Section 6.2.4 of the SRP. Compliance with the criteria in Section 6.2.4 of the SRP, as described in this section, constitutes an acceptable basis for satisfying the CIS requirements of GDC 1, 2, 4, 16, 54, 55, 56, and 57 and the additional TMI-related requirements of 10 CFR 50.34(f)(2)(xiv) and 10 CFR 50.34(f)(2)(xv).

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 appear 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, both titled "Contents of Applications; Technical Information," and added a new section, 10 CFR 50.46a, "Acceptance Criteria for Reactor Coolant System Venting Systems." 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. In 10 CFR 50.44(a)(1) "inerted atmosphere" is defined as "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 DBAs and significant beyond-design-basis accidents (BDBAs). 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 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, "Containment Atmosphere Cleanup," 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, "Testing of Containment Atmosphere Cleanup Systems," 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 ITAAC that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a facility that incorporates the design certification has been constructed and will be operated in conformity with the design certification, the provisions of the Atomic Energy Act, and the Commission's rules and regulations

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 6, states that the ESBWR meets the relevant requirements of the following:

- 10 CFR 50.44 and 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors," apply as they relate to BWR plants being designed to have containments with an inerted atmosphere.

- GDC 5, "Sharing of Structures, Systems, and Components," does not apply to the inerting function because there is no sharing of SSCs between different units.

- GDC 41, "Containment Atmosphere Cleanup," 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, "Inspection of Containment Atmosphere Cleanup Systems," and GDC 43, "Testing of Containment Atmosphere Cleanup Systems," 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 (oxygen concentration below the maximum permission limit of 4 percent during normal power operation) as discussed in DCD, Tier 2, Section 6.2.5.2. The containment inerting system can be used under postaccident conditions for containment atmosphere dilution to maintain 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 discussed in DCD, Tier 2, Section 6.2.5.3, 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 containment monitoring system 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: This system, discussed in DCD, Tier 2, Sections 6.2.5.3.1 and 6.2.5.3.2, respectively, 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 MCR 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.

Radiation Monitoring: This system, discussed in DCD, Tier 2, Section 6.2.5.3.3, consists of two channels per division (1 and 2) of radiation detector assemblies, radiation electronic assemblies and visual displays. The channels measure gross gamma radiation in the drywell and suppression chamber. The signals are carried back to the MCR where the signals are continuously displayed. The channels are equipped with an alarm to indicate channel malfunction. The radiation monitoring channels are divided into two redundant measurement 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, Tier 2, Section 6.2.5.3.4, discusses in detail how adequate mixing within the ESBWR containment system is assured based on the configuration of the containment, coupled with the dynamics of the design-basis LOCA and the mitigating components within the containment volume.

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 a hypothetical situation in which containment depressurization is required, manual operator action can perform this depressurization.

Containment Structural Integrity: DCD Appendix 19B presents the deterministic analysis performed and results obtained for the containment ultimate capability under internal pressure in accordance with the requirements in 10 CFR 50.44(c)(5) and SECY-93-087, "Policy, Technical,

and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs," dated April 2, 1993. Section 19.B of this report presents the evaluation of containment structural integrity.

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 sufficient time will be available 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 radiolytic oxygen concentration in containment was analyzed consistent with the methodology of Appendix A to SRP Section 6.2.5 and RG 1.7.

The analysis results show that the time required for the oxygen concentration to increase to the de-inerting value of 5 percent is 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 was, however, an open item concerning the placement of a 4 percent by volume limitation on containment oxygen concentration in the TS. In RAI 16.2-110, the staff asked GEH to propose TS Section 3.6, "Containment Systems," for containment oxygen concentration. GEH asserted that an operating restriction on oxygen concentration (to less than 4 percent 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, "Technical Specifications," and thus it is not included in the proposed TS.

However, both the NRC staff and the nuclear industry's Technical Specification Task Force have stated in the following that such TS are required:

- When the NRC revised 10 CFR 50.44 in 2003, the staff issued a model safety evaluation for implementation of the revised rule through the Consolidated Line Item Improvement Process, The model safety evaluation 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) include 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, "Elimination of Hydrogen Recombiners and Change to Hydrogen and Oxygen Monitors," dated July 18, 2003, which the staff has accepted, 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, the staff requested that GEH add a TS limiting containment oxygen concentration to less than 4 percent by volume.

GEH responded to RAI 16.2-110 and its Supplements 1 and 2 by agreeing to provide a new TS 3.6.1.8, "Containment Oxygen Concentration," and associated bases in DCD Chapters 16 and 16B, respectively. In addition, GEH deleted Availability Control 3.6.1, "Containment Oxygen," from Chapter 19.

GEH also proposed to incorporate a new special operation TS. TS 3.10.9, "Oxygen Concentration—Startup Test Program," will allow suspension of requirements of LCO 3.6.1.8 for the first 120 effective-full-power days, during performance of startup tests.

To allow containment entry for required startup tests without increasing personnel risks due to the oxygen-deficient atmosphere, GEH stated that the proposed TS 3.10.9 is generally consistent with NUREG-0123, "Standard Technical Specifications for General Electric Boiling Water Reactors," BWR/4 Standard Technical Specifications, and TS 3.10.5, "Oxygen Concentration," as modified and presented in NEDC-31681, "BWR Owner's Group Improved Technical Specification," for BWR/4 improved TS 3.10.12, "Oxygen Concentration—Startup Test Program."

The staff finds the GEH response adds a TS limiting containment oxygen concentration to less than 4 percent by volume as requested. RAI 16.2-110 was being tracked as an open item in the SER with open items. Based on the applicant's response, RAI 16.2-110 was resolved.

6.2.5.3.2 Mixed Atmosphere

The staff reviewed the ESBWR's capability to ensure a mixed atmosphere during DBAs and significant BDBAs.

In RAI 6.2-138, the staff requested that GEH provide additional description of the design's capability to ensure a mixed containment atmosphere. GEH was asked to address the following: 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 on 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.

In its response to RAI 6.2-138 and its supplements, GEH revised or proposed changes to DCD, Tier 2, Section 6.2.5. The following is the staff's evaluation of the containment mixing portion of the DCD.

The drywell and wetwell are inerted with nitrogen to meet 10 CFR 50.44. Containment mixing is not as critical for inert containments as it is for plants with mitigative features that recombine hydrogen and oxygen. In DCD, Tier 2, Section 6.2.5.3.4, GEH described the design features to ensure sufficient mixing for the drywell, wetwell, drywell head region, and RSA. The staff acknowledges that these features ensure that postaccident steam and entrained noncondensable gases will be transported to the PCCS heat exchangers. The PCCS heat exchangers are designed to condense the steam and transfer the majority of the noncondensable gases to the wetwell by the PCCS heat exchanger ventline. Another consideration with respect to the mixing process is the incorporation of passive autocatalytic recombiners (PARs) into both the drywell and wetwell. They have been included to assist in long-term pressure control and as defense-in-depth protection against the potential buildup of combustible gases generated by the radiolytic decomposition of water. DCD, Tier 2, Section 6.2.5.1, describes the PARs.

PARs are passive devices that operate when the surrounding atmosphere contains a stoichiometric mix of hydrogen and oxygen. The PARs contain a catalyst that facilitates the recombination of hydrogen and oxygen gases into water vapor. They also create convective air currents (recombination is an exothermic reaction), which further the recombination process and mixing within the drywell and wetwell atmosphere.

The number and size of PARs to be used in each containment compartment will be selected based on the nominal hydrogen depletion rate of each individual PAR unit such that the total depletion rate is twice the maximum hydrogen generation rate at 72 hours. The maximum hydrogen generation rate at 72 hours is 0.32 kilograms per hour, based on the methodology of RG 1.7 and the analytical assumptions in Section 6.2.5.5.2 of the DCD. The number and size of PARs specified will provide the minimum safety factor of 2 for each containment compartment (drywell and wetwell) to account for possible catalytic poisons.

The minimum capacity will be the equivalent of one full-size PAR unit specified for each containment compartment; however, because of other design considerations, more and smaller capacity units (with equivalent total capacity) will be specified. This will result in more complete coverage of the wetwell and drywell. The nominal hydrogen depletion rates for the full-size PAR will be a minimum of 0.8 kilograms per hour. The PARs are sited with consideration of factors such as protection from jet impingement, protection from containment spray and cooling fan discharge, protection from flooding and PS, discharged exhaust impacts, and accessibility for testing.

The staff reviewed the information provided by GEH in response to RAI 6.2-138 and its supplements and found it acceptable because the applicant revised the DCD to provide specific design criteria for the PARs consistent with RG 1.7. The ESBWR design meets 10 CFR 50.44(c)(1), based on DCD, Tier 2, Revision 6, Sections 6.2.5.1 and 6.2.5.3.4 4

because of passive features of the containment design for ensuring a mixed atmosphere during design-basis and significant beyond design basis accidents. RAI 6.2-138 was being tracked as an open item in the SER with open items. Based on the applicant's response, RAI 6.2-138 was resolved.

6.2.5.3.3 Oxygen Monitor

The regulation in 10 CFR 50.44(c)(4)(i) requires that equipment for monitoring oxygen 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.

In RAI 6.2-137 and its Supplements 1 and 2, the staff requested additional information concerning the range of measurement of the oxygen monitors and their functionality, reliability, and accuracy, and justification that the proposed monitors are adequate for their intended function. The RAI also inquired about functionality and reliability of the monitors when exposed internally to the temperature, pressure, humidity, and radioactivity of the containment atmosphere during a significant BDBA.

In its response to RAI 6.2-137, GEH stated that equipment chosen for oxygen monitoring will be specified to meet the environmental and radiological requirements for its location and for intended postaccident operations. GEH also stated that internal components will be evaluated to ensure that the instrument is qualified for the intended environmental and radiological conditions expected and for the required postaccident monitoring timeframe.

With respect to the accuracy of the oxygen monitors, GEH responded that it would comply with Table 2 in RG 1.97, Revision 3, "Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants," issued May 1983, where the required instrument range for a Type C variable is 0 to 10 percent volume for oxygen in inerted containments.

GEH stated that it would revise DCD, Tier 2, Section 7.5.2.1, and add a new table, Table 7.5-5 (the markup was provided in the GEH response) indicating the required instrument range. The staff confirmed that the applicant incorporated this change in DCD, Tier 2, Revision 6.
s described in Sections 6.2.5 and 7.5.2 of the ESBWR DCD, the oxygen monitors are a safety-related, seismic Category 1 system consisting of two redundant, physically and electrically independent postaccident monitoring divisions. The oxygen monitors are environmentally qualified (EQ). Section 19.3.4.2 of the ESBWR DCD identifies the oxygen monitors as equipment required for severe accident mitigation. Section 19.2.3.3.7 of this report evaluates the survivability of the oxygen monitors. The oxygen monitors are located outside the drywell and wetwell, as shown in Figure 7.5-1 of the ESBWR DCD.

The staff reviewed the information provided by GEH for oxygen monitor in response to RAI 6.2-137 and supplements and found it acceptable because the applicant revised the DCD to provide specific design criteria for the oxygen monitor consistent with RG 1.7 and RG 1.97. The ESBWR design of the oxygen monitors meets the regulation of 10 CFR 50.44(c)(4)(i) in accordance with the guidelines of SRP Section 6.2.5 and therefore is acceptable.. RAI 6.2-137 was being tracked as an open item in the SER with open items. Based on the applicant's response, RAI 6.2-137 was resolved.

6.2.5.3.4 Hydrogen Monitor

The regulation in 10 CFR 50.44(c)(4)(I) requires that 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 combustible gas control and accident management, including emergency planning.

In RAI 6.2-136 and its Supplements 1 and 2, the staff requested that GEH provide additional information concerning the range of measurement of the hydrogen monitors and their functionality, reliability, and accuracy, and justification that the proposed monitors are adequate for their intended function. The RAI also inquired about the functionality and reliability of the monitors when exposed internally to the temperature, pressure, humidity, and radioactivity of the containment atmosphere during a significant BDBA.

In its response to RAI 6.2-136, GEH stated that equipment chosen for hydrogen monitoring will be specified to meet the environmental and radiological requirements for its location and for intended postaccident operations. GEH also stated that internal components will be evaluated to ensure that the instrument is qualified for the intended environmental and radiological conditions expected and for the required postaccident monitoring timeframe.

With respect to the accuracy of the hydrogen monitors, GEH responded that it will comply with RG 1.97, Revision 3, Table 2, where the required instrument range for a Type C variable is 0 to 30 percent volume for hydrogen in inerted containments.

GEH stated that it would revise DCD, Tier 2, Section 7.5.2.1, and add a new table, Table 7.5-5(a) markup table was provided in the GEH response) indicating the required instrument range. The staff confirmed that the applicant incorporated this change in DCD, Tier 2, Revision 6.

As described in DCD, Tier 2, Sections 6.2.5 and 7.5.2, the hydrogen monitors are a safety-related, seismic Category 1 system consisting of two redundant, physically and electrically independent postaccident monitoring divisions. The hydrogen monitors are EQ.

DCD, Tier 2, Section 19.3.4.2, identifies the hydrogen monitors as equipment required for severe accident mitigation. Section 19.2.3.3.7 of this report evaluates the survivability of the hydrogen monitors. The hydrogen monitors are located outside the drywell and wetwell, as shown in Figure 7.5-1 of the ESBWR DCD.

The staff reviewed the information provided by GEH for hydrogen monitor in response to RAI 6.2-136 and supplements and found it acceptable because the applicant revised the DCD to provide the specific design criteria for the hydrogen monitor consistent with RG 1.7 and RG 1.97. The ESBWR design for the hydrogen monitors meets the regulation in 10 CFR 50.44(c)(4)(ii) in accordance with guidelines of SRP Section 6.2.5 and therefore is acceptable. RAI 6.2-136 was being tracked as an open item in the SER with open items. Based on the applicant's response, RAI 6.2-136 was resolved.

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 reaction accompanied by hydrogen burning. Systems necessary to ensure containment integrity must also be demonstrated to perform their function under these conditions.

In RAI 6.2-178, the staff requested that GEH identify the design-basis and special events that were considered in the analysis and provide the actual pressure that results from assuming a 100-percent fuel clad-coolant reaction, and whether this assumption includes hydrogen burning. If no hydrogen burning was assumed for any accident, GEH should justify this omission, with consideration of BDBA information from DCD, Tier 2, Chapter 19.

In its response, GEH stated that the design-basis and special events were those described in DCD, Tier 2, Chapters 6 and 15. The estimate of the internal pressure that results from assuming a 100-percent fuel clad-coolant reaction is 1.097 MPa (absolute) as described in the response to RAI 19.2-39 in DCD, Tier 2, Section 19B. The analysis did not consider burning of hydrogen because the containment is inerted.

RG 1.7, Revision 3, Section C.5, describes an analytical technique that is accepted by the staff. The applicant has used this technique in DCD, Tier 2, Section 19B, and concluded that the deterministic FE analysis demonstrates that the reinforced concrete containment vessel and liner maintain structural integrity according to the requirements of 10 CFR 50.44(c)(5) for pressures corresponding to 100-percent fuel clad-coolant reaction. Section 19B of this report presents the evaluation of DCD, Tier 2, Section 19B. The staff acknowledges that hydrogen burning was not considered because the containment is inert and analyses provided in Tier 2, Section 6.2.5.5.3, show that the time required for the oxygen concentration to increase to the de-inerting value of 5 percent is greater than 24 hours for a wide range of fuel clad-coolant interaction and iodine release assumptions up to and including 100 percent. The staff finds that the requirements of 10 CFR 50.44(c)(5) are met based on its evaluation in Section 19B of this report. RAI 6.2-178 was being tracked as an open item in the SER with open items. Based on the applicant's response, RAI 6.2-178 was resolved.

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 of this report 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, "Containment Atmosphere Cleanup," 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 passive.

- GDC 42, "Inspection of Containment Atmosphere Cleanup Systems," and GDC 43, "Testing of Containment Atmosphere Cleanup Systems," 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. Section 14.3.11 of this report addresses ITAAC related to containment and associated systems.

6.2.5.4 *Conclusions*

On the basis of its review, the staff concludes that the proposed ESBWR combustible gas control system in the containment, described in the DCD, complies with the acceptance criteria of Section 6.2.5 of the SRP.

Compliance with the criteria in Section 6.2.5 of the SRP, as described in this section, constitutes an acceptable basis for satisfying the combustible gas control system requirements in 10 CFR 50.44, GDC 41, 42, and 43, and the guidance in RG 1.7.

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 Crite*ria

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," 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, "Provisions for Containment Testing and Inspection," 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, "Systems Penetrating Containment," 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 of 1954, as amended; and the NRC's regulations

6.2.6.2 *Summary of Technical Information*

This section describes the testing program for determining the containment integrated leakage rate (Type A tests), containment penetration leakage rates (Type B tests), and CIV 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 ANSI/ANS 56.8-2002, "Containment System Leakage Testing Requirements."

Licensees perform Type A, B, and C tests before operation 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. Containment maintenance, including repairs of 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)

ILRTs (Type A tests) are conducted periodically, in conformance with Appendix J to 10 CFR Part 50, to ensure that 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 the COL licensee selects Option A or Option B of Appendix J to 10 CFR Part 50. If the COL licensee selects Option A, it will perform the ILRTs at least three times during each 10-year service period. If it selects Option B, the test interval will be in accordance with RG 1.163.

In addition, after the initial ILRT, the COL licensee will follow any major modification or replacement of components of the reactor containment with either a Type A or a Type B test, ensuring that the area affected by the modification meets 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, L_a is redefined as "containment leakage rate" in DCD, Tier 2, Table 6.2-1, which excludes the MSIV leakage rate. The treatment of an MSIV leakage pathway separately in the radiological dose analysis in DCD, Tier 2, Section 15.4.4.5.2, 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's accuracy. The measurements are acceptable if the correlation between the verification test data and the ILRT data demonstrates an agreement within $\pm 0.25\ L_a$. Appendix C to ANSI/ANS-56.8 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 the 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 the 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 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 the containment atmosphere) before performing the Type A test. In addition, any leakage pathways that were isolated during the test shall be factored into the performance determination. If the leakage can be determined by a local leak-rate test, the as-left minimum pathway leakage rate 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 the COL licensee selects Option A of Appendix J to 10 CFR Part 50, and if two consecutive periodic ILRTs fail to meet the acceptance criteria before corrective action, the COL licensee will perform an ILRT 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 COL licensee may resume the previously established periodic retest schedule.

If the COL licensee selects Option B of Appendix J to 10 CFR Part 50, and if the ILRT results are not acceptable, then the COL licensee should identify the cause of the unacceptable performance and determine appropriate corrective actions.

Once the COL licensee has determined the cause and has completed the corrective actions, it should reestablish acceptable performance by performing an ILRT within 48 months following the unsuccessful ILRT test. Following a successful ILRT, the surveillance frequency may revert to once every 10 years.

The additional criteria below will be met for ILRTs, if the COL licensee chooses Option A of Appendix J to 10 CFR Part 50:

- 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 a 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 CIVs 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 the 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. The results of local leakage rate tests of penetrations associated with these systems are added to the ILRT results.

The additional criteria below will be met for ILRTs if the COL licensee chooses Option B of Appendix J to 10 CFR Part 50. All Appendix J pathways must be properly drained and vented during the 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 purposes, or ALARA considerations, pathways that are Type B or C tested within the previous 24 calendar months that 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 TS 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 that depend 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 TS, 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 the reactor. 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 leak tight 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 CIVs 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 the equivalency of such valves is that they are located in lines designed to be, or remain, filled with water for at least 30 days after a LOCA. All test connections, vent

lines, or drain lines 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 they are used infrequently 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 the 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 by 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 TS 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 the containment atmospheric 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 TS 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 ensures that the leaktightness of the containment will be within the values specified in the facility TS and that offsite radiation doses in excess of the reference values specified in 10 CFR Part 100, "Reactor Site Criteria," 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 had two open items, RAI 6.2-90 and 6.2-91. RAI 6.2-90 and RAI 6.2-90 S01 asked GEH to (1) clarify in the DCD that "Type C" means testing with air or nitrogen and eliminating water as an allowed Type C test medium, and (2) for Options A or B, address CIV testing, in systems such as RWCU/SDC, under the requirements for seal systems.

RAI 6.2-91 and RAI 6.2-91 S01 asked that DCD revisions better reflect the regulatory requirements, as indicated, related to seal systems.

In its response, GEH stated that it will revise DCD, Tier 2, Section 6.2.6.3, to delete the option for water as a Type C test medium. It will also revise this section to clarify the requirements for testing CIVs with qualified seal systems, such as in the RWCU/SDC system, and to include the referenced provisions in Nuclear Energy Institute 94-01, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," issued December 8, 2005. DCD, Tier 2, Section 6.2.6.3, will be revised to include the regulatory requirements related to seal systems, as shown in the attached markup. The staff has reviewed the GEH response in DCD, Tier 2, Revision 6, Section 6.2.6.3, and finds it acceptable. GEH has clarified in the DCD that "Type C" means testing with air or nitrogen and eliminated water as an allowed test medium and addressed CIV testing, in systems such as in RWCU/SDC, under the requirements of seal systems as indicated in RAI 6.2-90. The revised DCD also includes the regulatory requirements related to seal systems as indicated in RAI 6.2-91.

RAIs 6.2-90 and 6.2-91 were being tracked as open items in the SER with open items. Based on the applicant's response, RAIs 6.2-90 and 6.2-91 were resolved.

6.2.6.4 *Generic Issues*

The staff's review of containment leakage rate testing includes one generic safety issue, Item A-23, "Containment Leak Testing" (see NUREG-0933, "A Prioritization of Generic Safety Issues," issued September 2007).

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

6.2.6.5 *Conclusions*

On the basis of its review, the staff concludes that the ESBWR DCD containment leakage rate testing program complies with the acceptance criteria of SRP Section 6.2.6, as described in this section, and thus constitutes an acceptable basis for satisfying the containment leakage rate testing requirements of GDC 52, 53, and 54, and Appendix J to 10 CFR Part 50.

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. It must prevent fractures of the containment pressure boundary. The ESBWR must address the following regulations:

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

- GDC 16, "Containment Design," requires that the reactor containment and associated systems establish an essentially leaktight barrier against the uncontrolled release of radioactivity to the environment and 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 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 a 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 the 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 DCD, Tier 2, Table 6.1-1, the applicant identified the containment vessel liner materials, which are in conformance with 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. These 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 a rapidly propagating fracture will be 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 used in 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, under operating, maintenance, testing, and postulated accident conditions, the ESBWR provides reasonable assurance that the materials used in the reactor containment pressure boundary will not undergo brittle fracture and that the probability of a rapidly propagating fracture will be minimized, thereby meeting the requirements of GDC 51.

6.3 <u>Emergency Core Cooling Systems</u>

6.3.1 Emergency Core Cooling Systems Design

6.3.1.1 *Regulatory Criteria*

The staff reviewed DCD, Tier 2, Section 6.3, in accordance with SRP Sections 6.3 and 15.6.5.

The staff based its acceptance criteria on the following requirements:

- GDC 2, "Design Bases for Protection Against Natural Phenomena," as it relates to the seismic design of SSCs where their failure could cause an unacceptable reduction in the capability of the ECCS to perform its safety function

- GDC 4, "Environmental and Dynamic Effects Design Bases," as it relates to the 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, “Emergency Core Cooling,” 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 the 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 1,204 degrees C (2,200 degrees F).

  (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, NUREG-0737:

- TMI Action Plan Item II.K.3.15, 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, 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, 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, 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, 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 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. Below is a brief summary of the GEH description.

Passive Core Cooling System

The passive core cooling system comprises the GDCS, the ADS, the 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 it 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 (PCTs) 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 ICs 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 or a sustained drywell high pressure 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 flowpath 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 C (1,000 degrees F). Once the squib valves are actuated, the GDCS deluge lines provide a flowpath from the GDCS pool to the 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, long-duration submersible, pyrotechnic-actuated, and nonreclosing, 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 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 and DPVs.

The DPVs are straight-through, squib-actuated, nonreclosing 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 eight 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. (Section 5.4.6 of this report contains 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. (Section 9.3.5 of this report contains a detailed description of the SLCS.)

Strainers

Section 6.2.1.7 of this report contains a description of the strainers.

6.3.1.3 *Staff Evaluation*

The staff's review of the ECCS uses SRP Section 6.3 as guidance. Because the ESBWR ECCS is quite different from the ECCS of the existing BWR designs, some SRP guidelines do not apply. The staff devoted the major portion of the review effort to the areas where the application is not identical to previously reviewed BWRs.

Emergency Core Cooling Systems

The ECCS is designed to provide coolant inventory to the reactor coolant system in the event of a LOCA. It has sufficient capacity to make up for the 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 cannot 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 ESBWR LOCA analysis. 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. Section 3.9.6 of this report contains the staff evaluation of the DPV, GDCS, and SLCS valve tests.

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 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 (1.0 m (3.28 ft)) above the top of active fuel (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.

To assess the equilibrium between the reactor decay heat and the condensate flow rate from the PCCS, in RAI 6.3-33, the staff requested additional information regarding the normal and postaccident water level in the GCDS pool. The applicant provided the information requested, and the staff concurs that the post-LOCA GDCS pool level depends on the type of pipe break and the break elevation. RAI 6.3-33 was being tracked as a confirmatory item in the SER with open items. The staff confirmed that this change was included in DCD, Tier 2, Revision 6. Therefore, RAI 6.3-33 was resolved.

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 6.2.1.7.3 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. Section 19.2 of this report contains the staff's evaluation of the severe accident mitigation features.

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 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 to address the applicability of the TRACG04 flow-choking computer model to the ESBWR RPV injection line and equalizing line nozzles.

By letter dated March 5, 2008, the applicant submitted the following additional information in its response to RAI 6.3-13, Supplement 1:

- TRACG has a subcooled choking model applicable to small length-to-diameter (L/D) throat conditions. The model prediction comparisons to data include choked flow for both smooth and abrupt area changes (i.e., orifices), thus validating the model for small L/D.

- TRACG is qualified over a range of 0.0–8.68 L/D through direct comparison to test data. GEH provided a table of tests that contains the L/D for the pressure suppression test

facility (PSTF) critical flow tests, Marviken, and the Edwards Pipe Tests used to qualify the TRACG critical flow model.

- GEH provided a table of L/Ds for break lines. The values of L/Ds of ESBWR break lines are within the ranges of the TRACG qualification database.

Recognizing that it is not possible to have continuous L/D values in the range of test data, the ESBWR break throat L/D values are within the range of tests used to qualify the TRACG code choking model. Based on the RAI response, the staff concluded that the TRACG model covers L/Ds for the ESBWR break lines. RAI 6.3-13 was being tracked as an open item in the SER with open items. Based on the applicant's response, RAI 6.3-13 was resolved.

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. Section 3.9.6.3.2.4 of this report contains the staff's evaluation of squib valve tests.

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 prevents backflow from the RPV to the GDCS, thereby mitigating the consequences of spurious GDCS squib-actuated valve operations. The check valve is classified as Quality Group A, seismic Category I, and ASME Code, Section III, Class 1. The MCR has a remote check valve position indication. The staff noted that the applicant changed the description of the check valves in DCD Revision 3, Section 6.3.2.7.2. To evaluate the changes to the check valves, the staff made the following requests in RAI 6.3-78:

- Describe the design differences between the old and new designs.

- Add the typical check valve figure in the DCD, as before.

- Confirm that the check valves used for injection and equalization are of the same design.

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

By letter dated March 27, 2008, the applicant submitted its response to RAI 6.3-78 regarding the GDCS check valve design and confirmed that the GDCS injection line and equalization line check valves are the same. Section 6.3.2.3.3 of this report provides a detailed evaluation. RAI 6.3-78 was being tracked as an open item in the SER with open items. Based on the staff evaluation of the applicant's description of the valve design in the response, RAI 6.3-78 was resolved.

Automatic Depressurization System

The ADS is part of the ECCS; it depressurizes the reactor so the low-pressure GDCS can supply makeup coolant to the reactor. Depressurization is achieved through the 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, as follows:

Group I (three DPVs), 50 seconds; Group II (two DPVs), 100 seconds; Group III (two DPVs), 150 seconds; and Group IV (one DPV), 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, compared to SRVs. The use of DPVs also reduces the number of SRVs and the need for SRV maintenance, periodic calibration, and testing. In addition, since DPVs discharge into the drywell atmosphere, their use 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 qualifications. The SRVs and DPVs are designed with flange connections to allow easy removal for maintenance, testing, or rebuilding. In addition, they are designed so 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 tests of the DPV to demonstrate its operation. Section 3.9.6.3.2.3 of this report contains the staff's evaluation of the DPV tests.

Each of the 10 ADS SRVs is equipped with a seismically qualified pneumatic accumulator and check valve. Normally, a high-pressure nitrogen supply system provides nitrogen gas to the SRV accumulators. Section 9.3.8 of this report contains the staff's evaluation of the high-pressure nitrogen supply system. The accumulators ensure that the valves can be opened following the failure of the gas supplying the accumulators. The accumulator capacity is sufficient to actuate the valve once at drywell design pressure and at least twice under accident conditions. The containment design pressure is approximately 45 psig. At the beginning of the accident, the containment pressure is much lower than 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 in 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. Section 20.4 of this report contains the staff evaluation of Item II.K.3.28.

The SRVs and DPVs are sized such that vessel depressurization and cooldown are slow enough to prevent the system from exceeding vessel integrity limits. GEH performed a thermal analysis that considered the effect of blowdown. Because of the ESBWR's unique design, depressurization is expected to be slower than in the current BWR operating reactors. The RPV and the containment are designed to maintain structural integrity during an ADS event. Thus, the applicant has met TMI Action Plan Item II.K.3.45 in 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 a rapid BWR cooldown.

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.

Similar to hydrogen accumulation in PCCS as described in Section 6.2.2.3, there is a potential for hydrogen accumulation in the IC tubes during post-LOCA conditions. To address this issue for IC, the applicant proposed to isolate the IC soon after IC injection. In response to RAI 6.2-202 S01 the applicant provided a design change where, upon the opening of any two DPVs, the ICS isolation valves are automatically signaled to close. Resolution of this RAI is discussed in Section 6.2.2.3 of this report.

Standby Liquid Control System

The SLCS also supplies 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 contains 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 Revision 3 of RG 1.29. 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 in Revision 3 of RG 1.26, 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 meets the environmental qualification requirements of GDC 4 regarding operation under normal and accident conditions. Chapter 3 of this report discusses these aspects of the ECCS design.

The ESBWR is proposed as a single unit design, and therefore, GDC 5, which concerns the sharing of SSCs 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 ability, in conjunction with poison added by the ECCS, to reliably control reactivity changes under postulated accident conditions, with an appropriate margin for stuck rods.

The GDCS, ICS, and SLCS provide abundant emergency core cooling, thus satisfying the requirements of GDC 35. All the ECCSs are designed to permit appropriate periodic inspection of important components, such as the 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 ISI 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 RAI 5.4-43, the staff requested a clarification about the applicability of TMI-2 action item II.K.3.15 to the ESBWR design. In its 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. The staff agrees with the applicant that this item is not applicable and hence the RAI 5.4-43 is resolved.

Manual activation of the ADS was originally required to provide adequate core cooling for transient and accident events that did not directly produce a high-drywell-pressure signal (e.g., stuck-open relief valve or steamline break outside containment), and that were further complicated by the loss of all high-pressure systems.

However, TMI Action Plan Item II.K.3.18 required all BWRs to modify their ADS actuation logic to eliminate the need for manual activation to ensure adequate core cooling. Instead, the ESBWR ADS equipment is activated automatically upon receipt of a signal of persistent low reactor water level with a delay of 10 seconds or sustained high drywell pressure with a delay of 60 minutes, without the need for operator action. Manual actuation is also possible. ADS complements manual actuation. RAI 6.3-10 requested additional information on the ADS control logic used to model the Level 1 setpoint in TRACG. By letters dated July 28, 2006, May 29, 2007, and May 1, 2008, the applicant submitted its response to RAI 6.3-10, Supplements 1 and 2. The applicant provided detailed ADS actuation logic. The ADS will be initiated when the water level reaches L1 (11.5 m from RPV bottom). The safety margins for LOCAs, SBO, and loss of feedwater are well maintained with this setpoint, as demonstrated in RAI 6.3-10, Supplement 1, and DCD, Tier 2, Sections 6.3, 15.2, and 15.5. In addition, the response to RAI 6.3-10, Supplement 2, clarified how the water level is calculated in the TRACG model. SER Section 6.3.2.3.5 contains further evaluation. RAI 6.3-10 was being tracked as an open item in the SER with open items. Based on the applicant's response, RAI 6.3-10 was resolved. Since the ADS logic includes a drywell high-pressure signal with a time delay, TMI Action Plan Item II.K.3.18 is satisfied.

Preoperational Tests

Preoperational tests will ensure the 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. Section 14.2 of this report notes the applicant's commitment to conformance to the guidelines in RG 1.68, "Initial Test Programs for Water-Cooled Nuclear Power Plants," for preoperational and initial startup testing of the ECCS.

Safe Shutdown

Establishing a safe-shutdown condition requires maintaining 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, "Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety Systems in Passive Plant Designs," dated March 28, 1994. This position accepts temperatures of 215.6 degrees C (420 degrees F) or below, rather than the cold shutdown (less than 93.3 degrees C (200 degrees F)) temperature specified in SRP 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 the 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 LOCAs. The IC and PCCS expansion pools have an installed capacity that provides at least 72 hours of reactor decay heat removal capability. Replenishing the IC and PCCS expansion pool inventory allows the heat rejection process to continue indefinitely. A safety-related independent FAPCS makeup line adds makeup water to the IC and PCCS expansion pools. A dedicated diesel-driven makeup pump system is connected to the FAPCS. This connection enables the site FPS to fill the upper IC and 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 system's 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. In response to RAI 6.3-12, Supplement 1, GEH provided a draft paragraph for inclusion in the DCD to clarify the mechanical separation provided in the design. Since there is adequate separation between the GDCS systems, the response was acceptable. RAI 6.3-12 was being tracked as a confirmatory item in the SER with open items. The staff confirmed that the applicant included this change in Revision 6 of the DCD. Based on the applicant's response, RAI 6.3-12 was resolved

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 floodup. The design reliability assurance program will include all risk-significant ECCSs, 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 20.4 of this report.

Since the ESBWR design does not include core spray or LPCI systems that can restart after a LOCA, TMI Action Plan Item II.K.21 is not applicable.

Inspections, Tests, Analyses, and Acceptance Criteria

DCD Tier 1 contains the ESBWR ITAAC, and Section 14.3 of this report includes the staff's evaluation of them.

In RAI 6.3-18, the staff asked GEH to provide the pool inventory in the ITAAC and a physical elevation inspection of the GDCS pool level. In response to RAI 6.3-18, GEH revised GDCS ITAAC Table 2.4.2.3 to include verifying the minimum drainable water volume and minimum water levels in GCDS pools. This RAI was being tracked as an open item in the SER with open items. Based on the applicant's response, RAI 6.3-18 was resolved.

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, 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 6.3 and Section 15.6.5, "Loss-of-Coolant Accidents Resulting from Spectrum of Postulated Piping Breaks within the Reactor Coolant Pressure Boundary." The approved LTR NEDC-33083P-A and its safety evaluation contain a detailed discussion of regulatory criteria.

6.3.2.2 *Summary of Technical Information*

6.3.2.2.1 Evaluation Model

GEH used the staff approved TRACG code, (NEDC-33083P-A), 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 PCT would not provide useful results. GEH stated that a bounding evaluation for the minimum water level in the chimney during a LOCA 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 LOCAs using the failure of one GDCS valve, one SRV, or one DPV. DCD, Tier 2, Table 6.3-1, shown below, identifies the most limiting combinations.

Table 6.3-1 Single-Failure Evaluation

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

6.3.2.2.4 Loss of Offsite Power

GEH analyzed the LOCAs with a loss of offsite power (LOOP) occurring at the same time as the initiation of the break.

6.3.2.2.5 Break Spectrum

Table 6.3-2 below 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 Drainline	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 to RAI 6.3-47, the staff requested the applicant to include a table in the DCD to show the ECCS line break sizes and elevations. GEH did so, and the staff confirmed that the table is listed in DCD Revision 6, as Table 6.3-5a. This RAI was being tracked as an open item in the SER with open items. Based on the applicant's response, RAI 6.3-47 was resolved.

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 (2 in.). 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:

- MSL 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/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 those for the large steamline break case. For small line breaks, GEH analyzed the GDCS injection line break and the bottom head drainline breaks.

In RAI 6.3-46, the staff requested the applicant to provide the technical bases for the selection of the most limiting break size cases In the 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 drainline

GEH stated that the limiting cases are the GDCS injection line and IC drainline 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	8.47 (27.8)
Steamline Inside Containment	0.09832 (1.058)	1 GDCS Valve	8.36 (27.43)
Steamline Inside Containment	0.09832 (1.058)	1 DPV	8.76 (28.74)
Feedwater Line	0.07420 (0.7986)	1 SRV	8.37 (27.47)
Feedwater Line	0.07420 (0.7986)	1 GDCS Valve	8.26 (27.09)
Feedwater Line	0.07420 (0.7986)	1 DPV	8.35 (27.3)
GDCS Injection Line	0.004561 (0.04910)	1 SRV	8.69 (28.52)
GDCS Injection Line	0.004561 (0.04910)	1 GDCS Valve	8.9 (29.19)

Break Location	Break Size m^2 (ft^2)	Single Failure	Minimum Chimney Static Head m (ft)
GDCS Injection Line	0.004561 (0.04910)	1 DPV	8.73 (28.64)
Bottom Head Drainline	0.004052 (0.04361)	1 SRV	8.35 (27.39)
Bottom Head Drainline	0.004052 (0.04361)	1 GDCS Valve	8.62 (28.29)
Bottom Head Drainline	0.004052 (0.04361)	1 DPV	8.42 (27.63)
ICS Drain Line	0.01824 (0.1963)	1 SRV	8.40 (27.55)
ICS Drain Line	0.01824 (0.1963)	1 GDCS Valve	8.55 (28.04)
ICS Drain Line	0.01824 (0.1963)	1 DPV	8.56 (28.08)

The values in the above table are from DCD, Tier 2, Revision 7, 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 ICS drainline 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)
ICS Drainline	0.01824 (0.1963)	1 SRV	8.33 (27.33)
ICS Drainline	0.01824 (0.1963)	1 GDCS Valve	8.19 (26.87)
ICS Drainline	0.01824 (0.1963)	1 DPV	8.31 (27.26)
GDCS Injection Line	0.004561 (0.04910)	1 SRV	8.82 (28.93)
GDCS Injection Line	0.004561 (0.04910)	1 GDCS Valve	8.34 (27.36)
GDCS Injection Line	0.004561 (0.04910)	1 DPV	8.87 (29.09)

The values in the above table are from DCD, Tier 2, Revision 7, Table 6.3-5.

6.3.2.2.6 Evaluation Model Parameters and Assumptions

GEH chose the evaluation model parameters and assumptions discussed below.

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

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

Axial Power Shapes. The applicant's TRACG model uses 35 axial nodes, with 32 representing the heated section of the channel. 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.

Initial Stored Energy. GEH assumes constant gap conductance throughout the LOCA. 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 the applicability of the GSTRM code to the ESBWR. The applicant's fuel thermal conductivity model is based on that used in the PRIME03 code. RAIs 6.3-54 and 6.3-55 asked GEH to address the applicability of the PRIME03 code to the ESBWR. After receiving responses to RAIs 6.3-54 and 6.3-55, the staff issued a supplement to RAI 6.3-54, asking for experimental evidence of the data provided in RAI 6.3-54. Section 6.3.2.3.6 contains the staff's evaluation of the response.

Control Rod Insertion. 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 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 the containment pressure response.

ECCS Strainer Performance Evaluation. Section 6.2.1.7.3 of this report discusses the ECCS strainer performance evaluation.

6.3.2.2.7 Reactor Protection and Emergency Core Cooling System Actions

The following sections give a narrative description of the sequence of events for the break locations presented in DCD, Tier 2, Revision 6, and the ESBWR ECCS and RPS response.

Gravity-Driven Cooling System Line Break. In DCD, Tier 2, Revision 6, 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 6, Section 6.3.3.4. DCD, Tier 2, Revision 6, Table 6.3-9, contains the operational sequence of the RPS and ECCS actions.

DCD, Tier 2, Figures 6.3-32a and 6.3-32b, show the static head in the chimney and the two-phase level in the chimney. During the first 30 seconds after the break, because of flashing, the collapsed chimney level increased relative to the bottom of the chimney.

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, the 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 of the break flow reaching the saturation temperature, when it begins voiding. In RAI 6.3-69, the staff requested the applicant to include figures of void fraction vs. time for the break flow for the breaks presented in DCD, Tier 2, Section 6.3. 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. This plot shows that the void fraction increases until about 24 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. RAI 6.3-69 was being tracked as a confirmatory item in the SER with open items. The staff confirmed that the applicant included this change in Revision 6 of the DCD. Since TRACG adequately simulates voids fraction vs. time behavior as expected in the accident scenario, RAI 6.3-69 was therefore resolved.

The IC drain valves open on the LOOP. The drain valves open after a 15-second delay. DCD, Tier 2, Figure 6.3-37b, shows the IC drain flow, which 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 oscillate. This flow is from the IC steam condensation. In RAI 6.3-74, the staff requested the applicant to explain the cause of the increase in steam flow at about 200 seconds for all breaks. 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.

Since the applicant adequately explained the steam flow changes during the accident scenario, the response is acceptable and hence RAI 6.3-74 is resolved.

The system reaches the Level 1 setpoint approximately 220 seconds after the break. Level 1 must persist for 10 seconds to be confirmed. ADS is confirmed at approximately 231 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 from 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 decreases 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 281 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 380 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 488 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 the 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 1450 seconds into the transient, the level starts to experience large oscillations. GEH states, in NEDC-33083P-A, that these are manometric 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. DCD, Tier 2, Figure 6.3-39, includes the reactor power as a function of time plots, and it clearly labeled each plot with the nominal or bounding conditions. RAI 6.3-70 was being tracked as an open item. Based on the applicant's response, RAI 6.3-70 was resolved.

Main Steamline Break. DCD, Tier 2, Revision 6, 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 6, 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 6, Figures 6.3-7a through 6.3-14b, describe the system's response to an FWLB with one GDCS valve failure. DCD, Tier 2, Revision 6, 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, because of the level swell, the Level 1 setpoint actuates later than the GDCS line break but sooner than the MSLB.

Bottom Drainline Break. DCD, Tier 2, Revision 6, Figures 6.3-23a through 6.3-30b, describe the system response of a bottom drainline break with one GDCS valve failure. DCD, Tier 2, Revision 6, 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 more slowly, 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 of 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 updated them when it submitted the responses to RAIs 6.3-64 and 21.6-98. In RAI 6.3-64,the staff requested the applicant to submit the plots of the core level demonstrating that the core remain covered for 72 hours for the limiting break 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. In MFN 08-545, dated August 29, 2008, GEH pointed out that RAI 6.3-79 discusses long-term cooling. The response discussed TRACG calculation results for the short-term (0-2,000s) and long-term (0-72 hours) core cooling. The discussion showed that, for all break locations, the water levels are above the reactor core and above the GDCS equalization line water injection setpoint Level 0.5 for 30 days. Section 6.3.2.3.1 of this report contains the evaluation of the response to RAI 6.3-79. Section 21.6of this SER discusses the closure of the other parts of RAI 21.6-98. The staff received a supplemental response to RAI 6.3-64 in April 2008, and Section 6.3.2.3.8 of this report discusses its evaluation.

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 through 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. RAIs 6.3-64 and 21.6-98 addressed concerns related to long-term core cooling. RAI 21.6-98 is discussed in Section 21.6 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 through 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 in a few 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 (between the suppression pool and the drywell) 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 several feet above the bottom of the reactor vessel, 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 through 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 ESBWR design, described in NEDC-33083-A. Section 21.6 of this report provides an evaluation of its applicability to the current ESBWR 4500 MWt design. The ESBWR LOCA analyses show that the core does not heat up 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 maintaining a static head of water above the TAF.

6.3.2.3.2 Uncertainty Analysis

Regulations in 10 CFR 50.46(a)(1)(i) state, 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." Furthermore, 10 CFR 50.46(a)(1)(ii) states, "Alternatively, an ECCS evaluation model may be developed in conformance with the required and acceptable features of appendix K ECCS Evaluation Models." The staff issued RAI 6.3-81 to request that GEH demonstrate how the LOCA analyses comply with these requirements. In response to RAI 6.3-8, GEH stated for ESBWR LOCAs, that because there is no core uncovery and no core heatup, a statistical analysis of the PCT does not serve any useful purpose. The best estimate PCT and the 95/95 PCT would both be close to the saturation temperature corresponding to the peak steam dome pressure reached in the accidents. In the case of ESBWR LOCAs, there is a margin of over 871 degrees C (1,600 degrees F) to the limit of 1,204 degrees C (2,200 degrees F) (acceptance criteria set forth in 10 CFR 50.46(b)). GEH further stated that the static head inside the chimney (in meters of water) is selected as the figure of merit for comparison and to evaluate the impact of uncertainties in model parameters and plant parameters. This collapsed level is defined as the equivalent height of water corresponding to the static head of the two-phase mixture above the top of the core. The TRACG model parameter uncertainties and plant parameter uncertainties have been identified (NEDC-33083P-A, Sections 2.4 and 2.5.3). GEH performed sensitivity studies by varying each of these parameters from the lower bound to the upper bound value.

The impact on the chimney static head is between -0.3 m (-0.98 ft) to +0.2 m (+0.66 ft) (NEDC-33083P-A, Section 2.4.4.2), which is less than the minimum static head in the chimney from the parametric studies. Therefore, GEH proposed that a simple calculation be made setting the most significant parameters at the 2-sigma values to obtain a bounding estimate of the minimum level. The staff finds this approach acceptable and concurs that the ESBWR LOCA results demonstrate that there is a high probability that there is no core uncovery or heatup and that the PCT would be close to the saturation temperature corresponding to the peak steam dome pressure reached in the accident. The staff concludes that the GEH LOCA results comply with the requirements in 10 CFR 50.46. RAI 6.3-81 was being tracked as an open item in the SER with open items. Based on the applicant's response, RAI 6.3-81 was resolved.

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, process diagram, and 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 ECCSs.

SRP Section 6.3 states that the long-term cooling capacity is adequate in the event of failure of any single active or passive component of the ECCS. In RAI 6.3-79, the staff requested the applicant to clarify whether the ESBWR design takes credit for any passive component during long term post LOCA (i.e. beyond 72 hours) cooling. In its response to RAI 6.3-79, dated August 24, 2007, GEH stated that, for the ESBWR design, conformance to the requirement of adequate long-term cooling is assured and demonstrated for any LOCA where the water level can be restored and maintained at a level above the top of the reactor core. The response discussed TRACG calculation results for a short-term (0–2,000 s) and long-term (0–72 h) calculation. These calculations used assumptions with possible single failures of ECCS components. GEH then qualitatively determined, from the TRACG long-term (0–72 h) calculation, that the water level will remain near an equilibrium level above the core. For break locations lower than the spillover hole, the water level will remain at the final equilibrium level at the spillover hole level. For break locations higher than the spillover hole, GEH estimated, from the TRACG long-term (0–72 h) calculation, the inventory loss to the lower drywell caused by drywell wall condensation. The estimation showed that, for all break locations, the water levels are above the reactor core and above the GDCS equalization line water injection setpoint Level 0.5 for 30 days. Furthermore, by design, if the RPV water level drops below Level 0.5, these equalization lines would be actuated. After actuation, these equalization lines provide the long-term post-LOCA makeup water to the RPV from the suppression pool. The suppression pool's normal water level is about 10 m (32.81 ft) from the RPV bottom, or 2.5 m (8.20 ft) above the TAF. The addition of the suppression pool water will ensure that the reactor core is covered at a level above the TAF for an indefinite long-term period. For these reasons, the staff concurs that the design provided adequate long-term cooling. RAI 6.3-79 was being tracked as an open item in the SER with open items. RAI 6.3-79 was therefore resolved.

<u>GDCS Single Failure</u>. In RAI 6.3-43, staff requested the applicant provide additional information on the single failure analyses for the GDCS. 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. Staff followed up with RAI 6.3-43 S01 and requested the applicant to document the December 2006, audit discussions related to the comparison of different break/valve failure combinations and explanations that the applicant modeled the worst single failure. Since there are multiple GDCS lines and multiple GDCS pools, there are multiple combinations of failed valve and broken injection line combinations that are possible. GEH responded by updating DCD, Tier 2, Chapter 6 with tables showing the modeling of the most limiting combination of break locations and valve failures. The staff confirmed that GEH chose the most conservative combination of valve failure and line break by reviewing the evaluation of initial injection flow rate and total long-term GDCS water volume. Therefore, RAI 6.3-43 is resolved.

The GDCS check valve must be closed upon initiation of the squib valves, since the RPV pressure is higher than that of the GDCS. In RAI 6.3-78, the staff requested that GEH evaluate

the possibility of this failure, because it could result in additional coolant loss. In its response, dated March 27, 2008, GEH stated that the old design was a biased-open, tilting disk check valve installed in a horizontal piping run. The new design is a normally open, piston check valve, installed in a horizontal or vertical piping run. Revision 5 of the DCD updated the GDCS check valve description. GEH confirmed that the GDCS injection and equalization line check valves are of the same design. GEH further stated that it added an ITAAC item to DCD, Tier 1, Section 2.4.2, and Table 2.4.2-3. The ITAAC is to use GDCS check valve testing to measure the fully open flow coefficient in the reverse flow direction, and to verify that the measured value is less than the value assumed in the LOCA analyses. This verification will confirm that the check valve will function as designed and the core would remain covered in the event of a GDCS check valve failure following a LOCA, despite back flow through the GDCS injection line. Since the valves will be functionally tested during ITAAC, the staff accepted the GEH response and, therefore, RAI 6.3-78 was resolved. However, the staff did not see an analysis with a back flow in the GDCS drainline. The staff issued a new RAI 6.3-84, asking GEH to analyze cases in the event the GDCS check valve failed to close; this was an open item. By letter dated February 10, 2009, the applicant submitted its response to RAI 6.3-84 and provided calculation results for the limiting IC drainline break, where one of the GDCS check valves failed to close. The calculation showed that the reactor minimum level in the internal chimney during the LOCA would be 85.7 cm (33.74 in.) above the TAF. The calculation showed that the core would be covered with water, and the staff found that the design would provide adequate cooling during this event. Therefore, RAI 6.3-84 was resolved.

GEH performed analyses of all design-basis LOCAs, 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, the DPV stub tube (DPV/IC steamline), the RWCU/SDC return line, and the IC return line breaks with one GDCS injection line failure.

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 LOCAs, 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, the DPV stub tube (DPV/IC steamline), the RWCU/SDC return line, and the IC drainline breaks, with a failure of either a DPV or an SRV.

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 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 inventory will be lost through the break. In RAI 6.3-65, the staff requested the applicant to evaluate the consequences of a break in the SLCS injection line with the worst single failure. 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 2,000 seconds of

the event and does not require SLCS injection. The staff requested, in a supplement to RAI 6.3-65, that GEH discuss the long-term results of the SLCS line break. The staff received the response to RAI 6.3-65, S01, Revision 1, on September 2, 2008. GEH provided a full analysis, using TRACG with an SLCS line break. The analysis showed that a late ADS open actuation caused by the smaller break size, compared to the other break scenarios, and the minimum water level is above the top of the active core. The applicant provided a sensitivity analysis, with and without ICS heat transfer modeling. With the ICS heat transfer modeling, the RPV pressure decreased more slowly, and this caused slower inventory loss. The calculation showed that the ADS initiated around 6,674 seconds. For the case without the ICS heat transfer modeling, after MSIV closure, the RPV pressure rose and reached the SRV setpoints. The SRV discharged RPV steam into the suppression pool, which resulted in more RPV inventory loss. The L1 ADS initiation setpoint is reached around 1,731 seconds. In both sensitivity cases, after ADS initiation, the GDCS recovered the water level. The staff found that the calculation plots showed the water level is above the TAF, which is an indication that the ECCS has provided adequate cooling. Based on the applicant's response, RAI 6.3-65 was resolved.

ICS Single Failure. GEH did not take credit for the heat removal capability of the ICS in DCD, Tier 2, Revision 6, but modeled the inventory in the ICS drain tanks during a LOCA. The condensate drain valve for the ICS is single-failure-proof. There is a bypass valve that may be actuated in the event the condensate drain 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 6, 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.

In RAI 6.3-46 S01, the staff requested the applicant to submit the technical bases for the limiting break in the break spectrum analyses. The applicant's response to RAI 6.3-46 S01 shows that the core remains covered for the GDCS equalizing line, the DPV stub tube (DPV/IC steamline), and the RWCU/SDC return line breaks with one inoperable IC. For the IC drainline break, the IC that is out of service may be a different IC than the one attached to the broken line, so GEH assumed only two ICs were available for this event. The results in Table 6.3-46-1 show that the core remains covered for this event. The historical account and resolution of RAI 6.3-46 is described later in section 6.3.2.3.5.

In RAI 6.3-65 S01, the staff asked GEH to verify how many ICs were operating during the SLCS break evaluation. The staff sent this request to GEH as a supplement to RAI 6.3-65, as further discussed in section 6.3.2.3.5 of this report. GEH responded that there are four ICs associated with an ESBWR; however, the analysis of this event takes credit for only three of them. This resolved the availability of ICs. RAI 6.3-65 was being tracked as an open item in the SER with open items. Based on the applicant's response, RAI 6.3-65 was resolved with regard to the availability of ICs. Complete RAI 6.3-65 resolution is described in previous paragraphs of section 6.3.2.3.3.

In RAI 6.3-66, the staff requested that GEH include a statement that the LOCA RPV level analyses take credit for the IC 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. The staff finds this approach acceptable. RAI 6.3-66 was being tracked as a confirmatory item in the SER with open items. The staff confirmed that, in DCD, Tier 2,

Revision 6, Table 6.3-1, the analysis included the ICs and CRD water inventory and that GEH took no credit for IC heat removal in Table 6A-1. The applicant made appropriate DCD changes requested by the staff and therefore, RAI 6.3-66 was resolved.

Vacuum Breaker Failure. There is a vacuum breaker between the drywell and the suppression pool that opens if the wetwell pressure exceeds that of the drywell. Failure of this valve to close after opening would cause steam to leak from the drywell to the wetwell bypassing the suppression pool at a rate higher than the design steam leakage value. Steam that enters the wetwell bypassing the suppression pool does not get condensed by the suppression pool and raises the wetwell pressure and eventually the drywell pressure. In DCD, Tier 2, Revision 3, Section 6.2.1.1.2, GEH added an alternate means to close this opening by adding a vacuum breaker isolation valve (VBIV) that would allow the vacuum breaker system to remain operable with a single active failure of one vacuum breaker. 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 was being tracked as an open item in the ESBWR SER with Open Items.

Staff reviewed GEH's response to RAI 6.3-63 and found that this design approach is acceptable. See evaluation for DCD, Tier 2, Section 6.2.1.1.2 under RAI 6.2-148 for further detail. RAI 6.3-63 is therefore resolved.

Bottom Drainline Isolation. The bottom drainline is open during normal operations for RWCU. In the event of a LOCA, it is possible that, if this line fails to isolate, additional loss of inventory may occur. In RAI 6.3-59, the staff requested the applicant to explain the signals which will isolate the bottom drain valves and the consequences if these valves were to fail to isolate during a LOCA. In response to RAI 6.3-59, GEH confirmed that there are two isolation valves in series; therefore, the 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 during 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 a 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 is negligible, compared to the other ECCS sources, and its failure will not provide limiting results. In RAI 6.3-66, the staff requested the applicant to revise the DCD to include a statement that they take credit for the IC heat removal capacity and the water addition from the Hydraulic Control Unit (HCU). 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. RAI 6.3-66 was being tracked as a confirmatory item in the SER with open items. The staff verified the DCD was revised to include the credit for HCU. The response regarding the HCU credit in RAI 6.3-66 is acceptable. However, since HCUs are classified as Safety Class-2 in DCD Table 3.2-1, they are not considered to be safety grade. In RAI 6.3-87, the staff requested

a justification for the use of HCU scram water in the LOCA analysis. By letter dated November 3, 2008, GEH submitted the response to RAI 6.3.-87. Also, the staff raised the CRD system classification issue in RAI 3.2-21. Since the applicant provided sufficient information justifying the classification and the qualification of the system in the two responses, RAI 6.3-87 is closed. Section 3.2 of this report discusses the staff's resolution of this issue.

Conclusion of Single-Failure Evaluation. The staff examined failure possibilities and their significance. The GEH design selected single failures of one GDCS injection valve, one DPV, and one SRV for the LOCA analysis. The staff concurs that the failure of a DPV or SRV results in the greatest reduction in the depressurization rate from ADS actuation and results in a delay in GDCS injection. The failure of one GDCS injection valve results in the greatest reduction in the GDCS reflooding rate. The staff agrees with the discussion in the DCD and found the single failure selection to be reasonable and acceptable.

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 LOOP. To demonstrate that the ECCS performance meets the design requirements, GEH assumed a LOOP occurs coincident with the break for each of the design-basis LOCAs analyzed. This causes the reactor to scram and the ICS to initiate upon the loss-of-power signal. If there were no LOOP 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 from a LOOP and assumes a loss of feedwater is more conservative than incorporating the delays. In DCD, Tier 2, Revision 3, GEH changed the feedwater isolation to be 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 a LOOP, coincident with the break, is a conservative assumption because of the feedwater isolation. For the FWLB, the high-drywell-pressure signal occurs before 1 second (as shown in DCD, Tier 2, Revision 6, Table 6.3-7) 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 reasonable assumption. Therefore, the staff finds the applicant's LOOP assumption to be appropriate for ESBWR LOCA analyses.

6.3.2.3.5 Break Spectrum

GEH showed the results of a LOCA 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 2,000 seconds after the break). GEH uses minimum static head in the chimney as a metric to determine the most limiting break. The staff finds the labeling of DCD, Tier 2, Revision 3, Table 6.3-5, 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, not considering the void fraction in the core. In RAI 6.3-77, the staff requested 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. In its response, dated June 22, 2007, GEH explained that the chimney static head level with reference to vessel zero is calculated by adding the equivalent height of water corresponding to the static head of the two-phase mixture inside the chimney to

the elevation (7.896 m (25.91 ft)) of the bottom of the chimney. RAI 6.3-77 was being tracked as a confirmatory item in the SER with open items. The staff confirmed that the applicant included the above change in Revision 6 of the DCD. Therefore, RAI 6.3-77 was resolved.

GEH performed each of the 24 calculations using nominal conditions. GEH stated that the ICS drainline and the gravity injection line breaks are the most limiting cases and performed calculations for these two break locations, using bounding conditions. For these two cases, the core still remains covered. GEH was asked to clarify the justification for the limiting cases in Items A and C of RAI 6.3-46. In DCD, Tier 2, Revision 5, GEH stated that the calculation results showed that the ICS drainline and GDCS return line break are the most limiting cases. Therefore, the staff had no further questions regarding RAI 6.3-46, Items A and C.

RAI 6.3-86 asked GEH to show the sensitivity calculation results to demonstrate that the ICS drainline is the limiting case. GEH provided the sensitivity calculation results in its response, dated February 10, 2009. The nominal sensitivity calculation showed that the ICS drainline is the most limiting in terms of the chimney level. GEH further committed to documenting the sensitivity results in the new DCD revisions. The staff confirmed that the applicant included this change in Revision 6 of the DCD. Based on the applicant's response, RAI 6.3-86 was resolved.

However, there are still inconsistencies about limiting breaks in Revision 5 of the DCD, and RAI 6.3-85 requested a clarification from GEH. GEH responded with corrected DCD markups in its November 3, 2008, letter. The response is satisfactory, and RAI 6.3-85 was resolved. RAI 6.3-83 asked GEH to provide consistent tables for the LOCA break sizes analyzed and the LOCA analysis results in Tables 6.3-5a and 6.3-5 of the DCD. GEH responded, in its November 3, 2008, letter, that it provided the nonlimiting LOCA results in its response to RAI 6.3-46 and that the DCD documents contained the most significant LOCA results. Based on the applicant's response, RAI 6.3-83 was resolved.

In RAI 6.3-46, the staff requested the applicant to submit the technical bases for the limiting break in the break spectrum analyses. In response to RAI 6.3-46, GEH performed a sensitivity study of the GDCS line break size. The break sizes for this study ranged from the full double-ended break to 80, 60, 40, and 20 percent of this size. GEH ran these cases using nominal conditions and the 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 clarified 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. In its response, dated January 8, 2008, GEH stated that the minimum water level difference between break sizes of 100 percent and 80 percent for bounding assumptions is about 0.01 m, which is negligible. Therefore, there is no need to judge why a 100 percent break case is more limiting than an 80 percent break case. The staff accepted this argument, noticing the similarity of the system response for 80 percent and 100 percent breaks. GEH also stated, in its response to RAI 6.3-46, Item B, that, since the chimney static head for the 60 percent, 40 percent, and 20 percent nominal cases is higher than for the 100 percent and 80 percent cases, it is not necessary to analyze those cases with the bounding conditions. Therefore, the selection of bounding cases is acceptable. RAI 6.3-46 was

being tracked as an open item in the SER with open items. Based on the applicant's response, RAI 6.3-46 was resolved.

GEH did not analyze a break in the SLCS injection line. The staff was concerned about this break since it 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 inventory will be lost through the break. In RAI 6.3-65, the staff requested the applicant to address the consequences of the SLCS line 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 2,000 seconds of the event. In a supplement to RAI 6.3-65, the staff requested additional information from GEH on the event after 2,000 seconds. Section 6.3.2.3.3 of this report contains the resolution of RAI 6.3-65.

In Section 6.3.1.3, RAI 6.3-10 describes staff's request about the ADS control logic used to model level 1 setpoint in TRACG. In RAI 6.3-10 S01, the staff requested the applicant to explain in detail why the RPV level 1.5 plus drywell high pressure and the level 1.5 plus delay timer were removed from the ECCS initiation logic. In a relevant RAI 6.3-16, staff requested DCD clarification on GDCS initiation. GEH's response directed the staff to the DCD Table 6.3-1 where initiating signals and levels are listed. In a followup RAI 6.3-16 S01, the staff requested the applicant to provide the technical basis for the settings of the timer delays associated with the ECCS initiation logic. In response to both request for RAIs 6.3-10 S01 and 6.3-16 S01, 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 double-ended break size. GEH demonstrated that, for each of these break sizes, the "minimum chimney static head level above vessel zero" remains above the TAF. In RAI 6.3-10 S02, the staff asked GEH to 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." RAI 6.3-10 S02 also 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 also requested in RAI 6.3-10 S02 that GEH address the break sizes below 10 percent and provide the maximum break size that does not exceed the makeup system. In a relevant RAI 6.3-77, as described at the beginning of section 6.3.2.3.5, the staff requested the applicant to explain the calculation method for determining the "Minimum chimney static head level above vessel zero per active single failure m (ft)." GEH explained, in the response to RAI 6.3-77, that "chimney static head level with reference to vessel zero" is calculated by adding the equivalent height of water corresponding to the static head of the two-phase mixture inside the chimney to the elevation (7.896 m) of the bottom of the chimney. Furthermore, in its response to RAI 6.3-10, S02, on May 1, 2008, GEH explained that the level calculation did not account for the void fraction of the core. Since the calculation showed that there is a certain amount of collapsed water above the active core, the method of showing that the core is covered by water is acceptable. GEH discussed the relationship of break sizes and the minimum water level and argued, by extrapolating, that the water level still covers the top of the core for smaller break sizes. The staff does not agree with extrapolating results for the smaller sizes. However, GEH further explained that the normal reactor water makeup system is the feedwater system, and its capability is sufficient to provide inventory makeup for an 80-percent MSLB. In reality, small-size breaks would be well within the makeup capability of the feedwater system. In the event a

small break occurs that does not cause containment pressurization, the break would be detected by the LD&IS. Considering the ESBWR makeup water capability and the ADS, the staff accepts that there is no need to further analyze break sizes below 10 percent of an MSLB. And, because there is no core uncovery and the containment pressure is within limits, the response for the smaller break sizes is satisfactory. The technical basis requested by the staff in RAI 6.3-16 for the settings of the timer delays associated with the ECCS initiation logic were incorporated into the response of RAI 6.3-10 as described above. RAI 6.3-10 was being tracked as an open item. Based on the applicant's responses, RAIs 6.3-10 and 6.3-16 were resolved.

In RAI 6.3-76, the staff asked 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 for 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 case, it occurs before.

In RAI 6.3-76, the staff requested the applicant to explain the reason for the minimum chimney static head for the steam line break inside the containment for the cases run with bounding values are higher than those run using the nominal values. Although the bounding steamline break inside containment gives a higher collapsed liquid level in the chimney than the nominal case, 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 applicant adequately explained the differences in timing and magnitude for the interaction between the downcomer level and chimney level, as well as the differences between the bounding cases and the nominal cases, and hence 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. The minimum collapsed chimney level for the bounding case happens just before the injection and before the core oscillations. The staff reviewed 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 and RAI 6.3-76 is resolved.

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 6, 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 6, Table 6.3-11, states that GEH is using a peak linear heat generation rate of 44.95 kW/m (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 43.96 kW/m (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 for the ESBWR will be limited by the fuel rod thermal-mechanical design or minimum critical power ratio; therefore, the staff finds the value used for the LOCA analysis acceptable. The staff understands that the core operating limits report will specify the maximum linear heat generation rate. In accordance with the requirements of 10 CFR 50.46, GEH will update the LOCA evaluations, in the event that the maximum linear heat generation rate specified in the core operating limits report is allowed to exceed that used in the current LOCA analyses of record.

Axial Power Shapes. In RAI 6.3-50, the staff requested the applicant to provide the axial power shape used to perform the nominal and bounding LOCA analysis and provide a discussion on how this shape was selected. In response to RAI 6.3-50, GEH submitted the power shape used for the LOCA analyses. 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. 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. As 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. In response to RAI 6.3-54 S01, the applicant provided evidence, including sensitivity studies, showing that the results analyzed with the GSTRM and PRIME models are very similar. The staff also performed fuel conductivity sensitivity LOCA confirmatory calculations using the TRACE mode, and the results showed that the minimum water level in the limiting LOCA is not sensitive to the 30-percent conductivity reduction. Therefore, the staff concludes that GEH modeled the initial stored energy properly. RAIs 6.3-54 and 6.3-55 were being tracked as open items in the SER with open items. Based on the applicant's response, RAIs 6.3-54 and 6.3-55 were resolved. Section 21.6.3.2.14 of this report contains a detailed discussion of the evaluation of RAI 6.3-54.

Control Rod Insertion. RAI 6.3-52 asks GEH to provide the scram time delay and justify the delay time selected. During a review 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.

The applicant submitted its response to RAI 6.3-52 in a letter dated December 21, 2007. The trip delay time of 2.25 seconds is based on 2.00 seconds for the sensor delay, 0.05 seconds for the sensor trip scram solenoid to de-energize (RPS logic), and 0.20 seconds for the scram solenoid de-energize rods to start to move (scram valve open). GEH used a TRACG trip card to model this trip delay time and provided sufficient detail on how it modeled the travel time of the rods. RAI 6.3-52 was being tracked as an open item in the SER with open items. Since the applicant adequately explained the bases for the set point of the timer delay and adequately modeled it the applicant's response is acceptable, RAI 6.3-52 was resolved.

RAI 6.3-80 requested clarification of decay heat selections. In earlier DCD revisions, GEH inconsistently described the decay heat standard used in the TRACG model. In its response to RAI 6.3-80, GEH clarified a typographical error, stating that it based the decay heat curve on the ANSI/ANS 5.1-1994 standard, "Decay Heat Power in Light Water Reactors," and that there were no inconsistencies in ECCS performance analysis in the DCD. RAI 6.3-80 was being tracked as an open item in the SER with open items. Based on the applicant's response, RAI 6.3-80 was resolved.

RAI 6.3-62 requested further details on decay heat modeling. By letter dated August 17, 2007, the applicant submitted its response to RAI 6.3-62. The response gave details regarding the power used in the LOCA analysis. The ESBWR decay heat calculations were generated based on the ANSI/ANS 5.1-1994 standard, with additional terms for a more complete shutdown power assessment. The heat sources in the model include decay heat from fission products, actinides, and activation products, as well as fission power from delayed and prompt neutrons immediately after shutdown. The model considered the effect of neutron capture in fission products. GEH assumed end-of-cycle exposure and a conservative irradiation time for decay heat calculations. The irradiation time is the most sensitive input in the decay heat model. Increasing the irradiation time results in increased contributions from the long-lived actinides, thus resulting in higher shutdown powers. Since the decay heat is calculated following the appropriate standard ANSI/ANS 5.2, the staff considers that the assumption used for the decay curve is conservative. In addition, GEH provided assumptions of scram delay times, which included instrument detection of the plant parameters and the delay from signal processing. In the RAI, the staff also asked GEH to justify using the same decay heat curve for both small- and large-break LOCAs. GEH provided a power comparison between the end-of-cycle MSIV closure transient and a decay curve used in the LOCA analysis. The MSIV closure transient experiences a power increase at the beginning, caused by negative void feedback, compared to the power response in a small-break LOCA. The comparison showed that the decay heat curve bounds the MSIV transient power curve, which implies that the decay curve will bound the small-break LOCA as well, and the decay curve used is conservative. However, from the RAI discussion, the staff noticed that the assumptions for the scram signal delay time in the MSIV closure transient differ from those in the LOCA event. The staff estimated additional energy for the small-break LOCA, taking account of the scram delay time. This additional energy could boil off an extra amount of water in the vessel. The staff estimated the extra amount of water and, by subtracting this amount from GEH's minimum level prediction, estimated a new minimum water level. The estimated minimum water level is still above the top of the active core. Considering that other conservative assumptions are in the decay power calculation, the staff accepts the GEH approach of a single decay curve for all LOCA analyses. RAI 6.3-62 was being tracked as an open item in the SER with open items. Based on the applicant's response, RAI 6.3-62 was resolved.

Boric Acid Precipitation. DCD, Tier 2, Revision 6, Figure 5.1-1, gives a core volume of 96 m^3 (which does not include the volume in the chimney, separator, and lower plenum). In RAI 6.3-60, the staff requested the applicant to provide the maximum volume of the SLCS inventory that will be injected so the staff can evaluate the possibility of boron precipitation. In its response to RAI 6.3-60, GEH gave the maximum volume of the SLCS inventory that can be injected into the core. The volume of each of the two SLCS tanks is 8.31 m^3 (293.5 ft^3), giving a total possible SLCS injection inventory of 16.62 m^3 (586.9 ft^3). The SLCS tanks are at ambient temperature, with a 12.5 weight-percent (wt%) sodium pentaborate solution. The volume of the core is more than 5 times that of the SLCS tanks. Since there is no core uncovery and the amount of boron is relatively small unlike in PWRs and will be diluted, the staff finds that it is unlikely that boron will precipitate during a LOCA event in the ESBWR and, therefore, finds that the failure of GEH to analyze this possibility is acceptable.

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

ECCS Strainer Performance Evaluation. Section 6.2.1.7.3 of this report addresses ECCS strainer performance.

6.3.2.3.7 Reactor Protection System and Emergency Core Cooling System Actions

The staff reviewed the timing, sequencing, and capacity of the RPS and ECCS in relation to the design-basis LOCA analyses. In Revision 6 of the DCD, GEH stated that the ICS drainline break with failure of one GDCS 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 sections below discuss the evaluation of the RPS and ECCS functions for the design-basis events presented in DCD, Tier 2, Revision 6.

In RAI 6.3-56, the staff asked for more details on the sequence of events for several pipe breaks than were provided in DCD, Tier 2, Tables 6.3-7 through 6.3-10. The staff asked GEH to include trip signals and setpoints for all RPS actions, as well as the actions necessary for long-term core cooling. In its letter dated March 18, 2008, GEH responded that it revised DCD, Tier 2, Tables 6.3-7 through 6.3-10, to include the detailed sequence-of-events information and signals for all expected RPS actions. The RPS trip signals included are high drywell pressure and reactor water Level 3. In addition, the subject tables include ECCS initiation signals. The analyses show that no operator actions are required to support long-term core cooling (e.g., opening the GDCS equalizing line valves from the wetwell suppression pool to the RPV) for the timeframe of the sequence-of-events tables. DCD, Tier 2, Section 6.3.2.7, describes the actions supporting long-term core cooling beyond the timeframe established in the sequence-of-events tables, if required. This section explained that the long-term portion of GDCS can begin operation following a longer equalization valve time delay initiated by a confirmed ECCS initiation signal and by the RPV level reaching Level 0.5, which is 1 m (3.28 ft) above the TAF. The response to RAI 6.3-56 provided sufficient information on the sequence of events and trip signals for RPS actions. RAI 6.3-56 was being tracked as an open item in the SER with open items. Based on the applicant's response, RAI 6.3-56 was resolved.

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., a LOOP that results in a loss of all feedwater). DCD, Tier 2, Revision 6, Section 7.2.1.2.4.2, gives a complete list of

reactor scram signals. Those that would likely cause the reactor to scram during a LOCA include the following:

- high drywell pressure
- loss of power generation buses
- reactor water level reaching Level 3 and indicating that 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 under 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., a LOOP that results in a loss of all feedwater). DCD, Tier 2, Revision 6, 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:

- 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. The MSIV closure in the limiting LOCA analysis (ICS drainline break with failure of one GDCS injection valve) is initiated on 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 can inject cooling water. The ADS is initiated upon confirmation of the Level 1 setpoint or drywell high pressure. Confirmation of Level 1 occurs when it persists for 10 seconds, and confirmation of high drywell pressure occurs when it persists for 60 minutes. Section 6.3.2 of this report discusses the ADS, including the 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 under 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 enable it to perform 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 or a sustained high drywell pressure. 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, the collapsed chimney level experiences large oscillations. In NEDC-33083P-A, GEH stated that these are manometric oscillations. These oscillations occur as the voids in the core are quenched and a larger static head is created inside the shroud that reduces the flow from the downcomer, leading 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. 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 updated these calculations to reflect the most recent design. GEH responded to the supplemental RAI 6.3-64 on April 18, 2008. The original RAI response provided the limiting case for the containment LOCA, which is a GDCS line break with one DPV failure. The supplement asked why GEH did not choose a vessel-level limiting case for the long-term safety analysis. GEH provided plots in the supplement for the limiting case and explained that the level response in the short term is more important. The long-term calculation showed that the core is covered with water. GEH further stated that it originally included the discussion of the treatment of noncondensable gases in the analysis coverage in its response to RAI 21.6-96 (MFN 07-348, dated June 21, 2007), and it would clarify it further in the pending response to RAI 21.6-96, Supplement 1. GEH also agreed to include a discussion of the GDCS bounding case in DCD, Tier 2, Section 6.3.3.7.9. The staff confirmed the inclusion of the GDCS bounding analysis in Revision 5 to the DCD. The staff concurs that the minimum water level is determined in the short term, after the break initiation, and agrees that the containment wall condensation has no major impact on the equilibrium RPV level and that the long-term level in the vessel will be filled up to the break location or spillover hole. The response to RAI 6.3-64, Supplement 1, provided analysis results showing that the reactor core is covered by water up to 72 hours. RAIs 6.3-64 and 21.6-98 were being tracked as open items in the SER with open items. Based on the applicant's response, RAI 6.3-64 was resolved. The staff documented its evaluation of the response to RAI 21.6-98 in Section 21.6 of this report.

In RAI 6.3-45, the staff asked GEH to explain the differences between the TRACG input decks used to calculate minimum water levels and those used to perform the containment peak pressure analyses. In a supplemental RAI, the staff asked GEH to justify its assertion that, even though the input deck for calculating minimum water levels lacks the modifications applied to the containment input deck, the results are still accurate and conservative for the long-term core cooling analysis. GEH responded to RAI 6.3-45 on June 20, 2007, and to the supplement on

March 25, 2008. GEH responded that the analyses in DCD, Tier 2, Revision 4, had reconciled the model differences described in its response to RAI 6.3-45. GEH used a consistent set of assumptions, the same TRACG model, and a consistent input deck to calculate minimum water levels and to perform containment peak pressure of nominal cases. However, the assumptions made for the bounding cases for the containment analysis and the RPV water level analysis were different. GEH updated the table in its response to RAI 6.3-45 and explained the differences for the bounding cases; these differences include the normal water level in the downcomer and suppression pool. The staff agrees with GEH that using the lower water level in the minimum water level calculation is bounding for the LOCA analysis and using a higher water level in the suppression pool is bounding for the peak containment pressure calculation. GEH clarified the difference between the minimum water level calculation and the peak containment pressure analyses. RAI 6.3-45 was being tracked as an open item in the SER with open items. Based on the applicant's response, RAI 6.3-45 was resolved.

6.3.2.3.9 Loss-of-Coolant-Accident Analysis under Feedwater Temperature Operating Domain

GEH submitted an LTR, "ESBWR Feedwater Temperature Operating Domain Transient and Accident Analysis," NEDO-33338, Class I, in October 2007. For plant operation with feedwater temperature maneuvering (increase and reduction), GEH evaluated the GDCS injection line breaks for the initial core at the increased and decreased feedwater temperature operating points. The applicant did not find any significant chimney-level differences in the LOCA analysis among the core performance setpoints at SP0, SP1, or SP2. However, in the LTR, GEH did not show the limiting IC drainline break analysis for the expanded operating domain. RAI 6.3-82 asked GEH to provide an analysis for limiting break cases in the high and low feedwater temperature operating points. In its response to RAI 6.3-82, dated November 3, 2008, GEH committed to providing the limiting break in the high and low feedwater temperature operating points. The staff verified that, in LTR NEDO-33338, Revision 1, GEH analyzed the limiting break at the requested operation points, SP1 and SP2, and demonstrated that the minimum chimney water levels are above the TAF. The staff concluded that the ESBWR LOCA analysis showed that the reactor can safely operate in the expanded feedwater temperature operating domain, and RAI 6.3-82 was resolved.

6.3.2.3.10 Independent Staff Calculations

Plant Model

The staff used the TRACE thermal-hydraulics code model and independently verified the ESBWR system response in the event of a LOCA. The staff based its confirmatory calculations on the ESBWR design documented in Revision 5 of the ESBWR DCD. The breaks examined were the MSLB, the FWLB, the IC line break (ICLB), the GDCS line break (GDLB), and the bottom drainline break (BDLB). The staff made the calculations with and without an IC heat transfer (ICHT) to investigate the GEH assumption of no ICHT in its safety analyses. The heat structures connecting the IC to the pool were removed for the calculations without ICHT. The water inventory of the IC is kept available to the RPV. In addition, the staff performed a fuel conductivity sensitivity study to examine how sensitive the minimum water level is to the stored energy.

Summary of Results

The staff's study found that the analyzed cases do not show a core uncovery or heatup. A significant difference is seen in the pressure and level response between the cases with and without ICHT. Two effects were observed when the ICHT was removed from the calculation. The first and obvious effect was that removing the heat exchangers reduced the amount of heat removal from the system. A second effect is that more water from the IC drain tanks enters the system in a short time without ICHT, since it is a constant volume draining process and condensation of the steam in the ICs limits the amount of water that can drain into the RPV. A summary of the results is given in Table 6.3-5. The GDLB is the limiting break for the cases with no ICHT.

The minimum collapsed chimney level was 2.4 m (7.9 ft) above the top of the active core. Applying the additional conservative assumption of maintaining atmospheric pressure in the wetwell gas space lowers the minimum chimney level to 2 m (6.6 ft). The selection of a limiting GDLB agrees with Revision 4 of the DCD but does not agree with Revision 5. Revision 5 of the DCD shows that the limiting LOCA is an ICLB. The reactor responses in ESBWR LOCAs have similar characteristics, as the ADS turns the LOCA into a situation similar to a large-break LOCA. The minimum water level prediction is sensitive to the timing of the ADS initiation signal. The level oscillation changed the timing of ADS initiations, which is why a minor parameter change can cause the limiting LOCA to change from one case to another. Finally, the fuel conductivity sensitivity study showed that, with a decrease of 30 percent in the conductivity value, the change in the minimum water level is minimal. The staff calculations confirmed that there is enough water inventory to cover the core in all LOCAs.

Table 6.3-5 Minimum Average Chimney Collapsed Level

Break	Minimum Level Base	Minimum Level No ICHT
MSLB	3.4	3.6
FWLB	3.0	3.1
ICLB	3.3	3.1
GDLB	3.1	2.4
GDLB Atmospheric WW		2.0
BDLB	3.3	2.7
0.5 * BDLB	3.3	2.9
0.25 * BDLB	3.6	

6.3.2.4 *Conclusions*

The staff reviewed DCD, Tier 2, Section 6.3, and other relevant material regarding the ESBWR ECCS design, including process diagrams. The staff reviewed the ESBWR design bases and design criteria for the ECCS, as well as the manner in which the ESF design conform to the

criteria and bases. The staff concludes that the ESBWR ECCS design meets the guidelines of SRP Section 6.3 and the requirements of the following GDC:

- GDC 2, "Design Bases for Protection Against Natural Phenomena," the ECCS is designed to meet the seismic Category 1 requirements and remain functional following a safe-shutdown earthquake (SSE).

- GDC 4, "Environmental and Dynamic Effects Design Bases," The ECCS design incorporates features that preclude water hammer and excessive dynamic loads.

- GDC 5, "Sharing of Structures, Systems, and Components," The ECCS is designed for a single nuclear power plant and is not shared between units.

- GDC 17, "Electric Power Systems," the ECCS performs its functions without relying on onsite or offsite ac power.

- GDC 27, "Combined Reactivity Control Systems Capability," and GDC 35, "Emergency Core Cooling," safety analyses of the design-basis transients and accidents were performed with the assumption that the most reactive control rod stuck out of the core, and the results demonstrate that the ECCS can provide abundant core cooling, so that (1) fuel and clad damage will not interfere with continued effective core cooling, and (2) the acceptance criteria specified in 10 CFR 50.46 for LOCAs are met.

- GDC 36, "Inspection of Emergency Core Cooling System," and GDC 37, "Testing of Emergency Core Cooling System," the ECCSs and components are designed to permit periodic inspection and testing of the operability of the system throughout the life of the plant.

The ESBWR design includes preoperational testing for the ECCS, as discussed in DCD, Tier 2, Section 14.2.8. In addition, DCD, Tier 1, Sections 2.1.2, 2.2.4, 2.4.1, and 2.4.2, specify (1) the design commitments of the ECCS, (2) the inspections, tests, or analyses to be performed by the COL applicants, and (3) the acceptance criteria to ensure that the COL applicants build the ECCS as designed. Therefore, the staff finds the ESBWR ECCS design acceptable.

Based on the TRACG analysis provided in the DCD and in its responses to RAIs, GEH demonstrated that there is no core uncovery or heatup for any design-basis LOCA. The fuel does not heat up during a LOCA; therefore, the PCT is expected to be within the acceptance criterion of 1,204 degrees C (2,200 degrees F). There is no additional oxidation of the cladding as a result of a 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 a LOCA that would prevent the core from being amenable to cooling. The ECCS conforms to the review guidelines and acceptance criteria of SRP Section 6.3. The staff concludes that the ECCS meets the acceptance criteria of 10 CFR 50.46 and the pertinent requirements of GDC 2, 4, 5, 17, 27, 35, 36, and 37.

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. These systems are the CRHA HVAC subsystem

(CRHAVS), the radiation monitoring subsystem (RMS), the lighting system, and the FPS. The ESBWR design includes features to ensure that the control room operators can remain in the control room and take actions both 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, lighting, personnel and administrative support, and fire protection.

6.4.1 Regulatory Criteria

The staff reviewed the ESBWR DCD, Tier 2, Revision 7, Section 6.4, 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 these systems:

- GDC 4, "Environmental and Dynamic Effects Design Bases," as it relates to SSCs important to safety being designed to accommodate the effects of, and to be compatible with, the environmental conditions associated with postulated accidents

- GDC 5, "Sharing of Structures, Systems, and Components," as it relates to ensuring that sharing among nuclear power units of SSCs important to safety will not significantly impair the ability to perform safety functions, including, in the event of an accident in one unit, an orderly shutdown and cooldown of the remaining unit(s)

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

- 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 (NUREG-0737), regarding protection against the effects of toxic substance releases, either onsite or offsite

- 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"

- 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 83, "Control Room Habitability (Revision 3)"

- SRM and SECY 94-084 as they apply to the use of RTNSS to address uncertainties as a defense-in-depth method

The generic safety issues can be found in NUREG-0933.

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 a high level of radioactivity is detected in the CRHA outside air supply duct, the RMS automatically isolates the normal air supply to the CRHA. 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. The CRHA is equipped with a variable orifice relief device to ensure that the amount of air exhausted from the CRHA is equal to that supplied. When ac power is unavailable, the CRHA is passively cooled by the CRHA passive heat sink.

The process 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.

The MCR provides storage capacity 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
- provisions for protection from airborne radioactive contaminants

The CRHA is contained inside a seismic Category I structure (the control building (CB)) 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 MCR environment suitable for prolonged occupancy for the duration of a postulated accident. In particular, the systems ensure the following:

- The MCR is designed to withstand the effects of an SSE and a design-basis tornado.

- The radiation exposure of MCR personnel for 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 fresh air requirements in American Society of Heating, Refrigeration and Air Conditioning Engineers (ASHRAE) Standard 621, "Ventilation for Acceptable Air Quality," issued 2007, for up to 21 MCR occupants.

- The habitability systems detect and protect MCR personnel from external fire, smoke, and airborne radioactivity.

- The individual systems that perform a habitability system function are automatically actuated. 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 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 valves, including supporting ductwork and 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 a loss of normal ac power.

- 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.

- Electrical power for safety-related equipment, including EFUs, dampers, valves, and associated instrumentation and controls, is supplied from the safety-related uninterruptible power supply. Active safety-related components are redundant, and their power supply is divisionally separated, such that the loss of any two electrical divisions does not render the component function inoperable.

The EFUs are redundant safety-related components that supply filtered air to the CRHA for breathing and pressurization to minimize inleakage. The EFUs and their related components

form a safety-related subset of the CRHAVS. Each train consists of an air intake, fan filtration housing, ductwork, and dampers.

The EFU delivery and a variable orifice relief device discharge system are optimized to ensure that there is adequate fresh air delivered and mixed in the CRHA. This is accomplished by using multiple supply registers, which distribute the incoming supply air with the control room air volume, and a remote exhaust to prevent any short cycling. The EFU-delivered supply air is distributed in the MCR area of the CRHA. The EFUs turn over the volume of control room air approximately seven to nine times per day.

This diffusion design (mixing and displacement), in conjunction with convective air currents (caused by heat loads or sinks) and personnel movement, ensures that the occupied zone temperature is within acceptable limits, the buildup of contaminants (e.g., carbon dioxide (CO_2)) is minimal, and the air remains fresh.

The “Occupied Zone” of the MCR region is normally occupied and is generally considered to be between the raised floor and 2 m (6.6 ft) above the floor. Short cycling refers to a poor design condition, where the outside air transits the served space and exhausts to the outside without mixing. This occurs when the outside air inlet and room exhaust are in close proximity. The fresh air for the CRHA is supplied at a high elevation and the exhaust for removing the air is below the floor, so the two are not in close proximity to each other.

Control Room Habitability Area

The CRHA boundary is located on elevation –2000 mm in the CB.

The CRHA envelope includes the following areas:

- Administration Area (Room 3270)
- Reactor Engineer/Shift Technical Advisor Office (Room 3271)
- Shift Supervisor Office (Room 3272)
- Kitchen (Room 3273)
- MCR (Room 3275)
- Restroom A (Room 3201)
- Restroom B (Room 3202)
- MCR Storage Room (Room 3204)
- Electrical Panel Board Room (Room 3205)
- Gallery (Room 3206)
- Auxiliary Equipment Operators Workshop (Room 3207)
- Air-Handling Unit (AHU) Room (Room 3208)

These areas constitute the operation control area, which can be isolated and remain habitable for the duration of a DBA if high radiation conditions exist. Potential sources of danger, such as steamlines, pressurized piping, pressure vessels, CO_2 firefighting containers, and the like, are located outside the CRHA.

Heat Sink

The function of a passive heat sink for the CRHA, which is part of the CRHA emergency habitability system, is to limit the temperature rise inside each room during the 72-hour period following a loss of CRHAVS operation.

The CRHA heat sinks consist of the following: the CRHA outer walls, floor, ceiling, and interior walls and access corridors; adjacent safety-related distributed control and information system (Q-DCIS) and non-safety-related DCIS (N-DCIS) equipment rooms and electrical chases; and CRHA HVAC equipment rooms and HVAC chases. After the 72-hour period, the EFU maintains the habitability of the CRHA using RTNSS power supplies. The recirculation AHU, with supporting auxiliary cooling units, removes heat to support MCR habitability after 72 hours.

Shielding Design

The design-basis radiological analysis presented in the DCD, Chapter 15, crediting the control room protective features, dictates the shielding requirements for the CRHA. DCD, Chapter 15, Section 15.4, contains descriptions of the design-basis LOCA source terms, MCR shielding parameters, and evaluation of doses to MCR personnel.

Component Descriptions

The EFU outside air supply portion of the CRHAVS is safety-related and seismic Category I. Two trains, which are physically and electrically redundant and separated, provide single active failure protection. If one train fails, it is isolated, and the alternate train is automatically initiated. Both trains have 100-percent capacity and are capable of supplying 99-percent credited efficiency filtered air to the CRHA pressure boundary at the required flow rate. The exhaust from the CRHA is through a variable orifice relief device, which is safety-related, and its location is optimized to ensure proper scavenging of the air from the control room in an amount equal to the supply. Backflow prevention through the controlled leak path, the variable orifice relief device, is not required, since the CRHA is at a positive pressure during normal and emergency operation. The EFU design uses a prefilter, a high-efficiency particulate air (HEPA) filter, a carbon filter, and a postfilter to provide radiological protection for the CRHA outside air supply.

The CRHA pressure boundary includes penetrations, dampers and valves (including the variable orifice relief device), interconnecting duct or piping, and related test connections and manual valves. The isolation dampers and valves are classified as Safety Class 3 and seismic Category I. The dampers and valves have spring return actuators that fail closed 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 valves will be demonstrated by pressure testing the CRHA and by inleakage testing.

Tornado protection dampers are a split wing or an equivalent type, designed to close automatically. The tornado protection dampers are designed to mitigate the effect of a design-basis tornado.

Each access to the MCR has two sets of doors, with a vestibule between them that acts as an air lock.

Leaktightness

The CRHA boundary envelope structures are designed with low-leakage construction. The CRHA is located in an underground portion of the CB. The boundary walls are adjacent to underground fill or underground internal areas of the CB. The construction consists of cast-in-place reinforced concrete walls and slabs to minimize leakage through joints and penetrations.

During normal operation, the CRHA is heated, cooled, ventilated, and pressurized by either of a redundant set of recirculation of AHUs and either of a redundant set of outside air intake fans for ventilation and pressurization purposes. During a radiological event or upon loss of normal ac power, an EFU maintains a positive pressure in the CRHA to minimize the infiltration of airborne contamination. The access doors are designed with self-closing devices, which close and latch the doors automatically.

There are double-door air locks for access and egress during emergencies. Interlocked, double-vestibule doors maintain positive pressure, thereby minimizing infiltration when a door is opened. The CRHA remains habitable during emergency conditions.

Emergency Habitability

The CRHA emergency habitability portion of the CRHAVS is not required to operate during normal conditions, with the exception of the variable orifice relief device. This device is in service to exhaust CRHA air during normal and emergency operation. The normal operation of the CRHAVS maintains the air temperature within a predetermined temperature range. This maintains the CRHA emergency habitability system's passive heat sink at or below a predetermined temperature. The normal 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: a high level of radioactivity in the MCR supply air duct or an extended loss of normal ac power.

Operation can also be initiated by manual actuation. If radiation levels in the MCR supply air duct exceed the high setpoint, the normal outside air intake and restroom exhaust are isolated from the CRHA pressure boundary by the 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 pascals (Pa) (⅛-in. WG) positive differential pressure with respect to the surroundings. The variable orifice relief device exhausts excess air from the CRHA. This device is a locked-in-place orifice or valve set up to maintain CRHA pressure at the delivered flow. The EFU system airflow rate is also sufficient to supply a fresh air requirement of 10.5 L/s (22 cfm) per person for up to 21 occupants.

Airflow in Emergency Mode

The following mechanisms mix the EFU-supplied inlet air with the general CRHA air:

(1) Supply or inlet registers—The mixing is continuous, as EFU-provided outside air is delivered to the CRHA. Each cfm delivered mixes with the control room air as it exits the supply registers. This is the most common type of space air diffusion, called a mixing

system. The supply air is delivered through the air inlet registers, which create an air jet that then mixes the outside air with the room air by entrainment (induction); this helps to reduce the jet velocity and equalize the supply air temperature as it enters the CRHA.

(2) Displacement (ventilation) supply or exhaust—As air is supplied to the CRHA, a similar amount is exhausted from the space. This displaced air is designed to exhaust at a remote location to prevent short cycling and ensure a properly scavenged control room.

(3) Equipment and personnel convective plumes caused by air differential temperature and density—The higher temperature of the air surrounding operating equipment and personnel generates convective air plumes that rise out of the occupied zone, along with any pollutants (e.g., body odors). The rising air is replaced by cooler air from below.

(4) Personnel movement—The airflow requirements are derived from the assumed activity level of the CRHA occupants. This activity generates mixing of the CRHA air.

(5) Molecular dispersion—CO_2 and other contaminants are moved across a space by molecular dispersion.

The airflow developed in the ESBWR control room during worst case (outside air temp of 117ºF) accident conditions when the CRHA is isolated and the EFU is in operation with passive cooling is as follows and is illustrated in DCD, Tier 2, Figure 6.4-2.

The EFU is operating and provides 220 L/s (466 cfm) of clean outside air into the CRHA. This is delivered to the occupied MCR area, primarily, since this area contains the personnel on duty and houses the active electronic equipment. This supply air exits the ductwork at supply air diffusers (4), which perform the mixing mechanism in (1) above. Depending upon the delivered air temperature, the combined mixed volume either rises or drops. During the worst-case accident conditions, where the outside air is 47.2 degrees C (117 degrees F), modeling shows that this air mixture rises above the ceiling, with a larger quantity of heated air in the MCR; the balance is driven primarily by the convective plumes of the equipment and personnel (mechanism (3) above). The combined air, rising above the ceiling tiles, draws the same quantity of air into the MCR from the area below the raised floor (mechanism (2)). This cooler, slow-moving air gradually spreads over the raised floor and displaces the warmer, stale air toward the ceiling, where it leaves the room. The MCR with the high ceiling becomes thermally stratified (i.e., warmer stale air is concentrated above the occupied zone and cool, fresher air is concentrated in the occupied zone). When the cool air encounters a heat source, such as a person or heat-generating equipment, the air heats up and buoyantly rises out of the occupied zone. The hot air, including CO_2 and body-generated odors, rises because of the air density difference, collects above the suspended ceiling, and spills over into the adjacent rooms. The heat is then released to the cooler walls and concrete. Cooler air in these adjacent rooms drops to the raised floor level and through to the common space below the floor. The discharge flow, 220 L/s (466 cfm), of this air exits the MCR at a remotely opposite location from the EFU supply, to prevent any short cycle of the supply air and ensure a constant turnover of the CRHA air. This air is then drawn into the MCR, and the circuit is complete.

A positive pressure is maintained in the CRHA. There is no buildup of CO_2, since these areas are scavenged continuously by the EFU supply and the exhaust airflow of 220 L/s (466 cfm). The exhaust is located in the lower common area of one of the adjacent rooms and is remote from the EFU supply.

With a source of ac power available, the EFU can operate and is controlled indefinitely through Q-DCIS. In the event that normal ac power is not available, a safety-related battery power supply is sized to provide the required power to the EFU fan for 72 hours of operation. The CRHA isolation dampers fail closed on a loss of normal ac power or instrument air.

One of two ancillary diesel generators provides backup power to the safety-related EFU fans (post-72 hours), if normal ac power is not available. These generators support operation of the control room EFU beyond 72 hours after an accident. For a period between 7 days and the duration of the DBA, the safety-related function of the EFU can be powered by offsite power, by an onsite diesel-generator-powered plant investment protection bus, or by continued use of the ancillary diesel generators. Appendix 19A describes the RTNSS requirements for the ancillary generators.

Upon a loss of normal ac power, the initial temperature in the CRHA ranges from 21.1 to 23.3 degrees C (70 to 74 degrees F), and the relative humidity ranges from 25 to 60 percent.

The CRHA temperature and humidity values calculated during the 72 hours following a DBA equal less than 32.2 degrees C (90 degrees F) wet bulb globe temperature (WBGT) index. The 32.2 degrees C (90 degrees F) WBGT index value is the acceptability limit for minimizing performance decrements and potential harm and preserving the well-being and effectiveness of the control room staff for an unlimited duration.

During the first 2 hours of loss of normal ac power, most of the equipment in the MCR remains powered by the same non-safety-related battery supply that powers the non-safety MCR equipment. Any time during a loss of normal ac power, once either ancillary diesel is available, it can maintain the environmental conditions indefinitely. This is accomplished through the continued operation of a CRHA recirculation AHU and the auxiliary cooling unit supplied with each recirculation AHU. If this cooling function is lost, the N-DCIS components in the MCR are automatically de-energized. This is accomplished through safety-related temperature sensors with two-out-of-four logic that automatically trips the power to selected N-DCIS components in the MCR, thus removing the heat load caused by these sources. The remaining CRHA equipment heat loads are dissipated passively to the CRHA heat sinks. The CRHA heat sinks limit the temperature rise by passively conducting heat into the concrete thermal mass.

System Safety Evaluation

Doses to MCR personnel are calculated for the accident scenarios where the EFU provides filtered air to pressurize the CRHA. Doses are calculated for the following accidents:

- 1000 Fuel Rod Failure Dose Results, Table 15.3-16
- Radwaste System Failure Accident Dose Results, Table 15.3-19
- LOCA Inside Containment Analysis Total Effective Dose Equivalent (TEDE) Results, Table 15.4-9
- Main Steam line Break Accident Analysis Results, Table 15.4-13
- Feedwater Line Break Analysis Results, Table 15.4-16

- Small Line Carrying Coolant Outside Containment Break Accident Results, Table 15.4-19

- RWCU/SDC Line Break Accident Results, Table 15.4-23

For all events, the control room dose is within the dose acceptance limit of 5.0 rem (50 millisieverts (mSv)) TEDE. Chapter 15 contains the details of the analytical assumptions for modeling the doses to the MCR personnel. No radioactive material storage areas are located adjacent to the MCR pressure boundary. The control room ventilation inlet distances from potential release points are maximized to the extent possible. However, the separation distances in SRP Section 6.4 are not always met. Failure to meet these distances is acceptable because the dose analyses developed for the CRHA used the actual plant layout of the CB intake louvers and potential release points.

As discussed and evaluated in SRP Section 9.5.1, 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 MCR atmosphere. SRP Section 9.4.1 discusses the operation of the CRHAVS in the event of a fire. The exhaust stacks of the onsite standby power diesel generators and ancillary diesel generators are located more than 48 m (157 ft) from the fresh air intakes of the MCR.

The fuel oil storage tanks for the onsite standby power system and the ancillary diesel generators are located more than 55 m (180 ft) feet from the MCR fresh air intakes. These separation distances reduce the possibility that combustion fumes or smoke from an oil fire would be drawn into the MCR.

Table 6.4-2 lists the typical sources of onsite chemicals, and Figure 1.1-1 shows their locations. The staff analyzed 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,” is to be applied on a site-specific basis (Section 6.4.9).

During emergency operations, the design of the passive heat sink for the CRHA emergency habitability system limits the temperature inside the CRHA to 33.9 degrees C (93 degrees F). This maintains the CRHA within the limits for reliable human performance (DCD, Tier 2, Section 6.4.10, References 6.4-1 and 6.4-2) 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.

Table 3H14 lists the input parameters assumed in the CB heatup analyses. The EFU portion of the CRHAVS provides 220 L/s (466 cfm) of ventilation air to the MCR and is sufficient to pressurize the control room to at least a positive 31 Pa (⅛ inch WG) differential pressure with respect to the adjacent areas. This flow rate also supplies the recommended fresh air supply of 10.5 L/s (22 cfm) per person for a maximum occupancy of 21 persons (DCD, Tier 2, Section 6.4.10, Reference 6.4-4).

The normal and emergency (EFU) outside air intake flows are adjusted as required to maintain a minimum flow and, in conjunction with a controlled leak path, maintain a 31 Pa (⅛ inch WG) minimum positive pressure in the CRHA, relative to adjacent areas. Flow instrumentation is provided for the fans and AHUs to ensure airflow is maintained above the minimum required.

A low-airflow alarm is provided. CRHAVS differential pressure transmitters are provided to monitor CRHA pressure with respect to adjacent areas and to ensure the pressure is maintained above the minimum positive pressure. A low CRHA differential pressure alarm is provided. A variable leakage device is located under the raised floor to facilitate air circulation and mixing, with sufficient adjustment to maintain the required airflow and CRHA positive pressure, relative to adjacent areas, under all normal and emergency conditions requiring operation of the CRHA AHU or EFU. The CRHA air intake flows and the positive CRHA differential pressure are periodically monitored during operation of the CRHA AHU or EFU.

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 takes into consideration the radiological decay of fission products and the infiltration and exfiltration rates to and from the CRHA pressure boundary. Chapter 15 fully describes the specific radiological protection assumptions used in the generation of post-LOCA radiation source terms.

The use of noncombustible construction and heat- and flame-resistant materials, wherever possible throughout the plant, minimizes the likelihood of fire and the consequential fouling of the control room atmosphere by smoke or noxious vapor. In the smoke-removal mode, a dedicated fan, intake, and exhaust path purge the control room with a high volume of outside air.

The EFU automatically starts during a radiological event, independent of the loss of normal ac power. Through the use of redundant EFU components and dampers, one EFU and supply path to the CRHA would be available during a loss of normal ac power, with failure of up to two divisions of safety-related power, to provide CRHA breathing air and pressurization during a loss of ac power, concurrent with a radiological event. Local, audible alarms warn the operators to shut the self-closing doors, if, for some reason, they are open.

Testing and Inspection

A program of preoperational and post-operational testing requirements is implemented to confirm initial and continued system capability. The CRHAVS is tested and inspected at appropriate intervals consistent with plant technical specifications. Emphasis is placed on tests and inspections of the safety-related portions of the habitability systems. Design changes to the CRHA will ensure key design assumptions are met such as:

- Heat sink / Heat source assumptions
- Air flow assumptions
- Heat transfer values

This will ensure that CRHA calculations and methodologies are maintained and updated throughout the life of the plant.

The applicant provided the following two COL Information Items:

6.4-1-A CRHA Procedures and Training

The COL Applicant will verify procedures and training for control room habitability address the applicable aspects of NRC Generic Letter 2003-01 and are consistent with the intent of Generic Issue 83, A Prioritization of Generic Safety Issues, NUREG-0933, October 2006. (ESBWR DCD, Tier 2, Reference 6.4-3), System Operation Procedures (ESBWR DCD, Tier 2, Subsection 6.4.4), including statements under Testing and Inspection (ESBWR DCD, Tier 2, Subsection 6.4.7).

6.4-2-A Toxic Gas Analysis

The COL applicant will identify potential site-specific toxic or hazardous materials that may affect control room habitability to meet the requirements of TMI Action Plan Item III.D.3.4 and GDC 19. The COL applicant will determine the protective measures to be instituted to ensure adequate protection for control room operators, as recommended in RG 1.78. These protective measures include features to (1) provide the capability to detect releases of toxic or hazardous materials, (2) isolate the control room if there is a release, (3) make the control room sufficiently leaktight, and (4) provide equipment and procedures for ensuring the use of breathing apparatus by the control room operators (COL Information item 6.4-2-A).

Testing and Inspection

A program of preoperational and postoperational testing requirements will confirm initial and continued system capability. The CRHAVS is tested and inspected at appropriate intervals, consistent with plant TS. Emphasis is placed on tests and inspections of the safety-related portions of the habitability systems.

Design changes to the CRHA will ensure key design assumptions are met, such as the following:

- heat sink and heat source assumptions
- airflow assumptions
- heat transfer values

This will ensure that CRHA calculations and methodologies are maintained and updated throughout the life of the plant.

Preoperational Inspection and Testing

Preoperational testing of the CRHAVS will verify that the minimum airflow rate of 220 L/s (466 cfm) is sufficient to maintain pressurization of the MCR envelope of at least 31 Pa (⅛ in. WG), with respect to the adjacent areas. The variable orifice relief device is set during this evolution to ensure that an equal amount of air is exhausted from the CRHA. The differential pressure transmitters monitor and confirm the positive pressure within the MCR.

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 certified as within the guidelines of ASHRAE Standard 62.1/2007 requirements for continued occupancy, by meeting the fresh air supply requirement of 10.5 L/s (22 cfm) per person for the type of occupancy expected in the CRHA. The capacity of the safety-related battery is verified to ensure it can power an EFU fan for a minimum of 72 hours. Heat loads within the CRHA are certified as less than the specified

values. Preoperational testing of the CRHAVS isolation dampers verifies the leaktightness of the dampers. Preoperational testing for CRHA inleakage during EFU operation is conducted in accordance with ASTM E741, "Standard Test Method for Determining Air Change in a Single Zone by Means of a Tracer Gas Dilution." SRP Section 11.5 discusses the testing and inspection of radiation monitors, while Chapter 14 discusses the other tests noted above.

Inservice Testing

Inservice testing of the CRHAVS includes operational testing of the EFU fans and filter unit combinations, EFU filter performance testing, automatic actuation testing of the CRHA isolation dampers and EFU fans, and unfiltered air inleakage testing of the CRHA envelope boundary. The CRHA boundary is pressure tested periodically to verify leaktightness on the envelope walls, doors, and boundaries. The integrity of the CRHA envelope is tested in accordance with RG 1.197 and ASTM E741.

The control room EFU supplies air with a design flow rate of 220 L/s (466 cfm), and it is designed to maintain the control room envelope at a positive pressure, with respect to adjacent compartments, during normal operation and radiological events. An intake filter efficiency of 99 percent is assumed for particulate, elemental, and organic iodine species. The system does not include filtered recirculation, and the design incorporates leaktightness requirements (SRP Section 6.4.3). Although the control room is maintained at a positive pressure, the dose analysis assumes an unfiltered inleakage rate of 5.66 L/s (12.0 cfm).

Based on the ESBWR CRHA design and ventilation system operation, the acceptance criteria for inleakage associated with the CRHA will be no greater than the amount of unfiltered leakage assumed in the dose consequence analysis minus 2.36 L/s (5 cfm), which is the amount of unfiltered inleakage allocated for ingress and egress.

Nuclear Air Filtration Unit Testing

The EFU filtration components are periodically tested in accordance with ASME AG-1-2003, "Code on Nuclear Air and Gas Treatment," to meet the requirements of RG 1.52. Periodic surveillance testing of safety-related CRHA isolation dampers and the EFU components are carried out in accordance with IEEE-338-2006, "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 the plant's normal and abnormal operating modes.

Instrumentation Requirements

The MCR contains alarms for the following CRHA/CRHAVS conditions:

- low airflow (each EFU fan, recirculation AHU, and outside air intake fan)
- high filter pressure drop (each EFU and normal outside air intake filters)
- high space room temperatures (non-safety-related temperature detection)
- high room temperature (safety-related temperature detection)
- low room temperature
- low recirculation AHU entering air temperature
- low CRHA differential pressure
- smoke detected
- high and low humidity in the CRHA

- CRHA airlock doors that are open during an SBO
- high radiation in the CRHA
- high radiation in the outside air intake duct

If the redundant, non-safety-related CRHAVS cooling is lost, and the CRHA temperature increases, safety-related sensors provide a trip signal through the safety-related system logic and control ESF to de-energize selected non-safety N-DCIS equipment located in the CRHA. Safety-related sensors monitoring CRHA temperatures provide the logic to trip selected N-DCIS loads in the CRHA. A common alarm is provided to indicate a high CRHA air temperature and a potential high thermal heat sink temperature. Furthermore, this high-temperature alarm setting is set below the N-DCIS trip setpoint. This early detection of rising CRHA and heat sink temperatures allows early operator attention and action before selected N-DCIS loads are tripped in the MCR and ensures operators will take appropriate actions before experiencing temperatures in excess of those assumed in the CRHA heatup calculation. CRHA heat sink temperatures are assumed to be within the specified limit if the average of the air temperatures in the heat sink has been within the specified limit. The temperature response of the materials in the CRHA heat sink area is slower than the response of the average air temperature on increasing temperature (i.e., a loss of normal cooling). If the average of the CRHA air temperatures exceeds the specified limit, restoration of the CRHA heat sinks is verified by an administrative evaluation, considering the length of time and extent of the CRHA heat sink average air temperature excursion outside of limits, or by direct measurement of the temperatures of the structural materials in the CRHA heat sink area.

6.4.3 Staff Evaluation

The staff reviewed the information in DCD, Tier 2, Section 6.4, and referenced sections, to determine compliance with the GDC, TMI Action Plan items, and other appropriate regulatory criteria and guidance documents.

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

The CRHAVS and its 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.

RAI 6.4-13 was being tracked as an open item in the SER with open items. In RAI 6.4-13, the staff asked the applicant to identify which intakes are protected against tornado damage and to provide an assessment of the impact of a sudden pressure drop resulting from a tornado. In MFN 08-288, dated March 31, 2008, GEH responded to the RAI. The applicant revised the DCD to state that all CRHA ventilation penetrations for outside air intake and exhaust openings have tornado protection. In addition, the CB ventilation systems outside air intake and return exhaust openings have tornado protection. Because the applicant revised ESBWR Tier 2, Subsection 9.4.1.1, Design Bases to include a design requirement that all CRHA ventilation penetrations for outside air intake and exhaust openings are provided with tornado protection, the staff finds that this Tier 2 design requirement provides assurance that the CRHAVS components located on the outside of the Category 1 structure will also be protected from

tornado and missile damage. Therefore, based on the applicant's response, RAI 6.4-13 was resolved.

The design of nonseismic pipe, ductwork for kitchen and sanitary facilities, and other nonessential components in the CRHA ensures that their failure during an SSE will not adversely affect essential components.

Potential sources of danger, such as pressure vessels and CO_2 firefighting containers, are located outside the CRHA.

There are no high-energy lines in the CB that could affect the CRHA; 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 I building and is protected from the effects of external environmental conditions, such as wind, flooding, pipe whip, and discharging fluids from high-energy piping.

GDC 5, "Sharing of Structures, Systems, and Components," requires that SSCs important to safety shall not be shared among nuclear power units, unless it can be shown that such sharing will not significantly impair their ability to perform their safety functions, including, in the event of an accident in one unit, an orderly shutdown and cooldown of the remaining units. The staff review determined that the CRHAVS meets the acceptance criteria of GDC 5. The ESBWR control room habitability design supports a single unit. SSCs important to safety are not shared among nuclear power units. Thus, the design satisfied the GDC 5 requirements.

GDC 19, "Control Room Habitability," requires that a control room be provided from which actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions, including LOCAs. It also requires that adequate radiation protection be provided to permit access and occupancy of the control room under accident conditions, without personnel receiving radiation exposures in excess of 5 rem (50 mSv) TEDE for the duration of the accident.

Implicit in GDC 19 is that the environmental conditions (such as temperature, humidity, lighting, air circulation, oxygenation, and atmosphere degradation) will be acceptable for personnel and equipment to function. The ESBWR passive reactor design has limited safety-related battery power sources and passive cooling features. The design relies on reducing electrical loads, including lighting, to a minimum, eliminating air recirculation, eliminating nonessential instrumentation and personnel, and other related heat sources, to control power consumption items in the period of 0–72 hours in which a loss of ac power from active sources is not credited.

The applicant justifies the passive control room's reduced function on the basis that, for the first 72 hours, essentially no operator actions are required, and non-safety-related instrumentation and equipment can be isolated and shut down. Forced air supply during the period of 0-72 hours is unconditioned air distributed by one of two redundant EFUs to occupied areas of the control room. Both control room recirculation AHUs are shut down, and no forced air is supplied to the kitchen, bathrooms, shift supervisor's office, and other areas deemed to be unoccupied. There are substantial concrete physical barriers between unoccupied and occupied areas, but these barriers have openings in the ceiling plenum and floor plenum spaces

of the CRHA. Convective air currents exist at some level, and these currents provide the potential for mixing. CRHA air temperatures and air mixing are interrelated and evaluated in a subsequent section.

The staff agrees that the ESBWR's passive design features reduce the requirement for operator action in the first 72 hours following an accident. Consequently, this would permit some reduction in the requirements for control room temperature and humidity during this period. However, the essential habitability function of the control room postaccident must still be satisfied. The staff interprets the postaccident function based on GDC 19 and the guidance in SRP Section 6.4 and NUREG-1242. The principal function is to provide a protected and acceptable environment where operators and others who may be present can monitor and maintain the reactor in a safe stable shutdown condition and take action, if necessary, to respond to any adverse performance of systems and components. The actions may involve planning; communicating with State and Federal officials; interfacing with the NRC; evaluating unexpected performance issues, such as a failed component or system; and taking direct physical actions to ensure public health and safety. The staff evaluated the protective and environmental control features discussed below.

In regard to GDC 19 as it applies to radiation protection, the CRHA is well shielded with its position below grade and its enclosure inside the CB. The two principal sources that affect operator dose in the control room are (1) the radiation that bypasses the filter, because of filter inefficiency in the EFU supply air, and (2) the unfiltered inleakage from all other sources.

RAI 6.4-11 was being tracked as an open item in the SER with open items. In RAI 6.4-11, the staff asked the applicant whether the EFU supply louver location, as shown in DCD, Tier 2, Revision 3, Figures 1.2.3.and 1.2.11, is consistent with SRP Section 6.4, Revision 3, Acceptance Criterion 5A; specifically, if the louvers are separated from potential release points by 30.5 m (100 ft) laterally and 50 ft (15.2 m) vertically, and whether the actual minimum distances are based on the dose analyses. In its response, MFN 07-687, dated December 21, 2007, GEH confirmed that the ESBWR does not always meet the SRP guidance for intake vertical and horizontal distances from potential release points; however, the dose analyses used actual plant layout data for the intake louver location and release points. The applicant included this information in DCD, Tier 2, Section 6.4.5. The staff reviewed the response and the DCD changes. Because ESBWR Tier 2, Subsection 6.4.5 was revised to clarify that the dose analyses developed for the CRHA used the actual plant layout of the Control Building intake louvers and potential release points, the staff finds that the separation of intake louvers from potential release points acceptable. Therefore, based on the applicant's response, RAI 6.4-11 was resolved.

The EFU filters are safety-related and designed and tested with appropriate TS surveillances, in accordance with RG 1.52. Other potential sources of leakage into the CRHA are from people entering or leaving and leakages that could occur through cracks and crevices around penetrations or other locations. These other potential sources of leakage are controlled by pressurization of the CRHA to a positive pressure of 31 Pa (⅛ inch WG) and by construction and design to ensure very low leakage. The applicant assumed 5.66 L/s (12 cfm) for this unfiltered inleakage in the DBA analysis.

RAI 6.4-14 was being tracked as an open item in the SER with open items. In RAI 6.4-14, the staff asked the applicant to include additional details of EFU supply and purge duct paths. In MFN 08-009, dated January 9, 2008, the applicant proposed revisions to Section 6.4.3 of the DCD that included these details. The staff has reviewed the DCD changes and finds that they

adequately clarify and describe the ductwork external to the CRHA associated with the EFU supply, the normal outside air supply, and the smoke purge pathways. Therefore, based on the applicant's response, RAI 6.4-14 was resolved.

RAI 14.3-152 was being tracked as an open item in the SER with open items. In RAI 14.3-152, the staff asked the applicant to provide an ITAAC to verify that the leaktightness of the CRHA had been achieved by testing, in accordance with the guidance in RG 1.197. The applicant clarified that DCD, Tier 1 Revision 4, Table 2.16.2-16 added ITAAC 5.b for confirming that Control Room Habitability Area in-leakage does not exceed the unfiltered in-leakage assumed by control room operator dose analyses. In addition Tier 2 chapter 16 Technical Specification Section 5.5, "Programs and Manuals," includes a section on CRHA boundary control, in which the applicant commits to periodic CRHA leakage testing, performed in accordance with RG 1.197, to verify that the inleakage would not exceed the value assumed in the design-basis analysis. The staff reviewed the RAI response and the referenced Tier 1 and Tier 2 sections and found that ITAAC and Technical Specification requirements ensure sufficient verification of the initial and periodic leak tightness of the CRHA. Therefore, based on the applicant's response, RAI 14.3-152 was resolved.

The value assumed in the analysis consists of two parts: the assumed leakage of the CRHA, and the value assumed for access and egress. The assumed access and egress value must be subtracted from the assumed unfiltered inleakage value used in the analysis to obtain the acceptance criteria for CRHA testing.

In RAI 6.4-22, the staff asked the applicant to clarify the DCD to clearly state that the ESBWR COL applicant is required to justify a near-zero value for the CRHA access and egress leakage limit.

In MFN 09-759, dated December 5, 2009, GEH responded to the RAI. The staff reviewed the response to RAI 6.4-22 S01, and requested that the applicant further clarify, in the DCD that the acceptance criteria for CRHA unfiltered inleakage will be no greater than the amount of unfiltered leakage assumed in the dose consequence analysis minus the amount of unfiltered inleakage allocated for CRHA access and egress. The staff requested that the applicant revise Section 6.4.4 of the DCD to include the value assumed for access and egress for CRHA unfiltered inleakage and to provide a basis for the number assumed, or alternatively, revise the DCD to indicate that this number must be specified and justified by the COL applicant.

In response to the RAI, the applicant revised DCD, Tier 2, Section 6.4.7, "Testing and Inspection, Inservice Testing," to specify 2.3 L/s (5 cfm) as the amount of unfiltered inleakage allocated for CRHA access and egress. The applicant revised DCD Chapter 16, Section 5.5.12, "Control Room Habitability Area (CRHA) Boundary Program," to indicate that the quantitative limit of unfiltered air inleakage will be the inleakage flow assumed in the licensing basis analyses of DBA consequences, less the amount designated for ingress and egress. The staff finds the proposed DCD changes acceptable because they conservatively allocate a minimum value of unfiltered leakage that is due to CRHA access and ingress and this value is in accordance with SRP Section 6.4. . The staff confirmed that the applicant had incorporated these changes in DCD, Tier 2, Revision 7. Therefore, based on the applicant's response, RAI 6.4-22 was resolved.

RAI 14.3-153 was being tracked as an open item in the SER with open items. In RAI 14.3-153, the staff requested that the applicant provide an ITAAC to verify that the unfiltered leakage is no greater than the value assumed in the dose analysis in DCD Chapter 15. Based on a review of

the RAI response and the response to RAIs 14.3-152 and 6.4-22, as discussed above, the staff found the responses acceptable because they confirm that DCD, Tier 1, Table 2.16.2-16 ITAAC 5.b exists which ensures that CRHA unfiltered inleakage will not exceed the unfiltered in-leakage assumed by the control room operator dose analyses. Therefore, based on the applicant's response, RAI 14.3-153 was resolved.

The unfiltered inleakage allocation of 2.3 L/s (5 cfm) is reasonable, because, as stated in the DCD, during a radiological event or upon loss of normal ac power, an EFU maintains a positive pressure in the CRHA to minimize infiltration of airborne contamination. The access doors are designed with self-closing devices, which close and latch the doors automatically. There are double-door air locks for access and egress during emergencies. Interlocked double-vestibule doors maintain the positive pressure, thereby minimizing infiltration when a door is opened.

The staff finds that the test acceptance criterion for CRHA unfiltered inleakage is in accordance with SRP Section 6.4 and RG 1.197 guidance.

It is acceptable to the staff for the applicant to test to the low-leakage criteria, if the assumptions are justified and if the applicant performs the test in accordance with the requirements of RG 1.197. The staff finds that, through control of inleakage from filter inefficiency or other unfiltered sources, and by acceptable results in the dose consequence analyses, the applicant has provided adequate radiation protection for control room operators.

In regard to GDC 19 as it applies to toxic gas protection, the RAIs 6.4-12 and 6.4-15 were being tracked as open items in the SER with open items. In RAIs 6.4-12 and 6.4-15, the staff requested that the applicant identify the design features in the ESBWR standard design that mitigate the consequences of a toxic gas event. In MFN 07-687, dated December 21, 2007, the applicant explained that the ESBWR does not make specific provisions for toxic gas control. Instead, the ESBWR identifies COL items whereby each COL applicant must review the potential effects of toxic gas spills on the specific site, near the site, or in transportation modes in the vicinity of the site, in accordance with RG 1.78. In the event toxic gas levels exceed guidance values in the CRHA, the COL applicant must submit a plan acceptable to the staff that provides for the protection of control room operators. The staff reviewed the RAI responses and COL information Item 6.4-2-A and found them acceptable because a COL information item requires a toxic gas review to be performed by an applicant that references the ESBWR standard design. The details of any required design provisions, required by the plan, to mitigate the consequences of a toxic gas event would be provided by a COL applicant. Therefore, based on the applicant's responses, RAIs 6.4-12 and 6.4-15 were resolved.

In regard to GDC 19 as it applies to air quality in the MCR, the number of occupants affects the freshness of the air and cooling or heating loads. The ESBWR designed the air supply to provide 220 L/s (466 cfm). A review of ASHRAE 62.1-2007 indicates that this is more than sufficient for the 11 personnel assumed to occupy the CRHA during postaccident isolation. The staff considered the guidance of NUREG-1242 and concluded that, postaccident, there would be an expanded control room occupancy that may include a utility executive, an NRC observer, a communications specialist, five operators, and potentially two individuals from the Technical Support Center staff, if the Center is not available, and that the air supply would be sufficient. In RAI 9.4-57, the staff asked the applicant to describe how the design-basis assumptions on CRHA occupancy will be controlled throughout the life of the plant. In response to the RAI, the applicant revised DCD, Tier 2, Sections 6.4.5 and 6.4.7, to identify critical key assumptions, such as heat sink values, that will be controlled through procedures. The applicant indicated that DCD, Tier 2, Section 17.4, "Reliability Assurance Program," ensures

that relevant aspects of plant operation are maintained. COL Item 6.4.1-A directs COL applicants to develop procedures to control such parameters for the CRHA. The staff finds the response, including the proposed DCD changes, acceptable because COL Item 6.4-1-A requires a COL applicant to develop procedures and training for control room habitability that specifically address statements under Testing and Inspection section of the DCD, ESBWR DCD, Tier 2, Subsection 6.4.7. ESBWR DCD, Tier 2, Subsection 6.4.7 states, among other things, that assumption for heat sources will be maintained throughout the life of the plant. The staff confirmed that these changes were incorporated in DCD, Tier 2, Revision 7. Based on the applicant's response, RAI 9.4-57 was resolved.

With regard to GDC 19 as it pertains to control room air quality, the staff reviewed provisions for temperature control, air supply distribution, and mixing. For normal operation the staff determined that the ESBWR design provided sufficient conditioned air with adequate recirculation by the non-safety-related supply fans and the RTNSS-qualified AHUs, with the associated heating and cooling coils. The recirculation AHU also provides humidity control. The system is powered from the station's ac system. The staff also determined that temperature control for postaccident operation was adequate, as long as ac power was available to operate an AHU and the associated heating and cooling equipment.

With regard to postaccident operation, the staff considered a LOCA that included a 0–72 hour operation with loss of offsite power (LOOP). AC power is not credited from non-safety-related sources for 72 hours following the accident. The applicant evaluated the impact on control room temperatures for both the 0 percent exceedance summer design condition of 47 degrees C (117 degrees F), with 20 percent residual humidity, and the winter design condition of -40 degrees C (-40 degrees F).

The staff acknowledges that the concurrence of a LOCA with a LOOP at the maximum or minimum design temperatures would be a statistically small occurrence. Also, the redundant ancillary diesel generators are RTNSS-qualified, and their availability is controlled through the Availability Controls Manual and the Maintenance Rule. In addition, it usually takes much less than 72 hours to restore a LOOP.

RAI 6.4-7 was being tracked as an open item in the SER with open items. In RAI 6.2.4-7, the staff asked the applicant to describe how the temperature is maintained for the entire 30-day accident period, to clarify the need for active CRHA cooling after 72 hours into the accident, and to identify CRHAVS non-safety-related systems and power supplies included in the RTNSS. In MFN 08-392, dated April 29, 2008, the applicant provided a more detailed description of these issues. The applicant submitted revisions to DCD, Tier 2, Sections 6.4.3 and 9.4.1.1 which added details on what Control Room Area Ventilation components are providing CRHA cooling at various times during the 30-day accident period. The changes also clarified the need for post-72 hour active cooling in the control room and how this will be provided.

In RAI 9.4-31 the staff requested the applicant clarify design details of the power source for the EFU mentioned in RAI 6.4-7 response, which was proposed to be used during the post-72-hour period. In its response to the RAI, the applicant modified the design such that the EFUs rely on ancillary diesel generators, which are RTNSS power supplies. As described in Section 9.4.1 of this SER, the staff has reviewed and found acceptable the RTNSS systems associated with the CRHAVS as a means to provide post-72 temperature control for the CRHA.

Because on the applicant's DCD changes clarified the role of various CRHA heat removal structures and systems and clarified what CRHA structures were operating for each phase of the entire 30 day accident period, RAIs 6.4-7 and 9.4-31 were resolved.

For the first 72 hours after a DBA with a loss of ac power, the CRHA zone is isolated. The unfiltered supply air system is shut down and isolated by safety-related dampers. One of two EFU fans starts and supplies filtered air to the CRHA at 220 L/s (466 cfm). The operating recirculation AHU is shut down. Power for the system is provided by a safety-related battery system. With the isolation of the recirculation AHU, normal temperature control is lost, and air circulation in the CRHA is driven only by the EFU supply fans and convective currents. Air circulation and supply distribution is important in maintaining a uniform bulk temperature throughout the multiroom CRHA and in ensuring fresh air for operators at any location.

The applicant states that the ESBWR uses a passive heat sink consisting of the walls, ceilings, and floors of the CRHA to maintain the temperature at less than 33.9 degrees C (93 degrees F).

RAI 6.4-8 was being tracked as an open item in the SER with open items. In RAI 6.2.4-8, the staff asked the applicant how it would ensure the initial passive heat sink temperature.

The applicant established the maximum normal operation temperature in the CRHA at 21.1 degrees C (74 degrees F). The maximum temperature in the CRHA is important, in that it establishes the basis for the initial concrete heat sink temperature used in the passive heat sink analysis. In MFN 08-288, dated March 31, 2008, the applicant explained that the heat sink temperature will be controlled by a TS 3.7.2 surveillance performed every 24 hours. The staff reviewed the RAI response and found it acceptable because the surveillance provides assurance that the actual temperatures of the heat sinks relied upon for passive cooling of the control room will be periodically monitored to ensure that they remain conservative with respect to the assumptions uses for these values in the passive heat sink design basis analysis . Therefore, RAI 6.4-8 was resolved.

The staff reviewed the maximum temperature of 33.9 degrees C (93 degrees F), with respect to environmental qualification requirements of safety-related components, and determined that this temperature was acceptable. The staff reviewed the maximum CRHA temperature value of 33.9 degrees C (93 degrees F), as stated in DCD Table 9.4-1, against the mild environment equipment qualification temperature of 50 degrees C (122 degrees F), as stated in the DCD, Appendix 3H, Table 3H-10. RAI 9.4-34 and RAI 3.11-28 were issued to the applicant to resolve staff questions in this area as described below.

In RAI 9.4-34, the staff asked the applicant to clarify whether the design considers the reduced airflow and locally increased temperature inside electrical cabinets during the period of passive cooling, and whether those temperatures pose a challenge to equipment operation.

In RAI 3.11-28, the staff asked the applicant to provide additional details on how the service temperature of electrical equipment, including computer-based instrumentation and control (I&C) systems, will be determined for the ESBWR. In particular the applicant was asked to provide details on this process for equipment that is planned to be located inside electrical cabinets or panels in the RB and the CB. The applicant was also asked to explain how the detailed design and testing of electrical equipment, including enclosures, would be carried out, so that the key assumptions of environmental bounding temperatures in these areas remain conservative.

In response to the RAIs, the applicant revised DCD, Tier 2, Sections 3.11.1.3, 3.11.4.3, and 3.11.3.1, to more fully explain the temperature qualification process.

The applicant revised the DCD, Section 3.11.1.3, definition of equipment, to indicate that computer-based I&C equipment is defined by the equipment plus its surrounding enclosure. It revised the DCD, Section 3.11.4.3, to indicate that system testing of computer-based I&C equipment within its cabinet or enclosure is preferred.

In the DCD, Section 3.11.3.1, the applicant states that the EQ equipment in the CRHA is to be tested at temperatures that are 10 degrees C (18 degrees F) higher than the maximum temperature to which the equipment is exposed for the worst-case abnormal operating occurrence, with the equipment at maximum loading. This temperature is given at 50 degrees C (122 degrees F), as stated in the DCD, Appendix 3H, Table 3H-10. In addition, DCD, Tier 2, Section 3.11.3.2, states that margins will be included in the qualification parameters to account for normal variations in the commercial production of equipment and reasonable errors in defining satisfactory performance, and that the environmental conditions shown in the Appendix 3H tables do not show such margins. The staff noted that, in DCD Section 3.11.3.2, the applicant stated that the program margin would be in accordance with the guidance in IEEE-323-2003, "Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations." The staff infers that the applicant used the +5 degrees C (+8 degrees F) value, as stated in the document.

Thus, since CRHA EQ equipment is to be tested at 60 degrees C (140 degrees F), there is some confidence that the equipment would not fail if actual local temperatures exceed the calculated maximum average CRHA bulk temperature of 33.9 degrees C (93 degrees F) by several degrees. Based on the margin in the assumed normal operating temperature used in the CB heatup analysis, and the conservatism inherent in the EQ process that establishes the equipment service temperature, the staff finds that local temperatures are not likely to challenge component operability before ac power is restored. The staff concludes that, independent of operator actions or offsite support, the CB ventilation system design maintains satisfactory environmental conditions for equipment to function for the first 72 hours after the onset of an accident that assumes that all ac power is lost for this period. Therefore, based on the applicant's responses, RAI 9.4-34 and RAI 3.11-28 were resolved.

The staff considered the impact on operators operating in an elevated-temperature environment. The applicant's passive cooling analysis indicates that the temperature in the CRHA would reach a 30 degrees C (86 degrees F) dry bulb bulk temperature in approximately 12 hours. After 12 hours, the temperature rate of change is much lower, reaching a CRHA bulk temperature of 33.5 degrees C (92.5 degrees F) at 72 hours. Humidity may also increase from moisture contained in the supply air. Based on a review of NRC and industry standards, the staff notes that human performance is most frequently assessed based on the WBGT index.

In RAI 6.4-24, the staff asked the applicant to justify the use of a psychrometric wet bulb temperature as a valid index to assess heat stress in the ESBWR CRHA, or alternatively, to amend the DCD to provide a heat stress acceptance criterion and index that is in accordance with NRC guidance. The staff also asked the applicant to demonstrate that such a criterion can be met for the ESBWR environmental footprint. The staff also asked the applicant to identify the associated ITAAC.

In response to RAI 6.4-24, Supplement 1, the applicant revised the DCD to state that the WBGT index would be the design-basis means by which a heat stress acceptance criterion would be

measured. The applicant stated that the CRHA is designed such that 32.2 degrees C (90 degrees F) WBGT would not be exceeded at the end of 72 hours of passive cooling. The applicant provided an accompanying CONTAIN 2.0 computer code demonstration and revised DCD, Tier 1, Table 2.16.2-4, to include an ITAAC 4iii that requires a COL applicant to demonstrate this, using an analysis updated with as-built design information.

The staff has compared the proposed DCD revisions and analysis result to NRC and industry guidance and has found that, although high, the applicant's chosen WBGT index acceptance criterion for heat stress at the end of 72 hours of passive cooling would not require compensatory actions, such as stay times. Therefore the staff concludes that the ESBWR CRHA temperature and humidity at the end of 72 hours of passive cooling is acceptable with regard to human performance. The staff confirmed that these changes were incorporated in DCD, Tier 2, Revision 7. Based on the applicant's response, RAI 6.4-24 was resolved.

The staff reviewed the analytical basis for evaluation for temperature in the control room habitability area for the 0-72 hour post-accident period. The applicant submitted a passive cooling analysis (Control Building Environmental Temperature Analysis) as part of the licensing basis that evaluates heat transfer by use of the CONTAIN computer code. The results indicate that the maximum bulk temperature reached in the CRHA during the 0–72-hour period is less than 33.9 degrees C (93 degrees F).

The staff finds that the applicant's use of an analytical approach as a method to demonstrate the passive heat removal mechanism and to show that the CRHA bulk temperature will not exceed design-basis limits is reasonable.[1]

RAI 6.4-16 was being tracked as an open item in the SER with open items. In RAIs 6.4-16 and RAI 9.4-32, the staff asked the applicant to discuss the need to provide cooling to non-safety-related heat loads in the CRHA following an accident. In response to these RAIs, the applicant explained that, as stated in DCD, Tier 2, Section 9.4.1.2, CRHA non-safety-related heat loads are automatically de-energized when the CRHA AHUs are not available during the first 2 hours, and discussed operator actions to isolate the non-safety-related heat loads. The staff found the RAI response acceptable because they clarify an analysis assumption on accident heat load: that non-safety heat loads in the CRHA will be de energized during such accidents. RAIs 6.4-16 and 9.4-32 were therefore resolved.

In RAI 9.4-33, the staff asked the applicant to provide sufficient information needed for the staff to evaluate the performance of the ESBWR passive cooling features. In response to RAI 9.4-33, the applicant provided analysis assumptions for the control room design and outside environmental conditions for a single-node model of the CRHA that demonstrates the mechanism by which heat is removed (i.e., the absorption of heat by thermal mass of concrete).

The staff noted some conservative parameters in the Control Building Environmental Temperature Analysis, such as the assumptions used for the heat transfer to the concrete, the conservative assumption regarding the initial heat sink temperatures, and the margin for assumed heat loads. In order to ensure the as-built CHRA design captured these assumptions, the staff asked the applicant in RAI 9.4-55 to incorporate the Control Building Environmental Temperature Analysis in the DCD and revise the ITAAC to specifically refer to this analysis.

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1 See Yilmaz, T.P., and Paschal, W.B., "An analytical approach to transient room temperature analysis," Nuclear Technology, 114:135–140.

In response to RAI 9.4-55, the applicant submitted Control Building Environmental Temperature Analysis , LTR NEDE-33536P, as Tier 2* information, and revised DCD, Tier 1, Table 2.16.2-4, to clearly link ITAAC 4i, 4ii, and 4iii to the submitted LTR. The staff confirmed that the changes were incorporated in DCD, Tier 2, Revision 7.

Because the DCD changes associated with the RAI response clearly establish the analysis methodology for the passive heat sinks, and because the applicant has made changes to that methodology and its assumptions subject to staff review, the staff found this acceptable and RAIs 9.4-55 and 9.4.33 were resolved.

The staff has reviewed the results of the applicant's Control Building Environmental Temperature Analysis as a basis for meeting the design requirements for the MCR HVAC systems, as stated in Chapter 9, Section 8.2.2.1, of the Utility Requirements Document, and in SRP Section 9.4.1. The staff conducted an audit of the applicant's calculation and performed confirmatory calculations using the same methodology and input assumptions. The staff obtained similar results.

The Control Building Environmental Temperature Analysis, model relies on EFU fan flow for air circulation. Because the applicant chose to model the CRHA as a single node, the design-basis analysis model does not demonstrate the convective mixing mechanism that would also be expected to occur. In addition, the design-basis model does not illustrate pressure changes in the CRHA caused by temperature differences between the supply and exhaust air during EFU operation.

In RAI 9.4-29 the staff requested that the applicant provide assumptions used to establish the minimum EFU fan flow rate criterion that is used to ensure adequate fresh air supply to the CRHA. The staff also requested additional information on how mixing of air would occur in the CRHA.

In response to RAI 9.4-29, the applicant provided the results of an analysis of a multinode GOTHIC model. The results demonstrated temperature stratification in the CRHA and convective mixing. The applicant included CRHA airflow design details obtained from this analysis, including a description and illustration of the airflow expected in the CRHA occupied zone in DCD, Tier 2, Section 6.4. Based on a review of the design of the CRHA air distribution system as described in the DCD, the staff finds that such mixing would occur and would improve the air quality and temperature in the CRHA. The staff considers the DCD design requirements for mixing and distributing the EFU-supplied inlet air sufficient to ensure that the air quality will be within ASHRAE Standard 62.1 guidelines. RAI 9.4-29 was, therefore, closed.

In RAI 9.4-49, the staff requested that the applicant provide additional information on the applicability of ASHRAE 62 to a tightly closed facility, such as the ESBWR MCR, and determine whether there are long-term indoor air quality effects on habitability that need to be addressed. The applicant responded that preoperational testing as described in DCD, Tier 2, Section 6.4.7 and surveillances as described in Generic Technical Specifications Section 5.5.13 in DCD, Tier 2, Chapter 16, will verify that the minimum air flow rate to the CRHA will be supplied. The applicant clarified that CO_2 and odors will be removed using the CRHA leakage paths, including the controlled leakage path. The applicant clarified the DCD to include a design requirement for 7 to 9 air changes to take place per day in the CRHA, and added details for air supply and exhaust location in the CRHA. The staff found the RAI response acceptable because the Tier 2 changes clarify the importance of design features to ensure adequate air supply and quality to the CRHA; therefore RAI 9.4-49 was resolved.

The staff reviewed the means by which the as-built CRHAVS heat sink will be analyzed to ensure that it will passively maintain the temperature in the CRHA within the design basis for the first 72 hours following a DBA. The means of verification of this design commitment is a CB temperature analysis, using the as-built heat sink dimensions, thermal properties, exposed surface area, as-built thermal properties of materials covering parts of the heat sink, and the as-built heat loads to confirm the results of the control room design-basis heatup analysis.

A CB temperature analysis will be used to confirm the control room winter design-basis heatup analysis to demonstrate that the CRHA bulk air temperature will not be below 12.8 degrees C (55 degrees F) on a loss of normal heating for 72 hours, given winter design-basis conditions.

The staff reviewed DCD, Tier 1, Table 2.16.2-4, ITAAC, 4i and 4ii, and verified that sufficient ITAAC exist to perform a thermal analysis, with as-built design details, that confirms the results of the MCR design-basis heatup analysis.

The staff has considered some use of conservative assumptions in the applicant's design-basis heatup model, such as the assumed thermophysical properties of CB concrete, the orientation of the CB for the highest solar radiation, a 15-percent margin in the assumed sensible heat load, an assumed CRHA failure 8 hours before the postulated accident (resulting in increased CRHA air and heat sink temperatures at the start of the analysis), and the applicant's use of higher heat sink temperatures for walls in contact with the ground than would be expected. Based on the use of these conservatisms, and the staff review of the applicant's model, as previously discussed, the staff finds that the applicant has adequately demonstrated that the CB passive heat sinks would likely limit the CRHA occupied zone bulk temperature to below the design-basis temperature of 33.9 degrees C (93 degrees F) for 72 hours, assuming no ac power sources are available for that period. The staff concluded that this bulk temperature would not significantly affect CRHA operator or equipment performance, and the ITAAC acceptance criteria for the summer maximum CRHA bulk average air temperature of 33.9 degrees C (93 degrees F) are acceptable.

The applicant evaluated the minimum CRHA temperature using ECOSIMPRO software, which its consultant developed and owns. The applicant benchmarked the ECOSIMPRO software against the CONTAIN software for the summer design case. The ECOSIMPRO code also assumes a single node for the CRHA. The ECOSIMPRO results showed a minimum bulk temperature in the CRHA of 16 degrees C (61 degrees F) at 72 hours. Based on a review of the analysis results, the staff concluded that the CB passive heat sinks would likely limit the CRHA occupied zone bulk temperature above this design-basis temperature value for 72 hours, assuming no ac power sources are available for that period. The staff concluded that this bulk temperature would not significantly affect the performance of CRHA operators or equipment, and the ITAAC acceptance criteria for the winter minimum CRHA bulk average air temperature of 12.8 degrees C (55 degrees F) are acceptable.

In summary, the staff concludes that the CONTAIN analysis adequately predicts the CRHA occupied zone's maximum and minimum bulk temperatures within the applicant's acceptance criteria. The CONTAIN analysis adequately demonstrates a mechanism of thermal absorption of heat in the CRHA. Verification of the analysis with as-built design and site environmental parameters provides adequate assurance that assumptions in the analysis remain valid. The applicant's maximum and minimum temperature acceptance criteria are adequate to ensure that the CRHA would have an acceptable environment for personnel and equipment in a postulated

accident. Thus, the staff concludes that the passive cooling design and associated acceptance criteria are acceptable.

The staff acknowledges that a certain degree of uncertainty remains concerning the performance of the CB ventilation system's unique passive features and the overall performance of the CRHA heat removal system, because of lack of a proven operational performance history. Although not credited by the applicant or the staff to function before 72 hours, the staff notes that the design and regulatory treatment of the ancillary diesels, as described and reviewed in Section 9.4.1 of this SER, make it likely that this non-safety-related source of ac power will be available for CRHA AHU operation before 72 hours. Section 8.3.1.1 of the DCD states that the ancillary diesel generator automatically starts upon sensing undervoltage on their respective busses. Based on the review of the functional capability and availability of these systems, the staff notes there is additional defense-in-depth protection in the CB ventilation system design to overcome this inherent uncertainty.

In regard to GDC 19 as it applies to air quality the staff reviewed EFU supply register location and provisions for air distribution. During normal operation or post-72 hour operation, the location of the EFU supply registers is not critical and the RTNSS-qualified AHU fully establishes air mixing. During the postaccident 0 to 72-hour operation with a LOOP, air mixing is important to keep localized temperatures from reaching the extremes and to ensure that fresh air is maintained in the operator breathing zone. The applicant located the EFU supply registers just underneath the false ceiling in the occupied zone of the CRHA. The heat or the cold added by the registers would likely be caught in the convective current updraft and distributed to all areas of the CRHA. The staff concluded that some convective currents are probable and that the supply registers in this location have a beneficial effect on mixing.

After 72 hours, the EFU and AHU can be powered by offsite power sources or by two redundant ancillary diesels that start automatically on a LOOP. The staff reviewed the temperature controls in the post-72-hour period for the duration of the accident. The applicant has made provisions to start one of the two RTNSS-qualified AHUs to increase circulation in the CRHA. In addition, the applicant has arranged for additional CRHA cooling to be connected to the AHU cooling water piping outside the CRHA by a valve arrangement.

In regard to GDC 19 as it applies to habitability, the ESBWR emergency lighting provides 10-ft candles at all workstations in the main operation areas. This is consistent with the recommendations for emergency lighting in NUREG-0700, "Human-System Interface Design Review Guidelines," issued May 2002. High-efficiency lighting will be used. The applicant assumed a heat load of 400 watts for the emergency lighting in the passive cooling analysis. Although the staff considered this to be a marginal lighting design, it realized that additional portable battery-operated lighting is readily available and could be used to supplement lighting, if needed, for the 0–72-hour postaccident situation with a LOOP.

In regard to GDC 19 as it applies to air supply, stratification, and mixing, the applicant designed the air supply for both normal and postaccident operation on the basis of ASHRAE 62.1—2007, which uses a combination of requirements for personnel and area to determine fresh air requirements. The applicant established 220 L/s (466 cfm) as the supply air flow rate. For normal operation with an AHU providing recirculation, the staff considers the flow rate to be adequate. For postaccident operation with a LOOP, the AHU is isolated. The air in the CRHA is mixed by convective currents, personnel movement, molecular dispersion, and the EFU supply air, with the EFU supply registers located in the MCR area of the CRHA.

The applicant has included design features to promote mixing. The staff finds that these features would promote mixing and mitigate stratification. The staff finds that the ESBWR designed in compliance with ASHRAE 62.1 air quality standards would limit the buildup of other contaminants, such as CO_2, and provide enough mixing to ensure that the CRHA remains at the bulk temperature calculated in the licensing-basis CONTAIN passive cooling analysis. The staff concludes that there would be some convective flow that would augment EFU flow and that air movement would be sufficient to keep the air mixed for freshness and to prevent the buildup of contaminants.

In regard to GDC 19 as it applies to CHRA pressurization and air discharge control, RAI 6.4-9 was being tracked as an open item in the SER with open items. RAI 6.4-9 requested additional information about the adequacy of the EFU system flow rate to maintain CRHA pressurization. In MFN 08-288, dated March 31,, 2008, the applicant provided an analysis to demonstrate that the control room makeup flow is sized for leakage from the control room boundary when the control room is pressurized to a positive pressure differential of 31 Pa (⅛ in. WG). The applicant revised DCD, Tier 2, Section 6.4.3, with the results of the analysis. Based on a review of the RAI response and proposed DCD changes, the staff found the RAI response acceptable because the applicant submitted an analysis, based on the planned leaktight design features that ensured the feasibility of maintaining the tested differential pressure with the design makeup airflow rate in accordance with Standard Review Plan Section 6.4, Revision 3, acceptance criteria item 3. The RAI was therefore resolved.

The staff noted that the applicant did not model the pressure changes in the CRHA caused by temperature differences between the supply and exhaust air during the passive cooling period with the EFU operating.

In RAI 9.4-30 the staff requested that the applicant clarify if changes in the outside environmental conditions such as air temperature and pressure over the accident period could significantly change the volumetric addition of air to the control room such that manual adjustments to the variable orifice device would be required in order to maintain acceptable CRHA positive pressure and makeup airflow rate.

In response to RAI 9.4-30, the applicant modified the CRHA design to add variable orifice relief to maintain a greater-than-31 Pa (⅛ in. WG) positive pressure at the minimum flow rate. DCD, Tier 1, ITAAC Table 2.16.2-6, Design Commitment 5a, states that the EFUs maintain the CRHA at the minimum positive pressure with respect to the surrounding areas at the required air addition flow rate. This commitment is verified by an ITAAC test.

The variable orifice relief device is manually adjusted, as needed, to maintain CRHA positive pressure. In response to follow-up questions from the staff under this RAI, the applicant demonstrated that required manual adjustments of the device during a postulated accident would be unlikely and would not likely be a burden on operators. The staff finds the ITAAC acceptable because it provides assurance that the design, when built, will supply the minimum required positive pressure at the minimum required air addition flow rate. The staff reviewed the ESBWR Technical Specification TS 5.5.12, Control Room Habitability Program, paragraph d. This paragraph requires periodic measurement at designated locations, of the CRHA pressure relative to all external areas located at the CRHA boundary while the EFU filter is supplying at least minimum airflow rate. The staff finds that this surveillance requirement provides assurance that the controlled leakage path setting will be monitored and adjusted as required during the life of the plant. As discussed in DCD, Tier 2, Section 6.4.8, the existence of alarms for low EFU airflow and low CRHA differential pressure assure that these parameters are

continually monitored when an EFU is in operation, and that operators would be alerted in a timely manner if any corrective action is required. Based on review of the applicants discussion of impacts to changes in outside air temperature on EFU volumetric flow the staff agrees that since the percent change of air specific volume is low for a relatively large outside air temperature swing, any changes in outside air temperatures would not affect the performance of the CRHA positive pressure and EFU flow rate parameters during an accident, Therefore frequent adjustment of the CRHA variable orifice relief device is not anticipated. RAI 9.4-30 was therefore, resolved.

The ESBWR provides a variable orifice relief device to maintain a constant 31 Pa (⅛ in. WG) positive pressure in the CRHA, while ensuring that the amount of air exhausted from the CRHA is equal to the amount supplied. This device location is optimized to ensure proper scavenging of air from the control room.

The CRHA has the fresh air supplied at a high elevation and the exhaust removed below the floor, so that the supply and exhaust are not in close proximity to each other. The CRHA has a differential pressure indication for monitoring under normal and emergency operation. A low-airflow alarm is provided. Pressure in the CRHA is monitored, and an alarm is actuated if the pressure falls below the setpoint level. In RAI 6.4-23, the staff asked the applicant to revise the DCD to clarify the function, seismic, and safety classification of the variable orifice relief device. In response to the RAI, the applicant revised DCD, Tier 1, Table 2.16.2-3, DCD, Tier 2, Sections 6.4.2, 6.4.4, 6.4.7, 9.4.11, and 9.4.1.2. The staff finds the proposed DCD changes acceptable because the applicant clarified that the device meets SRP Section 9.4.1 design and inservice testing guidance. The applicant's revision to DCD, Tier 1, Table 2.16.2-3 lists the CRHA Variable Orifice Relief Device as safety-related, seismic Category 1. These changes clarified the function, seismic and safety classification of the device. The staff confirmed that these changes were incorporated in DCD, Tier 2, Revision 7. Based on the applicant's response, RAI 6.4-23 was resolved.

In regard to GDC 19 as it applies to smoke purge, the applicant provided a system for rapid removal of smoke in the event of a fire inside the CRHA. The system is isolatable with safety-related and tornado-protected dampers. The assumption of a CRHA fire post-accident is not a requirement of the design.

To summarize the GDC 19 review, the ESBWR has incorporated design features that protect operators and equipment from radiation, temperature, humidity, and other environmental conditions, and these features are adequate, considering the low probability of adverse events and the availability of defense-in-depth measures.

The regulation in 10 CFR 50.34(f)(2)(xxviii) requires the applicant to evaluate potential pathways for radioactivity and radiation that may lead to control room habitability problems under accident conditions, resulting in an accident source-term release, and to make the necessary design provisions to preclude such problems (TMI Action Plan Item III.D.3.4). The design includes adequate protection from radiation, in compliance with GDC 19. The staff determined that this was acceptable.

TMI Action Plan Item III.D.3.4 requires that control room operators be adequately protected against the effects of the accidental release of toxic and radioactive gases and that the nuclear power plant be safely operated or shut down under DBA conditions (GDC 19 in Appendix A to 10 CFR Part 50).

RAI 6.4-17 was being tracked as an open item in the SER with open items. RAI 6.4-17 asked the applicant to state the following in the DCD, regarding testing the CRHA envelope for integrity: (1) that the test requirements and the testing frequency will be consistent with the guidance of RG 1.197, which establishes an inservice test program, and (2) that the test requirements appear in Chapter 16, "Technical Specifications," Section 5.5.12, "Control Room Habitability Area (CRHA) Boundary Program." In MFN 07-687, the applicant stated, in response to RAI 6.4-17, that DCD Sections 6.4.7 and 6.4.9 were revised to include this information. The staff found the RAI response and associated DCD changes acceptable because they clarify that testing to demonstrate the integrity of the Control Room Habitability Area envelope is performed in accordance with RG 1.197 and ASTM E741. This is in compliance with SRP Section 6.4, SRP Acceptance Criteria Item 1.E as it applies to CRHA envelope integrity testing requirements and testing frequency, and the RAI was therefore closed.

The staff concludes that GEH has met the TMI Action Plan Item III.D.3.4 requirements by adding COL Information Item 6.4-2-A in Revision 4 of the DCD, Tier 2, Chapter 6. This requires the COL applicant to identify potential site-specific toxic or hazardous materials that may affect control room habitability to meet the requirements of TMI Action Plan Item III.D.3.4. If high radioactivity is detected in the CRHA outside air supply duct, the CRHA normal air supply is automatically isolated, and the GDC 19 habitability requirements are met by an EFU. The EFUs provide emergency ventilation and pressurization for the CRHA. The staff determined that this was acceptable.

Task Action Plan Item B-36 required the development of design, testing, and maintenance criteria for atmospheric cleanup system air filtration and adsorption units for ESF systems and for normal ventilation systems. GEH meets the requirements of Item B-36 by complying with RG 1.52, for the safety-related EFU system, and RG 1.140, for the non-safety-related filter systems. RAI 6.4-10 was being tracked as an open item in the SER with open items. RAI 6.4-10 requested that the applicant include a reference in the DCD to ASME AG-1, including all addenda. The applicant included this reference in DCD, Tier 2, Table 1.9-22. The staff determined that this was acceptable, and RAI 6.4-10 was closed.

Task Action Plan Item B-66 addresses the magnitude of the control room air infiltration rate. RG 1.197 provides methods acceptable to the staff for determining air infiltration and is referenced in TS Section 5.5.12 on the control room habitability boundary. The staff therefore considers the concern of Item B-66 to be satisfied.

Generic Safety Issue 83, "Control Room Habitability" (Revision 3), addresses deficiencies in the maintenance and testing of ESFs designed to maintain control room habitability (e.g., inadvertent degradation of control room leaktightness, shortage of personnel knowledgeable about nuclear HVAC systems). It recommends increased training of NRC and licensee personnel in inspection and testing of control room habitability systems.

GL 2003-01 reemphasized this concern. GEH developed COL Information Item 6.4-1-A ("CRHA Procedures and Training"), which requires the ESBWR COL applicant to verify procedures and training for control room habitability. GEH also added the CRHA Boundary Program (Section 5.5.12) in the ESBWR DCD, Tier 2, Chapter 16 ("Technical Specifications") to establish the CRHA boundary test method and frequency. The staff determined that Generic Safety Issue 83 has been adequately addressed.

RAIs 6.4-5, 6.4-6, and 6.4-18 were being tracked as open items in the SER with open items. The RAIs requested editorial changes to the DCD to correct discrepancies. The staff has reviewed the responses to these RAIs, including the proposed DCD changes, and has found them acceptable. The staff confirmed that the applicant had incorporated these changes in DCD, Tier 2, Revision 7 therefore RAIs 6.4-5, 6.4-6, and 6.4-18 were resolved.

6.4.4 Conclusions

The staff finds that the ESBWR control room habitability systems meet the requirements of SRP Section 6.4 and associated guidance and regulations. There is reasonable assurance that passive cooling features will be sufficient to limit the control room environment temperatures under the summer and winter design conditions to a range that is acceptable for equipment and operator performance.

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 52.47(a)(2) and GDC 19.

The staff's bases its acceptance criteria for the atmosphere cleanup systems on the relevant requirements of the following regulations:

- GDC 19, "Control Room," as it relates to systems being designed for habitability of the control room during and following postulated DBAs

- GDC 41, "Containment Atmosphere Cleanup," as it relates to the design of systems to be used for containment atmosphere cleanup during and following postulated DBAs

- GDC 42, "Inspection of Containment Atmosphere Cleanup Systems," and GDC 43, "Testing of Containment Atmosphere Cleanup Systems," 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"

SRP Section 6.5.2, "Containment Spray as a Fission Product Cleanup System," and SRP Section 6.5.4, "Ice Condenser as a Fission Product Cleanup System," are not used because the

ESBWR design does not include either a safety-related containment spray system or an ice condenser.

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 that 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 0.35 wt% per day. The applicant stated that 0.01 wt% per day of a 0.35 wt% 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 RBVS isolation dampers will be tested as described in 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 141.6 L/s (300 cfm). The applicant stated that the RB envelope is not intended to provide a leaktight barrier against a radiological fission product release. The applicant further stated that the RB will be periodically tested 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 did not take credit for suppression pool scrubbing in the bounding accident scenario considered, as it is a low pressure event. The flow through the safety relief valves (SRVs) is negligible for low pressure events.

Control Room Emergency Filter Unit

In DCD, Tier 2, Revision 3, the applicant described the control room EFU (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 CB HVAC system, is a safety-related system and is located in the CB. The CB is designed to seismic Category 1 criteria. The CREFU replaces the passive control room emergency air breathing system provided in previous revisions to the DCD.

The CREFU consists of two redundant trains, each with a prefilter, HEPA filter, 4-in.-deep charcoal adsorber, and postfilter to remove fission products and to pressurize the control room to prevent any inleakage 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 MCR air supply duct or can be activated manually from the MCR.

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.

6.5.3 Staff 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 applicant's 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 removal of decay heat by 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

Based on the staff's review of the information provided by GEH, the staff concludes that the passive atmosphere cleanup systems provided in the ESBWR design, which are intended 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, meet the radiological consequence evaluation factors specified in 10 CFR 52.47(a)(2) and GDC 19.

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 SRP Section 6.6, "Inservice Inspection of Class 2 and 3 Components." 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, and 46 are specified in 10 CFR 50.55a and detailed in Section XI of the ASME Code as described below.

- 10 CFR 50.55a contains preservice and periodic inspection and testing requirements of the ASME Code for Class 2 and 3 systems and components.

- GDC 36, "Inspection of Emergency Core Cooling System," requires that the design of the ECCS permit appropriate periodic inspection of important safety components, such as spray rings, in the RPV.

- GDC 37, "Testing of Emergency Core Cooling System," requires that the design of the ECCS permit appropriate testing to ensure structural integrity, leaktightness, and the operability of the system.

- GDC 39, "Inspection of Containment Heat Removal System," requires that the design of the containment heat removal system permit inspection of important components, such as the torus and spray nozzles, to ensure the integrity and capability of the system.

- GDC 40, "Testing of Containment Heat Removal System," requires that the design of the containment heat removal system permit appropriate periodic pressure and functional testing to ensure 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.

- GDC 42, "Inspection of Containment Atmosphere Cleanup Systems," requires that the design of the containment atmospheric cleanup system permit appropriate periodic inspection of components such as filter frames and ducts to ensure integrity and capability of the system.

- GDC 43, “Testing of Containment Atmosphere Cleanup Systems,” requires that the design of the containment atmospheric cleanup system permit appropriate periodic pressure and functional testing to ensure the structural integrity of components and the operability and performance of active components of the system, such as fans, filters, and dampers.

- GDC 45, “Inspection of Cooling Water System,” requires that the design of the cooling water system permit appropriate periodic inspection of important components, such as heat exchangers, to ensure the integrity and capability of the system.

- GDC 46, “Testing of Cooling Water System,” requires that the design of the cooling water system permit appropriate pressure and functional testing to ensure 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.

ASME Class 2 and 3 components rely upon these design provisions to allow performance of an ISI. Compliance with these GDC ensures that the design of the safety systems will allow access to important components, so that periodic inspections can 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 the acceptance criteria in SRP Section 6.6, Revision 1, by conforming to the ISI requirements of the aforementioned GDC and 10 CFR 50.55a for the areas of review described in Section I of the SRP.

The applicant stated that all items within the Class 2 and 3 boundaries provide access for the examinations required by ASME Code, Section XI, Subarticles IWC-2500 and IWD-2500.

The physical arrangement of piping, pumps, and valves provides personnel access to each weld location for the performance of ultrasonic and surface (magnetic particle or liquid penetrant) examinations and sufficient access to supports for the performance of visual (VT-3) examinations. Working platforms in some areas 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. The design of weld locations permits ultrasonic examination from at least one side and access from both sides, where component geometry permits.

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 CIVs shall be 100-percent volumetrically examined at each inspection interval.

Piping systems that are ASME Code, Section III, Code Class 1, 2, and 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 or

corrosion shall be subject to a program of nondestructive examination (NDE) to verify a system's structural integrity. The examination schedule and methods shall be determined in accordance with the Electric Power Research Institute (EPRI) guidelines in Nuclear Safety Analysis Center (NSAC)-202L-R2, "Recommendations for an Effective Flow-Accelerated Corrosion Program," issued April 1999, 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 licensee will be responsible for developing the site-specific preservice inspection (PSI) and ISI program plans, which will be based on the ASME Code, Section XI, edition and addenda approved in 10 CFR 50.55a(b), 12 months before initial fuel load. The COL applicant is responsible for providing a full description of the PSI/ISI programs and augmented inspection programs for Class 2 and 3 components and piping by supplementing, as necessary, the information in DCD, Tier 2, Section 6.6. The COL applicant will provide milestones for program implementation (COL Information Item 6.6-1-A). The COL applicant is also responsible for providing a full description of PSI/ISI, and design activities for components that are not included in the referenced design, to preserve accessibility to piping systems to enable NDE of ASME Code Class 2 austenitic and dissimilar metal welds during ISI (COL Information Item 6.6-2-A).

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 provide access for the examinations required by ASME Code, Section XI, Subarticles IWC-2500 and IWD-2500.

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 inspections required by ASME Code, Section XI, and 10 CFR 50.55a. The staff developed RAI 5.2-62, which supersedes the aforementioned RAIs, regarding the accessibility and inspectability 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 an ISI of welds in ASME Code Class 1 and 2 austenitic and dissimilar metal welds, and (2) add COL information items to Sections 5.2.4 and 6.6 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 an 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 examinations 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 tell the staff 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 PSI 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 examinations during the construction phases of the component. The staff asked that the COL information items requested above reflect these considerations. The staff identified this issue in RAI 5.2-62.

The applicant responded by letter dated April 11, 2008, and indicated that it would modify DCD, Tier 2, Sections 5.2.4 and 6.6, to include a description of its design process to ensure that the accessibility of austenitic and dissimilar metal welds to perform UT or radiographic testing (RT). The staff reviewed the applicant's RAI response and modifications in DCD, Tier 2, Revision 5, Sections 5.2.4 and 6.6, and found them to be unacceptable because they did not address the design's accessibility, taking into account operational and radiological concerns. The staff issued RAI 5.2-62, Supplement 1, and requested that the applicant address this issue.

The applicant responded by letter dated October 6, 2008, and stated that it would modify DCD, Tier 2, Sections 5.2.4 and 6.6, to address the staff's concerns. Section 5.2.4 of this report addresses the accessibility of Class 1 components. The applicant proposed the modifications below to DCD, Tier 2, Section 6.6.2, which includes Tier 2* information, in lieu of the Tier 1 changes requested by the staff. Given that the COL applicant cannot depart from Tier 2* information without NRC approval, the staff considers that making the modifications in Tier 2* is acceptable.

> [The ESBWR design includes specific access requirements, in accordance with 10 CFR 50.55a(g)(3), to support preferred UT or optional RT examinations. The design of each component and system takes into account the NDE method, UT or RT, that will be used to fulfill preservice inspection and in-service inspection examination and will take into full consideration the operational and radiological concerns associated with the method selected to ensure that the performance of the required examination will be practical during commercial operation of the

plant. Additionally, the design procedural requirements for the 3D layout of the plant include acceptance criteria regarding access for inspection equipment and personnel]*. However, with respect to any design activities for components that are not included in the referenced ESBWR certified design, it is the responsibility of the COL Applicant to preserve accessibility to piping systems to enable NDE of ASME Code Class 2 austenitic and DM welds during in-service inspection (COL Information Item 6.6-2-A)

The staff finds that the proposed modifications to DCD, Tier 2, Section 6.6.2, discussed above, ensure that austenitic and dissimilar metal welds will be accessible to perform ASME Code-required inspections, taking into account operational and radiological concerns that could affect the practicality of the inspection method chosen for ISIs and PSIs. The staff subsequently confirmed that the applicant had made the above modifications to DCD, Tier 2, Revision 6, Section 6.6. RAI 5.2-62 was being tracked as an open item in the SER with open items. Based on the applicant's response, RAI 5.2-62 was resolved.

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 agree 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 during the 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 Categories C-H and D-A.

DCD, Tier 2, Section 5.2.4, indicates that the design regarding PSI 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.

It appeared that the applicant has made references to the 1989 edition of ASME Code, Section XI, regarding examination Category D-A. The staff noted that, in other instances, the applicant also 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 an updated 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 the 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.

The applicant responded to RAI 6.6-8 by letter dated December 21, 2007, and stated that it would modify the incorrect examination categories listed in DCD, Tier 2, Sections 6.6 and 5.2.4. The applicant also stated that a complete review of DCD, Tier 2, Sections 5.2.4 and 6.6, was

performed and appropriate changes would be made to DCD Section 5.2.4 and 6.6. The staff confirmed that the appropriate modifications, as discussed in the applicant's December 21, 2007 letter, were made in DCD, Tier 2, Revision 5. RAI 6.6-8 was being tracked as an open item in the SER with open items. Based on the applicant's response, RAI 6.6-8 was resolved.

The 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 the 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 IC heat exchangers are ASME Code Class 2 and the PCCS heat exchangers are ASME Code Class MC.

In RAI 5.4-56, the staff requested that the applicant confirm that the method or 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 RAI 5.4-58, the staff requested that the applicant discuss the results of inspections performed on Alloy 600 components in operating BWRs. 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 response to RAI 5.4-56, the applicant indicated that, because of the size of the IC tubes (nominal pipe size 2), they are exempt from volumetric and surface inservice examinations by ASME Code, Section XI, Subarticle IWC-1220, which exempts nominal pipe sizes 4 and smaller. The applicant indicated that the ICs are subject to leakage examination (VT-2) 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 prevent fully simulating the range of variables that may exist in the field (and that may be 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 the lack of inspection requirements. 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. The applicant responded by letter dated January 16, 2008, and stated that SCC is not plausible because of the IC pool temperature, control of water chemistry, use of modified Alloy 600, and lack of crevices in the IC heat exchanger assembly. The applicant also stated that the IC design takes general corrosion into account.

The staff identified two degradation mechanisms that could be of concern in the IC. They are SCC and general corrosion. With regard to SCC, the use of modified Alloy 600 greatly reduces

the risk. Pressure boundary welds, such as IC header and tube-to-header welds, are full penetration welds that do not contain crevices that could be initiation sites for SCC to occur. The low normal operating temperature of the IC pool and water chemistry controls, coupled with the use of niobium-modified Alloy 600, make the possibility of SCC unlikely. In the event that leakage were to occur because of a through wall flaw, it would be detected by radiation leakage monitoring equipment. General corrosion of modified Alloy 600 is considered negligible in the IC environment. In addition, the applicant stated that the IC design takes into account general corrosion and its effects. Based on the resistance of modified Alloy 600 to SCC in the BWR environment, the lack of crevices to act as SCC initiation sites and the low operating temperature of the IC pool, the staff does not consider augmented inservice examinations necessary, beyond current ASME Code requirements. RAIs 5.4-56 and 5.4-58 were being tracked as open items in the SER with open items. Based on the applicant's response, RAIs 5.4-56 and 5.4-58 were resolved.

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. The applicant responded by letter dated January 3, 2008, and stated that the PCCS heat exchangers are fabricated from 304L stainless steel, immersed in deionized water at ambient pressure and temperature, and only used post-LOCA. In addition, the post-LOCA environment is flowing steam at a maximum of 171 degrees C (340 degrees F) for 72 hours. The applicant stated that, under these conditions, corrosion of stainless steel is extremely limited, as are other forms of material degradation. Cracking of stainless steel in the PCCS heat exchangers is not expected to occur as it has in the past in stainless steel IC tubes, because the PCCS uses low-carbon stainless steels and is submerged in chemically controlled deionized water at ambient temperature and pressure for essentially its entire life. In addition, the applicant indicated that, because of the operating conditions and environment, no augmented inspections are necessary. The staff agrees that general corrosion of the PCCS heat exchangers fabricated from low-carbon stainless steel in the expected environment will be negligible. The potential for SCC is all but eliminated through the use of low-carbon stainless steel, the absence of crevices in the tube-to-header full penetration welds, extremely low normal operation temperature and pressure, a chemically controlled deionized water environment, and limited use for these components under post-LOCA conditions. Based on the corrosion resistance of 304L in the PCCS pool environment, which includes chemically controlled deionizer water, low operating temperature and low operating pressure, the staff finds that 304L will not be susceptible to stress corrosion cracking and therefore, the staff does not consider augmented inservice examinations necessary, beyond current ASME Code requirements. RAI 5.4-57 was being tracked as an open item in the SER with open items. Based on the applicant's response, RAI 5.4-57 was resolved.

In DCD Revision 7, the applicant modified DCD Table 6.1-1 to change the material used for the PCCS heat exchanger tubes from 304L to XM-19. XM-19, also known as NITRONIC 50, is a nitrogen strengthened austenitic stainless steel which has a higher yield and tensile strength than 304L stainless steel. In addition, XM-19 has superior corrosion resistance to 304L in the PCCS operating environment and is acceptable for use in accordance with ASME Section III materials specifications requirements. Therefore, degradation of XM-19 used in the PCCS heat exchanger tubes is expected to be negligible for the design life of the plant. The staff therefore finds the applicant's use of XM-19 PCCS heat exchanger tubes acceptable and no augmented inservice inspections are required.

6.6.3.4 *Examination Intervals*

The required examinations and pressure tests must be completed during each 10-year interval of service, hereafter designated as the inspection interval. In addition, the scheduling of the program must comply with the provisions of ASME Code, Section XI, Article IWA-2000, concerning inspection intervals.

DCD, Tier 2, Section 6.6.4, discusses inspection intervals for ASME Code Class 2 and 3 systems. Subarticles IWA-2400, IWC-2400, and IWD-2400 of ASME Code, Section XI, define inspection intervals. The inspection intervals specified for the ESBWR components are consistent with the definitions in Section XI of the ASME Code and, therefore, are acceptable.

6.6.3.5 *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, Section II.5, and is therefore acceptable.

6.6.3.6 *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 applicant responded by letter dated March 26, 2008, and indicated that it would modify DCD, Tier 2, Section 6.6.6, to state that the requirements of IWA-5000 and IWC-5000 will be met for Class 2 components, and the requirements of IWA-5000 and IWD-5000 will be met for Class 3 components. The applicant's response addressed requirements for Class 1 components, and Section 5.2.4 of this report discusses them. The staff reviewed Revision 5 to the DCD and verified that the appropriate modifications were made to Section 6.6.6. RAI 5.2-65 was being tracked as an open item in the SER with open items. Based on the applicant's response, RAI 5.2-65 was resolved.

6.6.3.7 *Augmented Inservice Inspection To Protect against Postulated Piping Failure*

The augmented ISI program for high-energy fluid systems piping between CIVs is acceptable, if ISI examinations completed during each inspection interval provide a 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 Section 3.6.2) between CIVs is subject to additional inspection requirements. Circumferential welds shall be 100-percent volumetrically examined at each inspection interval. The piping in these areas is seamless, thereby eliminating longitudinal welds. The applicant's augmented ISI program to protect against postulated pipe failure is consistent with SRP Section 6.6 and is, therefore, acceptable.

BL 80-08 “Examination of Containment Liner Penetration Welds,” identifies NRC concerns related to UT of primary piping containment penetration fluid-head (integral fitting) to outer sleeve welds, using backing bars, which form part of the containment pressure boundary. In RAI 20.0-5, the staff requested that the applicant address the NRC concerns identified in BL 80-08. The applicant responded by letter dated November 19, 2007, and stated that backing bars are not used in flued-head containment penetration assemblies or other penetration sleeves and process piping. The applicant also stated that it would modify DCD, Tier 2, Table 3.8-5, accordingly. The staff reviewed DCD, Tier 2, Revision 5, and confirmed that the applicant had made the appropriate modifications. Based on the above, the staff has determined that the ESBWR design appropriately addresses the NRC concerns identified in BL 80-08.

6.6.3.8 *Augmented Erosion/Corrosion Inspection Program*

BL 87-01, “Thinning of Pipe Walls in Nuclear Power Plants,” dated July 9, 1987, requested that operating reactor licensees submit information concerning their programs for monitoring the thickness of pipe walls in high-energy, single-phase and two-phase carbon steel piping systems. The staff subsequently issued GL 89-08, requiring operating reactor licensees to verify implementation of formalized procedures or administrative controls to ensure continued long-term implementation of the erosion and corrosion monitoring program for piping and components. The ESBWR design requires COL applicants to develop appropriate long-term monitoring for potential wall thinning of high-energy piping by erosion and corrosion, as described in GL 89-08. In addition, COL Action Item 6.6-1-A requires COL applicants to provide a full description of augmented inspection programs and milestones for program implementation. The staff therefore finds that BL 87-01 and GL 89-08 were resolved for the ESBWR design. GL 89-08 is discussed further below.

As described in GL 89-08, an appropriate long-term monitoring program for potential wall thinning of high-energy piping by erosion and corrosion must be implemented. The applicant has indicated that all piping systems that are ASME Code, Section III, Code Class 1, 2, and 3, as well as non-safety-related piping, and components described in GL 89-08 that are determined to be susceptible to erosion or corrosion shall be subject to NDE to verify system integrity. The applicant further stated that the 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 or 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 PSI/ISI and augmented inspection programs and to provide milestones for their implementation. The applicant appropriately addressed this issue in its letter dated April 11, 2008, and the staff’s detailed analysis of this response is found in Section 6.6.3.9 of this report.

6.6.3.9 *Combined License 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 information item to describe the PSI/ISI and augmented inspection programs and to provide

milestones for their implementation. The staff was concerned that the GEH reference to the COL holder does not make it clear that the COL applicant must provide a description of its PSI/ISI and augmented inspection programs with commitments for scheduled implementation of those programs identified in the COL application. It is understood that the COL licensee will fully develop and implement the actual programs. However, the COL applicant must fully describe the PSI/ISI and augmented inspection programs to allow the staff to make a reasonable assurance finding of acceptability.

The applicant responded to RAI 5.2-64, by letter dated April 11, 2008, and indicated that it would modify DCD, Tier 2, Section 6.6.11, to address the staff's concerns. The staff reviewed DCD, Tier 2, Revision 5, and confirmed that the appropriate modifications were made to Section 6.6.11. COL Action Item 6.6-1-A now states that the COL applicant is responsible for providing a full description of the PSI/ISI and augmented inspection programs for Class 2 and 3 components and piping, by supplementing, as necessary, the information in Section 6.6. The COL applicant will also provide milestones for program implementation (Section 6.6). The staff was tracking RAI 5.2-64 as an open item in the SER with open items. Based on the applicant's response, RAI 5.2-64 was resolved.

COL Information Item 6.6-2-A states that the COL applicant is responsible for developing a plan and providing a full description of its use during construction, PSI, ISI, and for design activities for components that are not included in the referenced certified design, to preserve accessibility to piping systems to enable NDE of ASME Code Class 2 austenitic and dissimilar metal welds during ISIs (Section 6.6).

6.6.4 Conclusions

The staff concludes that the ESBWR program for Code Class 2 and 3 components is acceptable and meets the inspection and pressure-testing requirements of 10 CFR 50.55a, as detailed in ASME Code, Section IX, and therefore satisfies the applicable requirements of GDC 36, 37, 39, 40, 42, 43, 45, and 46.

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