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<u>**No figures were included in this chapter.**</u>

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6 ENGINEERED SAFETY FEATURES

6.1 Engineered Safety Features Materials

6.1.1 Metallic Materials

6.1.1.1 *Introduction*

Engineered Safety Features (ESFs) must be compatible with the fluids to which they may be exposed during normal operation, maintenance, testing, and postulated accident conditions. In order to maintain the integrity of the reactor coolant pressure boundary (RCPB), ESF components that are part of, or interface with, the RCPB must be fabricated of materials that provide a low probability of significant degradation or rapidly propagating fracture. In addition to using appropriate fabrication materials, processes for welding, non-destructive examination, and cleaning of ESF systems must be controlled to assure initial quality and prevent deterioration.

6.1.1.2 *Summary of Application*

FSAR Tier 1: The Tier 1 information associated with this section for various ESF systems is found in final safety analysis report (FSAR) Tier 1, Section 2.2.2, “In-Containment Refueling Water Storage Tank (IRWST)”;Section 2.2.3, “Safety Injection System and Residual Heat Removal System (SIS/RHR)”; Section 2.2.7, “Extra Borating System (EBS)”; Section 2.1.1.1, “Reactor Building” (steel liner); Section 3.5, “Containment Isolation,” regarding Reactor Containment Building (RCB) penetrations; and Section 2.2.4, “Emergency Feedwater System (EFWS).”

The aforementioned Tier 1 sections address materials by committing to comply with the applicable American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code, Section III, articles for design, welding and testing, which include requirements for materials.

FSAR Tier 2: The applicant has provided a Tier 2 topic description in FSAR Tier 2, Section 6.1.1, “Metallic Materials,” summarized here, in part, as follows. Additional information for the containment steel liner and penetrations is provided in FSAR Tier 2, Section 3.8, “Design of Category I Structures.”

FSAR Tier 2, Section 6.1.1 discusses materials and the fabrication methods for ESF components including cleaning of components, compatibility of the materials with the specific fluids to which they are subjected, and thermal insulation used to insulate ESF components.

Materials used in the principal pressure-retaining ESF components, including the SIS/RHRS, IRWST, and EBS are given in FSAR Tier 2, Table 6.1-1, “Pressure-Retaining Material Specifications for Engineered Safety Features.”

Material selection criteria are described including consideration of exposure to corrosive fluids and service conditions as follows:

- Materials exposed to core coolant or borated water are corrosion resistant austenitic stainless steel materials.

- Materials that are not exposed to core coolant or borated water may be ferritic materials.

- Pressure retaining ESF materials are selected from material specifications permitted by the ASME Code, Section III.

Water chemistry is controlled to minimize negative impacts of water chemistry on materials integrity. Controls include routine analysis and passive pH control during accidents.

Procedures and specifications address handling, protection, storage, and cleaning of austenitic stainless steel materials to prevent stress corrosion cracking.

The use of non-metallic insulation is controlled. Non-metallic insulation is designed with low leachable chloride and fluoride concentrations. Limits are placed on sodium and silicate concentrations in order to prevent stress corrosion cracking.

ITAAC: The inspections, tests, analyses, and acceptance criteria associated with FSAR Tier 2, Section 6.1.1 are given in Tier 1 for various ESF systems including Section 2.2.2 for the IRWST, Section 2.2.3 for the SIS/RHRS, Section 2.2.7 for the EBS, Section 2.1.1.1 for the RCB steel liner, Section 3.5 for RCB penetrations, and Section 2.2.4 for the EFWS.

The aforementioned ITAAC sections address materials by verifying compliance with applicable ASME Code, Section III, articles for design and welding, which include requirements for materials

Technical Specifications: There are no Technical Specifications (TS) applicable to ESF metallic materials; however, related TS information can be found in FSAR Tier 2, Chapter 16, “Technical Specifications,” Section 3.6.8, “pH Adjustment.”

6.1.1.3 *Regulatory Basis*

The relevant requirements of Nuclear Regulartory Commission (NRC) regulations for this area of review, and the associated acceptance criteria, are given in NUREG-0800, Section 6.1.1, and are summarized below. Review interfaces with other SRP sections also can be found in NUREG-0800, Section 6.1.1.

1. General Design Criterion (GDC) 1, “Quality Standards and Records,” and Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.55a, “Codes and Standards,” as they relate to quality standards for design, fabrication, erection, and testing of ESF components and the identification of applicable codes and standards.

2. GDC 4, “Environmental and Dynamic Effects Design Bases,” as it relates to compatibility of ESF components with environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant accidents (LOCAs).

3. GDC 14, “Reactor Coolant Pressure Boundary,” as it relates to design, fabrication, erection, and testing of the RCPB so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture.

4. GDC 31, “Fracture Prevention of Reactor Coolant Pressure Boundary,” as it relates to designing the RCPB such that the boundary behaves in a non-brittle manner and there

is an extremely low probability of rapidly propagating fracture and of gross rupture of the RCPB.

5. GDC 35, “Emergency Core Cooling,” as it relates to providing adequate core cooling following a LOCA at such a rate that fuel and clad damage that could inhibit core cooling is prevented and that the clad metal-water reaction is limited to negligible amounts.

6. GDC 41, “Containment Atmosphere Cleanup,” as it relates to control of the concentration of hydrogen in the containment atmosphere following postulated accidents to assure that containment integrity is maintained.

7. Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, Appendix B, “Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants,” Criteria IX, “Control of Special Processes,” and XIII, “Handling, Storage and Shipping,” as they relate to establishing and controlling work and inspection instructions that prescribe the special cleaning processes and measures necessary to prevent material and equipment damage or deterioration in accordance with applicable codes, standards, specifications, criteria, and other special requirements.

Acceptance criteria adequate to meet the above requirements include:

1. Regulatory Guide (RG) 1.44, “Control of the Use of Sensitized Stainless Steel,” which describes acceptable criteria for preventing intergranular corrosion of stainless steel components of the ESF.

2. RG 1.31, “Control of Ferrite Content in Stainless Steel Weld Metal,” which describes acceptable criteria for control of ferrite in austenitic stainless steel welds.

3. RG 1.36, “Nonmetallic Thermal Insulation for Austenitic Stainless Steel,” which describes controls for the composition of non-metallic thermal insulation.

4. RG 1.37, “Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants,” which describes quality assurance requirements for cleaning of fluid systems and associated components of water-cooled nuclear power plants.

6.1.1.4 *Technical Evaluation*

6.1.1.4.1 Materials and Fabrication

To meet the requirements of GDC 1 and 10 CFR 50.55a to assure that plant structures, systems, and components (SSCs) important to safety are designed, fabricated, erected and tested to quality standards commensurate with the importance of the safety function to be performed, codes and standards should be identified and records maintained. The materials specified for use in these systems must be selected in accordance with the applicable portions of Section III, Division 1 or Division 2 of the ASME B&PV Code or an ASME code case listed in the most recent version of RG 1.84, “Design, Fabrication, and Materials Code Case Acceptability, ASME Section III,” incorporated by reference into 10 CFR 50.55a. Section III references applicable portions of ASME Code Section II, Parts A, B, C, and D.

FSAR Tier 2, Table 6.1-1 lists materials used in ESF systems. Table 6.1-1 shows that austenitic stainless steel is used for all ESF components with the exception of the reactor building liner and penetrations, portions of the annulus ventilation system and the shell secondary side of the low head safety injection heat exchangers.

The staff reviewed the material specifications given in FSAR Tier 2, Table 6.1-1, "Pressure-Retaining Material Specifications for Engineered Safety Features," and verified that the materials given meet ASME Code, Section III requirements and are, therefore, acceptable for use in the U.S. EPR design.

In accordance with SRP Section 10.4.9, the compatibility of Emergency Feedwater System (EFWS) materials is reviewed by the staff under SRP Section 6.1.1. EFWS materials specifications are listed in FSAR Tier 2, Table 10.4.9-2, "Emergency Feedwater Material Specifications." The staff reviewed the material specifications provided in FSAR Tier 2, Table 10.4.9-2, and verified that all materials specifications, including those for weld filler materials, meet the requirements of ASME Code Section III and are therefore acceptable.

Based on the above, the staff finds that the ESF materials comply with ASME Code, Section III and that the ESF materials meet the requirements of GDC 1 and 10 CFR 50.55a.

Austenitic Stainless Steel

The U.S. EPR 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 quality assurance (QA) requirements of 10 CFR Part 50, Appendix B. Designs may meet these requirements by following the guidance of RG 1.31, RG 1.37, and RG 1.44 and by providing controls over the use of cold-worked austenitic stainless steels.

RG 1.31 contains staff guidance pertaining to the delta ferrite content in austenitic stainless steel welds to minimize the presence of microfissures, which could have an adverse effect on the integrity of components. FSAR Tier 2, Section 6.1.1 states that "the delta ferrite content of the weld filler material is controlled as required by RG 1.31."

RG 1.44 describes recommended controls applied to the application and processing of austenitic stainless steels in order to minimize component susceptibility to stress corrosion cracking. For stainless steel components in the ESF systems, FSAR Tier 2, Section 6.1.1.1, "Materials Selection and Fabrication," states that unstabilized austenitic stainless steels used in ESF components are provided in the solution annealed and rapidly cooled condition to optimize the resistance to intergranular corrosion in accordance with RG 1.44. The staff notes that austenitic stainless steels used in ESF systems have a maximum carbon content of 0.03 percent to minimize the susceptibility of ESF components to stress corrosion cracking as recommended by RG 1.44. Also, FSAR Tier 2, Table 1.9-2, "U.S. EPR Conformance with Regulatory Guides," indicates that the applicant does not take any exception to the guidance provided in RG 1.44 for the fabrication of ESF components.

FSAR Tier 2, Section 6.1.1.1 states that a COL applicant that references the U.S. EPR design certification will review the fabrication and welding procedures and other QA methods of ESF component vendors to verify conformance to RG 1.44 and RG 1.31. FSAR Tier 2, Table 1.8-2, "U.S. EPR Combined License Information Items," identifies this as COL Information Item 6.1-1. The inclusion of this COL information item will ensure that ESF component vendors adhere to the staff recommendations described in RG 1.31 and RG 1.44. The staff finds this acceptable.

FSAR Tier 2, Section 6.1.1.1 states that abrasive work on austenitic stainless steel is controlled to minimize the cold working of surfaces and the introduction of contaminants that promote stress corrosion cracking as described in RG 1.37. The staff finds this acceptable, because the applicant will take appropriate precautions when grinding on austenitic stainless steel to minimize cold working and prevent contamination. Additional information related to the applicant's compliance with the recommendations in RG 1.37 is contained in Section 6.1.1.4.3 of this report.

FSAR Tier 2, Table 6.1-1 identifies the use of cast austenitic stainless steel (CASS). CASS materials can be susceptible to thermal aging embrittlement when exposed to operating temperatures greater that 482°F. Staff guidance related to thermal aging embrittlement of CASS materials is documented in a letter from Christopher I. Grimes of the NRC to Douglas J. Walters of the Nuclear Energy Institute, May 19, 2000. The ferrite content of CASS materials, which is used to determine its susceptibility to thermal aging embrittlement, should be calculated using Hull's equivalent factors as indicated in NUREG/CR-4513, "Estimation of Fracture Toughness of Cast Stainless Steels During Thermal Aging in LWR Systems," Revision 1, May 1994. FSAR Tier 2, Section 6.1.1.1 states that the applicant will use Hull's equivalent factors to calculate the ferrite content of CASS. In addition, the applicant places limits on the ferrite content of CASS components consistent with the NRC guidance referenced above for all CASS components that will experience service temperatures greater than (482 °F). The staff finds this acceptable because the applicant will follow staff guidance to reduce the susceptibility of ESF CASS components to thermal aging embrittlement.

Based on the above, the staff finds that the applicant conforms to the staff guidance for the fabrication of austenitic stainless steel components in ESF systems and meets all regulatory requirements applicable to such components.

Dissimilar Metal Welds

FSAR Tier 2, Section 6.1.1.1 states that the use of nickel-base alloys in the primary pressure-retaining ESF applications is limited to Alloy 52, 52M, and 152 (Ni-Cr-Fe) weld metals (Alloys 52, 52M, and 152).

Control of dilution of ferritic base material in Ni-Cr-Fe dissimilar metal welds (DMWs) is important because excessive dilution can result in a weld that contains a significantly reduced amount of chromium, thus increasing the weld's susceptibility to stress corrosion cracking. Ni-Cr-Fe welds are also susceptible to ductility dip cracking and other types of welding flaws. In addition, good weld quality of Ni-Cr-Fe welds is highly dependent on interpass cleaning. In RAI 81, Question 06.01.01-5, the staff requested that applicant describe its fabrication controls used to address dilution, welding flaws and interpass cleaning.

In a December 19, 2008, response to RAI 81, Question 06.01.01-5, the applicant stated that Alloys 52/52M and 152 weld filler metal are used for joining dissimilar metal ferritic or austenitic base material combinations such as the low head safety injection heat exchanger ferritic shell to the austenitic stainless steel tube sheet welds. The applicant also stated that the as-deposited weld chromium content is maintained at sufficient levels by controlling welding parameters affecting heat input and power ratio to maintain chromium content in production welds within current industry standards. The applicant further stated that vendors for ESF equipment will meet these standards. The staff finds this acceptable because the applicant will limit the amount of dilution in dissimilar metal welds, thus reducing the susceptibility of DMWs to stress corrosion cracking. To reduce the susceptibility of welds to ductility dip cracking and other flaws inherent in NI-Cr-Fe welds, the applicant stated that all heats of Alloy 52 and Alloy 52M used for

ESF equipment will undergo weldability testing prior to acceptance of the heat for production use. The testing process includes cross-sectioning and metallographic examination. Regarding interpass cleanliness, the applicant stated that interpass cleanliness is of primary importance, and the welders/welding operators are instructed to be attentive during welding to maintain interpass cleanliness as an AREVA NP (AREVA) standard practice. The applicant further stated that this practice will be imposed on subcontractors as applicable, when production welding using these filler metals is performed on ESF safety-related components.

The staff finds the applicant's description of its methods to control dilution and limit welding flaws is acceptable, because the applicant will take extra precautions to maintain the chromium content of DMWs, employ welding filler material testing and in process welding controls that will ensure acceptable weld quality and make DMWs less susceptible to service induced degradation.

Based on the above, the staff finds the applicant's selection of filler metals and fabrication process controls for DMWs in ESF systems acceptable.

Ferritic steel welding

To meet the requirements of GDC 1 and 10 CFR Part 50, Appendix B related to general quality assurance for control of special processes, and 10 CFR 50.55a, the amount of minimum specified preheat should be in accordance with the recommendations of the ASME Code, Section III, Appendix D, Article D-1000 and RG 1.50, "Control of Preheat Temperature for Welding Low-Alloy Steel," unless an alternative procedure is justified. Moisture control on low-hydrogen welding materials should comply with the requirements of ASME Code, Section III, Articles NB, NC, ND-2000 and 4000, and AWS D1.1, unless an alternative procedure is justified. For areas of limited accessibility to perform welding, RG 1.71, "Welder Qualification for Areas of Limited Accessibility," provides guidance acceptable to the staff to ensure the quality of welds.

The amount of specified preheat for the welding of ferritic steels should be in accordance with the recommendations listed in ASME Code, Section III, Appendix D, Article D-1000. Appendix D provides preheat temperature recommendations that the staff considers sufficient to prevent delayed hydrogen cracking in ferritic steel welds. Appendix D is supplemented by positions described in RG 1.50, which recommends that preheating of low-alloy steel welds be maintained until PWHT. FSAR Tier 2, Section 6.1.1.1 states that the minimum preheat for welding of carbon and low-alloy, ferritic materials is in accordance with ASME Code Section III, Division 1, Appendix D (Article D-1000), and RG 1.50. In addition, the staff notes that FSAR Tier 2, Table 1.9-2 indicates that the applicant takes no exceptions to the guidance provided in RG 1.50.

FSAR Tier 2, Section 6.1.1.1 states that moisture control of low hydrogen welding materials complies with the requirements of ASME Code, Section III, Articles NB, NC, or ND-2000 and 4000, and AWS D1.1. The staff determined this to be acceptable because the applicant will appropriately control moisture in low-hydrogen welding materials.

ASME Code, Section III, requires adherence to the requirements of ASME Section IX for welder qualification for production welds. However, there is a need for supplementing this section of the Code, because the assurance of providing satisfactory welds in locations of restricted direct physical and visual accessibility can be increased significantly by qualifying the welder under conditions simulating the space limitations under which the actual welds will be made. RG 1.71 provides the necessary supplement to ASME Code, Section IX, in this respect. FSAR Tier 2, Section 6.1.1.1 states that welder qualification for areas of limited accessibility, and the monitoring and certifying of such welds, are performed in accordance with RG 1.71. Therefore, the staff finds this acceptable.

6.1.1.4.2 Composition and Compatibility of ESF Fluids

To meet the requirements of GDC 4, GDC 14, and GDC 41, the composition of containment spray and core cooling water should be controlled to ensure a minimum pH of 7.0, as addressed in Branch Technical Position (BTP) 6-1, "pH for Emergency Coolant Water for PWRs." In addition, hydrogen generation resulting from the corrosion of metals by containment sprays during a design-basis accident should be controlled as described in RG 1.7, "Control of Combustible Gas Concentrations in Containment."

In order to reduce the probability of stress-corrosion cracking of austenitic stainless steel components, containment and core coolants should be maintained at a pH level of at least 7.0. FSAR Tier 2, Section 6.1.1.2, "ESF Fluids," states that in post-accident situations where the containment is flooded with water containing boric acid, pH is adjusted by releasing tri-sodium phosphate from storage baskets into the water draining to the IRWST (refer to FSAR Tier 2, Section 6.3, "Emergency Core Cooling System"). FSAR Tier 2, Section 6.1.1.2 also states that this raises the pH above 7.0, per the guidance of BTP 6-1, to reduce the probability of stress-corrosion cracking of austenitic stainless steel components. FSAR Tier 2, Section 6.1.1.2 further refers to FSAR Tier 2, Section 15.0.3.12, "Postaccident Reactor Building Water Chemistry Control," for an evaluation of post-accident Reactor Building water chemistry control. As part of its review of FSAR Tier 2, Section 15.0.3, "Radiological Consequences of Design Basis Accidents," the staff performed a confirmatory calculation of the post-accident IRWST pH (see Section 15.0.3 of this report), which verified the pH will be raised and maintained at a value greater than 7.0. This conforms to BTP 6-1 and, therefore, the staff finds this acceptable.

Materials such as aluminum and zinc are susceptible to corrosion in the post-accident environment resulting in the production of hydrogen. In accordance with staff guidance provided in RG 1.7, Regulatory Position 4, materials within the containment that would yield hydrogen gas by corrosion from the emergency cooling or containment spray solutions should be identified, and their use should be limited as much as practicable.

SRP Section 6.1.1, Acceptance Criterion 2.A, states that hydrogen generation from the corrosion of materials within containment, such as aluminum and zinc, depends upon the corrosion rate, which, in turn, depends upon such factors as the coolant chemistry, coolant pH, metal and coolant temperature, and surface area exposed to attack by the coolant. SRP Section 6.1.1, Acceptance Criterion 2.A further states that the assumed corrosion rates of materials in containment should conform to standard corrosion rate data. FSAR Tier 2, Section 6.1.1.2 states that the use of aluminum and zinc in components in containment that could be exposed to post-accident conditions is minimized to avoid hydrogen gas generation. In RAI 81, Question 06.01.01-12, the staff requested that the applicant discuss the methods used to conduct an inventory of and estimate corrosion rates for, zinc, aluminum, and other materials inside containment in order to calculate the amount of hydrogen generated as a result of corrosion during a design-basis accident (DBA) in order for the staff to evaluate the applicant's compliance with GDC 41. In addition, the staff requested that the applicant discuss the method used to determine the amount of hydrogen generated during a DBA. In a November 3, 2008, response to RAI 81, Question 06.01.01-12, the applicant provided the assumptions regarding the materials that contribute to hydrogen generation. The assumed amounts of materials provided in the November 3, 2008, RAI response are the same as those described in FSAR Tier 2, Section 6.2.5.3.1, “Post-LOCA Hydrogen Concentration,” with the addition that the amount of Hypalon and PVC cable were also quantified as 45,359 kilogram (kg) (100,000 pounds-mass (lbm)) and 1,814 kg (4,000 lbm). The response also provided the hydrogen release rate equations for the zinc-based paint, galvanized steel, and aluminum. The response did not explain how it would be assured that the actual amounts of materials such as aluminum and Hypalon cable would not exceed the assumptions of the hydrogen generation calculation.

Although FSAR Tier 2, Section 6.2.5, “Combustible Gas Control in Containment,” shows that materials such as aluminum contribute significantly to the total hydrogen production after a DBA, the combustible gas control system (CGCS) is designed with the capacity to control the amount of hydrogen resulting from a severe accident in which 100 percent oxidation of the fuel cladding occurs, as required by 10 CFR 50.44, “Combustible Gas Control for Nuclear Power Plants.” Regarding hydrogen production during a severe accident, FSAR Tier 2, Section 19.2.4.4.1.1, “Design Evaluation,” states that although hydrogen production due to radiolysis and corrosion occurs, the cladding reaction with water and molten core concrete interaction (MCCI) dominates the production of hydrogen. Therefore, control of the amount of aluminum, zinc, and other hydrogen producing materials is not critical, because the capacity of the CGCS is sufficient to address the hydrogen produced during a severe accident. Since the contribution of these materials to the total amount of hydrogen is not significant during a severe accident, the staff finds the applicant’s response to RAI 81, Question 06.01.01-12 acceptable.

FSAR Tier 2, Section 6.1.1.2 states that the reactor coolant system (RCS) water chemistry is controlled to minimize negative impacts of chemistry on materials integrity, fuel rod corrosion, fuel design performance, and radiation fields, and is routinely analyzed for verification, and that the water chemistry parameters are based on industry knowledge and industry experience as summarized in the Electric Power Research Institute (EPRI) pressurized water reactor (PWR) Primary Water Chemistry Guidelines. FSAR Tier 2, Section 6.1.1.2 refers the reader to FSAR Tier 2, Section 5.2.3, “Reactor Coolant Pressure Boundary Materials,” for more details on RCS water chemistry. This implies that the water chemistry of the ESF systems is identical to that of the RCS. If the impurities in the ESF water are maintained at the same levels as those in the RCS (<0.150 ppm for chloride, fluoride, and sulfate), the possibility of stress corrosion cracking will be minimized, thus supporting compliance with GDC 14. In RAI 314, Question 06.01.01-20, the staff requested that the applicant provide additional information to confirm that the ESF system chemistry for impurities is the same as those for that of the RCS. In a February 24, 2010, response to RAI 314, Question 06.01.01-20, the applicant stated that the IRWST limits for impurities will be the same as the impurity limits imposed on the RCS and the monitoring frequency for the IRWST will be once per month. The maximum IRWST impurity concentrations and the monitoring frequency are consistent with recommendations in EPRI PWR Primary Water Chemistry Guidelines, Appendix B.6, for refueling water storage tanks. The staff finds this acceptable because the applicant will monitor and control IRWST impurities to minimize the potential for stress corrosion cracking in ESF components.

SRP Section 6.1.1 also recommends that the reviewer examine the paths that the solutions would follow in the containment from sprays and emergency core cooling systems to the sump, for both injection and recirculation phases, to verify that no areas accumulate very high or low pH solutions and that any assumptions regarding pH in the modeling of containment spray fission product removal are valid, in accordance with the guidance of SRP Section 6.5.2. The U.S. EPR does not use containment sprays; however, the possibility of flow from certain pipe breaks bypassing the trisodium phosphate (TSP) baskets is evaluated in Section 6.2.2 of this report.

Based on the above, the staff finds that the applicant has met the requirements of GDC 4 and GDC 14, because ESF chemistry is controlled to minimize the possibility of corrosion induced failure of the RCPB, and has met the requirements of GDC 41 by limiting materials that could produce hydrogen in a post-accident situation.

6.1.1.4.3 Component and Systems Cleaning

RG 1.37 describes quality assurance measures for cleaning of fluid systems and associated components of water-cooled nuclear power plants. Per RG 1.37, the staff considers the provisions and recommendations included in ASME NQA-1-1994, “Quality Assurance Requirements for Nuclear Applications,” as generally acceptable for onsite cleaning of materials and components, cleanliness control, and preoperational cleaning and layup of water-cooled nuclear plant fluid systems. These provisions and recommendations provide an adequate basis for complying with the pertinent QA requirements of 10 CFR Part 50, Appendix B, and the regulatory positions given in RG 1.37. FSAR Tier 2, Section 6.1.1.1 states that abrasive work on austenitic stainless steels is controlled to minimize the introduction of contaminants that promote stress-corrosion cracking per RG 1.37. FSAR Tier 2, Section 6.1.1.3, “Component and Systems Cleaning,” states that to prevent stress-corrosion cracking, austenitic stainless steel materials used in the fabrication, installation, and testing of ESF components and systems are handled, protected, stored, and cleaned according to recognized and accepted methods, as identified in the applicable procedures and specifications. The applicant further states that procedures and specifications follow the guidance of RG 1.37. Accordingly, the staff finds the applicant’s component and systems cleaning measures acceptable. Additional information regarding the applicant’s compliance with ASME NQA-1-1994 is discussed in Section 17.5 of this report.

6.1.1.4.4 Thermal Insulation

To meet the requirements of GDC 1, GDC 14, and GDC 31, ESF systems should be designed, fabricated, erected, and tested such that there is an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture. The levels of leachable contaminants in non-metallic insulation materials that come into contact with 300 series austenitic stainless steels used in fluid systems important to safety should be carefully controlled so that stress-corrosion cracking is not promoted. In particular, the leachable chlorides and fluorides should be held to the lowest practical levels. The staff’s position is that following the guidance provided in RG 1.36 is an acceptable method to control leachable contaminants in non-metallic insulation materials.

FSAR Tier 2, Section 6.1.1.4, “Thermal Insulation,” states as follows: ESF systems insulation, from RCS to the first RCPB isolation valves, is primarily constructed of reflective stainless steel. Additional insulation for ESF systems inside containment, where required for plant personnel protection, is primarily constructed of reflective stainless steel. ESF systems insulation outside containment, where required for plant personnel protection, is primarily constructed of non-metallic insulation. In FSAR Tier 2, Section 6.1.1.4, the applicant further states that the use of non-metallic insulation is controlled in accordance with RG 1.36, which is acceptable to the staff as discussed above.

6.1.1.4.5 ITAAC

FSAR Tier 1 ITAAC for various ESF systems include FSAR Tier 1, Section 2.2.3 for SIS/RHRS, FSAR Tier 1, Section 2.2.2 for IRWST, FSAR Tier 1, Section 2.2.7 for EBS, FSAR Tier 1, Section 2.1.1.1 for the RCB steel liner, FSAR Tier 1, Section 3.5 for RCB penetrations, and

FSAR Tier 1, Section 2.2.4 for the EFWS. The staff's evaluation of FSAR Tier 1 ITAAC for ESF systems is located in Section 14.3 of this report.

6.1.1.5 *Combined License Information Items*

Table 6.1.1-1 provides a list of ESF metallic materials related COL information item numbers and descriptions from FSAR Tier 2, Table 1.8-2:

Table 6.1.1-1 U.S. EPR Combined License Information Items

Item No.	Description	FSAR Tier 2 Section
6.1-1	A COL applicant that references the U.S. EPR design certification will review the fabrication and welding procedures and other QA methods of ESF component vendors to verify conformance to RGs 1.44 and 1.31.	6.1.1.1

The staff determined the above listing to be complete. Also, the list adequately describes actions necessary for the COL applicant or holder. No additional COL information items need to be included in FSAR Tier 2, Table 1.8-2 for EST metallic metals consideration.

6.1.1.6 *Conclusions*

The staff concludes, based on the foregoing, that the U.S. EPR ESF materials specifications, controls on fabrication, and compatibility of materials with ESF fluids are acceptable and meet the relevant requirements of GDC 1, GDC 4, GDC 14, GDC 31, GDC 35, and GDC 41; 10 CFR Part 50; Appendix A; 10 CFR Part 50, Appendix B; and 10 CFR 50.55a.

6.1.2 Organic Materials

6.1.2.1 *Introduction*

Many surfaces within the U.S. EPR use organic and inorganic coatings for corrosion protection, or to facilitate surface decontamination. If these coatings were to detach from plant surfaces through delaminating, peeling, or flaking, they could be ingested into safety-related system components and impact the operation of engineered safety systems. Coating degradation can result from exposure to the following conditions: Abrasion or wear including high energy spray, corrosion in the presence of chemicals or liquids including chemical decontamination processes, localized high temperatures, and ionizing radiation. These degradation mechanisms could be present during plant operation, maintenance activities, or accident conditions. Coatings quality and its classification system provide assurance of coating integrity under operating and accident conditions. A listing of organic materials in the containment is maintained as equipment is installed, and the materials are evaluated for their potential interaction with ESFs.

6.1.2.2 *Summary of Application*

FSAR Tier 1: There are no FSAR Tier 1 entries for this area of review.

FSAR Tier 2: The applicant has provided an FSAR Tier 2 system description in Section 6.1.2, "Organic Materials," summarized here, in part, as follows:

Protective coatings are categorized according to their location, service conditions, and quality requirements. These categories are defined in the FSAR as follows:

<u>Inside Containment</u>

- Service Level I: Radiologically controlled areas (RCAs) with a direct path to the in-containment refueling water storage tank or other ESFs
- Service Level II: Radiologically controlled areas with no direct path to the IRWST or other ESFs

<u>Outside Containment</u>

- Service Level II: Radiologically controlled areas where coatings will not communicate with ESFs
- Service Level III: Areas with a potential path to ESFs
- No Service Level: Balance of plant (BOP), Non-radiologically controlled areas [non-RCAs] with no potential path to ESFs

Service Level I coatings for safety-related use within the plant are DBA tested and qualified in accordance with American Society for Testing and Materials (ASTM) D5144-00, "Standard Guide for Use of Protective Coating Standards in Nuclear Power Plants."

The Service Level II coating classification is normally restricted to coatings used in RCAs outside of containment. The FSAR states that failure of Service Level II coatings will not adversely affect the function of ESFs, and these protective coatings are classified as non-safety-related.

Service Level III coatings are applied in non-RCAs outside of containment where detachment could adversely affect the safety function of a safety-related SSC. The FSAR classifies these protective coatings as safety-related.

Coatings with no service level are classified as BOP and are used in non-RCAs. The FSAR states that failure of these coatings has no impact on the ESFs, and these coatings are classified as non-safety-related.

Coatings for different service classifications and locations are summarized in FSAR Tier 2, Table 6.1-2, "Coatings Classifications and Uses."

ITAAC: There are no inspections, tests, analyses, and acceptance criteria items for this area of review.

Technical Specifications: There are no Technical Specifications for this area of review.

6.1.2.3 *Regulatory Basis*

The relevant requirements of NRC regulations for this area of review and the associated acceptance criteria are given in NUREG-0800, Section 6.1.2 and are summarized below. Review interfaces with other SRP sections also can be found in NUREG-0800, Section 6.1.2.

- 10 CFR Part 50, Appendix B, as it relates to the quality assurance requirements for the design, fabrication, and construction of safety-related structures, systems, and components.

Acceptance criteria adequate to meet the above requirements include:

- A coating system to be applied inside containment is acceptable if it meets the regulatory positions of RG 1.54, "Service Level I, II, and III Protective Coatings Applied to Nuclear Power Plants," and the standards of ASTM D5144-00 and ASTM D3911-03, "Standard Test Method for Evaluating Coatings Used in Light Water Nuclear Power Plants at Simulated Design Basis Accident (DBA) Conditions."

6.1.2.4 *Technical Evaluation*

The U.S. EPR uses organic and inorganic coatings for corrosion protection or to facilitate surface decontamination. Ingestion of delaminating, peeling, or flaking coatings into safety-related system components could affect the operation of engineered safety systems. Coatings can degrade from abrasion, wear, and corrosion in the presence of chemicals, localized high temperatures, and ionizing radiation. As described in detail below, utilizing high quality coatings assures coating integrity under operating and accident conditions.

The staff reviewed FSAR Tier 2, Section 6.1.2, "Organic Materials," in accordance with SRP Section 6.1.2, "Protective Coating Systems (Paints) – Organic Materials," Revision 3, March 2007. The staff also reviewed FSAR Tier 2, Section 6.1.2 to verify the applicant has committed to use protective coatings that meet the guidance of RG 1.54, Revision 1, to the extent that failure of protective coatings could prevent SSCs from fulfilling their safety-related function.

The application provides the general description of Service Level designations used by the applicant depending on location and use of organic materials in the plant. RG 1.54 describes an acceptable method for complying with the requirements with regard to protective coatings applied to ferritic steels, stainless steel, zinc coated (galvanized) steel, concrete, and masonry surfaces of nuclear power plants. RG 1.54 recommends that qualified protective coatings have the capability of surviving a design-basis accident without adversely affecting safety-related SSCs. RG 1.54 references ASTM D5144-00, which provides guidance for the application and maintenance of protective coatings, including a basis for qualifying coatings for expected environmental conditions under both normal conditions and postulated accidents. ASTM D5144-00 references additional ASTM standards that provide guidance on practices and programs that are acceptable for the selection, application, qualification, inspection, and maintenance of protective coatings.

FSAR Tier 2, Section 6.1.2.1, "Description of Protective Coatings," describes the coating classification system used both inside and outside of the reactor containment. Service Level I coatings are applied inside containment where failure could impair safe-shutdown by adversely affecting the safety-related functions of post-accident systems. Service Level II coatings are used both inside and outside of containment where coating failure could impair, but not prevent, normal operating performance. Service Level II coatings are only used in containment in regions where transport of failed coatings to the sump during post-LOCA recirculation is unlikely. Service Level III coatings are used outside of containment where detachment could adversely affect the safety function of safety-related SSCs. Therefore, the staff finds that the coating classification system conforms to RG 1.54 and ASTM D5144-00.

The staff reviewed FSAR Tier 2, Section 6.1.2.1.2.1, “Inside Containment,” regarding the use of Service Level I qualified epoxy-type coating systems for locations inside containment expected to experience temperatures below 121 °C (250 °F) and qualified inorganic zinc (IOZ) coating systems for locations inside containment expected to experience temperatures above 121 °C (250 °F). The use of epoxy type coatings is acceptable because it conforms to the guidance in ASTM D5144-00. For the same reason, the staff finds the use of these coating systems for both carbon steel and concrete surfaces acceptable. For areas where painting is impractical, the staff finds the proposed use of hot-dipped galvanization for corrosion protection without a service level designation acceptable because it conforms to the guidance in ASTM D5144-00.

The staff reviewed FSAR Tier 2, Section 6.1.2.1.2.2, “Outside Containment,” regarding the use of Service Level II and III epoxy-type coating systems for temperatures below 121 °C (250 °F) and IOZ coating systems above 121 °C (250 °F) and found the use of these systems acceptable. The use of hot-dipped galvanization for Service Level II and III applications where painting is impractical was reviewed and found acceptable because it conforms to the guidance in ASTM D5144-00.

Qualified Service Level I coatings are used in containment where failure could impact safety-related systems such as the IRWST; however, additional design features such as screens ensure that in the case of failure, the amount of debris that could reach the IRWST is minimized. Additionally, in the vicinity of the IRWST and in the IRWST itself, stainless steel is used and the surfaces are not coated.

FSAR Tier 2, Section 6.1.2.2.2, “Coating Repairs and Limitations on Coating Thickness,” describes a maintenance program for coatings that ensures maintenance and repairs of coatings are performed following approved procedures. FSAR Tier 2, Section 6.1.2.3.5, “Protective Coating and Organic Materials Program,” states that the maintenance program complies with 10 CFR 50.65, “Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants.” The staff is not approving the repair and maintenance of coatings as part of this design certification application. The description and implementation of the coatings program are the responsibility of the COL applicant and are to be addressed by the COL applicant. Therefore, in RAI 480, Question 06.01.02-11, the staff requested that the applicant provide a COL information item to describe the coatings program and its implementation, including maintenance and repair of coatings. **RAI 480, Question 06.01.02-11, is being tracked as an open item.**

FSAR Tier 2, Section 6.1.2.4, “Exceptions to Regulatory Guide 1.54, Revision 1,” stated certain exceptions to standards referenced in RG 1.54. These exceptions included the use of more recent versions of approved ASTM standards, as well as the use of an alternative to ASTM D5962-96, “Standard Guide for Maintaining Unqualified Coatings (Paints) Within Level I Areas of a Nuclear Power Facility (Withdrawn 2008).” These exceptions were reviewed, and the staff finds that the exceptions conform to RG 1.54, Revision 2.

Technical Specifications and Surveillance

There are no Technical Specifications associated with this section.

ITAAC

There are no ITAAC associated with this section.

Preoperational Testing

There is no preoperational testing associated with this section.

Combined License Information Items

Table 6.1.2-1 provides a list of organic materials related COL information item numbers and descriptions from FSAR Tier 2, Table 1.8-2:

Table 6.1.2-1 U.S. EPR Combined License Information Items

Item No.	Description	FSAR Tier 2 Section
6.1-2	If components cannot be procured with DBA qualified coatings applied by the component manufacturer, a COL applicant that references the U.S. EPR design certification must do one of the following: procure the component as uncoated and apply a DBA-qualified coating system in accordance with 10 CFR Part 50, Appendix B, Criterion IX; confirm that the DBA unqualified coating is removed and the component is recoated with DBA-qualified coatings in accordance with 10 CFR Part 50 Appendix B, Criterion IX; or add the quantity of DBA-unqualified coatings to a list that documents those DBA-unqualified coatings already existing within containment.	6.1.2.3.2

The staff finds the above listing complete. Also, the list adequately describes actions necessary for the COL applicant or holder. No additional COL information items need to be included in FSAR Tier 2, Table 1.8-2 for organic materials consideration. As defined in the current design certification rules, COL information items typically identify matters that must be addressed in the site-specific portion of the FSAR and constitute information requirements but are not the only acceptable set of information in the FSAR. COL Information Item 6.1-2 does not appear to conform to this definition. Accordingly, in RAI 480, Question 06.02.01-10, the staff requested that the applicant revise the description of COL Information Item 6.1-2 for the COL applicant to describe its plans for addressing components that cannot be procured with DBA qualified coatings. **RAI 480, Question 06.01.02-10, is being tracked as an open item.**

6.1.2.5 *Conclusions*

Based on the above, the staff finds that the protective coating systems and their applications are acceptable and meet the requirements of 10 CFR Part 50, Appendix B. This conclusion is based on the applicant having met the quality assurance requirements of 10 CFR Part 50, Appendix B, since the coating systems and their applications meet the positions of RG 1.54, and the quality assurance standards of ASTM D5144-00, "Standard Test Method for Evaluating Coatings Used in Light Water nuclear Power Plants at Simulated Design Basis Accident.". Also, the staff evaluated the containment coating systems as to their suitability to withstand a postulated design-basis accident environment. The coating systems chosen by the applicant have been qualified under conditions which take into account the postulated DBA conditions. The item included in FSAR Tier 2, Table 1.8-2 is appropriate for deferral to the COL applicant.

6.2 Containment Systems

This section describes the staff's review of the U.S. EPR containment systems, including the containment, containment isolation system, combustible gas control system, containment heat removal systems, and secondary containment. These systems are designed to protect the integrity of the containment, and limit any releases of radioactive nuclides during plant operation and resulting from postulated accidents.

The U.S. EPR containment systems include the containment, containment isolation system, and containment combustible gas control system. These systems contain any radio-nuclides that are released from the fuel during postulated accidents, prevent further release of these nuclides to the balance of plant and environment, and limit accumulation of combustible gases in containment that are generated during an accident.

6.2.1 Containment Functional Design

FSAR Tier 2, Section 6.2.1, "Containment Functional Design," states that the U.S. EPR containment is a PWR dry containment with a number of innovative design features. It consists of a cylindrical concrete Containment Building (CB) surrounded by a cylindrical concrete outer Shield Building (SB). The SB protects the CB from external hazards. The annulus between the CB and the SB is filtered to reduce radioisotope release.

FSAR Tier 2, Section 6.2.1, states that the CB is separated into two rooms: The inner equipment space and the surrounding service space. The equipment space contains the RCS and supporting equipment. During power operation, temperatures and radiation levels are relatively high in the equipment space, making it inaccessible. In contrast, the service space is better protected, cooler, and accessible during power operation. The separation, which is complete, is accomplish by compartment walls, doors, foils and dampers.

FSAR Tier 2, Section 6.2.1, states that the U.S. EPR containment design does not utilize fan coolers common in many PWRs, and that the design does not rely on containment sprays for protection in case of DBAs. A containment spray system is included in the design; however, it is not safety related, and it is used only for the mitigation of severe accidents. The system can only be manually aligned and activated. Since multiple manual actions are necessary to activate the system, the U.S. EPR is not susceptible to potential damage because of rapid reduction of containment internal pressure due to inadvertent actuation of the spray system.

FSAR Tier 2, Section 6.2.1, states that containment heat removal is accomplished by two dedicated safety systems: the emergency core cooling system (ECCS) and the containment atmospheric circulation system (CONVECT). An important component of the ECCS is a large IRWST located below the RCS at the bottom of the containment. In addition to the refueling water, all spilled and condensed water, in case of a DBA, collects in this tank. The low head safety injection pumps circulate water from the tank through heat exchangers to the RCS. The subcooled water injected into the RCS will condense steam in the system, and thus reduce energy release to the containment. The rest of the cold water will spill into the containment, mix with the condensed water, remove additional heat, and then collect in the IRWST.

FSAR Tier 2, Section 6.2.1, states that the CONVECT system has two major safety-related functions: (1) Open up the containment into one large single volume in order to mitigate the consequences of DBAs; and (2) establish flow patterns within containment that enhance heat transfer to the containment walls and internal structures.

Additionally, FSAR Tier 2, Section 6.2.1, states that the CONVECT system consists of a large number of rupture and convection foils placed in the ceiling of the steam generator (SG) compartments. The foil location at the top of each steam generation compartment is referred to as the pressure equalization ceiling (PEC). In addition, a set of large mixing dampers is located in the wall of the IRWST above the water level. The rupture foils burst on a pressure differential, in either direction. The needed pressure differential is relatively small (less than 6.89 kilopascals (kPa) (1 psi)). The convection foils are designed to open, not only on differential pressure, but also on high temperature that occurs in case of both small and large breaks. The foils form a passive system; no actuation is necessary.

FSAR Tier 2, Section 6.2.1.1.1.2, “Design Features,” states that there are eight mixing dampers along the periphery of the IRWST. The dampers open either on a pressure differential signal between the equipment space and the service space of the containment, or on a set containment pressure signal. The set containment pressure is only slightly over atmospheric pressure, assuring prompt opening of the dampers for practically all DBAs. The dampers are spring loaded; they open if power is lost. The dampers can be operated manually from the control room.

FSAR Tier 2, Section 6.2.1.5.3, “Other Parameters,” states that the CB has a nominal net free volume of 8.178×10^4 m^3 (2.888×10^6 ft^3), and it is designed to withstand pressures and temperatures resulting from postulated DBAs. The containment design pressure is 427.5 kPa (62 psig). The design temperature is 170 °C (338 °F). The design leak rate at design temperature is less than 0.25 percent by volume per day.

The design of the U.S. EPR containment is evaluated in detail in the following five sections of this report:

- Section 6.2.1.1, “Containment Analysis,” describes and evaluates the containment pressure and temperature calculations performed by the applicant, discusses the effectiveness of the containment heat removal systems, and presents the results of NRC confirmatory calculations.

- Section 6.2.1.2, “Subcompartment Analysis,” reviews accident differential pressure calculated for individual containment subcompartments that might result from a high energy line break within the subcompartment.

- Section 6.2.1.3, “Mass and Energy Release Analysis for Postulated LOCAs,” discusses mass and energy release calculated for LOCAs, and evaluates the assumptions and models used for the mass and energy release calculations.

- Section 6.2.1.4, “Mass and Energy Release Analysis for Postulated Steam Line Break Accidents,” presents a similar evaluation for mass and energy release calculations performed for steam line break accidents.

- Section 6.2.1.5, “Minimum Containment Pressure Analysis for Emergency Core Cooling System Performance Capability Studies,” addresses minimum containment pressure used to show the adequacy of the ECCS analysis and conservatism used by the applicant in the analysis.

The containment heat removal function of the ECCS, as well as the CONVECT system, are discussed in more detail in Section 6.2.2, ”Containment Heat Removal Systems,” of this report.

6.2.1.1 *Containment Analysis*

6.2.1.1.1 Introduction

The U.S. EPR RCB consists of a cylindrical reinforced concrete outer SB and a cylindrical post-tensioned concrete inner CB with a steel liner. The SB is designed to protect the inner containment from external events and to collect and filter leakage from the inner containment in the event of an accident. The inner containment is designed to withstand the impact of postulated accidents involving the release of high energy fluids from the reactor coolant system and secondary systems. The containment SSCs that are important to safety are designed to withstand the environmental and dynamic effects associated with both normal plant operation and postulated accidents. The containment and its associated systems are designed to limit the release of radioactivity to the environment and are designed to remain functional during a design basis accident. In lieu of safety-related containment spray systems and containment air coolers, containment heat removal is provided by the ECCS and the CONVECT. The ECCS recirculates water from the in-containment refueling water storage tank through heat exchangers to the reactor coolant system. The CONVECT system establishes communication between the two rooms of the containment and promotes heat transfer to the containment walls and internal structures.

6.2.1.1.2 Summary of Application

FSAR Tier 1: In FSAR Tier 1, Section 2.1.1, “Nuclear Island,” the applicant states that the Reactor Containment Building is a Seismic Category 1 safety-related cylindrical concrete structure. The primary functions of the RCB are to protect the safety-related SSCs located within it, to prevent the release of radiation during plant operations, and to prevent the release of radiation and contamination in the event of an accident.

FSAR Tier 2: The applicant has provided an FSAR Tier 2 system description in Section 6.2.1.1, “Containment Structure,” summarized here, in part, as follows:

FSAR Tier 2, Section 6.2.1, states that the RCB consists of a cylindrical reinforced concrete outer SB and a cylindrical post-tensioned concrete inner CB with a steel liner. There is an annular space between the two buildings. The SB protects the CB from external hazards and collects and filters leakage from the inner containment in the event of an accident.

FSAR Tier 2, Section 6.2.1, states that the containment’s SSCs that are safety-related are designed to withstand the environmental and dynamic effects associated with both normal plant operation and postulated accidents. The environmental effects include the temperatures, pressures, and fluids encountered during normal and accident conditions such as pipe breaks. The dynamic effects include those arising from in-plant equipment failures or accidents (e.g., pipe breaks, missiles, jet impingement, and pipe whip forces), as well as those resulting from events and conditions outside the containment (e.g., tornadoes, earthquake, or aircraft impact). In the accident analysis, both RCS and secondary system pipe ruptures are assumed.

FSAR Tier 2, Section 6.2.1, states that the containment and its associated systems are designed to limit the release of radioactivity to the environment and are designed to remain functional during a design basis accident. By meeting these performance goals, the containment is designed to accommodate the calculated pressures and temperatures resulting from a LOCA, from a main steam line break (MSLB) accident or from a main feedwater line break (MFWLB) accident without exceeding its designed leakage limits.

FSAR Tier 2, Section 6.2.1.1, states that the U.S. EPR does not have an automatic containment spray system or containment air coolers for DBA mitigation, design features that are common to currently operating PWRs. Containment post-accident heat removal is performed by the ECCS and the CONVECT system. A design feature of the U.S. EPR is the IRWST at the bottom of the CB.

FSAR Tier 2, Section 6.2.1.1, states that in case of an accident, water spilled from the RCS and water condensed in the containment collect in the IRWST, in addition to the refueling water. The low head safety injection system recirculates this water into the RCS through heat exchangers. The CONVECT system enhances heat transfer within the containment.

ITAAC: The inspections, tests, analyses, and acceptance criteria associated with FSAR Tier 2, Section 6.2.1.1 are given in FSAR Tier 1, Section 2.1.1.

Technical Specifications: The Technical Specifications applicable to the containment systems can be found in FSAR Tier 2, Chapter 16, "Technical Specifications," Sections 3.6, "Containment Systems," and B 3.6, "Containment Systems."

6.2.1.1.3 Regulatory Basis

The relevant requirements of NRC regulations for this area of review, and the associated acceptance criteria, are given in NUREG-0800, Section 6.2.1.1.A, "PWR Dry Containments, Including Subatmospheric Containments," and are summarized below. Review interfaces with other SRP sections also can be found in NUREG-0800, Section 6.2.1.1.A.

1. GDC 16, "Containment Design," as it relates to the reactor containment and associated systems being designed to assure that containment design conditions important to safety are not exceeded for as long as postulated accident conditions require.

2. GDC 38, "Containment Heat Removal," as it relates to the containment heat removal system(s) function to rapidly reduce the containment pressure and temperature following any loss-of-coolant accident and maintain them at acceptably low levels.

3. GDC 50, "Containment Design Basis," as it relates to the reactor containment structure and associated heat removal system(s) being designed so that the containment structure and its internal compartments can accommodate the calculated pressure and temperature conditions resulting from any loss-of-coolant accident without exceeding the design leakage rate and with sufficient margin.

6.2.1.1.4 Technical Evaluation

The staff notes that the U.S. EPR containment is a new design, different from typical PWR dry containments, and is a two room design. Equipment rooms immediately surrounding the RCS are isolated from the rest of the containment. Beyond this inner region, personnel access can be provided during certain maintenance tasks. Separation is provided by structures and closed portals to minimize radiation exposure in the accessible space areas. During power operation, the inaccessible areas inside containment (the "equipment space") experience higher temperatures and radiation than the accessible areas, because they are exposed to the hot walls of the nuclear steam supply system. The cooler, accessible areas are the "service space."

The staff notes that another difference between the U.S. EPR containment and typical PWR dry containments is that the U.S. EPR design does not utilize safety-related fan coolers commonly found in many PWRs. The staff also notes a third difference in the design of the containment spray system. The spray system is not safety-related, and it is used only for the mitigation of severe accidents. The system can only be manually aligned and activated. It is not used for design basis accidents. Because of the number of steps required to manually align and activate the containment spray for the U.S. EPR, the U.S. EPR is not susceptible to inadvertent actuation of the spray system or to potential damage because of the rapid reduction of the containment internal pressure that would result from such an inadvertent actuation.

The staff notes that in the absence of containment atmospheric spray and fan coolers, the ECCS and the containment heat sinks (containment wall, internal structures) play a vital role in removing steam from the containment atmosphere following a high energy line break within containment. The ECCS recirculates water spilled from the RCS and water condensed on containment walls and internals back into the RCS through a heat exchanger. The spilled and condensed water collects in the IRWST, located at the bottom of the containment; the low pressure safety injection (LPSI) pumps take suction from the tank. Heat from the secondary side of the heat exchanger is transferred to the ultimate heat sink. Heat transfer to the in-containment heat sinks is promoted by the CONVECT system. Both the ECCS and the CONVECT systems are safety-related systems.

The staff notes that the CONVECT system consists of rupture foils, convection foils, mixing dampers, and related instrumentation and control equipment. Many rupture foils and convection foils are placed in the ceiling of each steam generator compartment. The total area of the foils is in excess of 18.6 m^2 (200 ft^2) per steam generator compartment. More than half of the foils are convection foils. The dampers are located in the lower part of the containment in the IRWST wall above the water level. There are eight of these dampers with a total area in excess of 5.57 m^2 (60 ft^2). Opening of the foils and dampers is designed to set up circulation patterns in both the equipment space and the service space.

The rupture foils are passive components which will burst open if the pressure differential on the foils exceeds a predetermined value. The rupture foils burst in either direction. The foils will rupture at a pressure differential less than 6.89 kPa (1 psi). The convection foils are rupture foils placed in a frame. The frame is kept in the closed position by a fusible link. Should temperature rise to a set level, the link will melt with a short delay, and the frame will swing open by gravity. The result is a foil that will burst open on a pressure differential and will also open if the compartment temperature reaches a certain level. Convection foils are passive components.

The mixing dampers open either on a differential pressure signal between the equipment space and the service space or on a preset containment pressure signal. The containment pressure signal is set just above atmospheric pressure, assuring fast opening of the dampers for most accident scenarios. The dampers open by horizontal rotation about a central pivot. When closed, the damper is held in position by an electromagnet against a compressed spring. In case of a power failure to the solenoid of the electromagnet, the spring will drive the damper open. A motor driven actuator operates each damper. When electric power is restored to the motor driven actuator, it is again available for normal operation. The mixing dampers can be operated from the control room.

The CONVECT system is discussed in more detail in Section 6.2.2 of this report. Both the rupture foils and the convection foils have been tested for performance verification and for establishing set points. Also, the effect of a postulated single failure on the performance of the mixing dampers was investigated. The staff's evaluation of the test results and the mixing damper failure mode analysis is reported in Section 6.2.2 of this report.

As mentioned above, rupture foils and convection foils are placed in the ceiling of each steam generator compartment. There is one compartment, the pressurizer compartment that houses high energy lines, which could be the location of a LOCA, which does not have foils installed in the ceiling. As described in the responses to RAI 368, Questions 06.02.01-66 through 06.02.01-69, postulated LOCAs in the pressurizer compartment have been analyzed to show compliance with regulatory requirements. The applicant added six safety-related doors to the containment design to vent the pressurizer compartment following a postulated piping rupture to ensure that regulatory requirements are met. In RAI 368, Questions 06.02.01-70 and 06.02.01-71, the staff has requested that the applicant describe these doors and address their safety-related features. **RAI 368, Questions 06.02.01-70 and 06.02.01-71 are being tracked as open items.**

The staff notes that separation between the equipment space and the service space is accomplished by compartment walls, foils and dampers of the CONVECT system, and access doors. The doors are kept closed during power operation. During shutdown, they provide access to various equipment compartments. As a result of accident conditions, some of the doors could open. The effect of opening of doors would be increased communication between the equipment space and the service space, and the stimulation of circulation and heat transfer. The staff notes that some of the the doors within the U.S. EPR containment are safety-related. The applicant considers the non-safety-related doors to be closed in the safety analyses for the plant and has demonstrated that the closed condition is most conservative for containment pressure and temperature analyses.

The applicant compared the U.S. EPR containment with three other PWR containments: Seabrook, St. Lucie, and Three Mile Island (TMI) (See "EPR Containment Design," AREVA public meeting slides, January 29, 2008). From the public meeting slides, the staff noted that on a generated thermal megawatt basis, the free volume of the U.S. EPR containment is significantly smaller than the free volume of the other containments (20 percent to 30 percent smaller). The design pressure is higher than for the other PWR containments, and the total heat sink area is larger than it is for the other plants.

Based on these observations, the staff's review focused on the following three questions:

- Would the higher containment design pressure compensate for the relatively smaller size of the containment?
- Can the ECCS and the CONVECT system without the help of fan coolers and containment spray terminate the pressure rise and reduce containment pressure as required?
- Would compartment designs and the design of the dampers and foils create effective circulation patterns within the containment to promote heat transfer?

The staff noted that the U.S. EPR containment is a new design, and that there is no experience accumulated with this design. Large scale integral tests of the U.S. EPR containment are not available to permit direct interpretation. Under these circumstances, both demonstration of the performance of the CONVECT system and demonstration of compliance with applicable regulations is done by analysis. In FSAR Tier 2, Section 6.2.1, the applicant described containment pressure and temperature calculations for two sets of postulated accidents, namely, for LOCAs and MSLB accidents. The applicant also addressed MFWLB accidents.

The purpose of this section is to review and assess the GOTHIC methodology used to model the reactor CB and predict its response to design basis LOCA and MSLB accidents, as well as the applicant's assessment of MFWLB accidents. In FSAR Tier 2, Section 6.2.1, Revision 0, the applicant originally proposed to evaluate the containment response to design basis events using a single-node representation in GOTHIC and contended that opening of the foils and dampers would make the two rooms of the U.S. EPR containment act as a single room. In RAI 82, Question 06.02.01-12, Part c.3, the staff raised concerns to the applicant that containment stratification would occur. Hot steam would rise to the containment dome and force cold air down to the lower regions of the containment. Most of the containment internal structures are located in the lower regions of the containment. The lower temperatures and the collection of air there would retard steam condensation on these surfaces. Temperature stratification and stratification of air and steam could not be evaluated using a single-node model.

The safety analyses of most operating PWRs have been performed using single-node containment models, and these safety analyses have been accepted by the staff. These designs include safety-related containment spray and fan cooler units which have been shown to produce containment mixing. In response to questions raised in RAI 1 and RAI 82, the applicant submitted technical report ANP-10299P, "Applicability of AREVA NP Containment Response Evaluation Methodology to the U.S. EPR for Large Break LOCA Analysis," Revision 0, on January 31, 2009. The technical report presents comparisons of predictions of multi- and single-node GOTHIC models with the experimental results from a number of large scale test facilities. After reviewing the comparisons of the GOTHIC models with the test data and other pertinent information, the staff concluded that development of a multi-node model of the U.S. EPR containment would be necessary for the following reasons:

- AREVA Technical Report ANP-10299P, Revision 0, Section 6.2, "Integral Test Benchmarks (EMDAP 18)," describes correlation of containment test data using single- and multi-node GOTHIC models for containment test facilities. For LBLOCA simulations, the single-node GOTHIC models generally predict conservative results for the first several minutes of the tests. After the first several minutes, the measured test pressures are under- predicted using the one-node GOTHIC models. A peak in the calculated containment pressure for the U.S. EPR is calculated to occur at 60 minutes into the accident for the limiting double-ended guillotine (DEG) - Cold Leg Pump Suction (CLPS) large-break LOCA. Therefore, the staff concluded that large-break LOCA analysis should be performed using a multi-node containment model.

- Findings presented in Nuclear Energy Agency/Committee on Safety of Nuclear Installations (NEA/CSNI), conference paper in June 1999 by Karwat, H., titled,"State of the Art Report on Containment Thermal Hydraulics and Hydrogen Distribution," indicates that break elevation within the containment is an important consideration. Breaks low in the containment were determined to promote atmospheric mixing,but breaks high in the containment were determined to result in steam stratification in the containment dome, which once established was very stable (See Section 4.4.3.3 of the conference paper). The steam lines of the U.S. EPR are located high within the containment. Therefore, analysis of steam line breaks with a one-node containment model would not be conservative, since the effects of stratification cannot be evaluated. Therefore, the staff concluded that containment analyses of steam line breaks should be performed using a multi-node containment model.

- In the case of the U.S. EPR, the adequacy of the location and size of the foils together with the location and size of the dampers to sufficiently mitigate the consequences of large breaks needs to be demonstrated. Furthermore, since the rupture foils might not rupture in case of small breaks, the size and location of the convection foils, and the size and location of the dampers, should be demonstrated to be sufficient to mitigate the consequences of small LOCAs and small MSLBs. AREVA Technical Report ANP-10299P, Revision 0, Section 6.2.1.5, "[Heissdampfreaktor] HDR Test E11.2," describes simulation of containment response to a small-break LOCA. The single-node GOTHIC model was found to under-predict both containment pressures and temperatures; whereas, the multi-node GOTHIC model made much better prediction compared to the available experimental data. Therefore, the staff concluded that containment analyses of small breaks should be performed using a multi-node containment model.

- There are at least five compartments in the U.S. EPR with closed ceilings. The four SG components are equipped with foils. The fifth compartment, the pressurizer compartment, has no foils in its ceiling. The staff was concerned about whether the design is sufficient if the break is in the pressurizer compartment. The staff indicated that multi-node capability would be needed for these analyses.

The staff issued RAI 209, Question 06.02.01-14, to document these concerns. To address the staff's concerns, the applicant developed a multi-node GOTHIC model of the U.S. EPR containment with 30 lumped parameter nodes and submitted AREVA Technical Report ANP-10299P, Revision 2. In ANP-10299P, Revision 2, the dome region was subdivided into a 5 x 5 x 19 mesh. The 30-node representation of the containment permits explicit modeling of the CONVECT system's foils and mixing dampers. The applicant stated that the arrangement of nodes is sufficiently detailed to permit development of natural circulation patterns in the containment. Both local atmosphere and local wall temperatures are calculated by the applicant and are used in heat transfer predictions. The applicant stated that the detailed, three-dimensional representation of the dome region is capable of predicting stratification, should it occur. The staff obtained the multi-node GOTHIC model from the applicant and performed detailed comparisons between the input and the applicant's assumptions as stated in the FSAR Tier 2, Section 6.2.1. The staff also performed calculations using the applicant's GOTHIC model. The staff confirmatory calculations were performed using the independently developed MELCOR code. The staff performed calculations with the applicant's GOTHIC model in order to evaluate the detailed code input and output, to provide graphical output of selected variables for comparison with the staff's confirmatory calculations and to evaluate the effect of changes in selected parameters on the result.

In RAI 368, Question 06.02.01-73, Part a, the staff requested that the applicant demonstrate that the level of detail of the 30-node GOTHIC simulation of the U.S. EPR containment is adequate to model the effects of containment stratification. In a December 6, 2010, response to RAI 368, Question 06.02.01-73, the applicant performed a sensitivity study for which the number of axial nodes in the service space was increased from three to eight. The increase in noding did not result in the calculated pressures and temperatures increasing above those previously calculated. This result was confirmed by the staff using the MELCOR code for which the noding in the service space was increased. In development of the multi-node MELCOR model, the staff noted inconsistencies between the applicant's GOTHIC model and the detailed containment design description described in FSAR Tier 2, Section 6.2.1. Therefore, in follow-up RAI 437, Questions 06.02.01-99 and 06.02.01-100, the staff requested that the applicant resolve these inconsistencies. **RAI 437, Questions 06.02.01-99 and 06.02.01-100 are being tracked as open items.**

In RAI 368, Question 06.02.01-73, Part b, the staff requested that the applicant demonstrate that the noding arrangement does not predict artificial circulation patterns as discussed in Numeral Applications, Inc., (NAI) 8907-02, "GOTHIC Containment Analysis Package, Users Manual," Revision 16, Section 22.10. Artificial circulation would have the effect of predicting mixing in the containment atmosphere in excess of that which would actually occur and might result in non-conservative calculated containment temperatures and pressures. In a December 6, 2010, response to RAI 368, Question 06.02.01-73(b), the applicant performed a null transient with the GOTHIC U.S. EPR containment model. A null transient assumes no blowdown source and no heat removal to the internal containment heat structures (internal walls and structures that are credited for removing heat). The staff notes that ideally, at the end of a null transient, flowrates, between the compartments should approach zero. The applicant calculated small residual flows at the end of the null transient. These residual flows were attributed to unequal axial noding between the subdivided dome region and the adjacent lumped parameter volumes. Unequal noding in the vertical direction between parallel node stacks of a compressible fluid results in artificial elevation pressure difference predictions, which, in turn, results in calculated residual flows during null transients. GOTHIC has an input option which allows the same reference pressure to be utilized for all volumes. Use of this option virtually eliminated the calculated residual flows. The applicant then used the same reference pressure option for the limiting LOCA and MSLB events. These analyses showed the peak containment pressures to be insensitive to the reference pressure option and, therefore, to the noding imbalances between the meshed containment dome region and the adjacent lumped parameter volumes of the applicant's 30-node GOTHIC containment model of U.S. EPR. The applicant's comparisons showed a small effect (approximately 5.6 °C (10 °F)) on the peak containment temperature for the MSLB. In RAI 368, Question 06.02.01-73, the staff requested that the applicant clarify if the temperature increase will be added to the EQ curve. The staff also requested that the applicant indicate what would be the effect of artificial flows for the later peaks of smaller breaks, and provide an explanation on how the EQ curve would be affected. **RAI 368, Question 06.02.01-73 is being tracked as an open item.** During a September 30, 2010, audit, the applicant stated that it will show that the EQ curve will not be significantly affected. The staff issued RAI 378, Question 06.02.01-93 to specifically track the development of a revised EQ curve. **RAI 378, Question 06.02.01-93 is being tracked as an open item.**

In AREVA Technical Report, ANP-10299P, the applicant elected to use the GOTHIC diffusion layer model (DLM) to calculate heat and mass flow from the containment atmosphere to the internal structures. The GOTHIC DLM is described in NAI 8907-06 "GOTHIC Containment Analysis Package, Technical Manual," Revision 17. Use of the GOTHIC DLM model has been accepted by the staff for safety analyses with the provision that the model not be enhanced to account for formation of waves in the condensate film (film roughening) and for the effect of small droplets (mist) in the boundary layer (See NRC staff letter to Nuclear Management Company, "Kewaunee Nuclear Power Plant – Issuance of Amendment (TAC NO. MB6408)," September 29, 2003). The model was validated by comparison with large scale tests as described in NAI 8907-09, "GOTHIC Containment Analysis Package, Qualification Report," Revision 10. The DLM with the film and mist enhancements is referred to as DLM-FM. Predictions made with the DLM-FM model show good agreement with the experimental data. Predictions were mostly within a 20 percent error band. Similar comparison with the Uchida correlation that is commonly used in single-node containment safety analyses resulted in a much larger error band. The DLM without enhancements underestimated the heat transfer observed in the tests, providing conservative results. Thus, elimination of the enhancements makes the approach sufficiently conservative for containment analysis. The applicant is using the DLM in the containment analysis of the U.S. EPR without enhancements.

The staff noted that the degree of circulation will depend on the flow losses for the circulation path through the rupture and convection foils, down through the service space, and through the dampers. Therefore, in RAI 221, Question 06.02.01-20, the staff requested that the applicant describe how the flow loss coefficients were determined. In a September 30, 2009, response to RAI 221, Question 06.02.01-20, the applicant stated that the flow loss coefficients were determined using a conservative application of the methods from the "Handbook of Hydraulic Resistance," by E. I. Idelchik, 3rd Edition, Begell House, 1996, ISBN 1-56700-074-6, (Diagram 4-18 on Page 225). The staff performed a sensitivity study using a multi-node MELCOR model of the U.S. EPR to assess the affect of increasing the flow resistance through the dampers by a factor of two. The increased flow resistance had little effect on the result. Therefore, the staff considers RAI 221, Question 06.02.01-20 resolved.

In a December 2, 2009, response to RAI 1, Question 06.02.01-1, Part b, the applicant confirmed that both the RELAP5-BW and GOTHIC codes were developed in accordance with a quality assurance program compliant with 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," such that when calculations are being performed, procedures for treating code input arising from the plant geometry and the assumed plant state at transient initiation are defined through methodology guidelines. Similarly, outputs are confirmed for accuracy and relevancy to the analysis objectives, and compared to the inputs for consistency.

In order to assess the capability of a code model to predict the important phenomena, evaluations against special effects test (SET) and integral effects test (IET) data should be performed. The GOTHIC code has been validated against an extensive set of test data by the code developers as described in NAI 8907-09, Revision 10. Since a prototype of a U.S. EPR plant is not available for destructive LBLOCA tests, the staff determined that scaling analysis on test facilities should be evaluated in order to properly assess distortions between the test results and the prototype. Therefore, the selection of tests should be such that the important phenomena in a PWR LBLOCA are captured. In addition, the staff determined that selection of tests should be scalable to both the LBLOCA phenomena and the U.S. EPR plant. In RAI 1, Question 06.02.01-1, the staff requested that the applicant address these concerns on the benchmarking of the code.

In a December 2, 2009, response to RAI 1, Question, 06.02.01-1, the applicant provided benchmarks against data from the Heissdampfreaktor (HDR) in ANP-10299, Revision 2. HDR is a decommissioned superheated steam reactor in the Federal Republic of Germany. The HDR test facility is characterized by a steel cylindrical containment consisting of a lower section divided into 70 subcompartments, and a large upper dome section. The characteristics of the HDR test facility are as follows:

- Large-scale test facility, cylindrical steel shell containment, typical of compartment arrangements; geometrical similarity not preserved.
- Volume scaling of approximately one-sixth compared to a full-size PWR.
- The energy release rates scaled to preserve power/volume ratio are as expected for a full-size PWR.

- The time of the mass and energy release rates is preserved, resulting in the preservation of typical pressurization transients.

- As-measured mass and energy release rates are cross-checked by supplementary blowdown calculations.

- The parameters available for comparison include five local compartment absolute pressures and eight pressure differences between compartments.

AREVA Technical Report, ANP-10299P, Revision 2, Section 12, “Appendix B – U.S. EPR Scaling Analysis,” presents a series of blowdown tests that were performed at the HDR facility beginning in the late 1970s and continuing through the late 1980s. One of the blowdown tests (T31.5) was selected by the Committee on Safety of Nuclear Installations (CSNI) as International Standard Problem (ISP) 23 for studying the containment responses of a typical PWR following a large pipe break. This test was selected by the applicant to perform a scaling analysis comparing ISP23 with a postulated DEG-CLPS break for the U.S. EPR.

In AREVA Technical Report, ANP-10299P, the applicant developed a set of equations to describe mass, momentum, and energy conservation within the RCS, within the equipment space, and within the service space for the various phases of a LOCA. These were divided into the blowdown phase (0-35 seconds), the pre-hot leg injection phase (35 seconds to 60 minutes), and the hot leg injection phase (after 60 minutes). The conservation equations were made non-dimensional with non-dimensional coefficients defined as various Π groups. The applicant then compared the magnitude of the Π groups calculated for U.S. EPR with those calculated for ISP23. The evaluation included containment parameters, as well as the magnitude of the mass and energy source.

The relative magnitude of the Π groups represents their importance for containment analysis. Those with the highest relative value were given a high importance (H). The relative magnitude of the U.S. EPR Π groups was compared with those of the HDR test facility. Those that were significantly different were given a distortion level of (H).

For the blowdown period, 10 of 38 containment Π groups were found to have an importance of H and a distortion of H.

For the period before hot leg injection (HLI) period, five of 38 Π groups were found to have an importance of H and a distortion of H

For the HLI period, one of 10 Π groups was found to have an importance of H and a distortion of H.

Given that several of the significant Π groups were not scaled well between the U.S. EPR containment and HDR containment test ISP23 HDR/T31.5, in RAI 368, Question 06.02.01-58, the staff requested that, for those Π groups that were found to have a high or medium significance and a high or medium distortion between U.S. EPR and HDR, the applicant provide justification that the U.S. EPR GOTHIC containment model adequately represents the associated phenomena represented by these Π groups.

In a June 9, 2010, response to RAI 368, Question 06.02.01-58, the applicant stated that the distortions resulted principally from the relatively smaller blowdown phase of the HDR experiment and the shorter duration of the steam release relative to the mass and energy releases assumed in the U.S. EPR large break containment analyses. In addition, distortions were attributed to the shorter flow path length and smaller open areas inside the HDR containment as compared with the U.S. EPR containment. The applicant contends that distortions in Π groups involving gas expansion are not significant because: (1) These relationships are well known; distortions involving heat structure area are not significant and because the DLM that is used in GOTHIC is not sensitive to heat structure geometry; (2) the distortions involving turbulence are not significant since the turbulent kinetic energy (k) and its heat dissipation rate (ε), that is the k Epsilon turbulence model, have been verified as described in the GOTHIC qualification report NAI 8907 09, Revision 10; and (3) distortions affecting momentum are not significant since the momentum equations used in GOTHIC are based on first principle and standard non-dimensional loss coefficients. The staff agrees with these statements based on the information provided by the applicant. The staff also performed confirmatory calculations using the independently developed MELCOR and the APROS computer codes, which confirmed the calculations by the applicant are conservative and acceptable for the safety analyses of U.S. EPR except for the unresolved issues noted in this report. The staff's confirmatory calculations are discussed in more detail below.

Containment Analysis for Loss of Coolant Accidents

Calculational Methods

For LOCA evaluation, the applicant elected to develop containment pressure and temperature calculations in two steps:

- The RELAP5-BW code is used to model the reactor system through blowdown, refill, reflood, and early post reflood (short term). The code calculates mass and energy release for the short term.

- The GOTHIC code simulates the containment. It is being used for mass and energy calculations for the remainder of the accident (long term) and for containment pressure and temperature calculations for both short and long terms. The short-term mass and energy releases are input into a GOTHIC simulation of the Containment Building as a table. The long-term mass and energy releases are internally transferred within GOTHIC between the reactor system and the containment simulations. Similarly, the containment pressure, which affects the long-term mass and energy release, is transferred between the GOTHIC containment and the reactor system simulations.

The LOCA calculations using RELAP5-BW for the short term, as well as the GOTHIC model for calculating the long-term mass and energy release, are discussed and evaluated in Section 6.2.1.3 of this report.

Multi-Node Analysis by the Applicant

Results of three large-break multi-node LOCA containment pressure and temperature calculations are presented in FSAR Tier 2, Section 6.2.1. All three breaks are double-ended guillotine breaks, one in the hot leg, one in the pump suction pipe of the cold leg, and one in the pump discharge pipe of the cold leg. The assumed single failure is loss of one ECCS division in each case. The applicant assumes offsite power is not available. Along with pressure results, the applicant presents temperature histories for the break node, various nodes in the dome, and for a few other selected nodes.

The staff notes that the hot leg break calculations predict a sharp pressure rise followed by continuous pressure decay until initiation of hot leg injection. With initiation of hot leg injection, some of the ECCS water spills onto the containment floor instead of traveling through the core and mixing with steam in the reactor vessel. This results in calculated containment pressure increasing for a limited time. The calculated peak containment pressure reached in case of the hot leg break is 480.6 kPa (69.7 psia) at 26 seconds into the accident. The peak temperature is calculated to occur in the containment dome at the 25.3 m (83 ft) elevation. The peak value is 170 °C (338 °F) at 12 seconds.

The staff notes that the calculation presented for the pump discharge cold leg break shows a fast pressure rise to approximately 448.2 kPa (65 psia) at 25 seconds followed by a decrease in pressure. At around 2 minutes, the calculated pressure rises again and reaches a new broad peak of 471.6 kPa (68.4 psia) around 30 minutes. The slow decay of pressure that follows accelerates when hot leg injection starts at 60 minutes.

The information provided by the applicant in FSAR Tier 2, Section 6.2.1, shows that there is a subsequent break in the calculated pressure curve around 4 hours. The second peak is the result of the heat balance; the model predicts that,temporarily, more heat is put into the containment than is removed. The second predicted peak is higher, and it is controlling. At least in this case, the predicted pressure rise is terminated before hot leg injection takes place. The predicted sharp decrease in containment pressure at 4 hours is due to the rapidly decreasing enthalpy of the steam released from the SG side of the break. The maximum predicted containment pressure of 471.6 kPa (68.4 psia) is reached at 31 minutes. The calculated peak containment temperature is 173.9 °C (345 °F) at 80 seconds in the break node. The maximum predicted containment dome temperature is 157.2 °C (315 °F).

The staff notes that the analysis of the double-ended pump suction line break shows similar behavior as the pump discharge break with one significant difference. Emergency coolant is assumed to fill the loop seals in the intact loops 20 minutes into the event. From there on, the analyses model predicts steam flow to the containment by a path that circumvents cold ECCS injection water. This produced a further increase in calculated containment pressure. The peak containment pressure is calculated to reach 477.8 kPa (69.3 psia). The highest predicted vapor temperature reported is 157.2 °C (315 °F). Acceptable mitigation of the containment consequences of large cold leg break events for which the loop seals become blocked depends on manual action by plant operators. The operators need to initiate hot leg injection within a particular time window in order for this action to succeed. The staff's evaluation of operator action to initiate hot leg injection is discussed in Sections 15.6 and 6.2.1.3 of this report.

The staff paid special attention to analyses of surge and spray line breaks in the pressurizer compartment. The pressurizer compartment is somewhat isolated from the rest of the equipment space, has no foils in its ceiling, and a break could be at a higher elevation than other LOCAs. Detailed containment design information was provided by the applicant dated March 23, 2010. "SUNSI and Proprietary Material Requested from the March 2-3 Audit Regarding U.S. EPR Design Certification Containment Analysis." This information revealed that the pressurizer rooms of this compartment at two different elevations have severe restrictions limiting upward flow. In RAI 368, Question 06.02.01-68, the staff requested that the applicant address the staff's concerns with the pressurizer compartment. As described in an October 13, 2010, response to RAI 368, Question 06.02.01-68, the applicant added descriptions of six safety-related pressurizer compartment doors to the multi-node GOTHIC containment model. The applicant performed analyses of a double ended surge line rupture and spray line at breaks both high and low elevationswithin the pressurizer compartment. Although the purpose of these analyses was to evaluate venting capability, peak containment pressures were calculated which, in all cases, were bounded by the result of the hot and cold leg LOCA and MSLB cases analyzed by the applicant.

Since the purpose of the pressurizer compartment analyses was to model vent capability, the dynamics of door opening was not modeled. The effect of door opening on the short-term pressure response within the pressurizer compartment was evaluated by the applicant as discussed in Section 6.2.1.2 of this report. Containment analyses were performed with two, three, four, and five doors opening in case of a surge line break, which is the largest high energy line within the pressurizer compartment. With two doors opening, pressure in the surge line room was predicted to increase rapidly, and the increase continued even after opening of the doors. In the analysis with five doors, once the doors are opened, the pressure was predicted to decrease. The six doors were identified as "safety doors" and were placed into the equipment qualification program. The safety doors are designed to open when the pressure differential on the doors exceeds a set value. The staff determined that the safety doors should be shown to be capable of meeting the safety goals assumed in the applicant's analyses. The doors should be subject toITAAC and Technical Specification requirements. In RAI 368, Questions 06.02.01-70 and 06.02.01-71, the staff requested that the applicant provide this information. **RAI 368, Questions 06.02.01-70 and 06.02.01-71 are being tracked as open items.**

The staff notes that most of the doors inside containment are non-safety-related. These doors are expected by the applicant to open at various pressures. The staff notes that opening of some or all of the doors is expected to produce more mixing in the containment atmosphere and increase heat transfer. Since the doors are non-safety-related, the applicant's multi-node analyses (except the pressurizer compartment break studies) were performed assuming all doors remain closed.

While the large-break LOCAs are limiting in terms of peak containment pressure and peak containment temperature, small-break LOCAs could play a limiting role in development of the temperature profile to be used for equipment qualification and in the design of the CONVECT system. FSAR Tier 2, Tables 6.2.1-6, "Containment Response to Hot Leg Breaks," and 6.2.1-8, "Containment Response to Cold Leg Pump Discharge Breaks," Revision 2, present peak containment pressure and peak containment temperature predictions from five small-break LOCA analyses. The purpose of the calculations is the development of the equipment qualification temperature profile. Small breaks are expected to produce a lower containment pressure than the double-ended guillotine breaks but tend to extend the time when the containment atmosphere would be at a high temperature. Three break sizes were analyzed (22.86 cm, 15.24 cm, and 7.62 cm (9 inches (in.)., 6 in., and 3 in.)) in the cold leg pipe, and one break in the hot leg pipe (7.62 cm (3 in.)). All but one of the calculations take credit for both rupture foils and convection foils opening when the burst pressure is reached. The calculations predict that peak temperatures occur much later for small breaks than for large breaks, especially for the 7.62 cm (3 in.) breaks. The results predict that the cold leg 7.62 cm (3 in.) break peak temperature is at 7,000 seconds, and the hot leg break peak temperature is at 12,000 seconds.

The applicant has provided only the peak pressure and temperature and the times of the peak values for the small breaks. In RAI 378, Question 06.02.01-90, the staff requested that the applicant provide additional details for these calculations including:

- Temperature and pressure profile for a typical small break case, including the temperature of the limiting node, temperature at the top of the dome, and the temperatures of the two lower annulus nodes

- A comparison of the pressure profiles of the five small break cases analyzed in the FSAR

- A comparison of the temperature profiles of the limiting nodes of the five cases

- The effect of break location on the results of small-break LOCA, in particular the effect of breaks in the service space as well as those in the equipment space

This information was provided by the applicant in the response to RAI 378, Question 06.02.01-90. The applicant provided the results from five small break cases for breaks in a reactor system cold leg and in a hot leg. Both peak pressure and peak temperature were calculated to decrease as break size decreased. The highest temperature was calculated to occur at the top of the containment dome but was limited to the temperature of saturated steam. The smallest break analyzed was a 3-inch hot leg break. For this postulated break, the applicant performed analyses (1) assuming that both the rupture and convection foils opened and (2) assuming that only the convection foils opened when the containment temperature rose sufficiently to melt the fusible length. These analyses resulted in virtually the same containment pressure and temperature response and demonstrate that the convection foils, when open, are capable of providing adequate containment circulation and mixing for very small break sizes that do not produce sufficient differential pressure to break open the rupture and convection foils.

The applicant evaluated reactor system piping traversing the service space. This piping consists of portions of the residual heat removal system and the safety injection system, and chemical and volume control system (CVCS) injection lines and the CVCS letdown line. The RHR is not operated when the reactor system is at high pressure. The SIS and CVCS injection lines are protected by check valves from potential breaks in the service space and the CVCS letdown line is protected by redundant fast closing isolation valves. Therefore, the staff agrees with the applicant's assessment that small breaks within the service space are not limiting accidents and need not be analyzed further.

Scoping Studies

In FSAR Tier 2, Tables 6.2.1-6, 6.2.1-7, "Containment Response Cold Leg Pump Suction Breaks," and 6.2.1-8, Revision 2, the applicant presents the results of 38 LOCA containment calculations performed using a single-node containment model. The calculations covered all three potential RCS pipe locations: hot leg, cold leg pump suction side, and cold leg pump discharge side. As explained below, the applicant's single-node GOTHIC containment model was found by the staff to be non-conservative for the calculation of the long-term containment response to a LOCA event.

The applicant states that the purpose of the calculations is to establish the limiting scenarios for the multi-node analysis. The scoping studies covered two different postulated single failures and three different break discharge coefficients. Also, calculations were performed by the applicant with the assumption of loss of offsite power and without this assumption.

The applicant states that containment pressure profiles following LOCAs typically exhibit two peaks: One peak shortly after the break (20 to 40 sec.) due to the initial blowdown of the RCS, and a second peak later (approximately an hour into the accident). The second peak depends on the balance between heat release into the containment from all sources and heat removal by the ECCS and various containment heat sinks. The single-node GOTHIC model provides reasonable results for the first pressure peak. In large break cases, this peak depends almost entirely on the energy stored in the water of the RCS, which is well known. The staff noted that the second peak is more complex, and depends not only on the momentary heat balance between heat sources and heat sinks, but also on the previous history of the heat balance. The single-node model over estimates heat transfer to the various heat sinks in the long term because complete mixing and condensation on all surfaces are inherent assumptions in the model. Results of the single-node calculations beyond the first peak are therefore not reliable and should not be used in the U.S. EPR containment safety evaluation, not even for scoping studies.

Since the single-node GOTHIC calculations are always first peak limited and the first peak is controlled by the energy stored in the RCS, these calculations give the same results when other parameters are varied. The results of the applicant’s scoping calculations, in terms of maximum containment pressure and temperature, show that all breaks (hot leg, cold leg suction side, cold leg discharge side) under all circumstances (single failure, discharge coefficient, offsite power) have nearly the same results. The calculated peak containment pressures are within 20.7 kPa (3 psi) for the 38 cases analyzed by the applicant. The corresponding calculated peak containment temperatures are within approximately 1.7 °C (3 °F).

Hot leg breaks result in a fast, sharp first peak which is always controlling. Accordingly, the staff notes that the single-node scoping studies are useful for these breaks. However, the staff also notes that the important part of the cold leg pressure and temperature profiles is the second peak. For these cases, the second peak can exceed the first peak, and in some cases, operator action is needed to keep containment pressure and temperature within acceptable limits. For these breaks, the single-node model is not sufficient to provide insight for establishing limiting scenarios. The staff determined that the applicant need to provide scoping calculations with the multi-node GOTHIC model to select the limiting cold leg break size, location, and single failure assumptions. The staff also determined that these calculations should not credit any equipment which is not safety-related. In RAI 378, Question 06.02.01-93, the staff requested that the applicant address these concerns. **RAI 378, Question 06.02.01-93 is being tracked as an open item.**

Staff LOCA Confirmatory Calculations

The unique features of the U.S. EPR containment design (two-room containment, lack of active containment atmospheric heat removal) raised issues which were not studied previously by NRC staff. The main issues raised by the staff were:

- Would the two-room containment respond as a single volume during the initial pressure rise?

- Would compartment designs and the design of the dampers and foils create effective circulation patterns within the containment to promote heat transfer?

- Can the ECCS and the passive heat sinks, without the help of fan coolers and containment spray, terminate the pressure rise and reduce containment pressure as required?

As described above, the U.S. EPR containment analysis presented in FSAR, Revision 0, was based on single-node GOTHIC calculations. The staff notes that the use of a single-node calculation raised a concern on whether the issues listed above could be properly addressed by single-node calculations.

On behalf of the NRC, VTT Technical Research Centre of Finland (VTT) performed a set of containment calculations for the U.S. EPR using the APROS computer code (reference VTT Research Report VTT-R-04252-09, "APROS Calculations of U.S. EPR Containment Behavior during Large Break LOCA and Main Steam Line Break Sequences," Revision 1.0). APROS is capable of describing a reactor CB using a multi-node simulation similar to the GOTHIC code used by the applicant and the MELCOR code used by the staff (reference NUREG/CR-6119, "MELCOR Computer Code Manuals," Version 1.8.3, Volumes 1 and 2). APROS contains methodology for calculating heat and mass transfer to the containment internal heat structures based on natural circulation correlations using local conditions, and on condensation determined from the heat/mass transfer analogy. VTT describes this model as a realistic approach. The APROS realistic approach is similar to the DLM programmed into the GOTHIC code and to the heat/mass transfer package in the MELCOR code. The purpose of the APROS study was development of a computer model that can predict flow distribution, flow mixing, and heat transfer in the containment following design basis accidents. Explicit representation of the CONVECT system in the model was a prerequisite. Sensitivity studies were run to evaluate the effectiveness of the foils and mixing dampers, opening of doors within the containment, the effect of heat transfer assumptions, and the effect of the elevation of the break.

Two different APROS nodalization schemes were developed: a single-node model and a 41 cell multi-node model. Two different accident scenarios were analyzed: (1) a large-break LOCA and (2) a main steam line break. The results were reported in terms of calculated containment pressure and calculated containment temperature. For the studied cases, when containment pressure increases, usually containment temperature also increases. The discussion below typically addresses predicted changes in containment pressure. It should be understood that similar changes occur in containment temperature.

In the LBLOCA sensitivity analysis, a double-ended break in the cold leg pump suction line was assumed. A number of useful insights were gained from the studies:

- In case of the LBLOCA, the foils were predicted to burst in all four steam generator compartments rather quickly (less than a second), and the mixing dampers were predicted to open within a few seconds. The analysis predicted that the entire free volume of the containment participated in the initial pressure rise. The first pressure peak was calculated to be reached in 30 to 40 seconds.

- Good mixing was predicted in most parts of the containment during the first hundred seconds of the accident. After 100 seconds, the pressures equalized, and local flows subsided. Nevertheless, certain flow patterns were predicted to continue. The pressurizer compartment (and possibly some equipment compartments) is an exception. Mixing in the pressurizer compartment was predicted to be limited. There are no foils in the ceiling of the pressurizer compartment.

- Safety injection was switched in the APROS simulations from the cold legs to the hot legs 90 minutes after the break occurred. (After completion of the APROS sensitivity studies, the applicant changed the hot leg injection time to 60 minutes after the break occurred resulting in a lower peak in the containment pressure.) The beginning of hot leg injection was found to have a very pronounced effect on the containment response. The rising containment pressure and temperature were predicted to be almost immediately turned over in the APROS calculations and drop sharply.

- Before the start of hot leg injection, containment pressure and temperature were predicted to be rising with no indication of a leveling off in the pressure increase. Initiation of hot leg injection calls for operator action. The predicted pressure rise would be terminated only if the operators manually transfer most of the safety injection flow to the hot legs in a timely manner.

- The time history of the analyzed LOCA shows two pressure and temperature peaks. The first peak is determined by the energy stored in the primary coolant that is modeled to blow into the containment during the first 40 seconds. This energy is well defined by the size and operating temperature of the RCS. The second peak occurs later, at the time when hot leg safety injection is modeled to start. The second peak depends on many variables including the performance of the CONVECT system and the time when hot leg injection is initiated. When doors within containment were modeled to open at the expected pressure differentials, flow patterns were predicted to change, and more mixing was predicted. Predictions showed both the first and second pressure peaks were reduced, as were peak temperatures, when compared to the results when the doors were modeled not to open.

- The two main components of the CONVECT system are the foils and mixing dampers. The foils are passive components. The applicant contends that the dampers are also passive components, as they are opened by springs and require no external power to perform their safety function. The APROS calculations indicate that should the dampers fail to open, containment pressure and temperature would increase by approximately 55.2 kPa (8 psi) and 13.9 °C (25 °F), respectively.

- The assumed LOCA break location was at a low elevation in the containment. An additional case was run placing the break arbitrarily in the top node of the steam generator compartment. The result was an increase (more than 68.9 kPa (10 psi)) in calculated peak containment pressure. These results led the staff to examine MSLBs which could occur at higher elevations.

- Thermal stratification was predicted between the dome and the lower levels. The higher the postulated break location was, the stronger the predicted stratification. Accumulation of non-condensables (air, nitrogen gas) was predicted in the lower nodes of the service space. The presence of non-condensables reduces condensation on the walls.

- Predictions made by the realistic 41-node model were compared with the results of the single-node model. The single-node model predicted a higher initial pressure peak but under-predicted the pressure rise for the second peak. This was expected, because the single-node model assumes complete mixing, and good heat transfer on all surfaces. Beyond the initial peak, the single-node model led to non-conservative results.

These insights, developed by the staff, were utilized in the review of the applicant's original single-node analyses and subsequent multi-node containment analyses.

Using the input from the VTT 41-node APROS model, the staff developed an equivalent input for use with the MELCOR containment analysis code (reference NUREG/CR-6119). The staff first benchmarked the MELCOR model against the APROS results for the postulated DEG-CLPS break assuming hot leg injection after 90 minutes. Similar results to those from APROS were obtained. The staff next compared the predictions of the 41-node MELCOR model with the applicant's GOTHIC predictions for the same postulated accident but assuming hot leg injection at 60 minutes. The staff's application of the MELCOR model obtained results similar to those of the applicant, which show the applicant's 30-node GOTHIC model to be conservative in predicting both peak containment temperature and pressure for the postulated DEG-CLPS break. The consistency of results between the applicant's GOTHIC model and the independently developed APROS and MELCOR models gives confidence in the accuracy of the result and shows the result to be neither computer code nor input dependant.

The staff developed a second MELCOR model based on the applicant's containment description in the 30-node GOTHIC input. Results from this model indicated that the applicant's evaluations using GOTHIC are conservative for the postulated DEG-CLPS large-break LOCA analyses in comparison with the MELCOR prediction. The staff noted that the applicant described the vertical circulation path in the service space using vertical stacks that were only three nodes high. The staff was concerned that additional noding detail would be needed in the service space in order to properly evaluate thermal stratification which might occur following a piping rupture. By studying containment drawings and compartment design information, the staff developed a 42-node MELCOR model with additional noding detail in the middle region of the service space. Basically, the middle service space node was split into four nodes on one side of the containment and into eight nodes on the other side of the containment. The revised model was used to evaluate the DEG-CLPS break. This break was selected since it provides an elevated long-term steam source to the containment and thus might be more affected by containment atmospheric stratification. The result was that calculated containment peak pressure was little affected by the additional noding. The peak pressure calculated by both the 30- and the 42-node MELCOR models remained below that calculated by the applicant for the U.S. EPR using GOTHIC. Staff confirmatory analyses for postulated main steam line breaks are discussed in the next section of this report.

Containment Analysis for Steam Line Break Accidents

Calculational Method

The MSLB containment analysis was performed by the applicant in two steps: (1) Mass and energy release were calculated by the RELAP5-BW code and (2) containment pressure and temperature were calculated using the GOTHIC code. This section addresses the GOTHIC methodology used for calculating containment response in case of a MSLB. The mass and energy release calculations are discussed and evaluated in Section 6.2.1.4 of this report.

The applicant used the same 30-node GOTHIC model of the containment for the MSLB analyses as was used for LOCA analyses. The CONVECT system was explicitly modeled. With the GOTHIC multi-node representation of the containment, the applicant used the DLM method for heat transfer. No enhancements to account for film roughness and mist formation in the condensing boundary layer were utilized, resulting in conservative, reduced heat transfer. The DLM method was accepted by the staff as described in the NRC letter to Nuclear

Management Company, "Kewaunee Nuclear Power Plant – Issuance of Amendment (TAC NO. MB6408)," September 29, 2003.

As described in NAI 8907-09, Revision 10, the GOTHIC computer code using the DLM with the film and mist enhancements was used to correlate the Carolinas-Virginia Tube Reactor (CVTR) test data. NAI 8907-09 provides graphs of both the CVTR test data and the GOTHIC code predictions. Both a two-node lumped parameter model and a multi-node model are described. Results that were generally conservative were obtained using both simulations with generally better results from the multi-node model. These results, as well as other experimental data comparisons presented in NAI 8907-09, Revision 10, confirm the validity of using the DLM to simulate heat and mass transfer to containment internal heat structures using a multi-node GOTHIC simulation.

Multi-Node Analysis by the Applicant

FSAR Tier 2, Section 6.2.1.4.1.1, "Plant Power Level," states that steam line breaks were postulated to occur with the plant in operating conditions ranging from hot shutdown to full power. Calculations were performed at seven initial power levels: Zero percent; 20 percent; 40 percent; 50 percent; 60 percent; 80 percent; and 100 percent and at four different break sizes (DEG, 0.093 m^2 (1.0 ft^2), 0.065 m^2 (0.7 ft^2), 0.048 m^2 (0.52 ft^2) and 0.028 m^2 (0.3 ft^2)). The U.S. EPR steam generators are equipped with an integral flow restrictor within the exit nozzles. Thus, the effective break area of the DEG break is 0.13 m^2 (1.4 ft^2). Both guillotine and split break types were considered. An extensive scoping study was conducted to find the worst break size and initial power level. Containment pressure and temperature calculations were performed for all 43 MSLB cases using the 30-node GOTHIC model.

FSAR Tier 2, Section 6.2.1.4.1.2, "Main Feedwater System Design," states that the single failure evaluation considered failure of a main feedwater (MFW) isolation valve, failure of a MFW control valve, failure of the MFW pumps to trip, and failure of a main steam isolation valve (MSIV). The U.S. EPR has redundant safety-related MFW isolation valves. Consequently, failure of one valve or pump trip failure has limited consequences. The worst single active failure was found to be failure of the MSIV in the broken line. All containment analyses were performed assuming this failure. Failure of a main steam relief control valve (MSRCV) was not considered in calculating the mass and energy releases for containment analysis. See the discussion below.

FSAR Tier 2, Section 6.2.1.4.1.3, "Emergency Feedwater System Design," states that actuation of the emergency feedwater (EFW) system increases the water mass available in the affected steam generator, cools the steam generator inventory, and absorbs heat from the steam generator tubes. These are compensating effects. In the case of large breaks, peak pressure and temperature occur early. Typically, the cool EFW reduces containment pressure and temperature. In small-break events, peak pressure and temperature occur later. The additional mass release to the containment over time increases peak temperature and pressure. The scoping cases were analyzed with EFW actuated. The effect of EFW isolation to the effected steam generator was investigated.

The staff notes that steam line breaks could occur either in the equipment space or in the service space. Typically, breaks at higher elevations produce higher temperatures. The scoping calculations assumed a failure in the equipment space, at the highest elevation of the steam line.

FSAR Tier 2, Section 6.2.1, shows that among the cases analyzed in the scoping study, the double-ended guillotine break starting from 20 percent initial power level produced the limiting temperature and pressure for the design of the U.S. EPR containment. Once this scoping study was completed, the limiting case (DEG, 20 percent initial power, EFW actuated, break in equipment space) was repeated with no EFW supplied to the affected steam generator. The applicant found both the pressure and temperature results to be less favorable for this break size and location when no EFW was supplied. An additional calculation was done with no EFW supplied, and the break placed in the service space at the highest point of the steam line. This last calculation produced the highest containment peak pressure and peak temperature.

In Tier 2, Table 6.2.1-9, "Peak Containment Pressure and Temperature for MSLB," the maximum peak containment pressure calculated by the applicant is 457.8 kPa (66.4 psia or 51.7 psig). The highest calculated containment atmospheric temperature is 248.6 °C (479.5 °F). This was calculated for a postulated DEG break at 20 percent power. The break was assumed to be where the main steam lines traverse the service space between the outer containment wall and the steam generator compartments. The containment design pressure and the peak containment design temperature are 427.5 kPa (62 psig) and 170 °C (338 °F), respectively. The calculated containment pressure compares favorably with the design pressure. The calculated vapor temperature exceeds the design temperature. Since heat removal is primarily by condensation on the walls, the applicant's position is that the wall temperature will not exceed the saturation temperature. At the calculated peak pressure, the saturation temperature is approximately 148.9 °C (300 °F), well below the peak design temperature of the containment. Wall temperatures calculated by the applicant were examined by the staff during an audit on February 9, 2010. The staff searched for wall temperatures in excess of the design temperature. None were found, nor were compartment wall temperatures calculated to be above the design temperature in the staff's confirmatory calculations. The high temperatures might affect the operation of vital electrical equipment within the containment. Therefore, in RAI 368, Question 06.02.01-80, the staff requested that the applicant update the equipment qualification curve by which the equipment is designed to operate. **RAI 368, Question 06.02.01-80 is being tracked as an open item.**

In RAI 368, Question 06.02.01-77, the staff requested that the applicant provide additional details concerning the postulated DEG MSLB within the service space from 20 percent power which was calculated to produce the limiting containment temperature and pressure. In a June 9, 2010, response to RAI 368, Question 06.02.01-77, the applicant provided the requested details and stated that for the calculation, the rupture and convection foils at the top of the steam generator compartments were assumed not to open on differential pressure. Instead, the equipment space containing the steam generators was assumed to be isolated from the service space until the convection dampers between the service space and the top of the IRWST pool opened at 18 seconds. Once the dampers open, steam from the break was calculated to enter the equipment space and to melt the fusible link of the convection foils which caused them to open. With both the dampers and convection foils open, circulation was calculated to begin throughout the containment, and the pressure and temperature were predicted to decrease. The staff believes that the delayed opening assumption for the foils which the applicant assumed for this analysis is very conservative. FSAR Tier 2, Revision 2, Section 6.2.2, "Containment Heat Removal Systems," states that the foils will open in either direction. The staff understands that the foils will be tested and verified to perform as described in the FSAR. The staff's review of the rupture and convection foils and the convection dampers is described in Section 6.2.2 of this report.

The applicant's MSLB analyses resulted in higher containment temperature than the LOCA analyses. This is at least partially due to the higher elevation of the break and the resulting reduced circulation, decreased mixing, and more pronounced stratification. Thermal stratification in the case of the limiting break was calculated to produce a temperature differential of 44.4 °C (80 °F) between the dome and lower portion of the containment and to persist for more than 24 hours. Air is forced down into the lower part of the containment where it interferes with heat transfer. The stratification and heat transfer rate calculated by the applicant using the GOTHIC code were compared to the staff's confirmatory calculations. GOTHIC under predicted the stratification differential in comparison with the staff's confirmatory calculations using MELCOR and APROS, both of which included more nodes in the annular region of the service space. The peak temperatures and pressures calculated by the staff were lower than those calculated by the applicant however, which indicates that overall the applicant calculations are conservative. Nevertheless, in RAI 368, Question 06.02.01-73, Part a, the staff requested that the applicant investigate the effect of additional noding detail for the service space. In a December 6, 2010, response to RAI 368, Question 06.02.01-73, Part a, the applicant performed a sensitivity study in which the number of axial nodes in the service space was increased from three to eight. The increase in noding did not cause the calculated maximum pressures and temperatures to increase above those previously calculated.

In development of the multi-node MELCOR model, the staff noted discrepancies between the applicant's GOTHIC model and the detailed containment design description. In follow-up RAI 437, Questions 06.02.01-99 and 06.02.01-100, the staff requested that the applicant address these discrepancies. **RAI 437, Questions 06.02.01-99 and 06.02.01-100 are being tracked as open items.**

The staff notes that small steam line breaks in the service space would act to increase the service space temperature but might not affect the temperature in the equipment space sufficiently to cause the convection foils to open. The convection foils contain a fusible link which is located below the PEC and which opens when the temperature below the PEC reaches that required to melt the link. Therefore, in RAI 378, Question 06.02.01-84, the staff requested that the applicant analyze small breaks outside the equipment space and that the time assumed for opening of the foils be provided along with the supporting analyses and justifications that conservative results are obtained. In an August 31, 2010, response to RAI 378, Question 06.02.01-84, the applicant provided an analysis of a small steam line break showing that opening only a few of the rupture foils would be sufficient to cause the convection foils to open on high temperature. Both the rupture and convection foils open on a differential pressure of 4.8 kPa (0.7 psia). The circulation dampers open at a total containment pressure of 120 kPa (17.4 psia). Calculations by the applicant show that even a very small break would be sufficient to actuate both the foils and dampers. The applicant analyzed small steam line breaks down to 4.65 cm^2 (0.005 ft^2) finding both the foils and dampers to open. Therefore, the staff considers RAI 378, Question 06.02.01-84 resolved.

In addition to the MSLB analyses presented in FSAR Tier 2, Section 6.2.1.1, the applicant presented another set of MSLB analysis in FSAR Tier 2, Section 15.1.5, "Steam System Piping Failures Inside and Outside of Containment (PWR)." The purposes of the two calculations are different. FSAR Tier 2, Chapter 15, "Transient and Accident Analyses," addresses potential fuel damage and radiological consequences, while FSAR Tier 2, Chapter 6 is concerned with containment integrity. Consequently, there are a number of differences between the two sets of MSLB calculations. In FSAR Tier 2, Chapter 15 cases, return to power was predicted, while FSAR Tier 2, Chapter 6 calculations showed no return to power. The FSAR Tier 2, Chapter 15 calculation assumes that the MSRCV on one of the unaffected main steam line fails in the fully opened position. FSAR Tier 2, Chapter 6 analysis is done with the assumption that the MSIV on the main steam line of the affected steam generator fails. Failure of an MSRCV is not addressed in FSAR Tier 2, Section 6.2.1. Failure of an MSRCV results in the partial blowdown of a second steam generator in addition to the affected steam generator, increasing the RCS cooldown. However, steam from the second steam generator is released outside containment. Due to these competing effects, it is not clear, a priori, which single failure will result in higher containment pressure or temperature or a more limiting temperature history to be used for equipment qualification. Therefore, in RAI 266, Question 06.02.01.04-4, the staff requested that the applicant analyze the containment pressure response from MSLBs using the same core reactivity and single failure assumptions as used in FSAR Tier 2, Chapter 15. In an August 4, 2010, response to RAI 266, Question 06.02.01.04-4, the applicant stated that the assumptions made in the FSAR Tier 2, Chapter 15 analyses were overly conservative for use in the containment analyses for several reasons. The principal excess conservatism was in the amount of main feedwater which can be added before feedwater isolation occurs. As previously stated for FSAR Tier 2, Chapter 6 containment analyses, the applicant assumes full feedwater flow until closure of the feedwater isolation valves using the longest delay time and the longest isolation valve stroke time of 40 seconds. The main feedwater addition assumed in the FSAR Tier 2, Chapter 15 analyses was much larger. Using the same assumptions for feedwater addition as assumed in FSAR Tier 2, Chapter 6 in the S-RELAP5 code, the applicant provided a comparison of the containment pressure response with those of the MSLB calculated to produce the highest containment pressure in FSAR Tier 2, Chapter 6. The FSAR Tier 2, Chapter 6 analysis was shown to be bounding. The staff notes that even for the containment analyses performed using the feedwater addition assumptions of the FSAR Tier 2, Chapter 15 analyses, the containment pressure was shown to be less than design by a considerable margin indicating the degree of conservatism in the U.S. EPR containment design. Although the peak calculated temperature was lower when the FSAR Tier 2, Chapter 15 assumptions were utilized, the long-term temperatures decreased more slowly than when the FSAR Tier 2, Chapter 6 assumptions were utilized. In RAI 437, Question 06.02.01-98, the staff requested that the applicant address how the EQ curve would be affected.

The staff notes that safety-related equipment located in the containment must be able to perform its function as long as needed. The applicant developed a temperature profile to be used for qualification of safety-related equipment. This temperature profile is the envelope of all individual temperature profiles generated for LOCAs and for MSLB accidents. The profile is presented in FSAR Revision 0. Since publishing of the original FSAR, the various LOCA and MSLB scenarios have been recalculated; new temperature curves have been generated. The calculated containment temperatures are higher than the original values, particularly for an MSLB. The equipment qualification temperature profile should be updated to envelope the updated calculations. Therefore, the applicant needs to prepare an updated equipment qualification temperature profile for equipment located in the containment. **RAI 437, Question 06.02.01-98, which is associated with the above request, is being tracked as an open item.**

Staff MSLB Confirmatory Calculations

As documented in VTT Research Report VTT-R-04252-09, VTT performed, on behalf of the NRC, MSLB confirmatory calculations. A double-ended break was postulated at the highest point of the steam lines within the containment dome. The same 41-node APROS model developed for LOCA analysis was used for the steam line break cases. The CONVECT system was explicitly represented. All doors were assumed to stay closed.

Heat and mass transfer to the containment internal structures was calculated using a realistic simulation based on natural circulation heat transfer correlations and condensation determined from the heat/mass transfer analogy. The APROS realistic approach is similar to the DLM programmed into the GOTHIC code and to the heat/mass transfer package in the MELCOR code (reference NUREG/CR-6119). The purpose of the APROS MSLB calculations was to gain an understanding of flow distribution, flow mixing, and heat transfer in the containment, to see if thermal stratification would occur, and to evaluate the effectiveness of the CONVECT system.

The APROS MSLB calculations yielded the following results:

- The foils ruptured within the first second of the calculations. The mixing dampers opened in approximately one second.

- During the short term (up to 100 seconds) superheated steam was released into the dome, and the warm steam remained in the dome. Some of the air from the dome was pushed down into the lower part of the containment through vertical flow paths.

- A clear vertical stratification formed during the short term between the dome and lower parts of the annular region. The stratification became even stronger later and lasted for more than 24 hours.

- Over the long term, a circulation pattern formed in the upper part of the containment; however, due to large temperature differences between the upper and lower part of the containment, the buoyancy effect inhibited gas flow downward into the lower portion of the containment.

- The single-node representation of the dome region in the APROS model prevented both circulation and stratification in the dome. With the dome being a large volume, the expectation is that stratification will occur in the dome. A more detailed modeling of the dome is needed to study this phenomenon. (Such detail for the dome is provided in the applicant's GOTHIC containment model.)

- Containment pressure peaked around 70 seconds, then started to decrease slowly. Later on when the structures were heated, condensation decreased, and a second pressure peak was observed, which was smaller than the initial peak.

- The CONVECT system opened to produce relatively uniform containment pressures. The CONVECT system was less effective in setting up circulation patterns between the containment equipment space and the service space that would act to reduce stratification. Circulation was affected by the elevated location of the steam line break within the Containment Building.

- Gas temperatures in the upper regions of the containment were significantly higher for the MSLB than for LOCAs. The difference existed for the duration of the calculation (24 hours).

- The single-node APROS model was not used for MSLB calculations. Since the multi-node MSLB calculations indicated little mixing, the expectation is that single-node MSLB calculations would not be appropriate for safety analyses. Using the 41-node MELCOR model that the staff developed from the APROS input, the staff performed additional sensitivity studies for the postulated DEG-MSLB at 20 percent power.
  The staff compared the predictions of the MELCOR MSLB model with the applicant's

GOTHIC predictions. Similar results to those of the applicant were obtained, in fact, which shows the applicant's calculations to be slightly conservative in predicting both peak containment temperature and pressure.

To investigate possible uncertainty in the prediction of loss coefficients in the applicant's model, the staff repeated the MELCOR MSLB calculation using twice the flow resistance for the dampers determined by the applicant. The increase in flow loss was found to have an insignificant effect on the calculated pressure or temperature.

In a similar manner, the staff investigated the effect of flow area into the service space. For the sensitivity analysis, the staff reduced the flow area in the 41-node MELCOR model between the containment dome and the service space by a factor of two. The decrease in flow area was found to have an insignificant effect on the calculated pressure or temperature.

The staff performed confirmatory analyses of the limiting MSLB using the 30-node MELCOR model and the 42-node MELCOR model for which additional noding detail was added to the service space. No increase in maximum calculated temperature resulted from the additional noding in the service space in the MELCOR analyses. The applicant's GOTHIC calculations were shown to be conservative in comparison with the MELCOR results. In developing the MELCOR models, the staff raised questions regarding the GOTHIC noding in comparison with more detailed plant information supplied by the applicant. **RAI 437, Questions 06.02.01-99 and 06.02.01-100, which are associated with the above request, are being tracked as open items.**

Evaluation of Feedline Break Accidents

FSAR Tier 2, Section 6.2.1.4, "Mass and Energy Release Analysis for Postulated Secondary Pipe Ruptures Inside Containment," Revision 2, presents an assessment of MFWLB accidents. The applicant considers MFWLBs be similar to MSLBs, but less limiting because the effective maximum break size is smaller and the energy release from the accident would also be smaller. Calculations for MFWLBs are not presented. Considering also the higher elevation of the steam lines, the staff agrees that MFWLBs will not be limiting with respect to containment integrity or for equipment qualification temperature data. Additional discussions comparing expected mass and energy release from MSLBs and MFWLBs are given in Section 6.2.1.4 of this report.

Negative Differential Pressure Analysis

The U.S. EPR design does not have an automatic containment spray system or containment air coolers for design basis accident mitigation. Thus, the U.S. EPR design is not susceptible to inadvertent automatic actuation of those systems, or the potential for damage because of the rapid reduction of the containment internal pressure that would result from such an inadvertent automatic actuation. The severe accident heat removal system (SAHRS) described in FSAR Tier 2, Section 19.2, "Severe Accident Evaluations," includes a manually actuated containment spray system dedicated to severe accident mitigation. This system is not used for DBAs. Because the SAHRS is manually aligned and manually actuated, it is not subject to a single failure that could cause inadvertent actuation of containment spray, thereby eliminating the need to analyze for this event.

In the response to RAI 266, Question 06.02.01-48 (SUP.4) the applicant concludes that there are no credible events, such as a temperature drop, that could produce a significant negative pressure differential within the containment. Evaluations by the staff support this conclusion, given the large heat capacities of materials in containment. Therefore, the staff agrees with the applicant that negative containment differential pressure is not a safety concern for U.S. EPR.

ITAAC

The ITAAC associated with the containment system are given in FSAR Tier 1, Section 2.1.1. The ITAAC specify containment design commitments, inspections, analyses and tests to be performed for the containment, and acceptance criteria to ensure that the containment is built by the COL applicant as designed. The staff's review of ITAAC related to the containment functional design and to the associated safety-related equipment is found in FSAR Tier 2, Section 14.3.11, "Containment Systems - Inspections, Tests, Analyses, and Acceptance Criteria," of this report.

ECCS-related ITAAC are discussed in Section 6.3 of this report. ITAAC for the CGCS are presented in FSAR Tier 1, Table 2.3.1-2, "Combustible Gas Control System ITAAC." The CGCS provides no safety-related function, but its subsystem, the CONVECT system, was re-classified as being safety-related. ITAAC related to the CONVECT system are discussed in Section 14.3.11 of this report.

Technical Specifications

The Technical Specifications and their bases applicable to the containment system are given in FSAR Tier 2, Chapter 16, Section 3.6 and B3.6. Required actions and surveillance requirements were reviewed together with the completion times allotted for corrective action and for surveillance frequencies. The staff finds the containment-related Technical Specification requirements are acceptable; however, additional technical specifications are needed for the safety-related pressurizer doors and for the CONVECT system.

FSAR Revision 0 did not identify any of the internal doors of the containment as being safety-related. Analyses performed to date indicate that at least six of the doors will be classified safety-related. Technical Specifications should be developed for the safety-related doors. **RAI 368, Questions 06.02.01-70 and 06.02.01-71 are being tracked as open items.**

Technical Specifications relating to the ECCS are presented in FSAR Tier 2, Chapter 16, Section 3.5 and B3.5. The staff's evaluation of the ECCS Technical Specification is presented in Section 6.3 of this report. Originally the CONVECT system was not classified as safety-related and had no Technical Specification. After re-classification of the system, in RAI 221, Question 06.02.01-24, the staff requested that the applicant develop the Technical Specification for the CONVECT system. See Section 6.2.2 of this report for the staff's review of the Technical Specification for the CONVECT system.

6.2.1.1.5 Combined License Information Items

There are no COL information items from FSAR Tier 2, Table 1.8-2 that affect this Section.

6.2.1.1.6 Conclusions

The unique design of the U.S. EPR containment two-room design and replacement of fan coolers and containment spray with the ECCS and the CONVECT systems represented a challenge in the review of the containment design. The applicant developed a 30-node GOTHIC code model for the plant, which explicitly representing the CONVECT system's foils and mixing dampers in the calculations. A significant number of LOCA and MSLB cases were analyzed with this model. The staff conducted independent confirmatory calculations with a 41-node APROS model and with two MELCOR models. No significant differences exist between the applicant's calculations and the confirmatory calculations. The staff has requested additional analyses with the multi-node GOTHIC model and has requested additional justifications as to the accuracy and completeness of the results. **RAI 437, Questions 06.02.01-99 and 06.02.01-100, associated with the above request, are being tracked as open items.**

The double-ended hot leg LOCA and the double-ended steam line break from an initial power level of 20 percent are the limiting accidents. The resulting calculated peak pressure and temperature are 379.2 kPa (55 psig) and 248.6 °C (479.5 °F), respectively. These values are compared against the U.S. EPR design pressure of 427.5 kPa (62 psig) and design temperature of 170 °C (338 °F). The calculated peak pressure is significantly lower than the design pressure. Also, the containment pressure drops below 50 percent of the peak value within 24 hours, which shows that the containment heat removal systems rapidly reduce containment pressure following a LOCA, as required by GDC 38. The calculated peak vapor temperature exceeds the containment design temperature; however, the calculated peak containment wall temperature is below 148.9 °C (300 °F), which is well below the containment design temperature.

The review of FSAR Tier 2, Section 6.2.1.1, Revision 2, was performed against GDC 16, GDC 38, and GDC 50. Based on the information above, and with the exception of the open items identified above, the staff finds that the applicant has demonstrated compliance with 10 CFR Part 50 Appendix A, GDC 16, GDC 38, and GDC 50, as follows:

GDC 16: Containment design conditions important to safety are not exceeded, and containment integrity is preserved under postulated design basis accident conditions.

GDC 38: Following a LOCA, containment pressure and temperature are rapidly reduced.

GDC 50: The containment structure and associated heat removal systems are designed to accommodate expected accident conditions from any LOCA with sufficient margin.

6.2.1.2 *Subcompartment Analysis.*

This section will be either delivered in a separate Phase 2 safety evaluation or evaluated in Phase 4 of the U.S. EPR design certification application review.

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

6.2.1.3.1 Introduction

The objective of the mass and energy (M&E) release analyses for postulated LOCAs is to provide conservative input for the containment design basis analyses as discussed in Section 6.2.1.1 of this report. The M&E releases for containment subcompartment analyses are discussed in Section 6.2.1.2 of this report. M&E releases from secondary system piping ruptures are discussed in Section 6.2.1.4 of this report. A large break LOCA is a rupture of a large reactor coolant system pipe. Such a postulated event causes a sudden increase in the containment pressure over a short period of time. Assumptions that are conservative for containment analyses lead to the rapid removal of heat from the reactor core, RCS piping, and steam generators, and the rapid transmittal of the reactor coolant to the containment. Assumptions that are conservative for containment analyses may not be the same as those evaluated in Section 15.6, "Decrease in Reactor Coolant Inventory Events," of this report which evaluates the performance of ECCSs in protecting the core following a LOCA.

6.2.1.3.2 Summary of Application

FSAR Tier 1: In FSAR Tier 1, Section 2.2.3, the applicant states that the safety injection system/residual heat removal system is a safety-related system. The SIS/RHRS has four divisions. The SIS/RHRS provides the following safety-related functions: Emergency core cooling; residual heat removal; protection of the reactor coolant pressure boundary; and containment isolation.

The SIS/RHRS is shown to deliver water to the reactor coolant system at the flow needed for core cooling during design-basis events.

FSAR Tier 2: The applicant provided a description of the M&E release analysis for postulated LBLOCAs in FSAR Tier 2, Section 6.2.1.3, "Mass and Energy Release Analyses for Postulated Loss of Coolant Accidents." Additional details are provided in AREVA Technical Report ANP-10299P, Revision 2, "Applicability of AREVA NP Containment Responses Evaluation Methodology to the U.S. EPR™ for Large Break LOCA Analysis." The analysis is summarized here, as follows:

FSAR Tier 2, Section 6.2.1.3, states that LBLOCA is characterized by the following phases which occur in sequence: initial blowdown; core refill; core reflood; and core post-reflood (long-term cooling). During the initial blowdown, there is a rapid depressurization of the RCS. Critical flow is established at the break location which is calculated based on the upstream thermal-hydraulic conditions. Containment pressure increases. Eventually, the core would become blanketed with steam, and the temperature of the fuel would increase.

FSAR Tier 2, Section 6.2.1.3, states that during refill, the safety injection system adds water to fill the reactor vessel lower plenum to the bottom of the core. The refill phase ends when the water begins to enter the core from the bottom so that reflood of the core can begin. During core reflood, the water level rises within the core producing a mixture of steam and water which is ejected from the top of the core. Chugging may occur as the cold water comes in contact with the hot fuel cladding. The reflood phase ends when all core locations are no longer blanketed with steam and begin to cool. The post-reflood phase starts as the core is being cooled and continues as decay heat and the remaining sensible heat are removed from the RCS in the process of long term cooling.

FSAR Tier 2, Section 6.2.1.3, states that the spectrum of LOCA break sizes analyzed included a range of hot leg (HL) breaks, cold leg pump suction (CLPS) breaks, and cold leg pump discharge (CLPD) breaks, ranging up to and including the largest postulated DEG break. The applicant evaluated postulated small break sizes equivalent to circular holes of 22.86, 15.24, and 7.62 centimeters (cm) (9 in., 6 in., and 3 in.) in diameter. The purpose of the small break evaluations was to investigate the containment circulation and stratification patterns if the rupture foils of the containment CONVECT system do not open and only the convection foils open on high local containment temperature. The staff's review of the applicant's containment circulation and stratification investigations, including those for which the rupture foils do not open following small breaks, is described in Section 6.2.1.1 of this report. The staff's review of the CONVECT system is described in Section 6.2.2 of this report. The break location studies include a break of the pressurizer surgeline inside the pressurizer compartment to confirm that venting from the pressurizer compartment to the accessible space is adequate. As a result of the pressurizer compartment evaluations, the applicant is adding six safety-related doors to the design to vent the pressurizer compartment space.

FSAR Tier 2, Section 6.2.1.3, states that a break in the HL piping would produce high containment pressure during the blowdown phase of the accident. A DEG break of this pipe will allow the reactor coolant to rapidly enter the containment early in the transient, before the passive heat sinks of the containment could effectively absorb the energy addition. The applicant found assumptions of minimum available safety injection and offsite power available also to be limiting. If offsite power is available, the reactor coolant pumps would continue to operate and add heat to the reactor system until they were tripped by the automatic pump trip logic.

In AREVA Technical Report, ANP-10299P, Revision 2, the applicant evaluated the consequences of a DEG-CLPS break. This is the break location which has often been determined to be limiting for operating pressurized water reactors and produce a large secondary containment pressure peak after the initial blowdown. The CLPS location results in a large post-blowdown pressure peak, because a path is opened for steam and water to escape the core, to pick up additional energy in the broken loop steam generator, and to flow into the containment without interacting with subcooled safety injection (SI) water in the cold legs. A DEG-CLPD break would produce a pressure response similar to that of a DEG-CLPS break and was included in the applicant's evaluations. Breaks at this location have generally not been found to be limiting in the containment analyses of operating plants.

Mass and energy release cases for postulated DEG breaks in the hot leg piping and the two cold leg locations were provided by the applicant in FSAR Tier 2, Section 6.2.1.3.

In AREVA Technical Report, ANP-10299P, Revision 2, the applicant stated that to maximize the containment peak pressure and temperature, the LOCA analyses use conservative assumptions that maximize the M&E released from the RCS. The sources of stored and generated energy, considered in the LOCA analyses include: Reactor power; decay heat; stored energy in the core; stored energy in the RCS fluid and metal, including the reactor vessel and internals; metal-water reaction energy and stored energy in the secondary system, including the SG tubing and secondary water. For the blowdown, refill, reflood and early post-reflood phases, M&E release rates are calculated by the thermal-hydraulic analysis code, RELAP5-BW. The applicant stated that the GOTHIC code was modified to maximize core heat removal, which would increase the calculated containment temperature and pressure response following a LOCA. The long-term post-reflood M&E release rates were determined by the GOTHIC computer code. The effect of single failures of various SIS/RHRS components on the M&E releases was included in the RELAP5-BW and GOTHIC M&E determinations.

ITAAC: In FSAR Tier 1, Section 2.2.3, the applicant states that FSAR Tier 1, Table 2.2.3-3 "SIS/RHRS Inspections, Tests, Analyses, and Acceptance Criteria," specifies the inspections, tests, analyses, and acceptance criteria for the SIS/RHRS.

Technical Specifications: The Technical Specifications and their bases applicable to M&E release analysis following postulated LOCAs can be found in FSAR Tier 2, Chapter 16, including Sections 3.5, "Emergency Core Cooling Systems," and B 3.5.

6.2.1.3.3 Regulatory Basis

The relevant requirements of NRC regulations for this area of review, and the associated acceptance criteria, are given in NUREG-0800, Section 6.2.1.3, “Mass and Energy Release Analysis for Postulated Loss-of-Coolant Accidents (LOCAs),” and are summarized below. Review interfaces with other SRP sections can also be found in NUREG-0800, Section 6.2.1.3.

The applicable regulatory requirements are as follows:

- GDC 50, requires that the containment structure and the containment heat removal systems be designed so that the containment structure 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.

The related acceptance criteria are as follows:

- 10 CFR Part 50, Appendix K, as it relates to sources of energy during the LOCA, provides methodologies to assure that all the energy sources have been considered.

While 10 CFR 50.46 requires the use of Appendix K methodologies in some cases for ECCS performance, the staff notes that 10 CFR Part 50 does not specify that Appendix K be implemented for containment analysis. However, SRP Section 6.2.1.3 provides guidance to consider the sources of energy described in Appendix K.

6.2.1.3.4 Technical Evaluation

Sources of Energy

LOCAs are among the design-basis accidents that must be evaluated to demonstrate adequate containment functional design. The staff notes that the calculated mass and energy release to containment is the main driving force behind the pressure and temperature rise in the containment. The staff notes that it is paramount that all significant sources of energy be included in the mass and energy release calculations.

Energy sources which should be considered are:

- Nuclear fission
- Decay heat
- Stored energy in the reactor core
- Metal-water reaction
- Stored energy in the RCS fluid
- Stored energy in metal (vessel, internals, piping)
- Secondary to primary heat transfer

All of the above energy sources were evaluated by the applicant in the mass and energy release rate calculations. The staff’s review of how the various energy sources were taken into account is described in the following sections.

Mass and Energy Release Rate Assumptions

In AREVA Technical Report, ANP-10299P, Revision 2, the applicant used a core inlet temperature of 296.4 °C (565.5 ºF) and a power level of 4,612 megawatts thermal (MWt) as initial conditions to maximize the energy content of the primary coolant. The primary system pressure was set at 15.76 MPa (2,286 psia) which is conservatively high. Maximum pressurizer volume was used to increase primary coolant mass. The SRP indicates that the initial mass in the coolant system should be calculated for the temperature and pressure conditions assuming that the reactor has been operating at a power level 1.02 times the licensed power level. The initial reactor power level used in these analyses is the rated thermal power level plus a calorimetric uncertainty of 0.5 percent as discussed in FSAR Tier 2, Section 6.2.1.1.3, "Design Evaluation." The use of the 0.5 percent calorimetric uncertainty is based on the applicant's commitment to install sufficiently accurate power measurement instrumentation. The review of the 0.5 percent calorimetric uncertainty can be found in the staff's safety evaluation report for AREVA Topical Report ANP-10278P, "U.S. EPR Realistic Large Break Loss of Coolant Accident Topical Report." The evaluation of the 0.5 percent calorimetric uncertainty is also discussed in Chapter 15 of this report.

Following a LOCA caused by a large piping rupture of reactor system piping, the sequence of events including the recovery of reactor core cooling has been divided into four phases, namely:

The blowdown phase: During this phase, coolant is rapidly ejected from the reactor system into the Containment Building and the reactor core temperature increases because of the lack of cooling.

The refill phase: During this phase, the reactor vessel lower plenum is refilled with emergency coolant to the bottom of the core. Little additional mass or energy release to the containment will occur during the refill period, and the core temperature will continue to increase.

The reflood phase: During this phase, the core is recovered with water and cooled below the initial temperature that occurred before the start of the accident. As the heated fuel comes in contact with the rising water level, experiments of simulated core reflooding have shown that a large fraction of the emergency coolant which reenters the bottom of the core will be carried out of the top of the core along with the exiting steam as water droplets. Depending on the break location, these droplets will be heated by the primary system piping and the stored energy within the steam generators to be turned into steam before flowing into the Containment Building.

The post-reflood or long-term cooling phase: Evaluations for this phase should consider the core decay heat and the removal of the remaining sensible heat from the reactor system and steam generators. During the early portion of this period, the core will still be boiling at a rapid rate. Void fraction evaluations have shown that a two-phase mixture will extend above the top of the core so that additional liquid will enter the steam generator tubes.

The flow of a two-phase mixture through the reactor system will remove the remaining sensible heat and will provide for additional steam formation to be added to the Containment Building.

As discussed in AREVA Technical Report, ANP-10299P, the RELAP5-BW computer code is used to determine the mass and energy addition rates to the containment during the blowdown phase, reflood phase, and early post-reflood phase of the accident. During the remainder of the post-reflood phase, the mass and energy addition to the containment is calculated using the GOTHIC computer code which also calculates the containment pressure as a function of time.

Using a single computer run for both the containment and the reactor system analysis avoids the need for iteration between computer codes.

Following a postulated LOCA, fission power generation in the reactor core will decrease. The reduction in fission power is a result of the decrease in coolant density within the core which increases the rate of neutron loss and suppresses the chain reaction. In addition, low pressure within the reactor system will cause the reactor protection system to trip the reactor so that the control rods will fall into the core. The staff has questioned the ability of control rods to perform their function during a large break LOCA because of the large forces that might exist within the reactor vessel. As stated in AREVA Technical Report, ANP-10299P, the applicant did not take credit for control rod entry into the core for the large break LOCA mass and energy release calculations. Conservative reactivity feedback assumptions were made to minimize the insertion of negative reactivity from the decrease in coolant density. The staff finds this approach acceptable. The staff gives credit for control rod entry during small break LOCA events. Adequate reactor shutdown margin was calculated. The staff's evaluation of control rod entry and reactor shutdown for small break sizes can be found in Section 15.6 of this report.

Decay heat during the blowdown phase was obtained using assumptions from 10 CFR Part 50, Appendix K, "ECCS Evaluation Models," which requires use of the 1971 American Nuclear Society (ANS) standard, "Decay Energy Release Rates Following Shutdown of Uranium-Fueled Thermal Reactors." The ANS standard was used with a multiplier of 1.2. This represents a sufficiently conservative account of decay heat generated in the core. In RAI 82, Question 06.02.01.03-1(p), the staff requested that the applicant describe the decay heat model used following the initial blowdown. In a proprietary December 17, 2008, response to RAI 82, Question 06.02.01.03-1(p), the applicant provided details on how decay heat was calculated for the U.S. EPR design. The applicant presented a comparison of the decay heat from the 1971 ANS standard using the method employed for the U.S. EPR with the 1979 ANS standard using an uncertainty of twice the standard deviation (2σ), which is specified by the standard. The 1979 ANS standard has a more extensive database than the 1971 standard and is considered to be the more accurate. The staff has recently accepted containment mass and energy release calculations using the 1979 ANS standard with a 2σ uncertainty. The applicant concluded that the core decay heat used in the mass and energy calculations for the U.S. EPR containment analysis is conservative compared with the 1979 ANS standard with a 2σ uncertainty and is, therefore, acceptable. The applicant extended the comparison to 10,000 seconds. The staff obtained a similar result for the first 10,000 seconds. At times greater than 20,000 seconds, the staff determined the 1979 ANS standard with a 2σ uncertainty to be the more conservative. Since long term decay heat removal has been an issue for the U.S. EPR containment evaluations, the staff requested additional justification in RAI 266, Question 06.02.01.03-2.

The staff noted that containment analyses should be extended over a 24-hour period following a LOCA to meet the acceptance criteria of SRP Section 6.2.1.1.A. In a February 25, 2010, response to RAI 266, Question 06.02.01.03-2, the applicant responded to the staff's question on the adequacy of the decay heat model by presenting a comparison of the total integrated energy over the 24-hour period for the two methods. The result was that the total energy was still greater after 24 hours using the applicant's method. Based on the above, the staff considers RAI 266, Question 06.02.01.03-2 resolved.

Energy stored in the reactor core is removed by heat transfer to the coolant during the blowdown, reflood, and post-reflood phases of the accident. The mode of core cooling during blowdown is stated by the applicant to be film boiling on the surface of the fuel rods once the critical heat flux is exceeded. SRP Section 6.2.1.3 states that calculations of heat transfer from surfaces exposed to the primary coolant should be based on nucleate boiling heat transfer. In RAI 221, Question 06.02.01.03-27(a), the staff requested that the applicant justify the conservatism of the heat transfer calculated by RELAP5-BW compared to the nucleate boiling assumption recommended by the SRP. In an August 27, 2009, response to RAI 221, Question 06.02.01.03-27(a), the applicant provided the requested information which demonstrates that when sufficient water is present within a core channel, RELAP5-BW calculates nucleate boiling to occur during the blowdown period, thus fulfilling the recommendation of the SRP and leading to a conservative result for containment analysis. Therefore, the staff considers RAI 221, Question 06.02.01.03-27(a) resolved.

Metal-water reaction between the heated fuel cladding and the reactor coolant was evaluated using the Baker-Just correlation which is programmed into RELAP5-BW. Use of the Baker-Just correlation is among the requirements of 10 CFR Part 50, Appendix K. The amount of the metal-water reaction was determined by the applicant to be insufficient to have a significant effect on the containment design basis evaluations.

Release of nitrogen gas from the accumulators is treated conservatively. The nitrogen is assumed to enter the containment starting at time zero and to be completely released at 20 seconds. This is more rapid than the expected release, and is conservative. The assigned temperature of the nitrogen, which is expected to be low following a LOCA, was selected to be 296.4 °C (565.5 °F), which is the RCS cold leg temperature. This temperature is also conservative.

According to the SRP, an acceptable approach for the refill phase is to assume a water level at the bottom of the active core at the end of blowdown, so there is no refill time. That occurs because little steam is released to the containment while the lower plenum is refilling. If a refill period is used, justification should be provided. The applicant calculated a refill period, and the applicant's justification was provided in AREVA Technical Report, ANP-10299P, Revision 2. As described in ANP-10299, Revision 2, Section 8.0, the refill period is shortened for containment mass and energy release calculations because the intent is to transport the core energy to the containment environment as quickly as possible. The refill period is shortened by direct injection of the accumulators into the lower plenum and downcomer. The staff finds the applicant's approach to shorten the refill period calculated by RELAP5-BW to be conservative, and therefore acceptable.

During reflood, the applicant's calculations show chugging, or rapid flow reversals, throughout the core as the incoming SI water contacts hot fuel surfaces in the core and flashes to steam. The steam is calculated to move away from the fuel area and then condense causing the chugging flow. In RAI 82, Question 06.02.01.03-1(c), the staff requested that the applicant provide an evaluation of the effect of chugging in the reactor core on the mass and energy release rate.

The staff's concern was that the chugging may be non-realistic and result in steam from the core being modeled as carried into the lower plenum and condensed in the injected ECCS water rather than flowing into the containment.

In a May 22, 2009, response to RAI 82, Question 06.02.01.03-1(c), the applicant claimed the chugging phenomenon to be real, particularly during the time of core reflood, as a result of water from the lower plenum surging into the core, producing steam which increases core pressure, and causing the core flow to temporarily reverse.

In RAI 221, Question 06.02.01.03-27(b), the staff requested that the applicant provide additional information concerning core heat transfer during the reflood period. In an August 27, 2009, response to RAI 221, Question 06.02.01.03-27(b), the applicant presented comparisons of RELAP5-BW with experimental data from FLECHT-SEASET experiments. Details concerning the comparisons with the FLECHT-SEASET experimental data can be found in AREVA Technical Report, ANP-10299P, Section 6.1.2. The comparisons showed good agreement with the test data. Core mass and energy releases during reflood are strongly correlated to the progression of the quench front. The test comparisons illustrate that core cooling and quench front progression is well-predicted by RELAP5-BW for a broad spectrum of initial conditions (i.e., reflood rates, pressures, and rod and liquid temperatures). The applicant further demonstrated that the flooding rate predicted for the U.S. EPR, as illustrated by the quench front progression during the reflood process, is within the range of the FLECHT-SEASET data. Therefore, the staff concludes that the core heat transfer processes are adequately calculated by the applicant during the reflood period for the purpose of containment analysis and consider RAI 221, Question 06.02.01.03-27(b) resolved.

To further assess the effect of core chugging on containment pressure, the staff performed confirmatory analyses using the RELAP5 Mod 3.3 computer code. One analysis was run by modeling a large reverse flow resistance at the entrance to the core which prevented negative core flow. The analysis was repeated with the design entrance losses. Using the design entrance losses, RELAP5 Mod 3.3 predicts significant core chugging similar to RELAP5-BW before the applicant modeled one accumulator to inject directly into the lower plenum. The results of the staff analyses showed the affect of chugging with the design entrance losses to result in a small reduction in calculated steam release to the containment relative to the case when reverse flow was prevented. The calculated reduction in steam flow was small, because RELAP5 predicted the water in the lower plenum to be at temperatures near saturation so that little steam condensation resulted. The staff determined the calculated reduction in steam flow to the containment to be more than entirely offset by the calculated increase in steam produced by the increase in accumulator water temperature in the applicant's revised RELAP5-BW reactor system model. The amount of chugging in the model has been reduced by the applicant and any reduction in the calculated steam release to the containment from chugging will be more than offset by the assumed increase in accumulator temperature. Therefore, the staff considers this issue resolved.

AREVA Technical Report, ANP-10299P, Revision 2, states that the secondary side of the steam generators is initially assumed to be isolated from the main steam system, and thus the steam generators remain at elevated temperatures and pressures relative to the reactor coolant system.

The staff noted that the water contained within the steam generator shell contains a considerable amount of sensible heat which must be conservatively treated. An important aspect of core reflood for containment analysis is the conservative prediction of liquid carried out from the core during the reflooding process. This liquid would be carried into the steam generators where the water would be turned to steam by the large heat transfer area of the steam generators before exiting the break and entering the Containment Building. The staff notes that tests at the FLECHT facility have shown that as much as 80 percent of the water that enters the core during the reflooding process is carried out the top of the core as liquid. The fraction of water that is carried out of the core is referred to as the carry out rate fraction (CRF). In RAI 221, Question 06.02.01-29, the staff requested that the applicant demonstrate that RELAP5-BW conservatively calculates the CRF for the U.S. EPR during design basis large break LOCA events. In a December 2, 2009, response to RAI 221, Question 06.02.01-29, the applicant provided comparisons with data from the FLECHT-SEASET test facility and revised AREVA Technical Report, ANP-10299P. The RELAP5-BW predictions of the CRF are in close agreement with the data from the FLECHT-SEASET test facility.

The CRF is a function of initial core temperature and linear fuel rod power at the beginning of reflood. The applicant demonstrated that both the initial temperature and linear fuel rod power predicted for U.S. EPR were within the range of the test data. Therefore, the staff considers RAI 221, Question 06.02.01-29 resolved.

Following the time when the core has been reflooded for postulated large cold leg breaks, the water level in the core and reactor vessel downcomer will increase to eventually reach the level of the break. Additional ECCS flow injected into the cold legs will spill out of the break, and ECCS flow into the bottom of the core will temporarily cease. Under these conditions, the water in the core will heat and eventually boil. As the core boils, additional ECCS water in the lower plenum will flow up into the core to make up for the water that was boiled away. The two-phase mixture of water and steam in the core will at this time have a lower density than that of the all-liquid downcomer. The uneven densities will cause the two phase level to rise above the core so that the static head in the lower plenum below the core will be equal to that below the downcomer.

As the two-phase level in the core rises into the upper plenum and into the hot legs, part of this frothy, two-phase mixture would be forced into the steam generators where it would make contact with the heated steam generator tubes. Energy remaining in the steam generator secondary fluid and metal will be transferred to the froth, causing it to become steam. The steam exiting the steam generator outlet that flows into the containment through the break might initially be super-heated. As the steam generator secondary side is cooled, additional steam will no longer be produced.

The height that the two-phase mixture would initially reach is dependent on the void fraction in and above the core. The higher the void fraction becomes, the higher the elevation to which the frothy mixture will extend above the core. RELAP5-BW calculates void fractions from the inter-phase drag between the liquid and the steam. The applicant replaced the original RELAP5-BW inter-phase drag model with one based on the applicant's FOAM2 computer code. The FOAM2 model was improved for better correlation of experimental level swell data at high void factions when it was programmed into RELAP5-BW (Reference (Babcock and Wilcox) BAW-10155-A, "FOAM2-Computer Program to Calculate Core Swell Level & Mass Flow Rated during Small Break LOCA," Babcock and Wilcox Co., October 1990, and BAW-10164P-A, "RELAP5/MOD2-B&W -An Advanced Computer: Program for Light Water Reactor LOCA and Non-LOCA Transient Analysis," Revision 6, AREVA NP, Inc., June 2007). As indicated in BAW-10155-A and BAW-10164P-A, both the FOAM2 computer code and its application in RELAP5-BW have been previously approved by the staff based on comparisons with experimental level swell data. The experimental data used in the validation ranged from atmospheric pressure to 2.76 MPa (400 psia), which encompasses the pressures expected in the reactor system during the post-reflood phase following a large break LOCA.

The applicant benchmarked the heat transfer models in RELAP5-BW for predicting steam generator heat flow by modeling the FLECHT-SEASET facility as described in AREVA Technical Report, ANP-10299P, Revision 2. These FLECHT-SEASET tests measured the energy transfer from the hot steam generator secondary fluid to the cooler two-phase mixture flowing through the steam generator tubes under LBLOCA, post-blowdown conditions. Two-phase mixtures of varying void fractions were forced into the steam generator test assembly at various flow rates to measure the transient heat transfer rates and fluid temperature distributions. The FLECHT-SEASET experimental test results showed the appearance of a quench front inside the steam generator primary side tubes. The dispersed two-phase flow above the quench front provided enough heat transfer and precursory wall cooling so that the quench front advanced up the tubes. Good agreement was shown between the RELAP5-BW model results and the FLECHT-SEASET test data. The staff requested that the applicant demonstrate that FLECHT-SEASET test conditions validate the RELAP5-BW model for the U.S. EPR. The applicant provided the results from Test No. 21909, which show that the steam generator heat transfer predicted for the U.S. EPR is high relative to the FLECHT-SEASET experiment test data, which is conservative for containment analysis. Furthermore, the heat transfer mode predicted by RELAP5-BW was nucleate boiling in the lower steam generator regions when wetted by coolant, which is in compliance with the recommendations of SRP Section 6.2.1.3. Therefore, the staff concludes that the RELAP5-BW model of the U.S. EPR is acceptable for prediction of secondary to primary heat transfer for containment analysis purposes. Therefore, the staff considers this issue resolved.

For postulated cold leg breaks downstream from the ECCS cold leg injection location, steam generated in the core and within the steam generator tubes will contact the injected ECCS water where it is injected into the reactor system cold legs. Some of the steam will condense in the reactor coolant system before entering the containment. For a break upstream of the injection location, steam from the core and from the broken loop steam generator can enter the Containment Building through the break without contacting the injected ECCS water. The SRP allows the staff to give credit for steam condensation, provided justification is provided. In RAI 221, Question 06.02.01-41, the staff requested that the applicant justify the efficiency of steam condensation predicted by RELAP5-BW. The applicant presented data from the Upper Plenum Test Facility (UPTF) (Reference MPR-1208, “Summary of Results from the UPTF Cold Leg Flow Regime Separate Effects Tests, Comparison to Previous Scaled Tests and Application to U.S. Pressurized Water Reactors,” MPR Associates, October 1992). A condensation efficiency of between 80 and 100 percent within the cold leg piping of the test facility was indicated by the data. This finding was supported by data from the Westinghouse 1/3 scale test facility (Reference EPRI 294-2 Final Report, “Mixing of Emergency Core Cooling Water with Steam: 1/3 - Scale Test and Summary,” Electric Power Research Institute, June 1975) which shows more than 90 percent of steam to be condensed in the presence of sufficient subcooled water in the cold legs. In contrast, the applicant determined that for the U.S. EPR LBLOCA analysis presented in AREVA Technical Report, ANP-10299P, only about a 25 percent condensation efficiency was predicted by RELAP5-BW at the cold leg ECCS injection nozzles. As the steam and water flow together down the intact cold legs and around the top of the upper plenum, RELAP5-BW calculates additional condensation so that a condensation efficiency approaching 90 percent is predicted before the fluid exits the break. The test data indicates essentially 100 percent condensation at this location. Based on the data comparison, the staff concludes that RELAP5-BW is conservative for predicting steam condensation within the reactor system cold legs.

Stored energy in the reactor vessel and vessel internals was included in the RELAP5-BW model of the U.S. EPR. In RAI 221, Question 06.02.01.28, the staff requested that the applicant indicate the mode of heat transfer and justify the conservatism of the heat transfer calculated by RELAP5-BW as compared to the nucleate boiling assumption recommended by SRP Section 6.2.1.3. In a December 18, 2009, response to RAI 221, Question 06.02.01.03-28, the applicant provided the requested information, which demonstrated that RELAP5-BW predicts nucleate boiling to be maintained as long as a liquid phase is present. The staff concludes that the SRP recommendation is met. Therefore, the staff considers RAI 221, Question 06.02.01.28 resolved.

In order to maximize heat transfer to the coolant, SRP Section 6.2.1.3 specifies that fuel cladding swelling and rupture should not be considered. In AREVA Technical Report ANP-10299P, the applicant stated that swelling and rupture of the fuel cladding was included in the RELAP5-BW model. In RAI 221, Question 06.02.01-32, the staff requested that the applicant evaluate the effect of the assumed cladding swelling and rupture on the U.S. EPR containment analysis.

In a December 18, 2009, response to RAI 221, Question 06.02.01-32, the applicant responded that the mass and energy release calculation considers an average core. As a result of the enhanced heat transfer from the core to the coolant in the applicant's RELAP5-BW RCS model, cladding swelling and rupture is not calculated to occur. Enhanced heat transfer in the RCS is conservative for containment analysis. Therefore, the staff considers RAI 221, Question 06.02.01-32 resolved.

NUREG-0800, Section 6.2.1.3, provides a number of additional recommendations for the mass and energy release calculations, namely:

- Water remaining in the vessel following blowdown should be assumed to be at the saturated temperature. This condition is predicted by the RELAP5-BW analyses.
- Steam leaving the steam generators should be superheated to the temperature of the secondary coolant. The steam temperature is calculated internally within RELAP5-BW, which will predict dry steam traveling through heated steam generator tubes to become superheated.
- Steam from decay heat boiling in the core should be assumed to flow to the containment by the path which produces the minimum amount of mixing with ECCS injection water. The applicant's modeling conforms to this recommendation which would be most significant for a large CLPS break. As discussed in AREVA Technical Report, ANP-10299P, the applicant assumes that for a large CLPS break, as the accident progresses, the intact cold legs become filled with water and blocked to flow. This models steam generated within the reactor core and within the broken loop steam generator as flowing directly to the containment without any mixing with ECCS injection water until hot leg injection is initiated. Thus, the staff finds that the applicant's approach complies with the above recommendations in NUREG-0800, Section 6.2.1.3.

Selection of Limiting Case

The applicant evaluated a range of potential single failures to determine which is limiting. The single failure of one pumped ECCS division in conjunction with another division of pumped ECCS being unavailable because of preventive maintenance provides the most limiting scenario for containment pressure response. Under these circumstances, only two divisions of pumped ECCS would be available to mitigate the postulated LOCA. The assessment also evaluated the effect of location of the operating ECCS divisions relative to the break. Little sensitivity was found; however, the most limiting condition was for each of the operating divisions to be associated with the intact loops.

FSAR Tier 2, Section 6.2.1.3, describes a spectrum of LOCA break locations that was analyzed, including breaks in a hot leg and breaks in both the suction side and the discharge side of a reactor coolant pump (RCP). Break size was varied from small break sizes up to DEG breaks in the hot leg and cold leg piping.

Loss of offsite power, break discharge rate, and containment back pressure were treated as parameters in order to have sufficient information for selection of the limiting LOCA scenario. Altogether, 52 cases were reported in FSAR Tier 2, Section 6.2.1. Mass and energy releases were generated for each of these scenarios and were used in the containment pressure and temperature evaluations discussed in Section 6.2.1.1 of this report. Since there are no systems to cool the containment atmosphere, these did not have to be considered in the single failure analysis. The partial loss of ECCS is significant, since condensation of steam by the ECCS water within the reactor system is an important consideration for U.S. EPR containment analysis. The staff finds that the selection of break location and break sizes, together with single failure considerations and additional sensitivity studies, provides an appropriate evaluation matrix for containment analysis.

Cold Leg Pump Suction Breaks

A break at the suction of an RCP has the potential of producing high containment pressures after the initial blowdown resulting from the release of energy stored in the steam generators. The pump suction location offers a flow path from the core through one steam generator with the minimum of flow resistance compared to other break locations so that the core is reflooded more readily, and steam and water leaving the core can pass to the containment through the broken loop steam generator where additional steam is generated. Postulated breaks at an RCP suction have been found to be the limiting break location for many operating PWRs. ANP-10299P, Revision 2, describes the applicant's methodology for calculating mass and energy release from a postulated DEG-CLPS break. In RAI 358, Questions 06.02.01-59 and 06.02.01-60, the staff requested, that the applicant describe how the methodology of AREVA Technical Report, ANP-10299P will be applied to other break sizes and locations and how the methodology will be conservative for the containment analyses of these break locations.

In a July 8, 2010, response to RAI 358, Questions 06.02.01-59 and 06.02.01-60, the applicant stated that the large break LOCA methodology in AREVA Technical Report, ANP-10299P, Chapter 8, "Electrical Power," and Chapter 9, "Auxiliary Systems," will be applied to CLPD and HL breaks. Specific methodology considerations related to the following break locations and break sizes were discussed as follows:

Hot Leg Breaks

Hot leg breaks offer a path for the hottest fluid in the RCS to enter the containment directly. Hot leg breaks are found to produce a large containment pressure peak during the blowdown period. Following the initial blowdown, all ECCS flow injected in the cold legs would be forced through the core.

Since the injected fluid is cooled by the RHR heat exchangers (HXs) before entering the reactor, considerable steam condensation would occur. Unlike cold leg breaks, for which the bulk of the ECCS injection will bypass the core and spill out the break, for a hot leg break all ECCS water injected in cold legs traverses the core before spilling out the break. This reduces the steaming rate and, as a result, the containment pressure continues to decline even before hot leg injection is activated at 3,600 seconds.

The applicant stated that ECCS models in AREVA Technical Report, ANP-10299P will be modified for the calculation of hot leg break mass and energy release. In RAI 358, Questions 06.02.01-59 and 06.02.01-60, the staff requested that the applicant provide a description of how the methodology in AREVA Technical Report, ANP-10299P was modified for the hot leg break mass and energy predictions and justify that conservative results are obtained. In a July 8, 2010, response to RAI 358, Questions 06.02.01-59 and 06.02.01-60, the applicant stated that for the analysis of hot leg breaks, the mass and energy release is calculated using the RELAP5-BW model of the reactor system until a quasi-steady state is reached in the reactor coolant system or hot leg injection is begun. A quasi-steady state is defined to exist when the core has quenched and there is no significant flow through the steam generator tubes. For a large hot leg break, these conditions occur in the first few hundred seconds as the primary reactor system flow path is ECCS injection into the cold legs, through the core and out the hot legs. Once the core has cooled and flow through the steam generators has stopped, the reactor system would behave essentially as a boiling pot with all ECCS water flowing through the core. The applicant continues to use RELAP5-BW after this time to calculate the mass and energy release or may switch to a simplified GOTHIC model of the reactor system. The staff agrees that use of either model is acceptable to calculate the mass and energy release until hot leg injection is begun at 3,600 seconds into the analysis. Loop seal formation is not an issue for hot leg breaks since steam and water exiting the core would flow directly to the containment through the hot leg break, whether or not the cold leg loop seals were filled with water.

Hot leg injection involves diverting a portion of the low head safety injection (LHSI) water to the hot legs. For a postulated hot leg break, the applicant assumes that the LHSI water injected into the hot legs is spilled into the IRWST without mixing with steam from the core either within the reactor system or within the containment. This is the bounding conservative assumption for containment analysis. The remaining LHSI and medium head safety injection (MHSI) flow that enters the cold legs is assumed to flow through the core and mix with any steam generated there in the applicant's GOTHIC model of the reactor system for calculation of the mass and energy release. The staff agrees with the applicant's modeling approach and concludes that with the bounding assumption that the hot leg injection spills to the IRWST without quenching steam from the core, the calculation is conservative. Therefore, the staff considers RAI 358, Questions 06.02.01-59 and 06.02.01-60 resolved.

Cold Leg Pump Discharge Breaks:

For a large break at the discharge of a reactor coolant pump, all of the cold leg loop seals could collect water. However, steam generated in the core would cause at least one loop seal to clear so that steam relief to the containment would occur. For a postulated pump discharge break, the rate of core reflood would be slower than for a break at the pump suction, since steam generated within the core and steam generators would have to pass through the rotor of an RCP in reaching the containment. The slower core reflood rate would be expected to produce a slower steam release to the containment in comparison to an equivalent break at the suction of a coolant pump.

In FSAR Tier 2, Section 6.2.1.3, the applicant stated that a break in the CLPD produces the same ECCS configuration as a break in the cold leg pump suction except that the ECCS delivered to the broken loop is assumed to be lost out of the break with no interaction with steam in the RCS.

For the CLPD break configuration, the applicant states that the delivery of coolant to the loops seal piping is prevented by the virtual weirs in the U.S. EPR reactor coolant pump design that rise above the top of the cold leg piping. As a result of the weirs, loop seals will not form as early in the CLPD scenario, and the transition from RELAP5/MOD2-BW to GOTHIC for the discharge breaks can be as late as 3,600 seconds coincident with the initiation of hot leg injection. In RAI 368, Question 06.02.01-61, the staff requested that the applicant provide justification that the cold legs will not be blocked following a CLPD break before 3600 seconds or provide analyses of the containment response resulting from earlier loop seal formation. In a September 30, 2010, response to RAI 368, Question 06.02.01-61, the applicant provided the dimensions of the virtual weirs. The applicant presented a detailed drawing of the RCP impeller to the staff at an audit on September 29 and 30, 2010, which confirmed the applicant's description. FSAR Tier 2, Figure 5.4.1 presents a similar drawing of the U.S. EPR pump impeller. For cold leg safety injection water to spill over these virtual weirs into the loop seals of the intact loops, the level in the cold leg and downcomer region will have to rise to a height above the virtual weirs. With the break at the discharge of one coolant pump a path is provided for coolant to spill from the break at an elevation that is lower than the virtual weirs. Because of the difference in elevation between the elevation of the break and that of the top of the virtual weirs, the staff concludes that it is unlikely that water will spill over the virtual weirs for a CLPD break. If some water did spill over into the intact cold leg loop seals, the containment response would be bounded by the analysis for the DEG-CLPS break for which the intact cold leg loop seals are artificially assumed to fill at a conservatively early time as described in the following discussion. The staff concluded this issue is adequately addressed by the applicant and, therefore, considers RAI 368, Question 06.02.01-61 resolved.

Split Breaks and Small Breaks

In an August 27, 2009, response to RAI 221, Question 06.02.01-46, the applicant states that as the break size decreases, the dynamics of the RCS change. Input considerations such as those for partial cooldown of the steam generators, loop seal formation, and hot leg injection that have a significant impact on the containment response may need to be modified. The applicant further states that methods and inputs will be evaluated individually for each break to verify a conservative calculated pressure and temperature response. In RAI 368, Question 06.02.01-62, the staff requested that the applicant provide further information including how the methods and input are evaluated as a function of break size and justification that the process yields conservative results. Further, the staff requested that the applicant describe how partial cooldown of the steam generators, loop seal formation and hot leg injection are evaluated for split and small breaks. In a September 30, 2010, response to RAI 368, Question 06.02.01-62, the applicant provided the RELAP5-BW noding detail that is used to model breaks that are smaller than double ended breaks. The applicant explained that partial cooldown of the steam generators is credited for small breaks and not for large breaks. Partial cooldown is the safety related function by which heat is removed from the reactor system via the steam generators following a LOCA as discussed in Chapter 7, "Instrumentation and Controls," of this report and is, therefore, appropriate for use in safety analyses. To neglect the partial cooldown for the containment analyses of large breaks is conservative. The applicant stated that assumptions for loop seal formation are dependent on the break location. As described in the preceding discussions, loop seal formation is not significant for hot leg breaks and has been reviewed for other break locations as discussed in the proceeding paragraphs. The staff reviewed the RELAP5-BW noding detail and modeling assumptions. The staff concluded that the RELAP5-BW model adequately describes the reactor system and the processes necessary to provide mass and energy release for containment analyses for postulated split breaks and small breaks. Therefore, the staff considers RAI 368, Question 06.02.01-62 resolved.

Pressurizer Compartment Breaks:

FSAR Tier 2, Section 6.2.1.3, describes the various break locations that were analyzed to include a break of the pressurizer surgeline inside the pressurizer compartment. This evaluation was stated to be used in the design of six safety-related pressurizer compartment doors. The methodology for evaluating this break was not included in the FSAR.

In RAI 368, Question 06.02.01-66, the staff requested that the applicant describe this methodology and justify that conservative assumptions were used. In addition, the staff questioned if the rupture of other high energy lines in the pressurizer compartment were evaluated. In a September 30, 2010, response to RAI 368, Question 06.02.01-66, the applicant provided additional information on the assumptions used in the methodology. The applicant calculated the mass and energy release from a broken surge line using RELAP5-BW input that was modified to describe the break location. A break in a surge line would be analyzed in the same fashion as a hot leg break since the surge line exits one of the hot legs. Conservatisms in RELAP5-BW for break flow, heat transfer from heated surfaces, and decay heat were applied to the surge line break. Therefore, the staff concludes that the applicant's approach is conservative, and considers RAI 368, Question 06.02.01-66 resolved.

Original Calculational Model

Two different computer methods are utilized in the LBLOCA analyses presented in FSAR Tier 2, Section 6.2.1.3. RELAP5-BW was used to determine mass and energy release during the blowdown phase, reflood, and initial post reflood phase. Long term post-reflood mass and energy release rates are determined by a one-node simulation of the RCS within GOTHIC which simultaneously calculates the pressure/temperature response of the Containment Building.

The staff compared the steam release to the containment originally predicted by the applicant using the RELAP5-BW/GOTHIC methodology to that in the FSARs of similar operating plants. The steam release calculated by the applicant for the post-reflood period was found to be significantly less than that which the staff had approved for the containment analyses for similar operating plants. The cause of the reduction in steam release in the applicant's calculations was determined to result from the assumptions made for steam condensation by the ECCS water injected into the cold legs. The staff questioned the degree of assumed steam condensation. The RELAP5-BW model of the reactor system assumed a major portion of the steam generated in the core to flow into the intact loops and to be condensed within the intact loop cold legs by the injected ECCS water. The original one-node GOTHIC model of the reactor system homogenized the reactor coolant, steam and injected ECCS water so that very little of the steam from the core and from sensible heat removal was added to the containment atmosphere. In RAI 82, Question 06.02.01.03-1(j), the staff requested that the applicant clarify concerns that the low points in the cold leg pump suction piping might fill with water (loop seals) and block steam flow in the intact loops. Under these conditions for a postulated break at the suction of a reactor coolant pump, all steam from the core and broken loop steam generator would flow into the containment with little or no interaction with the cold leg injection. With this increased steam flow to the containment, the mass and energy source would resemble that calculated for the operating plants.

The staff notes that the U.S. EPR containment design is unlike that of operating PWRs in the U.S., in that it does not include safety-related containment sprays or fan coolers. Although the staff has previously approved the RELAP5-BW and GOTHIC codes for use in safety analyses for core reload of operating plants (Reference: Thadani, Mohan C., NRC, letter to Gardner, Ronnie L., Correction to Letter Forwarding the Final Safety Evaluation for Framatome ANP

Topical Report BAW-10252P(A), Revision 0, “Analysis of Containment Response to Postulated Pipe Ruptures Using GOTHIC,” Framatome ANP, September 6, 2005), the staff did not review the methodology for application to the safety analysis of a plant without active safety-related systems able to cool and mix the containment atmosphere. Neither the RELAP5-BW RCS model nor the GOTHIC RCS/containment model was approved for the U.S. EPR post-LOCA containment evaluation. For this reason, the staff revisited the applicant’s methodology for computing mass and energy release following a postulated LOCA. In particular, the staff reexamined the applicant’s predictions of the long term steam release to the containment since it is in the extended period after the initial RCS coolant loss that operating plants most rely on the safety related atmospheric cooling systems to maintain containment pressure within design limits and eventually bring the containment to a stable and depressurized condition. Therefore, in RAI 1, Question 06.02.01-2, the staff requested that the applicant provide additional information concerning the application of both these models for the U.S. EPR. In a January 28, 2009, response to RAI 1, Question 06.02.02-2, the applicant responded to the staff’s requests by producing AREVA Technical Report ANP-10299P, which describes a new, more conservative methodology that takes less credit for steam condensation in the cold legs. The revised model takes credit for steam condensation in the upper plenum following the time when plant operators would redirect a portion of the LHSI flow into the hot legs.

Revised Calculational Model

Using the revised methodology, the applicant provided a sample calculation of the mass and energy release from a postulated DEG-CLPS break assuming loop seal plugging of the intact cold legs at a selected time after the break. With the intact loop seals plugged, all steam generated in the core will exit the reactor system into the containment without being condensed.

With the revised calculational model, RELAP5-BW is applied to calculate the mass and energy release until the intact cold legs are assumed to be plugged. The time assumed in the analysis was said to represent the earliest time following the occurrence of a LBLOCA when the loop seals could be filled and contain water so as to block flow. Continued steam flow through all cold legs was assumed in the original FSAR model such that the code modeled condensation of steam generated in the core, steam generators, and other reactor system piping in the cold leg safety-injection water for the entire 24-hour period of the analysis. With the revised methodology, after a selected time, steam flow through the intact cold legs ceases. Therefore, steam condensation in the intact loops is also modeled to cease. For a DEG-CLPS break, loop seal plugging at the break location cannot occur and the break will provide a path for steam from the reactor system to flow directly into the containment. The staff requested justification for the selected time assumed by the applicant.

The applicant used two independent approaches to justify the selected time as the assumed loop seal formation time. The first was a countercurrent flow limiting (CCFL) approach, and the second was a venting pressure drop approach. CCFL was evaluated using an empirical correlation by Hewitt and Wallis. Loop seal formation from CCFL was shown to occur much later than the selected time assumed for the sample problem, which demonstrates that the assumed time is conservative. For the venting pressure drop calculation, the applicant performed an assessment of the pressure difference across the upside of the CLPS loop seals in the intact loops. The results indicated a minimum loop seal formation time of 1,910 seconds, which confirms that the assumed loop seal formation time is conservative. The applicant’s conclusions were further confirmed by RELAP5 Mod 3.3 evaluations by the staff in which the CCFL correlation of Kutateladze was utilized to evaluate countercurrent flow in the cold leg loop seals. The staff’s evaluation predicted that the loop seals in the intact loops would begin to intermittently fill with water after about 3,200 seconds. The RELAP3 Mod 3.3 results were further confirmed by staff analyses using the TRACE computer code. The TRACE code predicted the first loop seal in an intact cold leg to fill at 4,800 seconds. The staff concludes that the applicant's assumed time at which the intact loop seals fill is conservative.

The applicant notes in ANP-10299P, Section 8.1, that coolant delivery to the loop seal piping segments following a LOCA would be significantly reduced because of a virtual weir in the U.S. EPR reactor coolant pump design that rises above the top of the cold leg piping. This configuration is said to significantly limit the amount of water that can drain from the discharge side of the pump. The RCP weirs were not modeled in either the applicant's or the staff's evaluations of loop seal plugging time for the DEG CLPS break, thus providing an additional conservatism in the assumed time of loop seal plugging.

In RAI 82, Question 06.02.01.03-1(o), the staff requested that the applicant provide more information on the possibility of air entering the RCS over the course of the accident. The applicant responded to RAI 82, Question 06.02.01.03-1(o) in Revision 2 to ANP-10299P. The applicant did not completely address the staff's concern that air might be drawn into the RCS over the course of the accident if the pressure in the RCS drops below the containment pressure from steam condensation. The staff noted that air inflow is not mathematically permitted in the applicant's evaluation model, since reverse break flow is not allowed. Inflow of air from the containment to the RCS can affect heat transfer in the RCS, for example, condensation. Therefore, in RAI 340, Question 6.02.01-56, the staff requested that the applicant provide justification that air flow from the containment into the RCS will not occur and affect the post LOCA containment calculations. In an April 4, 2010, response to RAI 340, Question 06.02.01-56, the applicant stated that a scoping study had been performed and that the resulting study did not indicate this problem. The staff reviewed this response and found it to be insufficient. In follow-up RAI 437, Question 06.02.01-97, the staff requested that the applicant provide additional information on this issue. **RAI 437, Question 06.02.01-97 is being tracked as an open item.**

In the revised mass and energy release model provided in AREVA Technical Report, ANP-10299P, Revision 2, the applicant takes credit for the safety-related partial cooldown of the steam generators that occurs following actuation of an SI signal. The partial cooldown is accomplished by lowering the main steam relief valve (MSRV) pressure control setpoints of the four steam generators at a rate corresponding to 82.2 °C/hr (180 °F/hr). The partial cooldown is completed at a time when the steam generator secondary side pressure reaches 6 MPa (870 psia).

At the time when the loop seals in the intact coolant loops are assumed to close, the applicant switches from the RELAP5-BW representation of the reactor system to a GOTHIC representation that is a revision to the original reactor system GOTHIC model. In the revised GOTHIC model, the intact loop seals are assumed to be filled with water so that all the steam generated by decay heat flows to the containment without interacting with the ECCS water injected into the cold legs. This assumption complies with the SRP Section 6.2.1.3 recommendation that, "steam from decay heat boiling should be assumed to flow to the containment by the path which produces the minimum amount of mixing with ECCS injection water."

In the revised GOTHIC model, the fluid entering the core is limited to that which is boiled away by decay heat plus an assumed fraction of ECCS flow. The selected fraction of ECCS flow is assumed to be carried out of the core as liquid. One quarter of this liquid flow is assumed to be turned to steam as the remaining sensible heat is removed from the reactor system metal above the reactor vessel nozzles and from the broken loop steam generator metal and fluid. This steam flows into the CB and acts to pressurize the containment. The remaining three quarters of the liquid flow is assigned to the intact loops where any steam produced is condensed by the ECCS water injected in the cold legs. Therefore, in RAI 221, Question 06.02.01-35(b), the staff requested that the applicant clarify the validity of the assumed equal flow split. In a December 18, 2009, response to RAI 221, Question 06.02.01-35(b), the applicant provided additional information on the assumed flow split. The staff reviewed this response and determined that it did not adequately address the issue. In follow-up RAI 398, Question 06.02.02-47, the staff requested that the applicant provide additional information regarding the assumed flow distribution. **RAI 398, Question 06.02.02-47 is being tracked as an open item.**

The ECCS water injected into the cold legs but which does not enter the core is assumed to be spilled onto the containment floor. The modeling of sensible heat sources in the GOTHIC model considers the principal sources from the primary system passive metal stored energy, the secondary-side fluid, and safety injection pump heat. This sensible heat is from metal below the reactor vessel nozzles, intact coolant loops, broken loop pump discharge piping, pressurizer, and intact steam generator fluid and metal. This heat addition will have little effect on the containment pressure but is necessary for conservation of energy. Therefore, in RAI 221, Question 06.02.01-35(a), the staff requested that the applicant provide additional justification for the 5 percent of total ECCS flow entrainment assumption. In a December 18, 2009, response to RAI 221, Question 06.02.01-35(a), the applicant provided additional information on the assumptions used for liquid entrainment. The staff reviewed this response and determined that it did not adequately address the issue. In follow-up RAI 398, Question 06.02.02-47, the staff requested that the applicant provide additional information regarding the assumed flow distribution. **RAI 398, Question 06.02.02-47 is being tracked as an open item.**

Effect of Hot Leg Injection

Following a large break LOCA, continued boiling in the core may cause the boric acid dissolved in the coolant of PWRs to concentrate so that core cooling might be adversely affected. To prevent this from occurring, operating PWRs include provisions within the emergency procedures to direct a portion of the ECCS flow from the cold legs to the hot legs. The time when hot leg injection would be procedurally implemented is determined from plant specific analyses.

For the U.S. EPR, the applicant proposed a procedure to call for hot leg injection implementation no later than 60 minutes following a LOCA. The staff noted that the operators need to initiate hot leg injection within a time window. In RAI 368, Question 06.02.01-72, the staff requested that the applicant provide a list of the actions that the operators take to perform this function, a discussion of when they begin these actions, and the identity of the control room signal in response to which the actions will be initiated. In an April 19, 2010, response to RAI 368, Question 06.02.01-72, the applicant responded that, to initiate hot leg injection, the operator must perform the following actions:

- Close the LHSI main containment isolation valves (CIVs) outside the Reactor Building (RB) to reduce the LHSI cold leg injection flow.

- Open the isolation valves of the connection line between the LHSI discharge line and the RCS hot leg suction line outside the containment.

- Open the first and second RCPB isolation valves of the RCS hot leg suction line inside the containment.

In an April 18, 2010, response to RAI 368, Question 06.02.01-72, the applicant stated that the first indication that hot leg injection may be necessary is when the operator receives a loss of subcooling margin alarm. A second indication that hot leg injection may be necessary occurs when the RCS pressure is less than 289.7 psia and the permissive P16 is present in the control room. The applicant stated that guidance will be provided by the emergency operating procedures (EOPs) to instruct the operator to initiate hot leg injection when conditions mandating hot leg injection are present. These conditions include criteria such as:

- Continued loss of subcooling margin.
- Reactor coolant pumps not running.
- Time-specific criteria to limit:
    - Containment pressure
    - Boron precipitation

The applicant stated that for the bounding containment pressure case, the operator will have at least 1 hour to complete the switchover to hot leg injection. The applicant stated that plant operators will be trained and qualified in the emergency procedures to initiate hot leg injection within the timeframe credited for that action in the safety analysis.

The staff's review of operator action to effect hot leg injection is discussed in Section 15.01.04 of this report. FSAR Tier 2, Section 13.5 indicates that a COL applicant who references the U.S. EPR design certification will provide site-specific information for Emergency Operating Procedures. The staff reviewed FSAR Tier 2, Section 13.5.2.1 regarding the EOPs and found that the description provided in FSAR Tier 2, Section 13.5 does not provide guidance to COL applicants to perform the necessary verification and validation of the EOPs that would assure the plant specific EOPs are consistent with the safety analyses assumption contained in FSAR Tier 2, Chapter 15. To assure that the safety analyses of record remain valid, the staff requested that the applicant propose a COL information item in FSAR Tier 2, Section 15.0 to call for the COL applicant to perform verification and validation of its plant specific EOPs to assure that the operator action times assumed in the safety analyses presented in FSAR Tier 2, Chapter 15, are achievable. The staff considers this issue an open item in the staff review to remain open until a satisfactory resolution is reached. **RAI 415, Question 15-9 is being tracked as an open item.**

In the context of containment analysis, the initiation of hot leg injection will have the effect of condensing steam leaving the core before it can enter the containment and affect containment pressure. The effect of hot leg injection in condensing steam is included in the GOTHIC mass and energy release model as an efficiency term describing the degree of mixing that the hot leg injection undergoes in the upper plenum before entering the core.

Only mixing in the upper plenum is credited in the GOTHIC reactor system simulation although considerable mixing is also expected in hot legs and in the core.

To determine the upper plenum mixing efficiency to be used in the GOTHIC analyses, in ANP-10299P, Revision 2, the applicant examined data from scaled model reactor vessel test facilities. The first facility investigated was the UPTF. The UPTF is a geometrically full-scale mock-up of a reactor pressure vessel. The applicant investigated four tests from this facility and determined the mixing efficiency from each. The applicant also investigated data from the cylindrical core test facility (CCTF) to determine the hot leg injection mixing efficiency. The CCTF facility is a scale model of the Trojan four-loop PWR. The scaled hot leg flow rates used in these test facilities bracket that of the U.S. EPR. The applicant chose to use the lower bound mixing efficiency from these tests in the GOTHIC mass and energy release model for the U.S. EPR, which is conservative for calculating containment pressure and temperature.

The applicant performed additional calculations of mixing efficiency using a wall plume model in ANP-10299P, Revision 2. The wall plume model describes separate down-flow and up-flow regions resulting from buoyancy differences. These calculations provided additional verification that the mixing efficiency derived from the scale model test facilities is conservative.

In ANP-10299P, Revision 2, the applicant performed an analytical prediction of hot leg injection mixing efficiency that would occur for the U.S. EPR using the STAR CD computer code which performs multi-cell computational fluid dynamics (CFD) evaluations. The code was first benchmarked by modeling one of the tests at the UPTF. This test was for all liquid subcooled flow conditions. Greater mixing is expected for two phase upper plenum conditions. When compared to data from the UPTF, the predictions of STAR CD for upper plenum mixing efficiency were found be slightly conservative. The STAR CD model of the U.S. EPR included the upper plenum of the reactor vessel, a portion of the core and the hot legs, and contained about 1.5 million hexahedral cells. A single phase liquid condition was assumed in the STAR CD analysis as was the case for the UPTF test, and the upper plenum mixing efficiency was calculated. The value of mixing efficiency selected for use in the GOTHIC analyses was again confirmed to be conservative, in comparison with that calculated by STAR CD.

In a December 18, 2009, response to RAI 221, Question 06.02.01-33, the applicant described an evaluation of reactor vessel mixing following hot leg injection using the CATHARE 3D computer code. CATHARE 3D is an advanced, best-estimate code used for nuclear power plant safety analyses and describes two-phase flows with a two-fluid, six-equation model. Development has been ongoing in France since 1979. CATHARE 3D is being used for EPR safety evaluations in Europe. The applicant states that the code has been successfully benchmarked against test data from the Nuclear Reactor Thermal Hydraulics (NURETH), UPTF, and Loss-of-Fluid Test (LOFT) test facilities. The CATHARE 3D model of the U.S. EPR reactor vessel provided axial as well as radial noding so that mixing of the hot leg injection water with the two phase mixture in the reactor vessel could be evaluated. CATHARE 3D showed efficient mixing within the reactor vessel after hot leg injection was initiated with circulation patterns established within the core. The applicant’s revised evaluation model for the RCS using GOTHIC does not take credit for mixing within the reactor core but only in the upper plenum. The overall mixing efficiency calculated by CATHARE 3D was evaluated and found to be greater than that assumed by the applicant in the GOTHIC analyses for the U.S. EPR.

The staff notes that although some steam was calculated to leave the break after hot leg injection began in the CATHARE 3D analysis, the rate of steam release was less than for the applicant’s GOTHIC analysis. At 10,000 seconds, CATHARE 3D predicted a steaming rate from the break of 0.3 kg/sec (0.66 lb_m/sec); whereas, the GOTHIC model predicted a steaming rate of 5.9 kg/sec (13 lb_m/sec) which indicates that the applicant’s GOTHIC model is conservative as applied to this scenario.

The staff performed evaluations of post LOCA mass and energy release using the RELAP5 Mod 3.3 computer code. The staff's RELAP5 analysis showed significant mixing of steam in the reactor system with the SI water for both that injected into the cold legs and that injected into the hot legs.

The staff analysis predicted that loop seals in the intact loops would begin to intermittently fill with water after about 3,200 seconds, which indicates that the applicant's assumption that the intact loop seals fill at an earlier time is conservative. Like the RELAP5-BW calculations performed by the applicant, the staff's RELAP5 calculations showed rapid flow reversals during the reflood and post reflood periods in both the core and coolant loops.

The staff performed additional analyses using the advanced TRACE computer code. Like CATHARE 3D, TRACE has the capability to model the reactor vessel with axial and radial noding. The coolant flow rates predicted by TRACE are much more stable than those predicted by RELAP5. The TRACE model for the U.S. EPR reactor system was developed by the staff to include the four steam generators, a pressurizer, hot and cold leg piping, a reactor vessel, four accumulators, and MHSI and LHSI systems (Reference Krotiuk, William J., "Analysis Report: U.S. EPR™ Double-Ended Pump Suction Break with Hot Leg Injection," U.S. NRC, July 2009). A detailed model of the reactor vessel is necessary to predict thermal-hydraulic behavior in the reactor core. Consequently, the TRACE model used for this analysis employed a three-dimensional simulation of the reactor vessel. The reactor vessel was divided into eight 45°azimuthal sections, 21 vertical levels, and 3 rings. The reactor core was composed of eight 45°azimuthal sections, 10 vertical levels, and 2 rings. The core vertical levels were located between the spacer grids. The study assumed a DEG CLPS break in the U.S. EPR.

One objective of the staff's TRACE study was to investigate when loop seal plugging would occur. The staff's RELAP5 analysis had predicted that water could flow into the loop seals during the post reflood period. The RELAP5 results are questionable because of the rapid flow reversals predicted in the coolant loops. The TRACE code, which was found to be much more stable than RELAP5, predicted the loop seal in one intact cold leg to fill at 4,800 seconds and to remain filled. These results indicate that the applicant's assumed time at which the loop seals fill and, therefore, are plugged to further flow is conservative.

The main objective of the staff's TRACE study was to assess the assumption that hot leg SI will reduce the break steam flow following a LOCA by calculating the degree of mixing of the injected SI water within the reactor vessel. The TRACE analysis predicted the filling of the core with liquid following the start of hot leg safety injection which results in a rapid reduction in calculated steam flow from the break to very low values after the start of hot leg SI. The staff's TRACE analyses confirmed that the applicant's assumption of hot leg SI mixing only in the upper plenum is conservative.

To assemble a reactor, the reactor internals are inserted into the reactor vessel. In order to accomplish this, a small gap must be present between the hot leg nozzles and the continuing hot leg flow path through the downcomer region. This gap is sized to be very small especially at the elevated temperatures and pressures present during normal operation. As the reactor system cools following a LOCA, the hot leg nozzle gaps can increase in size. The increase in gap size can result in some hot leg SI flow entering the downcomer region and bypassing the reactor core. Such flow is directed toward the break, making it unavailable for core cooling and increasing core steaming (negative effect).

On the other hand, steam from the upper plenum escaping through the gaps can condense in the downcomer, reducing the amount of steam reaching the containment (positive effect).

The staff requested in RAI 221, Question 06.02.01-26, that the applicant evaluate the effect of hot leg nozzle gaps on steam release to the containment following the time when hot leg injection is initiated.

In a December 8, 2009, response to RAI 221, Question 06.02.01-26, the applicant provided evaluations using a one-dimensional quasi steady-state analysis. Gap size and hot leg water level were varied. For the limiting design basis failure assumption, only two of the hot legs would be supplied with hot leg injection. A distinction is made between the sizes of the gaps in the two loops with hot leg injection versus the size of the gaps in the two loops without hot leg injection. In varying these assumptions, the applicant concluded that, as the water level rises in the hot legs (and the upper plenum), the gap benefit/detriment ranges from slightly beneficial to slightly detrimental. The applicant stated that even when the hot leg was assumed to become significantly full, only a slight increase in the steaming rate was determined. The applicant concludes that, at that time into the event, the steaming rate to the containment would be almost halted so that the bypass of ECCS water due to the presence of hot leg gaps would not be significant.

In order to assess the effect of hot leg nozzle gap and confirm the applicant's conclusion, the staff performed analyses using the TRACE computer code. The TRACE analyses modeled the entire reactor system during the postulated LOCA event and included phenomena that were not modeled in the applicant's more basic approach. It has long been recognized that the presence of a hot leg nozzle gap would enhance post LOCA core cooling. Core cooling is enhanced by establishing a circulation pattern through the nozzle gap between the upper plenum, the downcomer and reactor core, which allows ECCS water to more readily enter the core from the bottom. The staff has not specified credit for the presence of a hot leg nozzle gap in the ECCS performance analyses for operating plants. Two TRACE analyses were performed using different assumptions for the period after hot leg SI flow starts. One analysis assumed that no hot leg gaps were present, and the other assumed a gap size larger than that which would be expected, but bounding for the purpose of this sensitive study. The TRACE analysis of hot leg injection with the assumed hot leg nozzle gap predicted an enhanced reactor vessel circulation pattern to be set up. This would allow the hot leg SI flow to circulate through the gap to the downcomer and into the core, which would suppress core steam production. No significant difference in steaming rate was produced by the presence of the hot leg nozzle gap. Therefore, the staff concludes that the presence or absence of hot leg nozzle gaps is not significant for containment analysis and considers RAI 221, Question 06.02.01-26 resolved.

Mass and Energy Release Rate Results

FSAR Tier 2, Section 6.2.1.3, presents mass and energy release rates for three of the cases analyzed: A DEG-HL break; a DEG-CLPS break; and a DEG-CLPD break. The applicant determined the limiting conditions for these breaks based on a series of sensitivity studies. All 52 LOCA cases were described in the FSAR. Based on the results from the containment pressure analyses, the applicant developed a limiting set of assumptions for determining the mass and energy release.

The following parameters were investigated:

- Break size
- Break location
- Containment back pressure

- Single failure (ECCS division or LHSI/RHR HX)
- IRWST temperature
- Time of feedwater isolation
- Location of SI division failure in relation to break location
- Effect of partial cooldown
- Effect of RCP anti-rotation device
- Effect of hot leg nozzle gap during the reflood period

The three limiting cases for which mass and energy tables were presented in the FSAR assumed that the break was full and double ended, representing a complete severance of a coolant pipe. Minimum ECCS flow was assumed, the reactor coolant pumps were assumed to be operating until tripped by automatic trips, and the containment back pressure was determined from the GOTHIC containment simulation. These assumptions resulted in maximizing mass and energy flow rates to the containment so that the resulting containment pressure would be conservatively calculated. The applicant evaluated the long term containment temperature and pressure response for only one large break at each location. Therefore, the staff requested that the applicant justify that sufficient break sizes have been analyzed using the methodology in AREVA Technical Report, ANP-10299P including the multi-noded GOTHIC containment model and that the limiting break size at each location has been identified. In RAI 368, Question 06.02.01-63, the staff requested that the applicant clarify the limiting break size at each location. In a September 30, 2010, response to RAI 368, Question 06.02.01-63, the applicant stated that FSAR Tier 2, Table 6.2.1.1, Cases 1 through 41 (LBLOCA cases) were performed using assumptions consistent with Technical Report ANP-10299P, Sections 8 and 9. The methodology of ANP-10299 specifically describes how mass and energy release is calculated for a DEG-CLPS break. As described in the proceeding discussion of this section, the staff requested that the applicant describe and justify how the methodology was applied for other break sizes and locations. These included DEG-CLPD, DEG-HL, pressurizer line, and small and split breaks. As noted in the preceding discussions, the staff agrees with the applicant's approach and, therefore, considers RAI 368, Question 06.02.01-63 resolved.

ITAAC: There are no ITAAC related to this SRP section. However, the staff noted that the LPSI flow split between the cold legs and the hot legs is used in the applicant's safety analyses. The flow split directly affects steam release to the containment following a LOCA after hot leg injection is established. Therefore, in RAI 212, Question 06.03-11, the staff requested that the applicant develop an ITAAC to demonstrate that LPSI flow will reach the hot legs when hot leg SI is initiated, consistent with analysis assumptions. In a March 24, 2010, response to RAI 212, Question 06.03-11, the applicant modified FSAR Tier 1, Table 2.2.3-3, "Safety Injection System and Residual Heat Removal System ITAAC." The staff reviewed the March 24, 2010, response to RAI 212, Question 06.03-11, and determined that sufficient information was provided to resolve the staff's concern on ITAAC. The staff's review of ITAAC affecting the containment's capability to withstand a LOCA is described in Section 14.3.11 of this report.

Technical Specifications: In RAI 221, Question 06.02.01-38(b), the staff requested that the applicant develop technical specification surveillances to verify the safety analysis of LPSI flow split between the cold legs and the hot legs following hot leg injection remains achievable during the life of the plant. The staff's review of technical specifications is described in Chapter 16 of this report. **RAI 221, Question 06.02.01-38(b) is being tracked as an open item**.

6.2.1.3.5 Combined License Information Items

No applicable items were identified in the FSAR. No additional COL information items need to be included in FSAR Tier 2, Table 1.8-2 for M&E release analyses consideration.

6.2.1.3.6 Conclusions

The staff reviewed the applicant's predictions and methodology for determining mass and energy release to the CB resulting from a postulated loss-of-coolant accident. The staff performed its review in accordance with the guidelines of SRP Section 6.2.1.3. Adherence to the guidance of SRP Section 6.2.1.3 ensures that the requirements of GDC 50 will be met for this area of the review. The staff investigated the applicant's treatment of significant energy sources including reactor power, decay heat, stored energy in the reactor core, stored energy in the RCS fluid, stored energy in the RCS metal, and secondary to primary heat transfer. The staff finds that, except for the open and confirmatory items identified above, the applicant's methodology and results for U.S. EPR are in compliance with GDC 50 and are conservative and acceptable for containment design-basis evaluations. The staff's review of the applicant's methodology in calculating the short term mass and energy release for subcompartment analysis is discussed in Section 6.2.1.2 of this report.

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

6.2.1.4.1 Introduction

Secondary system piping ruptures inside a reactor containment structure may result in significant releases of high energy fluid into the containment environment, which produce high containment temperatures and pressures. The mass and energy release following an MSLB or MFWLB depends upon the configuration of the plant's main steam and feedwater systems, plant operating conditions, and the size of the pipe rupture. The objective of the M&E release analysis for postulated secondary pipe ruptures inside containment is to provide conservative input for the containment functional design analyses as discussed in Section 6.2.1.1 of this report.

6.2.1.4.2 Summary of Application

FSAR Tier 1: There are no FSAR Tier 1 entries for this area of review. However, there are FSAR Tier 1 entries for systems related to this area of review including FSAR Tier 1, Sections 2.8.2, "Main Steam System," 2.2.4, and 2.8.6, "Main Feedwater System."

FSAR Tier 2: The applicant has provided a description of its M&E release analysis in FSAR Tier 2, Section 6.2.1.4, "Mass and Energy Release Analysis for Postulated Secondary Pipe Ruptures Inside Containment," summarized here, in part, as follows:

FSAR Tier 2, Section 6.2.1.4, states that the applicant performed a series of MSLB safety analyses and tabulated the containment temperature and pressure response from each in the FSAR. Analyses of the containment response to MFWLBs were not presented. However, the applicant presented evaluations showing that the consequences to the containment from

MFWLBs would be bounded by breaks in a main steam line. In all scenarios, the applicant adds conservatism by assuming the longest expected delay time to actuate the isolation valves and the slowest isolation valve stroke time.

FSAR Tier 2, Section 6.2.1.4, states that actuation of the emergency feedwater system during a MSLB occurs on low steam generator level for the U.S. EPR. Emergency feedwater flow to the affected SG increases the SG mass available for release to containment. The EFW water being relatively cool, also acts to condense steam within the affected steam generator which is a competing effect. The EFW system isolates on high SG level. The applicant does not take credit for this signal in the analysis. EFW isolation is assumed to occur by operator action 30 minutes after the start of the event. The applicant has run analyses for both the condition that the EFW actuates and for when it does not, and conservatively selects the worst case. For the limiting large MSLB, which exhibits an early pressure peak, the worst case was determined to be with no EFW actuation by a fraction of a pound per square inch.

FSAR Tier 2, Section 6.2.1.4, states that most severe single active failure is the failure of a MSIV, which would provide additional fluid from the main steam system (MSS) and from the unaffected steam generators that could be released to the containment via the break. The applicant's analyses takes into account that as the steam generator tubes become uncovered during the course of the accident, this allows the exiting steam to become superheated before exiting the break to enter into the containment.

A description of the RELAP5-BW model used to determine the M&E released during the blowdown phase of a postulated MSLB is provided in BAW-10169P-A, "B&W Safety Analysis Methodology for Recirculating Steam Generator Plants," and BAW-10164P-A. The methodology in these topical reports has been previously approved by the staff for operating plants. The applicant has provided additional information to show that the methodology is appropriate for use with the U.S. EPR, which the staff reviewed as described in the Technical Evaluation of this section.

The applicant's containment analyses indicate the most limiting MSLB (highest calculated containment pressures and temperatures) is a double-ended guillotine break at 20 percent rated thermal power concurrent with the single active failure of an MSIV. See Section 6.2.1.1 of this report for a description of the evaluation of the Containment Building response to the secondary system piping breaks described in this section.

ITAAC: There are no ITAAC items for this area of review.

Technical Specifications: The Technical Specifications associated with FSAR Tier 2, Section 6.2.1.4 are given in FSAR Tier 2, Chapter 16, Section 3.7, "Plant Systems."

6.2.1.4.3 Regulatory Basis

The relevant requirements of NRC regulations for this area of review, and the associated acceptance criteria, are given in NUREG-0800, Section 6.2.1.4, "Mass and Energy Release Analysis for Postulated Secondary System Pipe Ruptures," and are summarized below. Review interfaces with other SRP sections also can be found in NUREG-0800, Section 6.2.1.4.

10 CFR Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants," GDC 50, as it relates to consideration of potential energy sources not otherwise included in calculating peak conditions for postulated loss of coolant accidents to ensure the reactor containment structure, including access openings, penetrations, and the containment heat removal system, shall be

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.

6.2.1.4.4 Technical Evaluation

SRP Section 6.2.1.4 states that the sources of energy that should be considered in the analyses of steam and feedwater line break accidents include the stored energy in the affected steam generator's metal, including the outer shell, tubing, feedwater line, and steam line; stored energy in the water contained within the affected steam generator; stored energy in the feedwater transferred to the affected steam generator before closure of the isolation valves in the feedwater line; stored energy in the steam from the unaffected steam generator(s) before the closure of the isolation valves in the steam generator crossover lines; and energy transferred from the primary coolant to the water in the affected steam generator during blowdown. The steam line break accident should be analyzed for a spectrum of pipe break sizes and various plant conditions from hot standby to 102 percent of full power.

The staff notes that calculations of the mass and energy release rates during a steam or feedwater line break accident should be performed in a conservative manner from a containment response standpoint. Steam and feedwater line break analyses should assume a single active failure. The single failure may be in the steam and feedwater systems or in the containment heat removal systems to maximize the containment peak pressure and temperature.

Sources of Energy

The staff notes that postulated steam line rupture accidents within containment release a large amount of secondary system coolant and a large amount of energy into the containment. There are numerous energy sources that must be considered in the analysis. Significant energy sources include:

- Steam generator fluid inventory
- Stored energy in metal components
- Feedwater flow to the steam generator
- Auxiliary feedwater flow to the steam generator
- Primary to secondary heat transfer

The staff notes that all of the above energy sources were included in the analyses of steam line break accidents for the U.S. EPR. Treatment of the various energy sources in the mass and energy release calculations is discussed below.

Although the FSAR lists SG fluid inventory among the significant parameters which must be considered in determining the M&E release following an MSLB, the applicant did not provide the steam generator fluid inventories that were used in the analyses or describe how these inventories were determined. Therefore, in RAI 378, Question 06.02.01-82, the staff requested that the applicant provide the steam generator fluid inventories that were used in the safety analyses, address how these values were made to be conservative for containment analysis, and describe how these values will be verified for the as-built plant. In an August 12, 2010, response to RAI 378, Question 06.02.01-82, the applicant stated that the inventories were determined from the geometric characteristics of the U.S. EPR SG, operating data and operating temperatures. The staff determined that the applicant's response did not address the methodology used to determine the steam generator inventories. The staff determined that additional detail was necessary as to how the steam generator inventories were determined using the geometric characteristics of the U.S. EPR SGs, and how these values were verified against operating data and operating temperatures given that no U.S. EPRs are currently now in operation. Therefore, in follow-up RAI 451, Question 06.02.01-04-8 the staff requested that the applicant provide the necessary additional information. **RAI 451, Question 06.02.01.04-8 is being tracked as an open item.**

Calculational Model

FSAR Tier 2, Section 6.2.1.4.2, "Description of Blowdown Model," states that the RELAP5-BW computer code was used to determine the mass and energy release during the postulated steam line break accidents. The calculational model and significant correlations used in the model were previously reviewed and approved by the staff for use in evaluating currently operating PWRs in BAW-10169P-A, "B&W Safety Analysis Methodology for Recirculating Steam Generator Plants," and BAW-10164P-A, Revision 6, "RELAP5/ MOD2-BAW – An Advanced Computer Program for Light Water Reactor LOCA and Non-LOCA Transient Analyses." Nevertheless, because the containment design of U.S. EPR differs from that of the operating plants in that active systems are not available to cool and circulate the containment atmosphere, the staff reexamined the calculation models to assure the release from each of the energy sources given in the preceding paragraphs is conservative.

Mass and Energy Release Determination

FSAR Tier 2, Section 6.2.1.4 states that following rupture of one of the steam lines, flow from the adjacent (affected) steam generator will increase. At the other side of the break, flow will reverse at the break as fluid is ejected from the remaining (unaffected) steam generators. The flow rates will be limited by the flow restrictor within the nozzle of the affected steam generator and by the smallest flow restriction within the steam lines between the break and the unaffected steam generators. The applicant conservatively assumes the flow restrictors, which have a flow area of 0.1301 m^2 (1.4 ft^2), to be limiting. Blowdown of the unaffected steam generators is terminated when the MSIVs close. The Moody critical flow model is used by the applicant to calculate the break flow rate. This is the recommendation of SRP Section 6.2.1.4 and, therefore, the staff finds this approach acceptable.

Following a postulated main steam line break, rapid depressurization of the steam generators increases the flow from the main feedwater system. This adds an additional volume of water to the steam generators. Closure of the feedwater isolation valves terminates the flow. Feedwater flow to the steam generators was maximized by using the longest delay time and the longest isolation valve stroke time of 40 seconds for closure of the feedwater isolation valves. As an additional conservatism, full feedwater flow was assumed until the isolation valves were fully closed. In RAI 82, Question 06.02.01.04-1(h), the staff requested that the applicant provide additional information on how the volume of water in the feedwater piping between the isolation valves and the affected steam generator was considered.

In a June 18, 2009, response to RAI 82, Question 06.02.01.04-1(h), the applicant stated that the RELAP5-BW model includes a partial representation of the main feedwater system. The piping downstream of the control and isolation valves is included. A portion of the piping upstream of the control and isolation valves is also modeled so that the injection of additional feedwater into the steam generator from the flashing or swelling of the water contained in the main feedwater piping can be evaluated.

The amount of water injected into the steam generator due to flashing and swelling varies with the feedwater temperature and the final steam generator pressure. The unisolated section amounts to 1.9 percent of the steam generator secondary side volume. The amount of water that flashes and is injected into the affected steam generator is available to flow into the containment as steam and will add to the containment pressure. The staff agrees with this treatment of the unisolated feedwater. Therefore, the staff considers RAI 82, Question 06.02.01.04-1(h) resolved.

FSAR Tier 2, Section 6.2.1.4 states that emergency feedwater system flow is included in the RELAP5-BW model. Actuation of the EFWS increases the steam generator mass but cools the steam generator inventory. The added cooler water also enhances primary to secondary heat transfer. The net effect of EFWS actuation is expected to be relatively small, but may increase containment pressure and temperature as a result of additional steam release. EFWS actuation occurs on a low steam generator level signal.

Emergency feed water activation is assumed to occur at a conservative time. The system isolates on high level in the steam generator. Isolation of the EFWS was conservatively assumed in the applicant's analysis by operator action 30 minutes after initiation of the event. Energy stored in the nuclear fuel, primary coolant, and primary system metal was included in the calculations. Once the primary coolant temperature falls below the bulk temperature of the unaffected steam generators, heat transfer reverses. Heat will flow from the unaffected steam generators to the primary coolant which is being cooled by the affected steam generator. This effect was included in the calculations. To maximize primary to secondary heat transfer, it was assumed that offsite power is available to operate the reactor coolant pumps. With the reactor coolant pumps operating, energy transfer to the affected steam generator, and eventually to the containment, was maximized.

In RAI 266, Question 06.02.01.04-2, the staff requested that the applicant provide additional information concerning how the stored energy of metal and fluid components of the reactor system is accounted for and requested that the applicant describe and justify the mode of heat transfer. The most significant locations of heat transfer are from the primary system to the two-phase mixture in the affected steam generator, reversed heat transfer from the unaffected steam generators to the primary system, and heat transfer from the core. In a February 25, 2010, response to RAI 266, Question 06.02.01.04-2, the applicant confirmed that within the affected steam generator, nucleate boiling continues until it dries out. With the reactor coolant pumps operating, the reactor system heat transfer mode from the shutdown core and from the unaffected steam generators is forced convection. The staff concludes that heat transfer from the reactor system components is calculated in a conservative manner which is in compliance with the SRP, and is therefore acceptable. Therefore, the staff considers RAI 266, Question 06.02.01.04-2 resolved.

The staff notes that SRP Section 6.2.1.4 does not address decay heat. Addition of decay heat from the core will increase the reactor system temperature which will cause additional heat to flow to the affected steam generator. Therefore, in RAI 266, Question 06.02.01.04-3, the staff requested that the applicant describe and justify the decay heat assumptions that were used. In a February 25, 2010, response to RAI 266, Question 06.02.01.04-3, the applicant stated that the decay heat model used in the analysis is 1.2 times the values for infinite operating time in the

1971 ANS Standard plus decay of actinides. This decay heat modeling is conservatively high and is, therefore, acceptable for containment analysis. Therefore, the staff considers RAI 266, Question 06.02.01.04-3 resolved.

During power operations, the two-phase level within the secondary side of the steam generators is maintained between prescribed limits so that the tubes are covered and excessive moisture carryover out of the steam lines does not occur. Accordingly, the liquid mass within the steam generators changes as a function of reactor power. The liquid mass is greatest at low power and less at higher powers. Larger liquid mass in the steam generators at low power can result in increased mass and energy release as the liquid water gains heat and is turned to steam as it passes the heated SG tube exterior surfaces. Higher reactor power level has the offsetting effect of making more reactor heat available to be transferred through the SG tubes and into the containment. Since the overall effect of power level on mass and energy release is complex, various power levels should be investigated so that the limiting conditions for the containment can be identified. Therefore, the applicant evaluated a spectrum of power levels from full power to hot zero power (HZP).

During the depressurization of the affected steam generator, the two-phase level within the steam generator secondary will rise. If the two-phase level increases to the level of the steam separators, liquid drops may be swept out of the break with the steam.

In addition, the increase steam velocity within the two-phase region may cause liquid to be entrained with the steam and to be carried out of the break as droplets even before the two-phase level reaches the level of the separators. The assumption that some of the initial steam generator water mass leaves the steam generator as liquid results in calculation of a lower containment pressure than would be the case if perfect steam/water separation occurred in the steam generator. The water remaining in the steam generator would continue to acquire heat from the reactor system and would eventually boil to flow into the containment as steam.

In order to conservatively maximize the energy released from the steam generator during time periods when entrainment is predicted, the applicant treats the entrained liquid in the break flow that is predicted by RELAP5-BW differently depending on the steam generator two-phase levels. The location of the two-phase level is defined by the applicant as when the liquid fraction in the steam separator region, as calculated by RELAP5-BW, exceeds a threshold value. The presence of entrained liquid maximizes the calculated mass flow rate over what would be calculated by the Moody model for the case of a pure steam flow. Thus, both energy and mass flow are maximized in the applicant's MSLB calculations.

For reactor power levels below 20 percent, RELAP5-BW calculates that the threshold value will be exceeded for double-ended steam line breaks. For this reason, the maximum containment pressure calculated by the applicant is for a DEG break at 20 percent power which is the lowest power level for which liquid from the steam generator is not assumed to enter the containment through the break.

In RAI 221, Question 06.02.01-39, and RAI 266, Question 06.02.01.04-5, the staff requested that the applicant clarify the basis for the liquid fraction threshold used in the entrainment model and requested the experimental evidence from which it was derived. In a February 25, 2010, response to RAI 266, Question 06.02.01.04-5, the applicant provided the results from a sensitivity study for which the containment response was calculated under two assumptions of liquid entrainment: (1) until the threshold liquid fraction is reached in the separator region and then liquid is released from the break (FSAR assumption); and (2) perfect steam separation is assumed in the upper steam generator so that only steam is released from the break. For the second scenario, the liquid falls back into the steam generator where it is turned to steam when sufficient energy has been transferred. Cases from HZP to 40 percent power were evaluated using the two sets of assumptions. For the HZP case, the steam generators contain the largest initial fluid mass. In each case, the applicant's FSAR assumptions were determined to be conservative, and the 20 percent power case in the FSAR remained the limiting case. Therefore, the staff considers RAI 221, Question 06.02.01-39 and RAI 266, Question 06.02.01.04-5 resolved.

Rapid blowdown of the affected steam generator reduces the temperature of the reactor coolant, which for a reactor core possessing a negative temperature coefficient of reactivity, adds positive reactivity. If sufficient positive reactivity is added, the shutdown margin may be overcome so that fission power generation may be resumed even though the reactor has tripped. Treatment of power generation following rupture is described in BAW-10169P-A which indicates that a stuck control element assembly is assumed in the analysis. In RAI 82, Question 06.02.01.04-1(d), the staff requested the applicant to explain if this assumption was included in the MSLB analyses for U.S. EPR. In a December 17, 2008, response to RAI 82, Question 06.02.01.04-1(d), the applicant confirmed that the most reactive control rod was assumed to be stuck out of the core for the MSLB analyses. Therefore, the staff considers RAI 82, Question 06.02.01.04-1(d) resolved.

Limiting Break Size and Power Level

FSAR Tier 2, Section 6.2.1.4 states that steam line breaks were postulated to occur with the plant in operating conditions ranging from hot shutdown to full power. Also, various break sizes were analyzed including small break sizes to determine the limiting pressure and temperature conditions within the containment. Five break sizes (DEG and split break sizes of 929 cm^2 (1.0 ft^2), 650 cm^2 (0.7 ft^2), 483 cm^2 (0.52 ft^2), and 279 cm^2 (0.3 ft^2)) at seven reactor power levels (100 percent, 80 percent, 60 percent, 50 percent, 40 percent, 20 percent, and 0 percent) were evaluated. Additional small break sizes were evaluated at zero percent power. These were postulated breaks of 186 cm^2 (0.2 ft^2), 139 cm^2 (0.15 ft^2), 93 cm^2 (0.10 ft^2), 9.3 cm^2 (0.01 ft^2) and 4.6 cm^2 (0.005 ft^2). The limiting break was determined to be a DEG break at 20 percent reactor power.

Single Failure Analyses

FSAR Tier 2, Section 6.2.1.4 states that the MFWS includes isolation valves and control valves that close upon receipt of an isolation signal. Similarly, the steam lines are also equipped with isolation valves. The single active failure of an MFW isolation valve or control valve, failure of the MFW pumps to trip, and failure of an MSIV were considered. Because the design includes redundant safety-related MFW isolation valves, the failure of one isolation valve or control valve has minimal consequences. Similarly, isolation of the feedwater lines negates feedwater pump trip failure as a concern. The applicant identified the worst single active failure to be the failure of an MSIV in the steam line from the affected steam generator. This failure results in the release of the total inventory of one steam generator into the containment. In addition, before the other MSIVs have closed, fluid comes from the blowdown of the steam piping between the break location and isolation valves in the intact loops and initially from the unaffected steam generators. The limiting MSIV failure location is thus properly identified.

The staff notes that the FSAR presents two sets of MSLB analyses, one in FSAR Tier 2, Chapter 6 and one in FSAR Tier 2, Chapter 15, "Transient and Accident Analyses." The Chapter 6 calculations are part of containment analysis, while the FSAR Tier 2, Chapter 15 calculations are mainly to evaluate core performance and radioactive material release.

Chapter 6 addresses a break inside containment and assumes failure of an MSIV to be the limiting single failure. Chapter 15 assumes that the single failure is an MSRCV. Failure of the MSRCV, in addition to the MSLB, results in the blowdown of two steam generators and the additional cooling of the reactor system so that the reactor returns to power. No return to power was indicated in the Chapter 6 analysis, which involves the blowdown of only one steam generator.

With the postulated failure of an MSRCV as the single failure, the return to power of the core would cause additional heat to be available which might add to the energy release to the containment. On the other hand, the MSRCV is outside containment, so that the energy removed through the blowdown of the second steam generator would be transferred to the atmosphere, not the containment. This is a compensating effect for containment analysis. In RAI 266, Question 06.02.01.04-4, the staff requested that the applicant perform a containment analysis for an MSLB inside containment with an assumed MSRCV failure as the single failure. Comparison of this analysis with the limiting analysis of FSAR Chapter 6 would confirm whether the limiting single failure assumption has been identified. In RAI 297, Question 06.02.01-51, the staff requested that the applicant perform a containment analysis for an MSLB inside containment with an assumed MSRCV failure as the single failure, run the case that resulted in the highest return to power in Chapter 15, but assume that the break flow enters the containment, run the calculations until steam flow to the containment ceases, and show that the results meet all applicable containment related requirements. In an October 4, 2010, response to RAI 297, Question 06.02.01-51, the applicant stated that the assumptions made in the Chapter 15 analyses were overly conservative for use in the containment analyses for several reasons. The principal excess conservatism was in the amount of main feedwater which can be added before feedwater isolation occurs. As previously stated for the FSAR Tier 2, Chapter 6 containment analyses, the applicant assumes full feedwater flow until closure of the feedwater isolation valves using the longest delay time and the longest isolation valve stroke time of 40 seconds. The main feedwater addition assumed in the Chapter 15 analyses was much larger. Using the same assumptions for feedwater addition as assumed in FSAR Tier 2, Chapter 6 in the S-RELAP5 code, the applicant provided a comparison of the containment pressure response with those of the MSLB calculated to produce the highest containment pressure in FSAR Tier 2, Chapter 6. The Chapter 6 analysis was shown to be bounding. The staff notes that even for the containment analyses performed using the feedwater addition assumptions of the Chapter 15 analyses, the containment pressure was shown to be less than design by a considerable margin, which indicates the degree of conservatism in the EPR containment design. The staff suggests that the equipment qualification EQ envelope curve should also be shown to be conservative. The staff has requested that the calculated temperatures be compared with the EQ curve. **RAI 437, Question 06.02.01-98(b), is being tracked as an open item.**

For smaller break sizes, the applicant observed that the existing steam generator isolation signals originating in the MSS were not adequate to initiate steam generator isolation. Without steam generator isolation, feedwater would continue to flow into the affected generator and produce a continued source of steam to the containment. In addition, steam from the unaffected steam generators would continue to flow into the containment through the break. Therefore, the applicant is including high containment pressure as one of the signals to initiate steam generator isolation. This signal has not yet been incorporated into the steam generator isolation discussions of FSAR Tier 2, Section 7.3, “Engineered Safety Features Systems.” The staff will monitor that this signal is properly included in the FSAR discussions. In RAI 389, Question 06.02.02-51, the staff requested that the applicant address this concern. **RAI 389, Question 06.02.02-51 is being tracked as an open item.**

Mass and Energy Release Rate Results

Mass and energy releases were provided in the FSAR for the double-ended guillotine break of the main steam line at 20 percent power concurrent with the single active failure of one MSIV. The applicant determined this break to be the limiting case resulting in the highest containment pressure and temperature of the steam line breaks examined. The staff utilized these M&E results to perform confirmatory calculations to investigate containment temperature and pressure as discussed in Section 6.2.1.1 of this report.

Although the applicant calculated other break sizes and conditions to produce lower containment pressures and temperatures, a smaller break might produce elevated temperatures for a longer duration which might have a more detrimental consequence on the operation of safety equipment than a higher temperature of shorter duration. In order for the staff to investigate containment temperature conditions for a small break, in RAI 378, Question 06.02.01-87, the staff requested that the applicant provide an additional M&E table. In an August 5, 2010, response to RAI 378, Question 06.02.01-87, the applicant provided M&E tables. The staff determined that more information concerning the details of the GOTHIC model was necessary to allow confirmatory calculations by the staff. In follow-up RAI 437, Questions 06.02.01-99 and 06.02.01-100 the staff requested that the applicant provide additional details, including nodalization of the GOTHIC model. **RAI 437, Questions 06.02.01-99 and 06.02.01-100 are being tracked as open items.**

Feedwater Line Break Accidents

The applicant did not specifically provide analyses of MFWLBs but presented arguments in FSAR Tier 2, Section 6.2.1.4 that the consequences of MFWLBs would always be bounded by those analyzed for an MSLB of equivalent size. The maximum effective size of an MFWLB would be limited by the critical flow area at the MFW ring which is 851 cm^2 (0.916 ft^2). This area is smaller than that of the flow restrictors in the steam generator exit nozzles, which is 0.13 m^2 (1.4 ft^2). Thus, the energy flow rate to the containment would be less for a large MFWLB than for a large MSLB. The integrated energy addition from an MFWLB would also be less than that for an MSLB for two reasons: (1) For an MFWLB, any unisolated feedwater in the affected feedwater line would enter the containment directly through the break without passing over the heated steam generator tubes; and (2) for an MSWLB, much of the liquid content of the steam generator would also reach the break without contacting the steam generator tubes. The direct injection of liquid for an MFWLB would result in less energy at the break than if additional heat were added from the steam generator tubes. The staff agrees that the detailed M&E evaluation for MSLBs is sufficient for evaluation of secondary system pipe ruptures.

ITAAC: There is no ITAAC associated with this SRP section.

6.2.1.4.5 Combined License Information Items

No applicable items were identified in the FSAR. No additional COL information items need to be included in FSAR Tier 2, Table 1.8-2 for M&E release analyses for postulated secondary pipe ruptures inside containment consideration.

6.2.1.4.6 Conclusions

The applicant evaluated all significant sources of energy that contribute to the mass and energy release in case of an MSLB and made conservative assumptions for both the mass and the energy release. Analyses for the mass and energy resulting from MFWLBs were not explicitly presented; however, the applicant provided convincing arguments that the consequences from these breaks would be bounded by those of the MSLB events that were analyzed.

Failure of an MSIV was identified as the single active failure that maximizes containment pressure and temperature resulting from an MSLB. Mass and energy releases were generated for a spectrum of break sizes and power levels to permit identification of the highest containment pressure and temperature that could result. Mass and energy releases were obtained using a calculational model that has been previously approved by NRC for operating plants. Except for the above open items, the staff finds that the calculated mass and energy release calculations meet GDC 50 as it relates to consideration of potential energy sources not otherwise included in calculating peak conditions for postulated loss of coolant accidents and is sufficiently conservative for the containment design basis of the U.S. EPR design. The staff's conclusions are in part based on U.S. EPR design-specific calculations. Use of this methodology for other designs is subject to additional staff review.

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

6.2.1.5.1 Introduction

Following a loss-of-coolant accident, the ECCS supplies water to the reactor system to flood and cool the core. The rate at which the core is flooded is governed by the capability of ECCS water to displace the steam generated in the reactor vessel. The core flooding rate is dependent on the containment pressure and is decreased by a reduced containment pressure. The core flooding rate is decreased by reduced containment pressure because the resistance to steam discharge from the core is increased by the lower pressure. As part of the overall evaluation of ECCS performance, a minimum containment pressure is calculated during the time when the core is being flooded. The analyses address the operation of heat removal systems, the effectiveness of structural heat sinks in removing energy from the containment atmosphere, and other heat removal processes such as containment steam mixing with ECCS water spilling from the break. It should be noted that the minimum containment pressure analyses for ECCS performance evaluation differ from the containment functional performance analyses that are described in Section 6.2.1.1 of this report in that the conservatisms and margins are taken in opposite directions. Thus, the conservatively high containment pressure analyses described in Section 6.2.1.1 of this report are not suitable for ECCS performance evaluation and cannot be used for that purpose.

6.2.1.5.2 Summary of Application

FSAR Tier 1: There are no FSAR Tier 1 entries for this area of review.

FSAR Tier 2: The applicant has provided a description in FSAR Tier 2, Section 6.2.1.5, "Minimum Containment Pressure Analysis for Performance Capability Studies on Emergency Core Cooling System," that is summarized here, as follows:

FSAR Tier 2, Section 6.2.1.5 states that the mathematical models used to calculate the mass and energy releases to the containment for the minimum containment pressure evaluation are those applied in FSAR Tier 2, Section 15.6, "Decrease in Reactor Coolant Inventory Events," to calculate peak cladding temperature following a LBLOCA. The containment pressure model is described in FSAR Tier 2, Sections 6.2.1.5 and 15.6.5.1.2, "Method of Analysis and Assumptions," in AREVA Topical Report, ANP-10278P, Revision 1, "U.S. EPR Realistic Large

Break Loss of Coolant Accident Topical Report," January 2010, and in responses to staff RAIs 1, 82, and 297. Containment pressure calculations are performed by the ICECON module within the S-RELAP5 computer code. The applicant has applied the RLBLOCA methodology permitted by 10 CFR 50.46(a)(1)(i), "Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors." In applying RLBLOCA to the containment, the applicant treats containment pressure as a statistically varied parameter with a random sampling of the containment temperature and volume.

FSAR Tier 2, Section 6.2.1.5 states that the initial conditions for the containment are assumed by the applicant to be 100 percent rated thermal power and a pressure of 101.35 kPa (14.7 psia). A service space temperature of 7.22 ^{o}C (45 ^{o}F) and relative humidity of 70 percent are assumed and modeled within the ICECON module. This is the minimum winter design value.

FSAR Tier 2, Section 6.2.1.5 states that inside the containment, the in-containment refueling water storage tank water temperature is expected to be at the containment temperature which could be as low as 15 °C (59 °F), but could range as high as 55 °C (131 °F) in accordance with the Technical Specifications. The containment volume temperature is sampled between 37.78 °C and 55 °C (100 °F and 131 °F). The containment vapor and liquid, including the liquid in the IRWST, are modeled at the same sampled temperature.

FSAR Tier 2, Section 6.2.1.5 states that the nominal or best-estimate containment volume is 8.178×10^4 m^3 (2.888×10^6 ft^3). The upper estimate value for the containment volume is 1.114×10^5 m^3 (3.934 x 106 ft^3) and represents the empty volume of the containment dome and cylinder and also neglects the volume displaced by the internal walls and structures. This latter value is most conservative, because a larger containment volume results in a lower containment pressure which produces a higher calculated peak cladding temperature. The RLBLOCA methodology documented in ANP-10278P samples containment volume between the nominal and upper estimate values.

FSAR Tier 2, Section 6.2.1.5 states that heat transfer between the in-containment refueling water storage tank water and the containment vapor space is considered in the analysis in a conservative manner. Water spillage rates from the accumulator in the broken loop are determined as part of the core reflooding calculation and are also conservatively included in the containment model. The applicant stated that the passive heat sinks and thermo-physical properties were derived in accordance with BTP 6-2, "Minimum Containment Pressure Model for PWR ECCS Performance Evaluation."

ITAAC: There are no ITAAC items for this area of review. The applicant has committed to provide an ITAAC in the FSAR to confirm the minimum heat sink surface area after construction. FSAR Tier 1, Section 2.1.1.1, Item 2.14 and Table 2.1.1-8, Item 2.14 will be added to require that deviations between as-built construction drawings and dimensions used in the containment analyses have been reconciled. **RAI 82, Question 06.02.01.05-1(c), is being tracked as a confirmatory item.**

Technical Specifications: The Technical Specifications associated with the ECCS and the containment are specified in FSAR Tier 2, Chapter 16, Sections 3.5, "Emergency Core Cooling Systems (ECCS)," and 3.6, "Containment Systems."

6.2.1.5.3 Regulatory Basis

The relevant requirements of NRC regulations for this area of review, and the associated acceptance criteria, are specified in NUREG-0800, Section 6.2.1.5, "Minimum Containment Pressure Analysis for Emergency Core Cooling System Performance Capability Studies," and are summarized below. Review interfaces with other SRP sections also can be found in NUREG-0800, Section 6.2.1.5.

- 10 CFR 50.46(a)(1)(i): For the evaluation of emergency core cooling performance, the applicant has selected the realistic methodology as described in 10 CFR 50.46(a)(1)(i). The requirements of 10 CFR 50.46(a)(1)(i) state that the analysis should be made with an acceptable ECCS evaluation model that realistically describes the behavior of reactor during LOCAs.

Acceptance criteria adequate to meet the above requirements include:

1. RG 1.157, "Best-Estimate Calculations of Emergency Core Cooling System Performance": The model used to determine the minimum containment pressure for ECCS studies should conform to RG 1.157, Position C.3.12.1. RG 1.157 states that cooling effectiveness during the post-blowdown phase of a loss-of-coolant accident should be calculated in a best-estimate manner and should include the effects of containment heat sinks. The calculation should include the effects of operation of all pressure-reducing equipment assumed to be available. Best-estimate models will be considered acceptable provided their technical basis is demonstrated with appropriate data and analyses.

2. BTP 6-2 provides guidance for prescribing conservative estimates of containment internal structures for new plants and for calculating the associated heat transfer from the containment atmosphere. These estimates are biased to be high for the purpose of minimizing the containment pressure.

6.2.1.5.4 Technical Evaluation

The staff notes that for PWR plants, during the core reflood phase of an LBLOCA, the containment pressure acts as a back pressure for the discharge of steam generated in the core in reaching the containment. For operating PWRs, the core flooding rate has been found to depend directly on containment pressure (i.e., the core flooding rate decreases with decreased containment pressure). The core flooding rate is decreased by reduced containment pressure because the resistance to steam discharge from the core is increased by the lower pressure. Thus, the effectiveness of the ECCS depends on containment pressure.

The purpose of the staff's review is to confirm that the pressure used as a boundary condition in the ECCS performance studies, as it relates to FSAR Tier 2, Section 15.6, is in compliance with the SRP or that another justified methodology is applied by the applicant. The acceptance criteria of the SRP are based on the relevant requirements of 10 CFR 50.46, which allow the use of either an acceptable ECCS evaluation model that realistically describes the behavior of the reactor during LOCAs or an ECCS evaluation model developed in compliance with 10 CFR Part 50, Appendix K, "ECCS Evaluation Models."

The applicant selected the realistic evaluation model. The model is described in,ANP-10278P, Revision 1. The staff's evaluation of the applicant's RLBLOCA methodology can be found in the safety evaluation report on ANP-10278P, Revision 1.

In the applicant's RLBLOCA methodology, containment backpressure calculations are performed using the ICECON module within S-RELAP5. Using this methodology, the containment pressure is treated as a statistically varied parameter, and minimum containment pressure is not calculated explicitly. When the containment volume and temperature are varied, different containment pressures are obtained. The mass and energy releases are generated as a part of the internal code calculations at each time step. The calculation of the containment pressure is related to FSAR Tier 2, Section 15.6 where the ECCS performance is evaluated. Other inputs are designed to be conservative.

Containment pressure depends on the following contributions:

- Initial internal containment conditions.
- Initial outside containment ambient conditions.
- Containment volume.
- Purge supply and exhaust systems.
- Active heat sinks. (There are no active heat sinks in U.S. EPR other than the IRWST pool surface.)
- Mixing containment steam with ECCS water and accumulator nitrogen release.
- Passive heat sinks: Surface area and heat transfer coefficients.

In the following paragraphs, each of the above contributors is discussed, including the conservatism of the applicant's assumptions.

According to FSAR Tier 2, Section 6.2.1.5.2, "Initial Containment Internal Conditions," the following initial conditions were used:

- Initial containment internal conditions
    - o Pressure 101.35 kPa (14.664 psia)
    - o Temperature 37.7 °C to 55 °C (100 °F to 1131 °F)
- Outside atmospheric conditions
    - o Temperature -6.67 °C (20 °F)
    - o Relative humidity 100 percent
- Service Space Conditions
    - o Temperature 7.22 °C (45 °F)
    - o Heat transfer to containment annulus 5 Btu/hr-ft^2 °F
    - o Relative humidity 70 percent

FSAR Tier 2, Section 6.2.1.5 states that the initial pressure is the normal atmospheric pressure. The initial temperature inside the containment ranges within the technical specification limits for the IRWST and is a sampled parameter in the realistic LOCA calculation. The external temperature is a conservative lower value. It was not clear to the staff how the cold outside temperature was taken into account in the analyses. Therefore, in RAI 82, Question 06.02.01.05-1(g), the staff requested that the applicant provide the initial temperature distribution through the containment wall and the range of variation that was assumed for the realistic LOCA calculation. In a May 22, 2009, response to RAI 82, Question 06.02.01.05-1(g), the applicant stated that an initial steady state temperature distribution in the concrete containment wall will be included in the analyses which will conservatively treat the effect of the cold outside temperature, and is therefore conservative. The staff considers RAI 82, Question 06.02.01.05-1(g) resolved.

In ANP-10278P, the applicant samples containment volume. The applicant uses values from 8.178×10^4 m^3 to 1.032×10^5 m^3 (2.888×10^6 ft^3 to 3.645×10^6 ft^3). The former represents the best estimate value, and the latter is the value for an empty containment. Since the sampled values of containment volume will be the same or larger than the expected value, the applicant's assumptions are conservative for minimum pressure calculations.

In RAI 82, Question 06.02.01.05-1(d), the staff requested that the applicant provide additional information on normally operating containment purge and exhaust systems and how their operation would affect the minimum containment pressure calculations. In a May 22, 2009, response to RAI 82, Question 06.02.01.05-1(d), the applicant provided information indicating that neither containment leakage nor leakage through the containment purge valves would produce a significant effect on containment back pressure or the peak cladding temperature. For the sensitivity study, the purge valves were assumed to remain open 10 seconds after the containment reaches its high pressure isolation setpoint. The valves are relied upon to close in 5 seconds. Therefore, the applicant's evaluation is conservative.

ANP-10278P states that mass and energy releases are generated as part of the internal calculation of the code at each time step. Water spillage rates from the accumulator in the broken loop are determined as a part of the core reflood calculation and are included in the containment pressure calculation. In a May 9, 2008, response to RAI 1, Question 06.02.01-9(b), the applicant stated that thermodynamic equilibrium is assumed between the spilled ECCS water and the containment steam. This assumption conservatively minimizes the containment pressure and is therefore acceptable to the staff.

Heat transfer between the IRWST water and containment vapor was treated in a conservative manner. First, the IRWST is assumed to be well mixed. This neglects heating of the surface water and maximizes the temperature differential for heat transfer. Secondly, structures that reduce the exposed surface of the IRWST were neglected.

The break flow generated by the S-RELAP5 module of the applicant's methodology does not transmit the nitrogen that would be released following accumulator discharge to the ICECON containment module. Since the release of accumulator nitrogen would act to increase containment pressure, its omission introduces conservatism in the applicant's methodology for calculating minimum containment pressure.

FSAR Tier 2, Section 6.2.1.5 states that the U.S. EPR containment is modeled in ICECON as a dry containment, which is described as a single lumped parameter volume. This is a limitation of ICECON and the CONTEMPT/LT-022 computer code on which it is based. In RAI 82, Question 06.02.01.05-1(h), the staff requested that the applicant justify that a single-node representation is appropriate for calculating minimum containment pressure. In a June 12, 2009, response to RAI 82, Question 06.02.01.05-1(h), the applicant addressed this and other staff concerns by showing that the ICECON model is conservative when compared to a best estimate version of the multi-node GOTHIC model that the staff has reviewed for the containment functional design analyses as described in Section 6.2.1.1 of this report. The resolution of this and other staff concerns is described in the following paragraphs.

According to FSAR Tier 2, Section 6.2.1.5.3, “Other Parameters,” the passive heat sinks and thermo-physical properties that were input to ICECON were derived in accordance with BTP 6-2. The BTP provides a chart of recommended heat sink inventory as a function of containment free volume. The BTP further states that the recommended passive heat sink chart was compiled from previous reviews of PWR power plants. The simplified model is stated to be acceptable for minimum containment pressure analyses for construction permit applications until a complete identification of available heat sinks can be made. In FSAR Tier 2, Table 6.2.1-5, “Containment heat sink inventory,” the applicant has shown the heat sink components for the U.S. EPR in detail, including their material, thickness and areas. In RAI 82, Question 06.02.01.05-1(c), the staff requested that the applicant confirm that the heat sink model of the BTP is appropriate for the U.S. EPR, since the containment heat sinks for the U.S. EPR appear to be well known. In a May 22, 2009, response to RAI 82, Question 06.02.01.05-1(c), the applicant provided a sensitivity study which demonstrated that use of the BTP heat sinks is more conservative for calculating minimum containment pressure for the U.S. EPR by approximately 13.79 kPa (2 psia) when compared to the use of the heat sinks in FSAR Tier 2, Table 6.2.1-4, “Containment Initial and Boundary Conditions.” Therefore, the staff considers this issue resolved. The applicant has committed to provide an ITAAC to confirm the heat sink surface area after construction. FSAR Tier 1, Section 2.1.1.1, Item 2.14 and Table 2.1.1-8, “Reactor Building ITAAC,” Item 2.14 will be added to require that deviations between as-built construction drawings and dimensions used in the containment analyses have been reconciled. The heat sink inventory used in containment overpressure analyses from LOCA and MSLB is given in FSAR Tier 2, Table 6.2.1.4 for containment overpressure analysis. The as-built heat sink inventory should also be shown to be less capable of heat removal than that which was assumed in the minimum containment pressure calculations that were used in the ECCS performance evaluations. For the minimum containment pressure calculations, the estimated heat sink formulation of BTP 6-2 of the SRP was used. The minimum containment pressure calculations are described in FSAR Tier 2, Section 6.2.1.5. In RAI 437, Question 06.02.01-96, the staff requested that the applicant provide an ITAAC requirement to ensure that the as-built containment heat structure inventory does not exceed the heat removal capability assumed in FSAR Tier 2, Section 6.2.1.5. **RAI 437, Question 06.02.01-96 is being tracked as an open item.**

BTP 6-2 also sets specific recommendations for heat transfer correlations and their use in the minimum containment pressure calculations. Instead of the recommendations of the BTP, the applicant applied other assumptions.

The BTP states that the heat transfer coefficients to be used are 4 times those that would be calculated using the Tagami correlation during the blowdown portion of the accident and then 1.2 times those from the Uchida correlation.

Instead, the applicant used heat transfer coefficients that were 1.7 times the Uchida correlation from the beginning of the accident.

The applicant produced a comparison with experimental test data showing the Uchida correlation with a 1.7 multiplier to be conservative with respect to experimental data presented in NAI 8907-09, Revision 10, “GOTHIC Containment Analysis Package Qualification Report Version 7.2b(QA).” The staff stated concerns that the test conditions in which the experimental heat transfer was determined might not represent the dynamic conditions within the U.S. EPR containment during a large break LOCA. Therefore, in RAI 82, Question 06.02.01.05-1(a), the staff requested that the applicant provide further confirmation that the ICECON modeling of U.S. EPR is conservative for minimum containment pressure analyses.

In an SER concerning reload of AREVA fuel into the North Anna Power Station (reference Stephen Monarque, NRC letter to David Christian, Virginia Electric and Power Company, “North Anna Power Station, Unit 1 – Issuance of Amendment Re: Use of Framatome ANP Advanced Mark-BW Fuel (TAC No. MB4714),” August 20, 2004), the staff concluded that AREVA methodology for calculating minimum containment back pressure using the ICECON code could be accepted if it is shown to be conservative for each case. Therefore, in follow-up RAI 297, Question 06.02.01-52, the staff requested that the applicant perform a benchmark for the U.S. EPR showing that, for a limiting large break LOCA, the ICECON methodology is conservative compared with a best estimate version of AREVA’s multi-node GOTHIC methodology.

In a December 17, 2009, response to RAI 297, Question 06.02.01-52, the applicant provided this comparison and made the following modifications in the multi-node GOTHIC model which was used to calculate conservatively high containment pressures for the containment functional design analyses of FSAR Tier 2, Section 6.2.1.1, “Containment Structure”:

- The mass and energy was changed to match that of the ICECON containment backpressure model.

- Interfacial heat transfer to the IRWST pool was included.

- The total containment volume was increased to match the volume in the ICECON model used in the comparison.

- The containment initial conditions were changed to match those in the ICECON model.

- The containment heat sink surface areas were increased to reflect those in the ICECON model, and additional heat sinks were added which had been conservatively left out for calculation of maximum pressures.

- Material properties (heat capacity and conductivity) were changed to match those of the ICECON model.

- Condensation and heat transfer were changed from the DLM option of the original model to the DLM-FM model which has been shown to yield best estimate predictions of containment test data as discussed in EPRI NAI 8907-09 Rev 10, “GOTHIC Containment Analysis Package Qualification Report Version 7.2b(QA),” March 2009.

With these changes, multi-node GOTHIC becomes a best estimate model for calculating minimum containment pressures with some assumptions selected to calculate pressures that are conservatively low.

In comparison with the ICECON prediction, revised multi-node GOTHIC calculated pressures are slightly lower than ICECON during the blowdown period and approximately 27.6 kPa (4 psi) higher during the reflood period.

Since it is during the reflood period when containment pressure affects the core reflooding rate, the staff concludes that the ICECON model as described by the applicant for U.S. EPR containment minimum pressure analyses is conservative for that purpose and is, therefore, acceptable to the staff.

6.2.1.5.5 Combined License Information Items

No applicable items were identified in the FSAR. No additional COL information items need to be included in FSAR Tier 2, Table 1.8-2 for minimum containment pressure analysis consideration.

6.2.1.5.6 Conclusions

As described in FSAR Tier 2, Sections 6.2.1.5 and 15.6.5.1.2, in ANP-10278P, and in the responses to staff RAIs 1, 82, and 297, the evaluation model used for minimum containment pressure analysis is a combination of a realistic LOCA uncertainty study and deterministic containment pressure input. Except for the open item described above, and based on the justifications provided by the applicant as discussed in Section 6.2.1.5.4 above, the staff concludes that the containment pressure methodology that is part of the applicant's RLBLOCA evaluation model conforms to the containment back pressure recommendations of RG 1.157, SRP Section 6.2.1.5, and BTP 6-2 and, therefore, meets the requirements of 10 CFR 50.46(a)(1)(i).

6.2.2 Containment Heat Removal Systems

This section will be evaluated in Phase 4 of the U.S. EPR design certification application review.

6.2.3 Secondary Containment Functional Design

6.2.3.1 *Introduction*

For the U.S. EPR design, the RCB is completely enclosed by the Reactor Shield Building (RSB), which provides a secondary containment barrier to the release of radioactive material from the RCB. The annulus space between the RCB and the Reactor Shield Building is maintained at sub-atmospheric pressure and is filtered by the annulus ventilation system (AVS). As a secondary containment, the RSB is designed to meet specific functional criteria as specified by SRP Section 6.2.3, "Secondary Containment Functional Design."

6.2.3.2 *Summary of Application*

FSAR Tier 1: The Reactor Shield Building is described in FSAR Tier 1, Section 2.1.1, "Nuclear Island." The annulus ventilation system is described in FSAR Tier 1, Section 2.6.3, "Annulus Ventilation System."

FSAR Tier 2: The applicant provided a Tier 2 description of the Reactor Shield Building and the annulus ventilation system in FSAR Tier 2, Section 6.2.3, "Secondary Containment Functional Design," summarized here, in part, as follows:

The RSB is an upright cylinder capped with a dome that encloses the Reactor Containment Building, leaving a 1.83 m (6 ft) wide annulus space between the buildings. The primary function of the RSB is to protect the RCB from external events. A secondary function is to provide a barrier against the uncontrolled release of radioactivity. This is achieved by the AVS, which maintains the annulus space at sub-atmospheric pressure under both normal and postulated post-accident conditions.

ITAAC: The ITAAC acceptance criteria for the annulus ventilation system are specified in FSAR Tier 1, Table 2.6.3-3, "Annulus Ventilation System ITAAC." Among other tests, the ITAAC will confirm that a containment isolation signal will isolate the AVS normal operations train and start the AVS accident filtration train. The ITAAC will also confirm that the AVS provides a negative pressure of at least 6.35 mm (0.25 in.) water gauge within 305 seconds from the initiation signal.

Technical Specifications: FSAR Tier 2, Chapter 16, Section 3.6.6, "Shield Building," provides limiting conditions for operation and surveillance requirements for the RSB. Similarly, TS 3.6.7, "Annulus Ventilation System (AVS)," provides TS requirements for the annulus ventilation system.

6.2.3.3 *Regulatory Basis*

The relevant requirements of NRC regulations for this area of review and the associated acceptance criteria are specified in NUREG-0800, Section 6.2.3, and are summarized below. Review interfaces with other SRP sections can be found in NUREG-0800, Section 6.2.3.

1. GDC 4, as it relates to SSCs important to safety being designed to accommodate the effects of environmental conditions of normal operation, maintenance, testing, and postulated accidents with protection against dynamic effects (e.g., effects of missiles, pipe whip, discharging fluids) that may result from equipment failures.

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

3. GDC 43, as it relates to reactor containment and associated systems being designed to permit periodic pressure and functional testing to assure structural integrity and operability.

4. 10 CFR Part 50 Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors," as it relates to secondary containment leakage rate testing being performed in accordance with procedures specified by technical specifications or associated bases, so that bypass leakage paths are identified and associated bypass leakage rates are determined.

5. 10 CFR 52.47(b)(1), "Contents of applications; technical information," which requires that a U.S. EPR 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 NRC regulations.

Acceptance criteria adequate to meet the above requirements include:

- 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," as it relates to meeting the applicable GDC 16 requirements.

6.2.3.4 *Technical Evaluation*

Annulus Ventilation System

The annulus ventilation system is designed to contain leakage from the Reactor Containment Building by maintaining sub-atmospheric conditions in the annulus. The AVS consists of three trains, one for normal plant operation and two trains for accident mitigation. The accident trains are also available as a backup if the normal operation train is unable to maintain sub-atmospheric pressure in the annulus.

One of the accident mitigation trains alone is sufficient to collect and process all leakage during an accident. The two trains are physically separated from each other to prevent common mode failures. The AVS components are located inside the Fuel Building, which is a Seismic Category I Structure.

Reactor Shield Building

The RSB is an upright cylinder capped with a dome that encloses the RCB with a 1.83 m (6 ft) wide annular space between the buildings. The primary function of the RSB is to protect the RCB from external events. A secondary function is to provide a barrier against the uncontrolled release of radioactivity. This is achieved by the AVS, which maintains the annulus at sub-atmospheric pressure under both normal and postulated post-accident conditions.

As stated in FSAR Section 3.8.4.1.1, the RSB is a Seismic Category 1 Structure. The RSB is a heavily reinforced concrete structure. The RSB completely encloses the RCB. The RSB protects the RCB from missiles and loadings resulting from external events (e.g., hurricanes, tornados, aircraft hazards, and explosion pressure waves). The RSB is surrounded by SBs 1, 2, 3, 4, and by the Fuel Building (FB), which are Seismic Category I safety-related structures.

GDC 16 Requirements for functional capability of the Secondary Containment

Analysis of secondary containment Pressure and Temperature Response

The staff reviewed the functional capability of the secondary containment against NUREG-0800, Section 6.2.3, Acceptance Criteria 1A through H. Acceptance Criterion 1 is intended to establish a conservative design basis for the secondary containment. It is not necessarily derived from a specific accident but is based instead upon a combination of worst conditions.

The applicant provided the results of an analysis of the pressure and temperature response of the U.S. EPR secondary containment. These results are presented in FSAR Tier 2, Table 6.2.3-2, "Secondary Containment Response Analysis."

In RAI 1, Question 6.2.1-10, the staff requested that the applicant provide an evaluation of the RSB post LOCA pressure and temperature. In a September 5, 2008, response to RAI 1, Question 6.2.1-10, the applicant provided details of the analysis.

In RAI 89, Question 06.02.03-3 and 06-02.03-4, the staff requested that the applicant provide details of the pressure and temperature analysis sufficient to address NUREG-0800, Section 6.2.3, Acceptance Criteria 1A through 1H. In an October 2, 2008, response to RAI 89, Question 06.02.03-3 and 06-02.03-4, the applicant provided details of the analysis.

Details of the analysis and the staff evaluation are summarized as follows:

An analysis was performed by the applicant using the GOTHIC code to demonstrate the ability of the AVS to depressurize the RSB and maintain sub-atmospheric pressure in the RSB (the annulus) following a design-basis LOCA. The LOCA is assumed to occur concurrently with a loss of offsite power and loss of one of the AVS accident trains. The applicant stated that heat loads in the annulus were not included in the analysis because the containment penetrations for high energy pipes are enclosed in guard pipes to minimize heat transfer between the piping and the annulus. The following assumptions were used in the analysis in order to maximize annulus temperature and pressure:

- As stated in FSAR Tier 2, Table 6.2.3-2, heat transfer is calculated by methods provided in BTP 6-2. NUREG-0800 Section 6.2.3 recommends these methods. Therefore, the staff finds that NUREG-0800, Section 6.2.3 AC 1A is met.

- As stated in U.S. EPR FSAR Tier 2 section 6.2.3.3, the total pressure expansion of the primary containment structure is assumed to occur prior to the start of the remaining accident train. Therefore the staff finds that the annulus volume decrease was modeled as a step change at the beginning of the transient in accordance with NUREG-0800, Section 6.2.3 AC 1C. Based on the above, the staff find that this AC is met.

- As stated in U.S. EPR FSAR Tier 2, Table 6.2.3-2, bounding inleakage from the containment and from the environment was used. NUREG-0800 Section 6.2.3 recommends in-leakage be considered in the analysis. Therefore, the staff finds that NUREG-0800, Section 6.2.3 AC 1D is met.

- The applicant described additional conservative assumptions in responses to RAI 89, Question 06.02.03-3 and 06-02.03-4. Although discussion of these assumptions is not related to staff findings on a specific SRP acceptance criterion, the assumptions support the overall the conclusion that the analysis conservatively predicts the secondary containment pressure response. The staffs review of these assumptions are as follows:

- The primary containment design temperature was used for the primary containment LOCA temperature. This is conservative because it maximizes the heat load on the secondary containment.

- The inner surface of the concrete RCB wall was assumed to be at the containment design temperature. This is conservative because it maximizes the heat load on the secondary containment.

- The containment design pressure was used as containment pressure. This is conservative because it maximizes the compression of the annulus volume, and consequently the pressure at the start of AVS drawdown.

As stated in FSAR Tier 2 Section 6.2.3.3 and Table 6.2.3-2, after an initial increase, the AVS draws down the annulus pressure to -6.35 mm (-0.25 in.) water gauge in 305 seconds and to-6.35 cm (-2.5 in.) water gauge in 565 seconds. Also, per the analysis throughout the accident, the annulus air temperature remains below 33.3 °C (92 °F), which is less than the AVS design temperature (i.e., 100 °C (212 °F)). During normal operation, conditioned air from the Nuclear

Auxiliary Building ventilation supply shaft is supplied in the bottom of the annulus. The service compartments in the containment are maintained at a temperature of 30 °C (86 °F) maximum. An initial annulus temperature of 30.3 °C (86.6 °F) was used for the applicant's analysis. The staff finds that the applicant's analysis complies with NUREG-0800, Section 6.2.3, Acceptance Criterion 1F, because the analysis assumes the loss of offsite power and the most severe single active failure in the emergency power system, the primary containment heat removal system, core cooling systems and the secondary containment filtration system. The staff also finds that the design complies with NUREG-0800, Section 6.2.3, Acceptance Criterion 1G, because there is no equipment in the annulus that would generate a heat load.

Based on the review of the AVS Accident Filtration Fan Flowrate as described in FSAR Tier 2, Table 6.2.3-1 and clarified by the response to RAI 89, Question 06.02.03-1, the staff finds that the fan performance characteristics are considered in the analysis.   In addition, as stated in the response to RAI 89, Question 06-02.03-4, the AVS filters were considered fouled and the AVS was activated 60 seconds after start of the LOCA. NUREG-0800 Section 6.2.3 recommends that fan performance characteristics be considered in the analysis. The staff considers these assumptions conservative and therefore, the staff finds that NUREG-0800, Section 6.2.3 AC 1H is met.

Heat conduction through metal penetrations was not included in the analysis. In an October 31, 2008, response to RAI 89, Question 06.02.03-3, the applicant stated that it would take approximately 23 hours post-accident for heat from the primary containment to diffuse through the 1.3 m (4.3 ft) pipe penetration segment into the annulus using a conservation diffusivity value of 2.5×10^{-5} m^2/s (2.7×10^{-4} ft^2/s) for steel. The applicant stated that since this time frame, 23 hours, is well beyond the calculated drawdown time of 305 seconds, heat loads in the annulus were not included in the analysis. The staff attempted to reproduce these conclusions and was not able to do so. The staff would expect heat to conduct though the large metal penetrations like the equipment hatch and the personnel hatch to contribute heat to the secondary containment earlier than 23 hours into an accident sequence. Therefore, in follow-up RAI 378, Question 06.02.03-6, the staff requested that the applicant provide a more substantive justification that heat will not transfer into the secondary containment from the primary containment or include the heat so transfered into the design basis for the sizing of the ventilation system. **RAI 378, Question 06.02.03-6 is being tracked as an open item.**

Based on the above discussion, the staff finds that the U.S. EPR design meets NUREG-0800, Section 6.2.3, Acceptance Criteria 1Ai, 1C, 1D, 1F, 1G, and 1H. The open item associated with RAI 378, Question 06.02.03-6 relates to NUREG-0800, Section 6.2.3, Acceptance Criteria 1Aii, 1Aiii, and 1B which have not been met which have not been met. In RAI 462, Questions 06.02.03-7 and 06.02.03-8, the staff requested that the applicant address NUREG-0800, Section 6.2.3, Acceptance Citerion 1E. **RAI 462, Questions 06.02.03-7 and 06.02.03-8 are being tracked as open items.**

GDC 4 Requirements to protect Structures, Systems and Components important to safety against dynamic effects

The staff finds that the requirements of GDC 4 are met because, as stated in FSAR Tier 2, Section 6.2.3.3, "Safety Evaluation," high-energy pipes passing through the RSB (annulus) has guard pipes to protect against pipe failures in the secondary containment. Guard pipes are shown on the piping and instrument diagrams (P&IDs) for the main steam and main feedwater headers. The staff's review of the design criteria for guard pipes is addressed in Chapter 3, "Design of Structures, Systems, Components, and Equipment," of this report. Therefore, the staff finds that because high energy lines passing through the annulus have guard pipes, the U.S. EPR design meets NUREG-0800, Section 6.2.3, Acceptance Criterion 2, and consequently GDC 4 is met as it relates to SSCs important to safety being designed to accommodate the effects of environmental conditions of normal operation, maintenance, testing, and postulated accidents with protection against dynamic effects that may result from equipment failures.

Annulus Ventilation System Normal Operation

The staff reviewed the functional capability of the secondary containment against NUREG-0800, Section 6.2.3, Acceptance Criteria 3A through E.

The full-capacity normal-operation filtration train is designed to maintain the annulus at sub-atmospheric pressure and to maintain annulus temperature between 7.2 °C (45 °F) and 45 °C (113 °F). Sub-atmospheric pressure prevents the escape of unfiltered air to the outside environment.

The annulus temperature is maintained above 7.2 °C (45 °F) to prevent boron precipitation in the extra borating system piping. Temperature control also provides air conditioning for personnel access.

The staff noted that the maximum nominal AVS fan flow was stated to be 33.3 cubic meters per minute (m^3/min) (1,177 cubic feet per minute (cfm)) in FSAR Tier 2, Table 6.2.3-1, "Design and Performance of Annulus Ventilation System." However, FSAR Tier 2, Chapter 16 TS Bases for surveillance requirement (SR) 3.6.6.3 and SR 3.6.6.4 stated the AVS fan flow required to satisfy the surveillance requirement must be ≤37.4 m^3/min (≤1,320 cfm). Therefore, in RAI 89, Question 06.02.03-1, the staff requested that the applicant explain which AVS train flow rate was correct. In an October 31, 2008, response to RAI 89, Question 06.02.03-1, the applicant clarified that the AVS fan design air flow ranges from 1.7 to 33.3 m^3/min (from 60 to 1,177 cfm), as stated in FSAR Tier 2, Table 6.2.3-1. FSAR Tier 2, Chapter 16, SR 3.6.6.4 and TS 5.5.10 are based on a nominal AVS fan flow of 33.3 m^3/min (1,177 cfm). The values given in TS 5.5.10 for AVS fan flow are ≥30.0 m^3/min and ≤36.7 m^3/min (≥1,060 cfm and ≤1,295 cfm), which are ± 10 percent of the nominal AVS fan flow. The applicant stated that the AVS fan flow rate stated in FSAR Tier 2, Chapter 16 TS Bases for SR 3.6.6.3 and SR 3.6.6.4 will be changed from ≤37.4 m^3/min (≤1,320 cfm) to ≤36.7 m^3/min (≤1,295 cfm). The October 31, 2008, response to RAI 89, Question 06.02.03-1, clarified the basis of the Acceptance Criteria of Chapter 16, SR 3.6.6.4 and TS 5.5.10, and made it consistent with FSAR Tier 2, Section 6.2.3. The staff confirmed that the proposed revisions were made in FSAR Tier 2, Revision 2, Section 6.2.3 and, therefore, considers RAI 89, Question 06.02-03-1 closed.

The normal operation flow path is from the nuclear auxiliary building ventilation supply shaft through a fire damper, a control damper, and two isolation dampers. Supply air is distributed at the bottom of the annulus. Exhaust air is taken from the top of the annulus by the Nuclear Auxiliary Building ventilation exhaust fans. A sub-atmospheric pressure of 20.3 mm (0.8 in.) of water gauge or greater is maintained by regulating the control damper in response to redundant pressure sensors in the annulus. If the normal ventilation train is unable to maintain sufficient negative pressure in the annulus, the two accident trains are available as backup.

Technical Specification 3.6.6 requires confirmation of a negative annulus pressure greater than 6.35 mm (0.25 in.) water gauge every 12 hours during plant operation. This strategy would satisfy the requirements of GDC 16. Since the system is designed to maintain a maximum annulus negative pressure of 20.3 mm (0.8 in.) water gauge, the TS requirement is well within the capability of the AVS.

Accident Operation

The accident filtration trains contain ESF filters that are used during post-accident conditions to mitigate releases to the environment. Exhaust air from the annulus is filtered before it is released to the plant stack and then to the outside environment.

Each of the two accident trains contains a motor-controlled damper, pre-filter electric heater, upstream high-efficiency particulate air (HEPA) filter, iodine absorber, downstream HEPA filter, motor-controlled damper, fan, and back-draft damper. Each of the trains is a full capacity train. Each is capable of reducing the annulus pressure below 6.35 mm (0.25 in.) water gauge within 305 seconds in case of a DBA, and each is capable of maintaining the annulus pressure below 6.35 cm (2.5 in.) water gauge following an accident.

Under accident conditions, a containment isolation signal causes the normal filtration train to automatically stop. The normal filtration train supply dampers close immediately and the exhaust dampers close with a time delay to maintain the annulus negative pressure during switchover to the accident trains. Both accident trains start on receipt of the containment isolation signal. An alarm sounds in the main control room when the accident trains start.

As stated in FSAR Tier 2, Section 6.2.3.2.2.2, “AVS Accident Trains,” the AVS accident trains are designed in accordance with RG 1.52. Each AVS accident train is designed to reduce and maintain annulus pressure as specified in the Technical Specifications, and is located in a Seismic Category 1 building. The staff finds that the AVS accident trains meet the guidance of RG 1.52. Details of the staff review of ESF filter systems, including the AVS accident trains and how they meet the guidance of RG 1.52, are in Section 6.5.1 of this report. NUREG-0800, Section 6.2.3 guidance states that a secondary containment depressurization and filtration system should meet RG 1.52. Because the AVS meets the recommendations of RG 1.52, the staff finds that NUREG-0800, Section 6.2.3, Acceptance Criterion 3A is met.

The staff reviewed the design of the secondary containment as it relates to review guidance for NUREG-0800, Section 6.2.3, Acceptance Criterion 3B. This guidance states that the negative pressure differential to be maintained in the secondary containment and other contiguous plant areas should be no less than 6.35 mm (0.25 in.)water gauge, compared to adjacent regions under all wind conditions up to the wind speed at which diffusion becomes sufficient to assure site boundary exposures less than those calculated for the design basis accident even if exfiltration occurs. If the leakage rate exceeds 100 percent of the volume per day, there should be a special exfiltration analysis. The applicant’s response to RAI 233, Question 6.5.3-1, stated that the U.S. EPR shield building has no penetrations exposed to the environment, and is surrounded by Nuclear Island buildings. Therefore, the secondary containment would not be subject to wind driven exchanges. Based on the above, the staff finds that NUREG-0800, Section 6.2.3, Acceptance Criterion 3B is met.

As stated in FSAR Tier 2, Section 6.2.3.3, doors and hatches leading to the annulus are maintained under administrative control. As stated in FSAR Tier 2, Section 6.2.3.5, indication of the position of AVS dampers is provided in the main control room (MCR). FSAR Tier 2, Section 14.2.12.8.5, “Annulus Ventilation System (Test No. 077),” shows the AVS will be tested to demonstrate the ability to lower pressure. Technical Specification SR 3.6.6.1 requires a verification of the negative pressure in the Shield Building on a 12-hour frequency. NUREG-0800, Section 6.2.3 recommends administrative control and indication of openings in the secondary containment. Therefore, based on the FSAR information described above, the staff finds that NUREG-0800, Section 6.2.3, Acceptance Criterion 3C is met.

NUREG-0800, Section 6.2.3 provides guidance for review of secondary containments that enclose only portions of the primary containment. Since the secondary containment completely encloses the primary containment, the staff find that NUREG-0800, Section 6.2.3, Acceptance Criterion 3D is not applicable.

As stated in FSAR Tier 2 Section 3.8.4, the RSB is a safety-related, Seismic Category 1 reinforced concrete structure designed in accordance with the guidance of RG 1.142. Conformance with RG 1.142 provides assurance to the staff that the external design pressure of the secondary containment structure provides adequate margin above the maximum expected external pressure. The design of the RSB and the staff evaluation and findings as they relate to the external design pressure margin above the maximum expected external pressure of the RSB outer shield building are documented in Section 3.8.4 of this report. Based on conformance with RG 1.142, the staff finds that NUREG-0800, Section 6.2.3, Acceptance Criterion 3E is met.

Therefore, the staff finds that the US EPR meets the NUREG-0800, Section 6.2.3, Acceptance Criterion 3.

GDC 43 and 10 CFR Part 50, Appendix J, criteria for secondary containment system testing.

The staff reviewed the U.S. EPR design as it relates to GDC 43 and 10 CFR Part 50 criteria for secondary containment system testing against NUREG-0800, Section 6.2.3, Acceptance Criteria 4A and 4B.

Bypass Leakage

Bypass leakage is leakage that bypasses the filtered annulus and escapes directly to the environment. Potential bypass leakage paths exist through the double seals of the equipment hatch, personnel airlocks, fuel transfer tube, and containment ventilation system isolation valves. As described in FSAR Tier 2, Section 6.2.3.2.3 and the applicant’s December 15, 2008, response to RAI 89, Question 06.02.03-5, the sub-atmospheric pressure in the annulus is the driving force behind a leak-off system that provides a means to capture bypass leakage and route it back to the annulus to be processed. The leak-off system has no active components.

The leak-off system functions during both normal operations and postulated accident conditions. During normal operation, the system collects leakage from various components inside the Reactor Building, Fuel Building, and Safeguard Buildings 2 and 3 in leak-off lines and routes the leakage to the annulus. During DBAs, all valves in the leak-off system are open. Leaks from devices (e.g., valves, hatch seals) are collected and drained to the annulus by the pressure differential created by the AVS. The leak-off system forms part of the secondary containment bypass leakage barrier and is leakage rate tested in accordance with 10 CFR Part 50, Appendix J.

Containment penetrations that provide paths for potential bypass leakage terminate in the Fuel or Safeguard Buildings, which are supplied with ESF ventilation and filtration systems.

Containment leakage testing, including special testing requirements for bypass leakage, is in accordance with 10 CFR Part 50, Appendix J, and is reviewed in Section 6.2.6 of this report.

FSAR Tier 2, Section 15.0.3.11.2 describes the assumed primary containment leakage pathway. Following the end of the 305 second drawdown, all primary containment leakage is directed to the atmosphere at the base of the vent stack and is filtered by the AVS at 99 percent efficiency.

The treatment of bypass leakage for containment leakage rate testing is described in FSAR Tier 2, Section 6.2.6. The fraction of primary containment leakage that is assumed to bypass the secondary containment and escape directly to the environment is specified in FSAR Tier 2, Section 6.2.6.5. The U.S. EPR has no primary containment penetrations or seals that penetrate outside the secondary containment to the general environment. The maximum combined bypass leakage rate for all bypass penetrations is assumed to be zero (0.00 L_a) at a primary containment pressure of P_a.

NUREG-0800 Section 6.2.3, Acceptance Criterion 4A calls for documentation of the fraction of primary containment leakage that bypasses the secondary containment. Acceptance Criterion 4A also calls for the description of the periodic leakage testing program to address bypass leakage. Because the fraction of bypass leakage is specified in FSAR Tier 2, Section 6.2.6 and because primary containment leakage testing program described in FSAR Tier 2, Section 6.2.6 describe special testing requirements for bypass leakage, the staff finds that the U.S. EPR design meets the guidance of NUREG-0800, Section 6.2.3, Acceptance Criterion 4A. Based on conformance with NUREG-0800, Section 6.2.3, Acceptance Criterion 4A, the staff finds that the U.S. EPR design meets the requirements of 10 CFR Part 50, Appendix J, as it relates to requirements that secondary containment leakage rate tests conform to procedures specified in the technical specifications so that bypass leakage paths are identified and associated bypass leakage rates are determined.

Inspections and Monitoring

NUREG-0800 Section 6.2.3, Acceptance Criterion 4B calls for provisions in the design of the secondary containment system for inspections and monitoring of functional capability. Preoperational and periodic test programs should verify key assumptions such as depressurization time and the secondary containment inleakage rate. Surveillance requirements for the AVS are specified in FSAR Tier 2, Chapter 16, TS 3.6.6.3 and TS 3.6.6.4. Based on the review of these surveillances, FSAR Tier 2, Section 14.2.12.8.5, “Annulus Ventilation System (Test No. 077), and ITAAC acceptance criteria given in FSAR Tier 1, Table 2.6.3-3, the staff finds that the preoperational and periodic tests verify depressurization time and verify that the secondary containment inleakage rate is within acceptable limits. Therefore, the staff finds that the U.S. EPR design meets the guidance of NUREG-0800, Section 6.2.3, Acceptance Criterion 4B.

Based on compliance with the NUREG-0800, Section 6.2.3 Acceptance Criterion 4, the staff finds that the requirements of GDC 43, as it relates to reactor containment and associated systems being designed to permit periodic pressure and functional testing to assure structural integrity and operability are met.

ITAAC: The staff reviewed conformance of the ITAAC given in Tier 1 against the guidance provided in RG 1.206, Section C.II.1, “Inspections Tests Analyses and Acceptance Criteria.” The review also followed the review procedures and acceptance criteria in SRP Section 14.3 Inspections Tests Analyses and Acceptance Criteria.” The staff finds that the AVS ITAAC meet the guidance of SRP Section 14.3 AC 1 through 5. Therefore the staff finds that ITAAC acceptance criteria given in FSAR Tier 1, Table 2.6.3-3 will adequately demonstrate the acceptability of the AVS design, which assures the safety function of the secondary containment. The requirements of 10 CFR 52.47(b)(1) are met in that the FSAR application contains proposed ITAAC that are necessary and sufficient to provide reasonable assurance that a plant that incorporates the design certification is built and will operate in accordance with the design certification.

Technical Specifications: Technical Specifications relating to the RSB and the AVS are presented in FSAR Tier 2, Chapter 16, TS 3.6.6 and TS 3.6.7, respectively. Both the required actions and surveillance requirements were reviewed together with the completion times allotted for corrective action and surveillance frequencies. The staff finds the Technical Specification requirements adequate to verify that the RSB and AVS remain capable of performing their functions as credited in the safety analysis.

6.2.3.5 *Combined License Information Items*

No applicable items were identified in the FSAR. No additional COL information items need to be included in FSAR Tier 2, Table 1.8-2 for secondary containment functional design consideration.

6.2.3.6 *Conclusions*

Information provided in FSAR Section 6.2.3 was reviewed against the requirements of GDC 4, GDC 16, and GDC 43. Information presented in FSAR Tier 1, Sections 2.1.1 and 2.6.3 and in FSAR Tier 2, Chapter 16 was also utilized.

Guard pipes placed on high energy lines in the RSB protect against dynamic effects that may result from pipe failures, thus, the staff finds that with respect to high energy lines in the RSB, the U.S. EPR design meets GDC 4 requirements as they relate to SSCs important to safety being designed to accommodate the effects of environmental conditions of normal operation, maintenance, testing, and postulated accidents with protection against dynamic effects that may result from equipment failures.

A determination regarding compliance with GDC 16 cannot be made until the applicant has adequately addressed conductive heat transfer through the primary containment structure and convective heat transfer to the secondary containment atmosphere. Therefore, in RAI 378, Question 06.02.03-6, and RAI 462, Questions 06.02.03-7 and 06.02.03-8, the staff requested that the applicant provide a more detailed explanation as to why it will take 23 hours to conduct heat through the steel components **RAI 378, Questions 06.02.03-6, 06.02.03-7, and 06.02.03-8 are being tracked as open items.**

The AVS will be periodically tested to demonstrate its operability and ability to lower pressure and maintain pressure below specified limits credited in the safety analysis. Consequently, the staff finds that the RSB and AVS design, and the specified test program and surveillance program meet the requirements of GDC 43 and 10 CFR Part 50 Appendix J, as they relate to reactor containment and associated systems being designed to permit periodic pressure and functional testing to assure structural integrity and operability.

6.2.4 Containment Isolation System

The containment isolation system (CIS) allows the normal and emergency passage of fluids through the containment boundary while preserving the ability of the boundary to prevent or limit the escape of fission products from postulated accidents. The CIS is not a discrete system, but is comprised of isolation barriers (e.g., valves, closed systems, blind flanges), system piping between the isolation barriers, and associated instrumentation and control circuitry for fluid systems that penetrate the reactor building containment.

6.2.4.1 *Introduction*

The U.S. EPR containment isolation function is typically provided by two valves at each containment penetration with one valve inside and one valve outside the containment. An exception exists for component suction piping sources from the IRWST which are arranged with guard pipe inside containment in lieu of an inside CIV. Tables in FSAR Tier 1, Section 2.2, "Nuclear Island Systems," FSAR Tier 1, Section 2.3, "Severe Accident Systems," and FSAR Tier 1, Section 3.5, "Containment Isolation," give containment isolation valves associated with their respective systems. FSAR Tier 2, Table 6.2.4-1, "Containment Penetration, Isolation Valve, and Actuator Data," gives all CIVs and other containment openings (e.g., personnel airlocks, electrical and spare penetrations, hatches, fuel transfer tube, and the dedicated penetration). FSAR Tier 2, Section 6.2.4.2.4, "System Actuation," states that containment isolation actuation signals are generated in two stages (i.e., Stage 1 or Stage 2) based on containment pressure, high containment activity, or a safety injection signal. Essential systems which are relied upon to perform post-accident functions that do not receive automatic containment isolation signals can be manually isolated remotely.

Design-basis information in FSAR Tier 2, Section 6.2.4, "Containment Isolation System," is reviewed per the applicable guidance provided in SRP Section 6.2.4, "Containment Isolation System," Revision 3. The requirements that pertain to the CIS as described in FSAR Tier 1, Section 3.5, and the requirements that pertain to CIVs as specified in FSAR Tier 1, Sections 2.2 and 2.3 are also reviewed to confirm that this information is consistent with the CIS design bases as described in FSAR Tier 2, Section 6.2.4. The staff's review of FSAR Tier 1 information focuses on the overall adequacy of the FSAR Tier 1 contents and related ITAAC. As specified by SRP Section 6.2.4, the staff's review includes the following general areas: CIV design, including number, location, and valve arrangements; actuation and control features; normal and post-accident valve positions, closure times; electrical power supplies; general design criteria, including missile, pipe whip, earthquake, and environmental conditions; and detection, isolation, and atmospheric release considerations. Details of the results of these reviews are presented in Section 6.2.4.4, "Technical Evaluation," of this report.

6.2.4.2 *Summary of Application*

FSAR Tier 1: FSAR Tier 1, Section 3.5 states that the functional arrangement of the containment isolation equipment is as shown in FSAR Tier 1, Figure 3.5-1, "Representative Containment Isolation Valve Arrangement," and as indicated in FSAR Tier 1, Table 3.5-1, "Containment Isolation Equipment Mechanical Design." Table 3.5-1 states that CIVs, as well as piping for the containment isolation configurations, are designed, welded, and tested to ASME Code Section III specifications.

Other mechanical design features including location, safety function, and seismic category are as shown in FSAR Tier 1, Table 3.5-1. CIV instrumentation and control (I&C) design features, displays, control and electrical power supplies are shown in FSAR Tier 1, Table 3.5-2, "Containment Isolation Equipment I&C and Electrical Design." ITAAC requirements are contained in FSAR Tier 1, Table 3.5-3, "Containment Isolation ITAAC."

In addition to the information provided in FSAR Tier 1, Section 3.5, containment isolation valves are also described in FSAR Tier 1, Sections 2.2.2, "In-Containment Refueling Water Storage

Tank System," through Section 2.2.7, "Extra Borating System," and Section 2.3.3, "Severe Accident Heat Removal System." The Design of these valves is further described in tables within these sections that are similar to FSAR Tier 1, Table 3.5-1.

FSAR Tier 2: Information provided by the applicant in FSAR Tier 2, Section 6.2.4, "Containment Leakage Testing," is summarized here, in part, as follows::

The applicant states that the piping systems penetrating the containment are provided with leak detection, isolation, and containment isolation capabilities having redundancy, reliability, and performance capabilities that reflect the importance to safety. The FSAR concludes that the systems are designed with the capability to periodically test for valve operability and to determine valve leakage in accordance with GDC 54, "Piping Systems Penetrating Containment." In FSAR Tier 2, Section 6.2.4.1, "Design Bases," the applicant also states:

- Piping that is part of the RCPB and penetrates containment is provided with containment isolation valves as required by GDC 55, "Reactor Coolant Pressure Boundary Penetrating Containment."

- Piping that connects directly to the containment atmosphere and penetrates containment is provided with containment isolation valves as required by GDC 56, "Primary Containment Isolation."

- Piping that penetrates containment and is neither part of the RCPB nor connected directly to the containment atmosphere is provided with containment isolation valves as required by GDC 57, "Closed System Isolation Valves."

FSAR Tier 2, Section 6.2.4.1 also includes the following information regarding the design of CIVs and associated equipment:

- Containment isolation system components, including associated control systems, are designed in accordance with 10 CFR 50.34(f)(2)(xiv) and 10 CFR 50.34(f)(2)(xv), and conform to the guidance provided in RG 1.141, "Containment Isolation Provisions for Fluid Systems," July 2010, by adhering to American Nationals Standards Institute (ANSI)/ANS-56.2, "Containment Isolation Provisions for Fluid Systems After a LOCA," American National Standards Institute/American Nuclear Society, 1989.

- Containment isolation valve power supplies satisfy station blackout requirements in accordance with 10 CFR 50.63, "Loss of All Alternating Current Power."

- Instrumentation and control lines that penetrate containment conform to the guidance provided in RG 1.11, "Instrument Lines Penetrating Primary Reactor Containment," March 2010.

Isolation valve closure times are as described in FSAR Tier 2, Section 6.2.4.2.6, "Isolation Valve Closure Times," and as given in FSAR Tier 2, Table 6.2.4-1. Penetration overpressure protection is provided as shown in FSAR Tier 2, Figure 6.2.4-2, "Containment Isolation Valve Arrangements for Overpressure Protection."

ITAAC: FSAR Tier 1 ITAAC for containment isolation equipment are shown in Table 3.5-3 of the application, and in corresponding tables in FSAR Tier 1, Sections 2.2 and 2.3 for CIVs that are described in those sections.

Technical Specifications: Technical specifications for containment isolation valves are as described in FSAR Tier 2, Chapter 16. Per proposed Technical Specification Section 3.6.3, “Containment Isolation Valves,” each containment isolation valve shall be operable in Modes 1, 2, 3, and 4. With an inoperable valve, isolation of the affected flow path is required within 4 hours. Valves, except for the full flow purge flow paths, may be un-isolated intermittently under administrative controls.

6.2.4.3 *Regulatory Basis*

The relevant requirements of NRC regulations for this area of review, and the associated acceptance criteria, are given in NUREG-0800, Section 6.2.4 and are summarized below. Review interfaces with other SRP sections can also be found in NUREG-0800, Section 6.2.4.

1. GDC 1, “Quality Standards and Records,” GDC 2, “Design Bases for Protection Against Natural Phenomena,” and GDC 4, “Environmental and Dynamic Effects Design Bases,” in that CIVs are important to safety and therefore must be designed and fabricated to appropriate quality standards and must be designed to withstand the effects of natural phenomena and the dynamic effects of missiles, piping failures, and resulting environmental conditions without losing the capability to perform their safety functions.

2. GDC 16, “Containment Design,” as it relates to reactor containment and associated systems establishing an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment.

3. GDC 54, insofar as it requires that piping systems that penetrate primary containment be provided with leak detection, isolation, and containment capabilities having redundancy, reliability, and performance capabilities which reflect the importance to safety of isolating such systems, 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.

4. GDC 55 and GDC 56 as they relate to isolation valves for lines penetrating the primary containment boundary as parts of the reactor coolant pressure boundary (GDC 55) or as direct connections to the containment atmosphere (GDC 56) as follows:

    - One locked-closed isolation valve inside and one outside containment; or

    - One automatic isolation valve inside and one locked-closed isolation valve outside containment; or

    - One locked-closed isolation valve inside and one automatic isolation valve outside containment; or

    - One automatic isolation valve inside and one outside containment, noting that a simple check valve is not an acceptable automatic isolation valve for use outside containment.

5. GDC 57 as it relates to the requirement that lines penetrating the primary containment boundary and that are neither part of the reactor coolant pressure boundary nor connected directly to the containment atmosphere have at least one locked-closed, remote-manual, or automatic isolation valve outside containment.

6. Isolation valves outside containment should be located as close to it as practical, as required by GDC 55, GDC 56, and GDC 57.

7. 10 CFR 50.34(f)(3)(iv), which requires one or more dedicated penetrations to be provided, equivalent in size to a single 3-foot diameter opening, in order to allow for future installation of systems that may be necessary to prevent containment failure.

8. 10 CFR 50.63(a)(2), as it relates to ensuring that appropriate containment integrity is maintained in the event of a station blackout for a specified duration.

9. 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 is 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 NRC regulations.

10. 10 CFR 52.47(a)(8), as it relates to demonstrating compliance with any technically relevant portions of the TMI-related requirements set forth in 10 CFR50.34(f)(2)(xiv) with respect to containment isolation and 10 CFR 50.34(f)(2)(xv), in particular with respect to containment purge isolation for design certification and COL reviews.

Acceptance criteria adequate to meet the above requirements include:

1. RG 1.141, July 2010

2. BTP 6-4, “Containment Purging During Normal Plant Operations,” Revision 3

3. RG 1.155, “Station Blackout,” August 1988

4. RG 1.11, “Instrument Lines Penetrating Primary Reactor Containment,” March 2010

6.2.4.4 *Technical Evaluation*

The CIS allows the normal or emergency passage of fluids through the containment boundary while preserving the ability of the boundary to prevent or limit the escape of fission products from postulated accidents. The CIS is not a discrete system, but is comprised of isolation barriers (e.g., valves, closed systems, blind flanges), system piping between the isolation barriers, and associated instrumentation and control circuitry for fluid systems that penetrate the reactor building containment.

The U.S. EPR containment isolation function is typically provided by two valves at each containment penetration with one valve inside and one valve outside the containment.
An exception exists for component suction piping sources from the IRWST that are arranged with a guard pipe inside containment in lieu of an inside containment CIV. FSAR Tier 1, tables in Section 2.2, 2.3, and 3.5, and FSAR Tier 2, Table 6.2.4-1, list all CIVs and other containment openings (e.g., personnel airlocks, hatches, fuel transfer tube). Containment isolation actuation signals are generated in two stages (i.e., Stage 1 or Stage 2) based on containment pressure, high containment activity, or a safety injection signal. Essential systems that are relied upon to perform post-accident functions and do not receive automatic signals can be manually isolated remotely.

The staff reviewed the design of containment isolation valves in accordance with SRP Section 6.2.4. The acceptability of the system design is based on meeting the requirements of GDC 1, GDC 2, GDC 4, GDC 16, GDC 54, GDC 55, GDC 56, and GDC 57. The results of the staff's review and conclusions that were reached are as follows.

The staff finds that all CIVs including annulus penetrations and guard pipes in CIV applications (FSAR Tier 1, tables in Sections 2.2, 2.3, and 3.5, and FSAR Tier 2, Section 6.2.4.1) are designed and tested to ASME Code Section III, Quality Group B, specifications per RG 1.26, "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants," March 2007. CIVs are also designed to withstand the effects of natural phenomena without loss of the capability to perform their specified safety functions. As shown in the FSAR Tier 1 tables, CIVs are designed to satisfy the guidance specified in RG 1.29, "Seismic Design Classification," March 2007. Accordingly, the staff finds the containment isolation valves to be designed to quality standards in accordance with their safety function.

The staff reviewed the arrangements and power supplies for CIVs and related devices given in the tables included in FSAR Tier 1, Sections 2.2, 2.3, and 3.5, and in FSAR Tier 2, Table 6.2.4-1. The staff reviewed system-specific functional arrangement drawings provided with the application properly show the locations and configurations of CIVs. The staff did not identify any single fault that would preventing isolation of the containment. Class 1E power sources supply powered CIVs. Two isolation barriers associated with each containment penetration are provided, and where electrical power is needed for actuating the CIVs, power sources to valve operators are provided via independent emergency diesel generators (EDGs). In a July 3, 2008, response to RAI 12, Question 06.02.04-1, the applicant confirmed that all CIVs are provided with power during normal operations. Post-accident reliability of CIVs is enhanced by the capability of powering the CIVs with alternate power sources as shown in the tables that give I&C and electrical design specifications for the FSAR Tier 1 sections referred to above.

The staff finds the combinations of isolation valves and devices are in accordance with GDC 55, and GDC 56, by providing locked-closed, automatic, or a combination of locked-closed and automatic valves on both sides of the containment barrier. The staff finds the systems which are closed inside containment are provided with a single isolation valve outside containment, either locked-closed, automatic, or capable of remote–manual actuation, and thereby meet the GDC 57 requirements.

GDC 55, GDC 56, and GDC 57 require that isolation valves outside containment be located as close to the containment as practical. FSAR Tier 2, Section 6.2.4, commits to this.

Based upon a review of the FSAR, the staff noted that:

1. The distances from the outside containment CIVs to the containment are not included in FSAR Tier 2, Table 6.2.4-1.

2. There are no FSAR Tier 1 commitments in Sections 2.2, 2.3, and 3.5 calling for outboard CIVs to be located as close to the containment wall as is practical.

3. No COL information items have been identified (FSAR Tier 2, Table 1.8-2, "U.S. EPR Combined License Information Items") associated with locating the subject valves as close as is practical to the containment wall or defining maximum allowable distances.

4. No ITAAC for FSAR Tier 1, Sections 2.2, 2.3, and 3.5 have been identified to confirm that the outboard CIVs are located as close to the containment wall is as practical or within the allowable distances that are specified.

Thus, the staff could not conclude that the subject GDC requirement is satisfied, because the proximity of CIVs to the containment has not been addressed; and no FSAR Tier 1 ITAAC and no COL information items have been established to address this item. Therefore, in RAI 410, Question 06.02.04-10, the staff requested that the applicant provide the distance from each CIV outside containment to the containment, add this information to the FSAR Tier 2, Table 6.2.4-1, "Containment Isolation Valve and Actuator," and provide an ITAAC for each outboard CIV for this distance in the respective sections in FSAR Tier 1. **RAI 410, Question 06.02.04-10 is being tracked as an open item.**

The containment design must include one or more dedicated containment penetrations that are equivalent in size to a single 3-foot diameter opening as specified by 10 CFR 50.34(f)(3)(iv). FSAR Tier 2, Section 19.2.3.3.8, "Containment Venting," states that an existing containment penetration will be used to meet this requirement. In RAI 181, Question 19-271, the staff requested that the applicant identify the systems or components that could accomplish the function of a dedicated containment penetration. In a May 13, 2009, response to RAI 181, Question 19-271, the applicant stated that the dedicated penetration will have a diameter of 91.44 cm (36 in.). The staff finds that the proposed 91.44-cm (36-in.) diameter dedicated penetration meets the regulatory requirement as specified by 10 CFR 50.34(f)(3)(iv). FSAR Tier 2, Table 6.2.4-1, lists and describes all containment penetrations, including the dedicated penetration, which is identified as 60BQ064. However, the required size of the penetration is not included in the table and needs to be specified. Therefore, in RAI 372, Question 06.02.04-9, the staff requested that the applicant identify the 91.44 cm (36 in.) penetration by equipment number and add the 91.44 cm (36 in.) penetration to the FSAR Tier 2, Table 6.2.4-1. **RAI 372, Question 06.02.04-9 is being tracked as an open item.**

The staff reviewed the special valve configurations as shown in FSAR Tier 2, Figure 6.2.4-1, "Representative Containment Isolation Valve Arrangement." Specifically, suction piping from the IRWST sumps is not provided with an inboard containment isolation valve. In lieu of the inboard valve, ASME Class III guard pipe is used as the proposed equivalent.

The staff reviewed the use of a concrete embedded guard pipe for the IRWST sump piping inside containment with a single outboard CIV in lieu of using inboard and outboard CIVs. Given the design of such systems, an inboard valve would be impractical and the proposed alternative is in conformance with the guidance provided in SRP Section 6.2.4. The embedded guard pipe provides a dual pressure boundary which provides greater assurance of leak-tightness; the IRWST suction piping and the guard pipe itself are designed as ASME Code Class III and are seismically designed; and the embedded nature of the IRWST suction piping and guard pipe makes them less susceptible to loss of integrity. In addition, FSAR Tier 2, Section 6.2.4.2.2, "Isolation of Lines Serving as Part of the RCPB or Connected Directly to the Containment Atmosphere," states that the SIS outside containment is protected from missiles and seismic events, is constructed to seismic Category I design and Group B quality standards, and has a design temperature and pressure rating at least equal to that of the containment. Similar configurations exist for the SAHRS and CVCS suctions from the IRWST. While portions of these systems are non-safety-related, each penetration is provided with two redundant outside CIVs, which satisfies SRP Section 6.2.4, Acceptance Criterion 4. These configurations are specified in SRP Section 6.2.4 as acceptable ways of meeting the regulations and, therefore, the staff finds them acceptable. Accordingly, the staff finds the use of a concrete embedded guard pipe for the IRWST sump piping inside containment with a single outboard CIV acceptable in lieu of using inboard and outboard CIVs.

The staff reviewed the post-accident, normal and shutdown CIV positions shown in FSAR Tier 1, Sections 2.2, 2.3, and 3.5; and in FSAR Tier 2, Table 6.2.4-1, and concludes that they are consistent with their respective system functions. In general, power-operated valves in fluid systems having no post-accident safety function (non-essential systems) are designed to close automatically on a containment isolation signal. Also, position indication for CIVs is provided in the main control room as indicated in the tables that are included in FSAR Tier 1, Sections 2.2, 2.3, and 3.5.

The staff reviewed actuation and control features for the CIVs. Valves are either automatically isolated by the protection system (PS) through Stage 1 or Stage 2 containment isolation signals, or they are remotely operated for essential systems that are necessary for accident mitigation. The majority of CIVs are closed upon receipt of a Stage 1 isolation signal. Exceptions include CIVs in the component cooling water system (CCWS) that must remain open to provide reactor coolant pump seal injection and seal cooling and those valves associated with the IRWST sump suction lines for the SAHRS, which isolate on a Stage 2 isolation signal. The staff finds that all of the CIVs are appropriately isolated based on their function and role in accident mitigation. In addition, the staff finds that FSAR Tier 2, Table 6.2.4-1 provided the proper classifications for systems containing CIVs as essential or non-essential in accordance with 10 CFR 50.34(f)(2)(xiv).

Instrumentation and control lines that penetrate containment are designed to satisfy the guidance provided by RG 1.11. Conformance to this document is an acceptable approach to limit the rate of coolant loss due to instrument line ruptures outside containment. However, no means of automatic or remote-manual isolation as suggested by RG 1.11 are provided for 12 instrument lines that penetrate containment. FSAR Tier 2, Section 6.2.4.2.1, "General System Design," states that for this class of line, the design is acceptable based on "other defined bases" as permitted by RG 1.11. The bases cited are from ANSI/ANS 56.2, Section 3.6.2, "Instrument Lines," which indicates that instrument lines are acceptable without isolation valves provided they are closed systems, are designed to withstand the maximum containment structural integrity test pressure and the containment design temperature, and are protected from missiles and dynamic effects. The staff confirmed that FSAR Tier 2, Section 6.2.4.2.1, includes these additional criteria. The staff finds this to be an acceptable basis for not having automatic or remote-manual isolation capability for the subject lines. Per FSAR Tier 2, Table 6.2.4-1, Sheets 10 & 11, each line is provided with an outboard safety-related manual containment isolation valve. The radiological consequences associated with limiting failures of small lines outside of containment are addressed in this report under Section 15.0.3, "Radiological Consequences of Design Basis Accidents."

The staff reviewed the signals from the plant protection system for initiating containment isolation. The U.S. EPR provides containment isolation in two stages to isolate nonessential components based on the size of the break, as described in detail below. The staff finds that the diversity of parameters that are sensed for isolating containment (e.g., two levels of containment pressure, high range radioactivity, and SIS signal) is sufficient to ensure timely isolation of non-essential penetrations, in accordance with SRP 6.2.4, Acceptance Criterion 12. FSAR Tier 2, Section 7.3.1.2.9, "Containment Isolation," states that there are no operating bypasses associated with containment isolation. While the sense and command outputs for containment isolation can be reset manually (i.e., restoration of non-essential systems post-accident), reset does not result in a change of state of the containment isolation actuators.

This demonstrates that the 10 CFR 50.34(f)(2)(xiv) requirement is satisfied with respect to preventing automatic opening of CIVs when the containment isolation signals are reset.

FSAR Tier 2, Section 6.2.4.2.4 states that Max1p is set to a minimum value compatible with normal operating conditions. FSAR Tier 2, Chapter 16, proposed Technical Specification Table 3.3.1-2, “Acquisition and Processing Unit Requirements Referenced from Table 3.3.1-1,” states that Stage 1 containment isolation occurs at a containment pressure of 128.9 kPa (18.7 psia), and Stage 2 containment isolation occurs at a containment pressure of 250.3 kPa (36.3 psia). Normally containment is kept at a negative pressure and is controlled by the non-safety containment ventilation system. The staff concludes that 128.9 kPa (18.7 psia) is a reasonable pressure to initiate containment isolation and is as low as compatible with normal operation, and that the requirements of 10 CFR 50.34(f)(2)(xiv)(D) are met.

Essential systems having post-accident functions are provided with remote-manual isolation valves, with the exception of the twelve instrument lines discussed above. FSAR Tier 2, Section 6.2.4.2, “System Design,” states that sumps in the Safeguard Buildings, Fuel Building, and Reactor Building are monitored and alarms or indications are provided to the control room operators to detect and alert operators to the presence of water in these areas. This, coupled with the methods available to detect intersystem leakage described in FSAR Tier 2, Section 5.2.5.3, “Detecting and Monitoring Intersystem Leakage,” including monitoring level, pressure or temperature, and increasing radioactivity, supports the conclusion that there are adequate leakage detection provisions to enable the operators to detect leakage and identify lines that should be isolated.

Leak testing of individual isolation barriers, as well as periodic operability testing of power-operated isolation valves and the containment isolation system, and the adequacy of Technical Specification related surveillance requirements are evaluated in Section 6.2.6, “Containment Leakage Testing,” of this report.

The staff reviewed individual valve closure times for the CIVs given in FSAR Tier 2, Table 6.2.4-1. In FSAR Tier 2, Section 6.2.4.2.6, the applicant proposed an exception to valve closure time specifications associated with the four containment full flow purge valves. It states that since these valves are maintained closed in accordance with TS 3.6.3 during normal plant operation (Modes 1, 2, 3, and 4), no closure time requirements are associated with these valves. This meets the guidance of BTP 6-4 and is acceptable to the staff. Consistent with the guidance contained in SRP Section 6.2.4, all other specified valve closure times are less than 1 minute regardless of the valve size. The staff finds that the closure times associated with CIVs used for low-flow purging of the containment are conservative with respect to accident assumptions. The normal purge CIVs should be closed within 5 seconds or less per FSAR Tier 2, Table 6.2.4-1. The staff finds that the normal valve positions for the CVS supply lines and the 5 second closure times, assuming CIVs used for purging are open, are in conformance to the guidance provided in BTP 6-4 and the analysis referred to in FSAR Tier 2, Table 14.3-2, “Radiological Analysis (Safety Significant Features),” Items 2-10 and 2-14, that was performed to satisfy the requirements as specified in 10 CFR Part 50, Appendix K, “ECCS Evaluation Models.” At the start of the accident, the safety analysis assumes the containment is in the purge mode, and further assumes this mode is terminated within 10 seconds. Since the 10 second closure time is conservative with respect to the 5 second closure time recommended in the guidance with respect to radiological consequences, the staff finds the specified closure times for the purge CIVs acceptable.

FSAR Tier 2, Section 6.2.4.2.7, "Penetrations Overpressure Protection," states that overpressure protection is provided for liquid-filled piping between containment isolation barriers to prevent damage when the piping is isolated unless it can be demonstrated that the pressure between the isolation barriers cannot exceed the design pressure of the isolation barriers or the piping between the isolation barriers. The staff reviewed selected systems and containment isolation configurations in FSAR Tier 2, Figure 6.2.4-2 and concluded that the CIV arrangements for the U.S. EPR meet the overpressure protection guidelines as stated in RG 1.141 by either providing a relief valve inside containment between the CIVs, or a check valve inside containment, either as a bypass to the inboard CIV or as the inboard CIV.

As discussed in FSAR Tier 2, Section 8.4.2.6.2, "RG 1.155 C.3.2-Evaluation of Plant-Specific Station Blackout Capability (Station Blackout Coping Capability)," containment integrity is ensured during a station blackout (SBO) event by the two independent alternate alternating current (Aac) station blackout diesel generator sets which can supply power to all four vital 1E buses. Thus, the Aac diesel generator sets and the ASME Code Class 1E uninterruptible power supply provide indication and closure power for the CIVs so the valves can be monitored from the control room. If any CIVs are open at the beginning of an SBO event, they can be closed by the operators. The following categories of valves do not include SBO provisions for closure such that the operators may not be able to close them:

- Valves normally locked closed during operation
- Valves that fail closed on a loss of power
- Check valves
- Valves in non-radioactive closed-loop systems not expected to be breached during an SBO event (excluding lines that communicate directly with the containment atmosphere)
- Valves of less than 7.62 cm (3 in.) nominal diameter

This arrangement satisfies the guidance provided in RG 1.155 for being able to establish and maintain containment integrity during an SBO event, thereby establishing compliance with the requirements specified by 10 CFR 50.63 in this regard. Therefore, the staff finds this arrangement acceptable.

Technical Specifications

The staff reviewed Technical Specification requirements that are specified for the CIVs in FSAR Tier 2, Chapter 16. CIVs are addressed in proposed Technical Specification Section 3.6.3, and the staff found this TS section to be similar in content to existing specifications for currently operating U.S. reactors. All CIVs are required to be operable in Modes 1, 2, 3, and 4, and there is a 4-hour action-time to isolate and secure penetrations that have an inoperable CIV, with a subsequent 31-day verification of valve position. The staff finds the technical specifications acceptable.

ITAAC

The staff reviewed the proposed ITAAC requirements specified in FSAR Tier 1, Sections 2.2, 2.3, and 3.5 with respect to CIV considerations. As discussed in Section 6.2.4, "Technical Evaluation," of this report, the staff determinded that the proximity of the CIVs to the containment has not been adequately addressed; and appropriate ITAAC have not been established in this regard. Consequently, in RAI 410, Question 06.02.04-10, the staff requested that the applicant address this issue. **RAI 479, Question 06.02.04-11 is being tracked as an open item.** In all other respects, the staff finds the proposed ITAAC acceptable.

6.2.4.5 *Combined License Information Items*

No COL information items are specified by the applicant for this area of review. However, as discussed in Section 6.2.4, "Technical Evaluation," of this report, the staff determined that a COL information item may be needed for COL applicants to address containment proximity considerations for the outside CIVs. The staff has requested that the applicant address this issue in RAI 410, Question 06.02.04-10. **RAI 479, Question 06.02.04-11 is being tracked as an open item.**

6.2.4.6 *Conclusions*

The staff evaluated the proposed design of the CIS for the U.S. EPR as described primarily in FSAR Tier 2, Section 6.2.4, and reviewed proposed design requirements that pertain to the CIS and CIVs as specified in FSAR Tier 1, Sections 2.2, 2.3, and 3.5. Other FSAR Tier 2 information was also evaluated with respect to CIV considerations, such as the Technical Specification requirements provided in FSAR Tier 2, Chapter 16, radiological release assumptions described in FSAR Tier 2, Section 14.3, and severe accident design provisions discussed in FSAR Tier 2, Section 19.2. The staff's evaluation was performed in accordance with the guidance provided in SRP Section 6.2.4 and included the following general areas: CIV design, including number, location, and valve arrangements; actuation and control features; normal and post-accident valve positions, closure times; electrical power supplies; general design criteria, including missile, pipe whip, earthquake, and environmental conditions; and detection, isolation, and atmospheric release considerations.

Containment isolation actuation signals are generated in two stages (i.e., Stage 1 or Stage 2) based on containment pressure, high containment activity, or a safety injection signal. Essential systems which are relied upon to perform post-accident functions that do not receive automatic containment isolation signals can be manually isolated remotely from the control room if necessary. The CIS includes redundant leak-tight barriers at the containment penetrations, and redundant motor-operated CIVs are powered from separate safety-related electrical divisions that receive emergency power from their respective EDG sets. Alternate electrical feeds provide additional defense-in-depth to close these valves and to satisfy station blackout considerations. As set forth above, and except for the open items specified above, the staff finds that the proposed design of the CIS can be relied upon to isolate containment when called upon to do so, with and without an SBO event, and in the event of any concurrent single active failure that might occur.

Compliance with GDC 1, GDC 2, and GDC 4 is satisfied by design provisions that invoke ASME Section III, Seismic Category I, and Quality Group B specifications for the CIVs and barriers that are credited for establishing containment isolation. As described above, the staff finds that the CIV configurations meet or are the equivalent of those described in GDCs 55, GDC 56, and GDC 57. In addition, because all CIVs including annulus penetrations and guard pipes in CIV applications are designed and tested to ASME Code, Section III, Quality Group B, the staff finds that GDC 16 is satisfied. The staff finds that the design features and provisions that enable the CIS to isolate containment when called upon to do so with and without off-site power available, with a concurrent single active failure, and by controlling the use of purge valves in conformance to the guidance provided in BTP 6-4, meet the guidance of SRP Section 6.2.4. As explained above, the CIS design conforms to RG 1.141 by establishing containment isolation set points that are commensurate with normal plant operating conditions, and by properly distinguishing between essential and non-essential containment penetrations, thereby satisfying the TMI-related requirements specified in 10 CFR 52.47(a)(8), 10 CFR 50.34(f)(2)(xiv), and 10 CFR 50.34(f)(2)(xv). The design includes a dedicated 91.44-cm (36-in.) diameter penetration which satisfies the requirement specified in 10 CFR 50.34(f)(3)(iv). The design conforms to Regulatory Position C.3.2.7 of RG 1.155 and includes an alternate source of power for CIV control and indication, thereby satisfying the SBO requirements specified in 10 CFR 50.63.

Except for the open items identified above, the staff also finds that the Technical Specifications and ITAAC proposed for the CIS are acceptable. The proposed Technical Specifications for the CIS are consistent with those used at current operating plants, and the ITAAC ensure that critical CIS design details are satisfied by the as-built plant. Therefore, the staff finds the proposed ITAAC acceptable and adequate to ensure that the plant will be built in accordance with the design specifications, thereby satisfying the requirements specified in 10 CFR 52.47(b)(1). The staff also finds that the Technical Specifications are adequate to ensure that the CIS will remain capable of isolating containment during plant operation consistent with accident analysis assumptions.

6.2.5 Combustible Gas Control in Containment

6.2.5.1 *Introduction*

Control of combustible gases in containment is described in FSAR Tier 2, Section 6.2.5, “Combustible Gas Control in Containment.” The CGCS in conjunction with the hydrogen monitoring system (HMS) mitigate the consequences of postulated accidents by mixing, monitoring, preventing, or removing combustible gas concentrations that may be released into the containment atmosphere in the event of a DBA or a significant beyond design-basis accident. Following a postulated accident, hydrogen and oxygen may accumulate inside the containment. Combustible gas is predominantly generated within the containment as a result of reactions between fuel-clad and reactor coolant and also by interactions between molten core and concrete. Since significant amounts of combustible gas can be generated, these gases must be mixed and removed to prevent the uncontrolled hydrogen/oxygen recombination (detonation) and possible damage to the containment structure.

6.2.5.2 *Summary of Application*

FSAR Tier 1: FSAR Tier 1 information for the CGCS is provided in FSAR Tier 1, Section 2.3.1, “Combustible Gas Control System,” relies on safety related foils and dampers, as well as non-safety-related passive autocatalytic recombiners. The major equipment and the associated locations for the CGCS are shown in FSAR Tier 1, Table 2.3.1-1, “CGCS Equipment Design,” and the inspections, tests, analyses, and acceptance criteria are provided in FSAR Tier 1, Table 2.3.1-2, “Combustible Gas Control System ITAAC.” FSAR Tier 1 information for the HMS is provided in FSAR Tier 1, Section 2.4.14, “Hydrogen Monitoring System”. The HMS measures the hydrogen concentration both for design basis accidents and for beyond design basis accidents.

FSAR Tier 2: Information provided by the applicant in FSAR Tier 2, Section 6.2.5 is summarized here, in part, as follows:

The CGCS and the HMS collectively are designed to mix, monitor, prevent, or remove combustible gases from the containment atmosphere, thereby preserving containment integrity and mitigating the consequences of core damage accidents, including severe accidents that might occur. The CGCS consists of passive autocatalytic recombiners (PARs), rupture and convection foils, and hydrogen mixing dampers. The CGCS design and performance parameters are given in FSAR Tier 2, Table 6.2.5-1, "CGCS Design and Performance Parameters." The HMS is used to monitor hydrogen concentrations in containment during severe accidents. The HMS consists of a low-range subsystem for DBAs and a high-range subsystem for severe accidents. The design and performance parameters for the HMS subsystems are given in FSAR Tier 2, Table 6.2.5-2, "HMS Design and Performance Parameters." The concentration of hydrogen in containment from all sources following a design-basis loss of coolant accident (DB-LOCA) with various combinations of PARs operating is shown in FSAR Tier 2, Figure 6.2.5-9, "Concentration of Hydrogen in the Containment."

ITAAC: The ITAAC associated with combustible gas control in containment are given in FSAR Tier 1, Section 2.3.1 and Table 2.3.1-2; and FSAR Tier 1, Section 2.4.14 and Table 2.4.14-2.

6.2.5.3 *Regulatory Basis*

The relevant requirements of NRC regulations for this area of review, and the associated acceptance criteria, are given in NUREG-0800, Section 6.2.5, "Combustible Gas Control in Containment," and are summarized below. Review interfaces with other SRP sections can also be found in NUREG-0800, Section 6.2.5.

1. 10 CFR 50.44(c), "Requirements for Future Water-Cooled Reactor Applicants and Licensees," as it relates to pressurized water reactor plants being designed to accommodate hydrogen generation equivalent to 100 percent fuel clad-coolant reaction while limiting containment hydrogen to less than 10 percent and maintain containment structural integrity and appropriate accident mitigating features; and the capability to ensure a mixed atmosphere during design-basis and significant beyond design-basis accidents.

2. 10 CFR 52.47(b)(1), "Contents of applications; technical information," 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 certified design is built and will operate in accordance with the design certification, the provisions of the Atomic Energy Act, and NRC regulations.

3. In meeting the requirements of 10 CFR 50.44(c)(3) regarding equipment survivability, equipment necessary for achieving and maintaining safe shutdown of the plant and maintaining containment structural integrity should perform their safety functions during and after being exposed to the environmental conditions attendant with the release of hydrogen generated by the equivalent of a 100 percent fuel clad-coolant reaction including the environmental conditions created by activation of the combustible gas control system.

6.2.5.4 *Technical Evaluation*

The main regulatory requirements for combustible gas control of future nuclear power reactors are specified in 10 CFR 50.44(c), namely:

1. A mixed containment atmosphere must be ensured; this applies to both design-basis and significant beyond design-basis accidents. A 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.

2. The concentration of hydrogen must be limited, both globally and locally, to less than 10 percent.

3. Equipment and systems needed to maintain containment integrity must be able to perform their functions during and after a hydrogen burn; detonations of hydrogen must also be included unless it can be shown that such detonations are unlikely to occur.

4. Equipment must be provided for continuously measuring hydrogen concentration inside containment following an accident.

5. A structural analysis must be completed that demonstrates containment integrity will be maintained during and after a hydrogen burn that ignites all of the hydrogen that is released by the fuel clad-coolant reaction.

The latter four criteria are based on the limiting conditions that are created by significant beyond design-basis accidents, which bound the conditions that are generated during a DBA. All of the significant beyond design basis analyses must consider the amount of hydrogen that is equivalent to that generated from a 100 percent fuel clad-coolant reaction.

<u>Containment mixing for design basis accidents</u>

In accordance with the requirements stated in 10 CFR 50.44, "Combustible Gas Control for Nuclear Power Reactors," all containments must have the capability to ensure a mixed atmosphere during design-basis and significant beyond design-basis accidents.

The sources of hydrogen generated during an accident include reaction of steam with fuel cladding, radiolysis of the reactor coolant system or emergency core cooling system water, corrosion of metals present in the containment (e.g., aluminum, zinc, etc.), and molten core-concrete interaction (for severe accidents only). These sources are not distributed uniformly throughout the containment.

FSAR Tier 2, Section 6.2.5.3.1,"Post-LOCA Hydrogen Concentration," states that for the design basis analysis, hydrogen generation is based on 1 percent oxidation of the fuel cladding; radiolysis of water from the RCS and the IRWST and jacketed cable; and corrosion of aluminum, zinc on painted surfaces, and galvanized steel structures.

Mixing of the containment atmosphere is accomplished by the CGCS. The CGCS consists of (1) rupture foils and convection foils installed in the ceiling of the steam generator compartments, (2) hydrogen mixing dampers installed between the refueling water storage tank and the annular compartments of the containment, and (3) 47 PARs distributed throughout containment. The foils consist of two types: rupture foils and convection foils. The rupture foils open bi-directionally on a differential pressure of 4.8 kPa (0.7 psid). The convection foils open downward on a temperature of 82.5 °C (180.5°F), or bi-directionally on a differential pressure of 4.8 kPa (0.7 psid). The mixing dampers open on a differential pressure of 3.4 kPa (0.5 psid) or on an absolute pressure of 120 kPa (17.4 psia), or they fail open on a loss of offsite power. The mixing dampers can also be opened manually from the main control room. The PARs reduce hydrogen concentration passively by means of an exothermic reaction. When the hydrogen and oxygen are combined, convection currents are created in the vicinity of the PARs.

As discussed in FSAR Tier 2, Section 6.2.5.3, "Safety Evaluation," the applicant analyzed the hydrogen concentration in containment that results following a design-basis accident and found that the hydrogen concentration does not exceed 4 percent in the 24-hour period following the accident. Even without taking credit for operation of the PARs, FSAR Tier 2, Figure 6.2.5-9, shows that the hydrogen concentration only reaches 2½ percent in 24 hrs. Combustion will not occur at hydrogen concentrations that are less than 4 percent.

The staff performed a confirmatory that showed the hydrogen was well mixed in the containment following a design-basis accident, with the hydrogen concentration remaining below 4 percent for the 24 hr post-accident period, thereby satisfying the requirement for a design basis accident in 10 CFR 50.44(c)(1).

<u>Containment mixing for severe accidents</u>

In accordance with the requirements stated in 10 CFR 50.44, all containments must have the capability to ensure a mixed atmosphere during significant beyond design-basis accidents.

An inerted atmosphere means a containment atmosphere with less than four percent oxygen by volume. (While nitrogen is typically used to inert a boiling water reactor (BWR) containment, the presence of steam in high enough concentrations can be used to inert a containment.) The U.S. EPR is a non-inerted containment. All non-inerted containments must limit hydrogen concentrations in containment during and following an accident that releases an equivalent amount of hydrogen that would be generated from 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.

The acceptance criterion for the CGCS, which provides for a mixed and homogeneous gas atmosphere in the containment, is to maintain the hydrogen concentration in containment (both globally and locally) below 10 percent by volume.

The applicant used the MAAP4 code to demonstrate the extent of containment mixing and hydrogen recombination by the PARs. The results are shown in FSAR Tier 2, Figure 19.2-5, "Hydrogen Concentrations through the U.S. EPR Containment." The results indicate a global concentration of hydrogen following a severe accident as being controlled below 10 percent for up to 50 hrs.

The staff performed a confirmatory calculation using MELCOR. The results of the MELCOR analysis show a containment atmosphere less well mixed than the AREVA MAAP4 analysis, including several local, non-steam inerted, compartments with hydrogen concentrations greater than 10 percent. The MELCOR analysis included several sensitivity studies showing that the PAR efficiency is the dominant sensitivity.

In order to understand the difference in the results between the applicant's MAAP analysis and the staff's confirmatory calculation, the staff issued RAI 471, Questions 06.02.05-20 through 06.02.05-23. These questions address the PAR efficiency that was credited in the analysis, the amount of hydrogen generated by MCCI, and the PAR recombination rates that were used. Resolution of the difference in results between the applicant's MAAP analysis and the staff's MELCOR analysis is an open item. **RAI 471, Questions 06.02.05-20 through 06.02.05-23 are being tracked as open items.**

In a beyond design-basis accident, the PARs are required to maintain the global and local hydrogen concentration below 10 percent. The PARs must be shown to be capable of operating effectively under the conditions expected during a severe accident. These conditions include initial ambient gas temperatures and pressures; steam and nitrogen as inert gases; the effects of dome spray and direct spray on PAR start-up; the effect of fumes that are generated from burning electric cable insulation; and the release of core melt aerosols, including elements such as iodine, tellurium, cesium, antimony, and other fission products, which have the potential to poison the PAR catalysts.

In RAI 323, Questions 06.02.05-8 through 06.02.05-14, the staff requested that the applicant clarify PAR performance in a severe accident environment. In a March 3, 2010, response to RAI 323, Questions 06.02.05-8 through 06.02.05-14, the applicant's cited many industry test results on PAR performance in a severe accident environment. The staff issued follow-up RAI 410, Questions 06.02.05-16 through 06.02.05-19 requesting that the applicant provide these test reports. In a November 3, 2010, response to RAI 410, Questions 06.02.05-16 through 06.02.05-19, the applicant stated that the test reports would not be provided, as they are vendor-specific and a vendor for the PARs has not been selected; and the PARs are not credited in the FSAR Tier 2, Chapter 6, "Engineered Safety Features," design basis. In follow-up RAI 474, Question 06.02.05-25, which was issued as a follow-up to RAI 323 and RAI 410, the staff requested that the applicant address PAR performance in a severe accident environment. The staff considers demonstration of adequate PAR performance in a severe accident environment by the applicant as an open item. **RAI 474, Question 06.02.05-25 is being tracked as an open item**.

<u>Equipment Survivability</u>

10 CFR 50.44(c)(3) requires that, in a containment that does not rely on an inerted atmosphere to control combustible gases, all equipment and instrumentation in containment needed to establish and maintain safe shutdown and containment structural integrity must also be capable of performing their functions during and after exposure to the environmental conditions created by the burning of hydrogen, in an amount equivalent to that generated from a fuel clad-coolant reaction involving 100 percent of the fuel cladding.

Equipment and instrumentation associated with the following items/functions are positioned inside containment and must withstand the conditions expected to occur during a severe accident, including hydrogen burning:

- Containment isolation valves and position sensors
- Primary depressurization system valves and position sensors
- Containment pressure sensors
- Hydrogen monitors
- Hydrogen mixing dampers and position sensors

- PARs, convection and rupture foils
- IRWST water level and temperature
- Dose rate measurement (i.e., gamma-sensitive detector)
- Severe accident sampling system
- Reactor pressure vessel thermocouples
- Passive flooding valves and position sensors
- Passive flooding line isolation motor operated valves (MOVs) and position sensors
- Thermocouples in core catcher main cooling channel and steam chimney
- Containment spray nozzles

FSAR Tier 2, Section 19.2.4.4.5.2, "Inside Containment," gives the equipment and instrumentation in containment which must withstand the conditions expected to occur during a severe accident. This equipment is also identified in FSAR Tier 2, Table 19.2-2, "SAHRS Design and Operating Parameters," and in FSAR Tier 2, Table 19.2-3, "Severe Accident Instrumentation and Equipment" (in containment equipment and instrumentation only).

The passive flooding line motorized isolation valves, 30JMQ42AA004 and 30JMQ42AA006, have been added to the SAHRS design. These valves are normally closed with electric power removed. In order to address both passive flooding and active cooling in the corium spreading area, the operator restores power to the valves and opens them. These valves are identified in an October 15, 2010, response to RAI 390, Question 09.02.02-106, and in FSAR Tier 1, Table 2.3.3-1, "SAHRS Equipment Mechanical Design," and FSAR Tier 1 Figure 2.3.3-1, "SAHRS Functional Arrangement." These valves should be added to FSAR Tier 2, Table 19.2-3. This issue is being pursued in RAI 460, question 19-350. **RAI 460, Question 19-350 is being tracked as an open item.**

FSAR Tier 2, Section 19.2.3.3.7.1, "Equipment and Instrumentation Necessary to Survive," states that the containment isolation valves, containment penetrations, air locks, hatches, and gaskets are required to maintain their leak tightness during a severe accident. Equipment needed to ensure the integrity of the containment in case of significant beyond design-basis events need not be safety grade. However, this equipment should be designed to provide reasonable assurance that it will operate in the accident environment including a hydrogen burn in the containment for the time period for which it is credited. These estimated time periods are depicted in FSAR Tier 2, Figure 19.2-21, "Course of Primary Events during a Severe Accident."

It is not necessary to demonstrate that equipment which is only credited to function before a hydrogen burn will function during or after a burn. For example, closing the containment isolation valves will occur before sufficient hydrogen is generated for burning. However, it is still necessary to demonstrate that the containment isolation valves will remain leak tight following a hydrogen burn. It is likely that the primary depressurization system (PDS) valves will have been opened on a high core outlet temperature prior to the creation of conditions for a hydrogen burn. Once opened, they need not remain functional. The convection and rupture foils installed at the ceiling of the steam generator cubicles will likely have opened either on pressure differential or elevated temperature. The open position for the foils is the position for mixing of the containment atmosphere. The hydrogen mixing dampers will likely open either on a pressure signal early in the accident or on loss of offsite power. Their position sensor will still provide useful information to the operator if they remain closed.

Passive equipment whose integrity would not likely be affected by the elevated temperature or pressure of a hydrogen burn would include the SAHRS spray nozzles, the passive flooding valves, and the PARs. The remainder of the equipment, sensors, and instruments discussed above needs to be evaluated for survivability during and following a hydrogen burn.

In an August 28, 2010, response to RAI 372, Question 06.02.05-15, the applicant stated that their primary source for performance expectations of equipment similar to that described above can be found in EPRI NP 4354, “Large-Scale Hydrogen Burn Equipment Experiments,” December 1985. In this report, equipment types including pressure transmitters, MOV valve operators, limit switches, containment penetrations, resistance temperature detectors (RTDs), and electric cables were tested in the presence of hydrogen burns. The increase in temperature of the tested equipment was noted and the operability post burn was noted. The relevance of the EPRI test results depends on two factors: the similarity of the design between the AREVA specified equipment and the devices tested by EPRI; and the equivalence of the EPRI burn conditions with those of the U.S. EPR.

FSAR Tier 2, Section 19.2.4.4.5, “Equipment Survivability,” provides the adiabatic isochoric complete combustion (AICC) pressures and temperatures of 724 kPa (105 psia) and 890 °C (1,634 °F) resulting from bounding scenarios. FSAR Tier 2, Figure 19.2-7, “Tolerance Limit Plot of AICC Pressure,” depicts the AICC pressure in containment. Although the AICC temperatures and pressures are a theoretical maximum, it is clear from FSAR Tier 2, Figure 19.2-7, that the pressure, and therefore temperature, maximum values are not momentary spikes but are conditions in the containment to which the needed equipment will be subjected. The peak temperatures in the EPRI tests ranged from 316 °C (600 °F) to 1,149 °C (2,100 °F), increasing with hydrogen concentration. In an August 28, 2010, response to RAI 372, Question 06.02.05-15, the applicant indicated that the maximum temperature is a momentary peak of 288 °C (550 °F) with an average temperature of 149 °C (300 °F) in the containment dome, which appears to be non-conservative.

In follow up RAI 473, Question 06.02.05-24 the staff requested that the applicant:

- Provide justification, item by item, that the equipment identified above as requiring a survivability assessment is sufficiently similar to the equipment tested by EPRI in 1985
- Provide the temperature and pressure vs. time throughout the containment during a hydrogen burn in an amount equivalent to that generated from a fuel clad-coolant reaction involving 100 percent of the fuel cladding, including a description of the scenario modeled, and describe to what extent the PARs were credited as functioning

Therefore, the staff considers documentation of an adequate basis for demonstrating equipment survivability and for establishing the containment temperature and pressure profiles following a severe accident unresolved. **RAI 473, Question 06.02.05-24 is being tracked as an open item.**

<u>Hydrogen Monitoring</u>

In accordance with 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 beyond design-basis accident for accident management, including emergency planning.

FSAR Tier 2, Section 6.2.5.1, “Design Bases,” indicates that the HMS measures the hydrogen concentration in containment during and after the accident, and remains functional during and after exposure to accident environmental conditions.

The low range system consists of seven hydrogen sensors whose measurement range is 0-10 percent hydrogen, arranged in the following containment areas:

- Upper dome
- Upper pressurizer compartment
- Upper steam generator compartments 1/2 and 3/4
- Annular rooms

A hydrogen concentration measurement that exceeds 1 percent by volume actuates an alarm in the main control room to indicate a release of hydrogen to the containment atmosphere, while a hydrogen concentration measurement that exceeds 4 percent by volume actuates an alarm in the main control room indicating that the flammability limit in air has been exceeded.

The high range HMS system consists of two redundant trains of four gas samplers each. The measurement range of the gas samplers is 0-30 percent hydrogen, and 30-70 steam volume percent.

This information is used for accident management measures, for assessing the efficiency of the CGCS, and for estimating the risk of deflagrations in containment. The high range gas samplers are located in the following containment areas:

- Upper dome
- Upper steam generator compartments 1/2 and 3/4
- Annular rooms

Alarms in the main control room indicate when the design value for hydrogen concentration has been exceeded at the sampling point, and when the hydrogen concentration for flammable mixtures has been exceeded. A combined hydrogen and steam concentration alarm indicates a possible threat to containment integrity.

While the high range monitors are not safety-related, they are safety significant, similar to the PARs and the SAHRS. Since both the low and the high range HMS equipment are part of the CGCS design basis, the high-range HMS equipment should be added to FSAR Tier 1, Table 2.4.14-1, “Hydrogen Monitoring System Equipment,” to establish an ITAAC for verifying that both the low range and the high range monitors display and alarm in the main control room and at the remote shutdown station. The non-safety-related function of the high range monitors should also be added to FSAR Tier 1, Section 2.4.14 “Hydrogen Monitoring System.” In RAI 104, Question 14.3.11-1(b), the staff requested that the applicant address these considerations. The applicant submitted a February 11, 2009, response to RAI 104, Question 14.3.11. The staff considers inclusion of the high-range HMS monitors within the scope of FSAR Tier 1, Section 2.4.14, and the establishment of appropriate ITAAC for the high-range HMS functions in FSAR Tier 1, Table 2.4.14-1, as an open item. **RAI 411, Question 14.3.11-4 is being tracked as an open item.**

<u>Containment Integrity</u>

In order to meet 10 CFR 50.44(c)(5) requirements, a structural analysis must be completed to demonstrate containment integrity. The structural analysis should include hydrogen detonation loads, unless they are shown to be highly unlikely. In FSAR Tier 2, Section 19.2.4.4.1.4, “Hydrogen Combustion,” the applicant cites experimental results which show that for flame acceleration, and consequently deflagration to detonation transition (DDT), the most important parameter is the expansion ratio. This property will be related to the sigma (σ) criterion.

The σ criterion relates the expansion ratio σ (density of the gas before combustion divided by the density of the gas after non-isochoric combustion) to a limit value obtained through experimentation. Combustion risk is analyzed using the σ criterion, which states that there is no risk of flame acceleration as long as the expansion ratio σ remains below an experimentally based limit (σ*). The sigma index is the sigma value normalized to the critical sigma value, defined as σ/σ*. For scenarios where the sigma index is less than 1.0, flame acceleration can be excluded. The applicant has calculated this ratio, which is less than 1.0, and the results are shown in FSAR Tier 2, Figure 19.2-8, “Tolerance Limit Plot of Sigma Index for the Pump/SG Compartment.” The staff‘s confirmatory calculation also finds that the sigma index is less than 1.0 and, therefore, the staff agrees that detonation following a severe accident is highly unlikely.

To address the structural loads due to global deflagration of hydrogen, the AICC pressure was used as a bounding value for the pressure that would result should a single large deflagration occur. FSAR Tier 2, Figure 19.2-7, shows the global maximum AICC pressure is 724 kPa (105 psia) for all of the uncertainty cases. This does not exceed the containment ultimate pressure capacity as described in FSAR Tier 2, Section 3.8.1.4.11, “Containment Ultimate Capacity,” of approximately 820 kPa gauge pressure (119 psig). Therefore, the structural capability of the containment will not be exceeded due to the global burning of hydrogen, and containment integrity will remain intact.

Inspection, Tests, Analysis, Acceptance Criteria

The ITAAC associated with combustible gas control in containment are given in FSAR Tier 1, Section 2.3.1, “Combustible Gas Control System,” and Table 2.3.1-2; and FSAR Tier 1, Section 2.4.14, “Hydrogen Monitoring System,” and Table 2.4.14-2.

FSAR Tier 1, Section 2.3.1 describes the function and location of all the components in the CGCS and provides ITAAC confirming the existence and location of all the components. The performance of the mixing dampers following a loss of power is verified.

FSAR Tier 1, Section 2.2.14 describes the function of the low range and the high range monitors in containment, and provides ITAAC to verify the design, equipment qualification, and functional testing for the safety-related low range monitors. Open Item 6.2.5-4 addresses the ITAAC for the non-safety-related high range monitors.

With the exception of the open item, these ITAAC comply with the requirements of 10 CFR 52.47(b)(1).

6.2.5.5 *Combined License Information Items*

There are no COL information items related to this area of review. The staff determined that no COL information items need to be included in FSAR Tier 2, Table 1.8-2, "U.S. EPR Combined License Information Items," for combustible gas control in containment consideration.

6.2.5.6 *Conclusions*

Combustible gas control inside containment is provided by the CGCS and the HMS. The staff evaluated the proposed designs of the CGCS and HMS for the U.S. EPR as described primarily in FSAR Tier 2, Section 6.2.5, and FSAR Tier 1, Sections 2.3.1 and 2.4.14. The staff also reviewed information provided in FSAR Tier 2, Section 19.2, "Severe Accident Evaluations." The staff's evaluation to confirm compliance with the provisions of 10 CFR 50.44(c) and 10 CFR 52.47(b)(1) was performed in accordance with the guidance provided in SRP Section 6.2.5. The results of the staff's evaluation are presented below, along with the open items that were identified.

- In order to satisfy the requirement 10 CFR 50.44(c) to provide adequate mixing of the containment atmosphere, the CGCS design relies on global convection enhanced by rupture and convection foils, dampers, and passive autocatalytic recombiners. The 47 recombiners distributed throughout containment also serve to reduce hydrogen concentration. For the reasons described above, the staff finds that the hydrogen will be well mixed in the containment following a DBA, with the hydrogen concentration remaining below 4 percent for the 24-hr period following the DBA. However, based on independent analysis, the staff has determined that the applicant's analysis for the control of hydrogen in containment following a severe accident has not been adequately demonstrated. **RAI 471, Questions 06.02.05-20 through 06.02.05-23 are being tracked as open items.**

- In order to satisfy the requirement 10 CFR 50.44(c) to use equipment that is capable of surviving the environmental conditions inside containment that results from severe accidents, the applicant credited vendor test data for the PARs. However, the applicant was unable to provide documentation that demonstrates that the PARs are capable of surviving the containment atmosphere and functioning following a severe accident. **RAI 474, Question 06.02.05-25 is being tracked as an open item**.

- In order to satisfy the requirement stated in 10 CFR 50.44(c) to use equipment that is capable of surviving the environmental conditions that results inside containment from severe accidents, the applicant credited equipment test results from EPRI NP 4354 for equipment inside containment that must remain functional (except for the PARs, which is discussed above). However, the applicant has not demonstrated that the EPRI information is applicable to the U.S. EPR design. **RAI 473, Question 06.02.05-24 is being tracked as an open item.**

- The high range hydrogen monitors are safety significant, similar to the PARs and the SAHRS. Since both the low- and the high-range HMS equipment is part of the CGCS design basis, the staff found that the high-range HMS equipment should be described in FSAR Tier 1, Section 2.4.14; and ITAAC should be established and included in FSAR Tier 1, Table 2.4.14-1, per the requirement stated in 10 CFR 52.47(b)(1). **RAI 411, Question 14.3.11-4 is being tracked as an open item.**

- In order to meet 10 CFR 50.44(c)(5) requirements, a structural analysis must demonstrate containment integrity. The AICC pressure was used as a bounding value for the pressure that would result should a single large deflagration occur. The applicant determined that the global maximum AICC pressure that results is 724 kPa (105 psia), which does not exceed the containment ultimate pressure capacity of approximately 820 kPa gauge pressure (119 psig). These results were confirmed by the staff by independent analysis. Therefore, the global burning of hydrogen will not exceed the design capability of the containment and containment integrity will be maintained. Consequently, the staff finds that the requirement specified by 10 CFR 50.44(c)(5) is satisfied.

Based on the results of this evaluation and in particular, as discussed above, the applicant has not adequately addressed 10 CFR 50.44(c) requirements in that the control of hydrogen in containment following a severe accident was not adequately demonstrated, and the survivability of PARs and other necessary equipment relied upon to function in the severe accident containment atmosphere also was not adequately demonstrated. Additionally, the requirement specified by 10 CFR 52.47(b)(1) to establish ITAAC was not fully satisfied in that suitable ITAAC were not established for the high-range HMS function. Therefore, the staff is unable to conclude that the requirements specified by 10 CFR 50.44(c) and 10 CFR 52.47(b)(1) are fully satisfied with respect to combustible gas control in containment. The issues that remain have been identified as open items above.

6.2.6 Containment Leakage Testing

6.2.6.1 *Introduction*

Containment leakage testing assures that leakage through the primary reactor containment and systems penetrating primary containment do not exceed allowable leakage rate (L_a) values as specified in the technical specifications. Periodic surveillances of containment penetrations and isolation valves are performed so that proper maintenance and repairs are made during the service life of the containment and the penetrating systems or components. FSAR Tier 2, Section 6.2.6, “Containment Leakage Testing,” states that the containment leakage testing program complies with 10 CFR Part 50, Appendix J, “Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors,” Option B, “Performance-Based Requirements,” and follows the guidance of Regulatory Guide (RG) 1.163, “Performance Based Containment Leak Rate Test Program,” September 1995.

6.2.6.2 *Summary of Application*

FSAR Tier 2: Information provided by the applicant in FSAR Tier 2, Section 6.2.6, is summarized here, in part as follows:

FSAR Tier 2, Section 6.2.6 states that the containment, containment penetrations, and isolation barriers are designed to permit periodic leakage testing in accordance with GDC 52, “Capability for Containment Leakage Rate Testing,” GDC 53, “Provisions for Containment Testing and Inspection,” and GDC 54, “Systems Penetrating Containment.” FSAR Tier 2, Section 6.2.6 also states that the containment leakage rate testing program complies with 10 CFR Part 50, Appendix J, Option B, “Performance-Based Requirements,” and follows the guidance of RG 1.163. This program is implemented in accordance with ANSI N45.4, “Leakage Rate Testing of Containment Structures for Nuclear Reactors,” 1972 and ANSI/ANS 56.8, “Containment System Leakage Testing Requirements,” August 1994.

For the Type A test, the allowable containment leakage rate does not exceed 0.25 percent of containment air mass per day at accident pressure (P_a). The value of P_a is 379.2 kPa (55 psig), which is the calculated peak internal pressure associated with a design basis loss-of-coolant accident (LOCA). For Type B and Type C testing, containment penetrations and related testing specifications are identified in FSAR Tier 2, Table 6.2.4-1, "Containment Penetration, Isolation Valve, and Actuator Data."

Type B and Type C testing is also performed at pressure P_a. The combined leakage rate acceptance criterion for all penetrations and valves subject to Type B and C tests is less than 0.60 L_a. Tests are performed by local pressurization. Each tested valve is closed by normal means without preliminary exercising or adjustments, and pressure is applied in the same direction as would be present during the design-basis accident (DBA).

Technical Specifications: The TS applicable to containment leakage testing can be found in FSAR Tier 2, Chapter 16, "Technical Specifications," Section 5.5.15, "Containment Leakage Rate Testing (CLRT) Program."

6.2.6.3 *Regulatory Basis*

The relevant requirements of NRC regulations for this area of review, and the associated acceptance criteria, are given in NUREG-0800, Section 6.2.6, "Containment Leakage Testing," and are summarized below. Review interfaces with other SRP sections can also be found in NUREG-0800, Section 6.2.6.

1. 10 CFR 50.54(o), "Conditions of licenses," in that the primary reactor containment must meet the leakage-rate test requirements specified in either 10 CFR Part 50, Appendix J, Option A or B "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors."

2. 10 CFR 52.47(b)(1), "Contents of applications; technical information," in that design certification applications must contain the proposed inspections, tests, analyses, and ITAAC that are necessary and sufficient to ensure that the as-built plant is in conformance with the approved design.

3. GDC 52 in that the reactor containment and related equipment must be designed so that periodic integrated leakage rate testing can be conducted at containment design pressure.

4. GDC 53 in that the reactor containment must be designed to permit appropriate periodic inspection, surveillance, and testing at containment design pressure of penetrations that have resilient seals and expansion bellows.

5. GDC 54 in that piping systems penetrating primary reactor containment must be provided with leak detection capability and must be designed with the capability to periodically test isolation valves and associated apparatus to determine if valve leakage is within acceptable limits.

6. 10 CFR Part 50, Appendix J, in that the primary reactor containment must be designed such that the maximum allowable leakage rate, La, as specified in the TS or associated bases when tested at the calculated peak internal containment pressure, Pa, related to the design basis accident is not exceeded, with margin, as measured with containment overall integrated leakage rate (Type A) tests, local leakage rate (Type B) tests across

pressure retaining, leakage limiting boundaries, and containment isolation valve leakage rate (Type C) tests.

Acceptance criteria adequate to meet the above requirements include the following:

1. RG 1.163 endorses Nuclear Energy Institute (NEI) 94-01, “Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J,” July 1995, which provides the following guidance:

   - No repairs or adjustments should be made prior to the performance of the containment integrated leak rate test (CILRT). Instrument lines that penetrate containment can be isolated for the CILRT with measured leakage rates from the local leakage rate tests (LLRTs) added to CILRT results.

   - Where valves are tested with pressure in the direction opposite to post-accident pressure, the applicant should commit to justify that such testing will result in equivalent or more conservative results.

   - Acceptable test intervals for applicants planning to comply with 10 CFR Part 50, Appendix J, Option B, are specified in NEI-94-01, Section 11.0, “Bases For Performance and Risk-Based Testing Frequencies for Type A, Type B, and Type C Tests.” All other technical methods and techniques for performing Types A, B, and C tests contained in ANSI/ANS 56.8 are acceptable to the staff.

2. SRP Section 6.2.6 states that all leakage rate tests performed by either pneumatic or hydraulic means should have the capability to quantify leakage rates either explicitly or by a conservative bounding method.

3. NEI 94-01, Section 6.0, “General Requirements,” and ANSI/ANS 56.8, Section 3.3.1, “Local Leakage Rate Testing Requirement, General,” state that Type B and Type C tests are not required for the following cases:

   - Containment boundaries that do not constitute potential containment atmospheric leakage pathways during and following a design-basis loss-of-coolant accident (DBA-LOCA), noting that lines that terminate below the minimum post accident water level in a recirculation sump inside containment would be an example of lines that do not constitute potential leakage pathways

   - Containment boundaries sealed with qualified seal systems, as further defined in ANSI/ANS 56.8, noting that qualified systems are capable of sealing leakage with a liquid at a pressure of no less than 1.1 Pa for at least 30 days after a DBA LOCA, including the most limiting single active component failure

   - Test connections, vents, and drains between containment isolation valves

6.2.6.4 *Technical Evaluation*

Containment leakage testing assures that leakage from the primary reactor containment and systems penetrating primary containment do not exceed allowable leakage rate values as specified in the TS. Periodic surveillance of containment penetrations and isolation valves is performed so that proper maintenance and repairs are made during the service life of the containment and the penetrating systems or components.

For the Type A tests (CILRT), temporary compressors are connected via permanent piping to the containment. The compressors raise containment internal pressure to the test pressure (P_a) of 379.2 kPa (55 psig), which is the calculated peak accident pressure. Using pressure, temperature and humidity measurements from inside containment, containment leakage is calculated. The maximum allowable L_a is less than 0.25 percent of containment air mass per day at test pressure P_a, the assumed leakage for the accident analyses.

Type B tests measure leakage through containment penetrations that contain resilient seals, gaskets, or sealing compounds. Type B tests also measure leakage through air locks, door seals, equipment hatches, access hatches, and associated seals. Air lock leakage is measured at P_a and is limited to 0.05 L_a. Leakage for each door when pressurized at a minimum of 68.95 kPa (10 psig) must be ≤0.01 L_a.

Type C tests measure containment isolation valve leakage. Like the Type B tests, Type C testing is performed by local pressurization. The valves are closed by normal means with no exercising or other preconditioning. The combined leakage rates from the total of all Type B and Type C tests must be less than 0.6 L_a.

As stated in the FSAR Tier 2, Section 6.2.6, the applicant's CLRT program is based on the requirements stated in 10 CFR Part 50, Appendix J, Option B, and the guidance provided by RG 1.163. This approach is in compliance with the guidance provided in SRP Section 6.2.6 and, as described in more detail below, is therefore acceptable to the staff. Consequently, the staff focused its review on the requirements specified by 10 CFR Part 50, Appendix J, Option B, and the guidance provided by RG 1.163.

Type A Testing

Type A testing is conducted in accordance with the methodology specified in ANSI/ANS 56.8, using the mass-point method, in which the containment is pressurized up to the test pressure and then either the decrease in pressure or the makeup rate is measured and a leak rate derived.

After pressure stabilization, the test duration is established in conformance to ANSI/ANS 56.8. Changes in containment air mass are calculated using periodic measurements of containment pressure, dry bulb temperature, and dew point temperatures (i.e., water vapor pressure). Accuracies are verified by supplemental verification test. On completion of the CILRT, valves are realigned to return isolated instrument lines to their normal configuration.

The maximum allowable L_a is 0.25 percent of the containment air mass per day with the containment pressurized to P_a. A conservative value of 379.2 kPa (55 psig) is used for P_a, which is the calculated peak internal pressure associated with the DBA-LOCA for the U.S. EPR. As soon as practical after completion of a Type A test that identifies leakage, and prior to the containment inspection for the subsequent Type A test, repairs or adjustments are made to components that exceeded their individual leakage limits.

Both the preoperational and periodic (in-service) Type A tests are conducted at P_a (379.2 kPa (55 psig)) to obtain the as-measured leakage rate (L_{am}). The acceptance criterion for the as-found condition is ≤1.0 L_a, and the acceptance criterion for the as-left condition is (after repair or adjustment) ≤0.75 L_a. These criteria apply to both preoperational testing as well as subsequent periodic Type A testing. Because the acceptance criteria that are specified for L_{am} are within the limits required by L_a, the staff considers specified acceptance criteria for L_{am} acceptable. The Type A testing acceptance criteria are shown in Table 6.2.6-1 of this report.

The staff notes that in the preoperational test case for the as-left peak pressure acceptance criteria, if local leakage measurements are taken to effect repairs in order to meet the acceptance criteria, these measurements are taken at pressure P_a.

Table 6.2.6-1 Type A Testing Acceptance Criteria

Criteria	Preoperational Test	In-Service Test
As-Found Peak Pressure Acceptance Criteria	L_{am} <1.0 L_a at P_a	L_{am} <1.0 L_a at P_a
As-Left Peak Pressure Acceptance Criteria	L_{am} <0.75 L_a at P_a	L_{am} <0.75 L_a at P_a
Type A fails to meet criterion	Retest per Appendix J	Test schedule for subsequent tests adjusted per Appendix J, Option B

The applicant did not propose the use of any alternative tests or analyses as substitutes for as-found tests. The staff finds that this satisfies the restriction recommended on alternative tests or analyses by RG 1.163.

FSAR Tier 2, Section 6.2.6.4, “Scheduling and Reporting of Periodic Tests,” states that visual inspections will be conducted in accordance with NEI 94-01 prior to periodic Type A tests. The staff reviewed the list of prerequisite items that were given in FSAR Tier 2, Section 6.2.6.1, “Containment Integrated Leakage Rate Test (Type A),” to confirm that the NEI guidance was properly reflected. In particular, the staff confirmed that containment visual inspections are specified for two refueling outages when Type A test intervals have been extended to 10 years. This inspection frequency is in accordance with the NEI 94-01 standard, Section 9.2.1, “Pretest Inspection and Test Methodology.” Accordingly, the staff finds the frequency of the visual inspection program acceptable.

Certain instrument lines that penetrate the containment are isolated by the Type A test plan. For the isolated lines, LLRT are performed. Results from these LLRT are added to the CILRT results. The staff finds this acceptable, because it provides Type A test results for these instrument lines in as close to the as-found condition as practical without exposing sensitive instruments to the potentially damaging conditions that are established for the CILRT. Also, as stated in FSAR Tier 2, Section 6.2.6.4, “Scheduling and Reporting of Periodic Tests,” Type A tests are performed at intervals as specified in NEI 94-01, Section 11.0. This is in compliance with the guidance in SRP Section 6.2.6 for satisfying 10 CFR Part 50, Appendix J, Option B, and is therefore acceptable to the staff.

In view of the evaluation set forth above, the staff finds the Type A testing program proposed by the applicant acceptable, because it satisfies the staff’s acceptance criteria referred to above in Section 6.2.6.3 of this report and, therefore, is in compliance with 10 CFR Part 50, Appendix J, Option B.

Type B Testing

Preoperational and periodic testing of containment penetrations (Type B) are performed in accordance with 10 CFR Part 50, Appendix J, Option B. A list of specific containment penetrations subject to Type B tests is provided in FSAR Tier 2, Table 6.2.4-1. The following penetrations receive preoperational and periodic Type B tests:

- Penetrations designed with resilient seals, gaskets, or sealant compounds
- Air locks and associated door seals
- Equipment and access hatches and associated seals
- Electrical penetrations

Portable test panels are used for testing penetrations and isolation valves using air or nitrogen. Panels include pressure regulators, filters, pressure gauges, and flow instruments as needed to perform the tests.

Type B testing is performed at P_a. FSAR Tier 2, Section 6.2.6.2, “Containment Penetration Leakage Rate Tests (Type B),” states that the combined leakage rate for all penetrations and valves subject to Type B and C tests shall be less than 0.60 L_a. The staff considers this sufficient margin from L_a for satisfying the margin requirement specified in 10 CFR Part 50, Appendix J, Option B. The following acceptance criteria for air-lock testing are also specified in TS 5.5.15:

- Overall air lock leakage $\leq 0.05\ L_a$ tested at $\geq P_a$
- Leakage for each air-lock door $\leq 0.01\ L_a$ when pressurized to ≥10 psig

In view of the above, the staff considers the Type B testing program proposed by the applicant acceptable and meets the requirements of 10 CFR Part 50, Appendix J, Option B.

Type C Testing

Preoperational and periodic testing of containment isolation valves (Type C) are performed in accordance with 10 CFR Part 50, Appendix J, Option B. A list of CIVs subject to Type C tests is provided in FSAR Tier 2, Table 6.2.4-1. Valve arrangements and test connections are shown on piping and instrumentation diagrams (P&IDs) for the various systems that penetrate containment.

In FSAR Tier 2, Section 6.2.6, the applicant committed to the guidelines of RG 1.163. In so doing, Type C testing intervals will not be extended beyond 60 months, since RG 1.163 does not currently endorse such an extension. Similarly, Type C testing for containment purge and vent valves will not be extended beyond 30 months.

Portable test panels are used for testing penetrations and isolation valves using air or nitrogen. Panels include pressure regulators, filters, pressure gauges, and flow instruments to perform the tests. Valves are pressurized with either air or nitrogen at a pressure of P_a. However, FSAR Tier 2, Section 6.2.6.3, “Containment Isolation Valve Leakage Rate Test (Type C),” was not clear as to whether any Type C hydraulic testing would be performed. Therefore, in RAI 12, Question 06.02.06-2, the staff requested that the applicant clarify when hydraulic testing might be used in lieu of pneumatic testing. In a June 3, 2008, response to RAI 12, Question 06.02.06-2, the applicant stated that all Type B and Type C tests are planned to be performed pneumatically. Since RG 1.163 states that pneumatic testing is acceptable under these circumstances, the staff finds that this aspect of the test program with this clarification acceptable.

FSAR Tier 2, Section 6.2.6.3, states that Type C tests are performed in the same direction as accident pressure. It also states that where testing is performed in the direction opposite to accident pressure, a justification is provided. However, the staff noted that FSAR Tier 2, Table 6.2.4-1 does not identify which valves are tested in the direction opposite to accident pressure. Therefore, in RAI 12, Question 06.02.06-2, the staff requested that the applicant identify those valves to be tested in a direction that is opposite to accident pressure. In a June 3, 2008, response to RAI 12, Question 06.02.06-2, the applicant stated that no containment isolation valves are tested with pressure opposite to the post-accident pressure. Based on the above, the staff finds that this aspect of the test program is acceptable, and therefore considers RAI 12, Question 06.02.06-2 resolved.

FSAR Tier 2, Table 6.2.4-1, identifies some 52 containment isolation valves that do not require Type C testing. The basic categories of these valves are shown below:

- Refueling transfer tube to the spent fuel pool
- Guard pipe
- Main steam, feed water, emergency feed water, and related secondary systems that are not connected to containment atmosphere

In response to the identification of refueling transfer tubes above, in RAI 12, Question 06.02.06-02, the staff requested that the applicant clarify the basis for not performing Type C testing on the refueling transfer tube valves. In a June 3, 2008, response to RAI 12, Question 06.02.06-2, the applicant stated that the fuel transfer tube is sealed at one end by an airtight and watertight manually operated gate valve and at the other end by a blind flange. The valve contains a double seal gasket system with one set on the flange mated to the transfer tube and the other set on the valve seat. The volume between the seals is continuously monitored during operation by a leak-off system, and the seals are tested (preoperational and periodic) by Type B testing in accordance with ANSI/ANS 56.8. The staff concurs that based on this design, Type C testing of the transfer tube is not necessary.

FSAR Tier 2, Table 6.2.4-1, “Containment Penetration, Isolation Valve, and Actuator Data,” identifies containment isolation valves for essential systems that normally operate post-LOCA and, therefore, are vented prior to Type A testing. CIVs for systems that do not normally operate post-LOCA (non-essential systems), in contrast, are not vented before Type A testing. ANSI/ANS 56.8, section 3.2.5, “Pathway Venting and Draining,” states that this is the correct treatment for essential and non-essential CIVs. As documented in Section 6.2.4 of this report, the staff finds that the applicant correctly classified the CIVs for both essential and non-essential systems. The staff notes that the provisions of NEI 94-01, Chapter 8, “Testing Methodologies for Type A, B and C Tests,” are acceptable as a basis for compliance with the guidance contained in SRP Section 6.2.6.

Based on the above, the staff finds that the Type C testing proposed by the applicant acceptable and that it meets the requirements of 10 CFR Part 50, Appendix J, Option B. Therefore, the staff considers RAI 12, Question 06.02.06-2 resolved.

Test Intervals

In accordance with 10 CFR Part 50, Appendix J, Option B, test intervals are performance-based and are established by criteria in NEI 94-01 in accordance with the regulatory positions stated in RG 1.163, which satisfies the Option B requirement. A summary report is issued to document the preoperational and periodic test results. The report includes a schematic arrangement of the leakage rate measurement system, instrumentation and supplemental test methods that were used, and the applicable test program that was used.

Technical Specifications

TS 5.5.15, “Containment Leakage Rate Testing Program,” states that a testing program shall be established as required by 10 CFR 50.54(o) and 10 CFR Part 50, Appendix J, Option B; and in accordance with the guidelines specified in RG 1.163. Key TS parameters and the bases on which the staff accepts these parameters are summarized in Table 6.2.6-2 below:

Table 6.2.6-2 Key TS Parameters and Bases

Technical Specification	TS Value	Acceptable	Acceptance Basis
Test Pressure (P_a)	379.2 kPa (55 psig)	Yes	Pa is the peak calculated accident pressure of 379.2 kPa (55 psig). (FSAR Tier 2, Table 6.2.1-6).
Test Pressure (P_a)	379.2 kPa (55 psig)	Yes	Pa is less than containment design pressure of 427.5 kPa (62 psig) (FSAR Tier 2, Section 6.2.1.1.2).
Maximum allowable leakage rate (L_a) at P_a	0.25% per day by weight	Yes	FSAR Tier 2, Table 15.0-49 shows containment leakage at 0.25 percent per day starting 10 seconds into DBA-LOCA.
Leakage rate acceptance criteria [as-found condition]	$\leq 1.0\ L_a$	Yes	ANSI 56.8-1994 and NEI 94-01 (Paragraph 9.2.5) identify 1.0 L_a as “as found.”
During first startup following testing per this program [as-left condition]	$\leq 0.75\ L_a$	Yes	ANSI 56.8-1994 and NEI 94-01 (Paragraph 9.2.5) say 0.75 L_a is acceptance for “as left.”
Type B/Type C Testing	$\leq 0.6\ L_a$ total for Type B and Type C combined	Yes	Appendix J requires combined Type B and C leakage results shall be less than L_a with margin.

In RAI 12, Question 06.02.06-1, the staff requested that the applicant provide a clarification of the wording to TS 5.5.15.d.1 in that the words, “during the first startup following testing in accordance with this program,” could be interpreted as applying to only the first startup and not startup from the subsequent Type A tests. In a July 3, 2008, response to RAI 12, Question 06.02.06-1, the applicant stated that a change of wording would conflict with the standard technical specifications provided in NUREG-1431, "Standard Technical Specifications — Westinghouse Plants," Revision 3, for Westinghouse plants. The applicant proposed to clarify the as-found and as-left criteria in FSAR Tier 2, Section 6.2.6.1, and to state that these requirements apply to both the preoperational test and the subsequent periodic Type A tests. The staff agreed with the proposed clarifications and therefore considers RAI 12, Question 06.02.06-1 resolved.

FSAR Tier 2, Section 6.2.6 and Section 6.2.8, "References," Reference 10, referred to ANSI/ANS 56.8, 1987 and 2002. The NRC has not yet reviewed or accepted the 2002 version. In RAI 82, Question 06.02.06-3, the staff requested that the applicant provide a correct reference to the ANSI standard endorsed by RG 1.163, or justify why the referenced standard is an acceptable alternative. In a November 3, 2008, response to RAI 82, Question 06.02.06-3, the applicant proposed to change Reference 10 of FSAR Tier 2, Section 6.2.8, to refer to ANSI/ANS 56.8-1994. The staff confirmed that Revision 2 of the U.S. EPR FSAR, dated August 31, 2010, contains the changes committed to in the RAI response. Accordingly, the staff finds that the applicant has adequately addressed this issue and, therefore, the staff considers RAI 82, Question 06.02.06-3 resolved.

FSAR Tier 2, Table 1.8-2, provides a listing of COL information items to be addressed by the COL applicants, as further discussed in Section 6.2.6.5 of this report. COL Information Item No. 6.2-1 is applicable to containment leakage testing. The staff reviewed FSAR Tier 2, Table 1.8-2 and finds this item to be acceptable and did not identify any additional COL information items that are needed with respect to containment leakage testing.

In view of the evaluation described above, the staff finds that the containment leakage testing program described in FSAR Tier 2, Section 6.2.6, and the requirements stated in TS Section 5.5.15, satisfy the provisions of RG 1.163 and the requirements specified by 10 CFR Part 50, Appendix J, Option B. Therefore, the proposed containment leakage testing program for the U.S. EPR satisfies the requirements of 10 CFR 50.54(o), GDC 52, GDC 53, and GDC 54.

6.2.6.5 *Combined License Information Items*

Table 6.2.6-3 lists a containment leakage testing related COL information item showing the designated item number and description from FSAR Tier 2, Table 1.8-2, "U.S. EPR Combined License Information Items."

Table 6.2.6-3 U.S. EPR Combined License Information Items

Item No.	Description	FSAR Tier 2 Section
6.2-1	A COL applicant that references the U.S. EPR design certification will identify the implementation milestones for the CLRT program described under 10 CFR Part 50, Appendix J.	6.2.6

The staff finds the above listing complete. Also, the list adequately describes actions necessary for the COL applicant or holder. No additional COL information items need to be included in FSAR Tier 2, Table 1.8-2, for containment leakage testing consideration.

6.2.6.6 *Conclusions*

The staff evaluated the CLRT program for the U.S. EPR and determined that the CLRT program conforms to RG 1.163 and the applicable provisions of NEI 94-01 and ANSI/ANS 56.8, which are endorsed by the staff; and that proposed TS 5.5.15 includes a requirement for the CLRT program to satisfy the provisions of RG 1.163. On this basis, the staff finds that the proposed CLRT program for the U.S. EPR complies with the requirements specified by 10 CFR Part 50, Appendix J, Option B. Therefore, the staff finds that the requirements specified in GDC 52, GDC 53, GDC 54 and 10 CFR 50.54(o) are also satisfied.

In summary, the following values for key parameters are specified by the proposed CLRT program for the U.S. EPR:

1. The containment test pressure of 379.2 kPa (55 psig) (P_a) is the peak accident pressure as calculated in a post DBA-LOCA scenario but less than containment design pressure (427.5 kPa (62 psig)).

2. The maximum allowed containment integrated leakage rate (L_a) is 0.25 percent per day by weight of the containment air volume. This value is an input for analyzing the radiological consequences of a DBA-LOCA starting at 10 seconds into the postulated accident.

3. The maximum allowed as-found and as-left leakage rates for both preoperational and periodic Type A tests are 1.0 L_a and 0.75 L_a, respectively.

4. The maximum allowed combined leakage rate from Type B and Type C LLRT is 0.60 L_a.

6.2.7 Fracture Prevention of Containment Pressure Vessel

6.2.7.1 *Introduction*

The Reactor Containment Building is a Seismic Category I safety-related cylindrical concrete structure, with a 0.635 cm (0.25 in.) thick steel liner. The primary functions of the RCB are to protect the safety-related SSCs located within it, to prevent the release of radiation during plant operations, and to prevent the release of radiation and contamination in the event of accident conditions. The reactor containment pressure boundary consists of those ferritic steel parts of the reactor containment system which sustain loading and provide a pressure boundary in the performance of the containment function under operating, maintenance, testing, and postulated accident conditions. Within this context, the purpose of this review is to assess the ferritic materials of components such as freestanding containment vessels, equipment hatches, personnel airlocks, containment penetration sleeves, process pipes, end closure caps and flued heads, and penetrating-piping systems connecting to penetration process pipes and extending to and including the system isolation valves.

6.2.7.2 *Summary of Application*

FSAR Tier 1: The FSAR Tier 1 information associated with this section is found in FSAR Tier 1, Section 2.1.1.1, “Reactor Building,” which states that the RCB is designed to retain its pressure boundary integrity associated with the design pressure.

FSAR Tier 2: The applicant has provided an FSAR Tier 2 description of the design features that prevent fracture of the containment pressure vessel in FSAR Tier 2, Section 6.2.7, “Fracture Prevention of Containment Pressure Vessel,” summarized here, in part as follows:

The application references the description of the steel containment in FSAR Tier 2, Section 3.8.2, “Steel Containment,” and states that the material used in the ferritic steel parts of the reactor containment pressure boundary provides sufficient margin to assure that under operating, maintenance, testing, and postulated accident conditions ferritic materials behave in a non-brittle manner and the probability of rapidly propagating fracture of the pressure boundary is minimized.

The application also states that meeting the ASME Code fracture toughness requirements will ensure that the reactor containment pressure boundary will remain intact during the harshest expected conditions, thereby precluding release of radioactivity to the environment. In FSAR Tier 2, Section 3.8.1.6.4, “Liner Plate System and Penetration Sleeve Materials,” the applicant states that the materials used for the carbon steel liner plate meet the fracture toughness requirements of Subsection CC-2520 of the ASME B&PV Code, Section III, Division 2. This meets the requirements of GDC 51, “Fracture Prevention of Containment Pressure Boundary.”

ITAAC: The ITAAC associated with FSAR Tier 2, Section 6.2.7 are given in FSAR Tier 1, Section 2.1.1.1. Item 2.5 in Table 2.1.1-8, “Reactor Building ITAAC,” requires that analyses, inspections, and a structural integrity test be performed on the RCB and its penetrations to ensure that they comply with ASME Code requirements.

Technical Specifications: The Technical Specifications associated with FSAR Tier 2, Section 6.2.7 are given in FSAR Tier 2, Chapter 16, Section 3.6,

“Containment Systems.” These TS relate to the operability and leak rate testing of the containment, containment air locks, and containment isolation valves.

6.2.7.3 *Regulatory Basis*

The relevant requirements of NRC regulations for this area of review, and the associated acceptance criteria, are given in NUREG-0800, Section 6.2.7, “Fracture Prevention of Containment Pressure Boundary,” and are summarized below. Review interfaces with other SRP sections can also be found in NUREG-0800, Section 6.2.7.

1. GDC 1, as it relates to the quality standards for design and fabrication.

2. GDC 16, “Containment Design,” as it relates to the prevention of the release of radioactivity to the environment.

3. GDC 51, as it relates to the reactor containment pressure boundary being designed with sufficient margin to assure that under operating, maintenance, testing, and postulated accident conditions (1) its ferritic materials behave in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized.

Acceptance criteria adequate to meet the above requirements include:

1. RG 1.26, “Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants,” as it relates to the quality group classification of components.

2. NUREG-0577, Revision 1, “Potential for Low Fracture Toughness and Lamellar Tearing on PWR Steam Generator and Reactor Coolant Pump Supports,” as it relates to the metallurgical characterization of materials and fracture toughness data.

6.2.7.4 *Technical Evaluation*

The staff reviewed the U.S. EPR measures involving fracture prevention of ferritic materials used in the containment pressure boundary in accordance with the guidelines of SRP Section 6.2.7. Ferritic materials used in the containment pressure boundary are acceptable if they meet the requirements of GDC 51, as they relate to the reactor containment pressure boundary being designed with sufficient margins to assure that under operating, maintenance, testing, and postulated accident conditions, the ferritic materials will behave in a non-brittle manner, and the probability of rapidly propagating fracture is minimized.

The U.S. EPR containment pressure boundary utilizes SA516, Grades 55, 60, 65, or 70 material, which is an ASME Code-approved material appropriate for the intended containment vessel application, as a non-load-bearing liner plate. In specifying use of this material for the carbon-steel liner plate, the design meets the fracture toughness requirements of Subsubarticle CC-2520 of the ASME B&PV Code, Section III, Division 2. The Division 1 attachments and appurtenances meet the fracture toughness requirements of Subarticle NE-2300 of the ASME B&PV Code Section III, Division 1. The staff finds that meeting the fracture toughness requirements of the 2004 Edition of the ASME B&PV Code, Section III, Subsections NE and CC provide reasonable assurance of the containment’s fracture prevention capability. These fracture toughness requirements ensure that the containment pressure boundary design satisfies GDC 1, as it relates to the quality standards for design and fabrication; GDC 16, as it relates to the prevention of the release of radioactivity to the environment; and GDC 51, as it relates to providing sufficient margins to preclude fracture of the containment in a non-brittle manner. On this basis, the staff determined that the fracture-prevention measures used in the U.S. EPR containment design are acceptable.

6.2.7.5 *Combined License Information Items*

There are no COL information items from FSAR Tier 2, Table 1.8-2 that affect this section.

6.2.7.6 *Conclusions*

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

Therefore, the staff concludes that reasonable assurance will be provided that the materials of the reactor containment pressure boundary, under operating, maintenance, testing, and postulated accident conditions, will not undergo brittle fracture and that the probability of rapidly propagating fracture will be minimized, so that the requirements of GDC 1, GDC 16, and GDC 51 will be met.

6.3 Emergency Core Cooling System

6.3.1 Introduction

The ECCS provides a safety function of supplying borated water as emergency core cooling for the U.S. EPR. The ECCS limits fuel assembly damage and supplies emergency short and long term borated water for core cooling following a LOCA. The ECCS removes post-accident decay heat from the RCS and provides post LOCA containment cooling by way of the LHSI heat exchangers. The design of the ECCS uses four independent and separated trains, each housed and protected in its own seismically qualified Safeguard Building to perform its safety function. The ECCS performs a safety function by providing borated water to the RCS in the event of certain postulated transients, accidents, and operational events including a main steam line break, steam generator tube rupture, small-break LOCA, large-break LOCA, inadvertent opening of a pressurizer safety relief valve (PSRV), and RCS loop level decrease during shutdown or midloop operations.

6.3.2 Summary of Application

FSAR Tier 1: Tier 1 information associated with the ECCS is found in FSAR Tier 1, Sections 2.2.2, "In-Containment Refueling Water Storage Tank System," and 2.2.3, "Safety Injection System and Residual Heat Removal System."

FSAR Tier 2: Information provided by the applicant in FSAR Tier 2, Section 6.3, "Emergency Core Cooling System," is summarized here, in part as follows:

The ECCS is comprised of four identical piping and component supply and return circuits, or trains, one for each of the four RCS loops. Each of the piping and components trains can supply 100 percent of the credited core cooling capability following a DBA. The four ECCS trains each serve the safety function of injecting borated water into the RCS through parallel paths using a LHSI pump, an MHSI pump, and a pressurized accumulator. The water supply for the LHSI and MHSI pumps is from the IRWSTs.

The MHSI pumps and the accumulators of each train inject directly into one of the cold leg piping sections of each of the four RCS loops. The LHSI pumps inject through the LHSI heat exchangers to the cold legs. Closed loop cooling (supply and return trains) using the LHSI pump for post-accident heat removal is also available by aligning the suction piping of the LHSI pumps to the RCS hot legs. The LHSI system train may also be realigned subsequent to a DBA to inject borated water into the hot leg piping of the RCS loops to minimize boron precipitation and assist in further cooling down of the reactor core to mitigate steam releases into the Containment Building.

ITAAC: The ITAAC associated with FSAR Tier 2, Section 6.3 are given in FSAR Tier 1, Section 2.2.2, Table 2.2.2-3, "In-Containment Refueling Water Storage Tank System ITAAC," and in Section 2.2.3, Table 2.2.3-3, "Safety Injection System and Residual Heat Removal System ITAAC."

Technical Specifications: The TS applicable to the ECCS can be found in FSAR Tier 2, Chapter 16, "Technical Specifications," Section 3.5, "Emergency Core Cooling System (ECCS)."

Initial Plant Testing: Initial testing of the ECCS is discussed in FSAR Tier 2, Section 14.2, "Initial Plant Testing Program," which specifies testing that is applicable to the ECCS in

FSAR Tier 2, Section 14.2.12.2.2, "Medium Head Safety Injection System (Test #014),"
FSAR Tier 2, Section 14.2.12.2.3, "Safety Injection Accumulator System (Test #015),"
FSAR Tier 2, Section 14.2.12.2.4, "Residual Heat Removal System (Test #016)," FSAR Tier 2, Section 14.2.12.13.15, "Pre-Core Safety Injection Check Valve Test (Test #175)," and
FSAR Tier 2, Section 14.2.12.13.17, "Pre-Core Safety Injection Initiated at HZP (Test #177)."

6.3.3 Regulatory Basis

The relevant requirements of NRC regulations for this area of review, and the associated acceptance criteria, are given in NUREG-0800, Section 6.3, "Emergency Core Cooling System," and are summarized below. Review interfaces with other SRP sections can also be found in NUREG-Section 6.3.

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

2. GDC 4, as it relates to dynamic effects associated with the environmental conditions associated with normal operation and accident conditions, among other things, anc that include flow instabilities and loads (e.g., water hammer).

3. GDC 27, "Combined Reactivity Control Systems Capability," as it relates to the reactivity control system design having the capability in conjunction with poison addition by the ECCS, to assure that under postulated accident conditions and with appropriate margin for stuck rods, the capability to cool the core is maintained.

4. GDC 35, "Emergency Core Cooling," as it relates to the ECCS being designed to provide 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.

5. GDC 36, "Inspection of Emergency Core Cooling System," as it relates to the ECCS being designed to permit appropriate periodic inspection of important components, such as spray rings in the reactor pressure vessel, water injection nozzles, and piping, to assure the integrity and capability of the system.

6. GDC 37, "Testing of Emergency Core Cooling System," as it relates to the ECCS being designed to permit appropriate periodic pressure and functional testing.

7. GDC 38, "Containment Heat Removal," as it relates to the ECCS safety function of reducing 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.

8. 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors," in regard to the ECCS being designed so that its cooling performance, as predicted by an acceptable evaluation model, is in accordance with specified acceptance criteria.

9. 10 CFR 52.47(a)(8), as it relates to demonstration of compliance with any technically relevant portions of the Three Mile Island requirements set forth in 10 CFR 50.34(f)

10. 10 CFR 50.34(f)(2)(xi), as it relates to the requirements that direct indication of reactor coolant system relief and safety valve position be provided in the control room.

11. 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 NRC regulations.

Acceptance criteria adequate to meet the above requirements include the following:

1. RG 1.1, "Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal System Pumps (Safety Guide 1)," November 1970.

2. RG 1.47, "Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems," February 2010.

3. RG 1.29, "Seismic Design Classification," March 2007.

4. RG 1.157, "Best-Estimate Calculations of Emergency Core Cooling System Performance," May 1989.

5. Design certification/COL-Interim Staff Guide (ISG)-019, "Review of Evaluation to Address Gas Accumulation Issues in Safety Related Systems," July 2010.

6.3.4 Technical Evaluation

<u>Functional Design Bases</u>

The ECCS provides protection for accidents including LOCAs, MSLBs, SGTRs, inadvertent opening of a pressurizer safety relief valve, and RCS loop level decrease during shutdown or mid-loop operation. The ECCS must be designed to perform each of the following functions:

- Core cooling to prevent fuel damage
- Post accident decay heat removal
- Post accident containment cooling to preserve containment integrity
- Control of reactivity changes to maintain core subcriticality

During normal operation, the ECCS is idle but configured for rapid automatic response. The accumulator isolation valves are open. Only isolation check valves separate the pressurized borated water of the accumulators from the RCS. Both the MHSI pumps and the LHSI pumps are aligned for injection from the IRWST into the cold legs of the RCS. Both the MHSI and LHSI pumps, which share a three-way valve from the IRWST suction line, are idle. In addition, each pump train has one valve closed in the pump suction line and two valves closed in the discharge line. In addition, there is a check valve in each pump's injection line to prevent over pressurization of the injection system. A safety injection actuation signal will open the closed motor-operated valves and start the pumps. The ECCS is composed of four parallel trains that are independent of one another and are identical in design.

Loss-of-Coolant Accident

In case of a LOCA, the RCS pressure drops due to the breach in the system. When RCS pressure drops below a gauge pressure of 4,585 kPa (665 psig), the accumulators' borated water starts to discharge into the cold legs of the RCS. Low pressurizer pressure initiates a safety injection actuation signal that opens the appropriate valves, and starts the MHSI and LHSI pumps. The MHSI pumps start to deliver flow at a gauge pressure of 9,446 kPa (1,370 psig), and the LHSI pumps deliver flow at a gauge pressure of 2,275 kPa (330 psig). The accumulators are designed so that nitrogen stored in the accumulators gets discharged from the accumulator tank at a pressure lower than LHSI pump discharge pressure. Thus, the accumulators do not inject nitrogen into the RCS prior to commencement of LHSI injection.

If the break is large, the reactor core will uncover. The continued safety injection will eventually reflood the core, thus preventing unacceptable fuel damage. Heat generation in the core continues due to the decay of radioactive materials. The safety injection flow will remove the decay heat from the core, and through the break deposit it in the containment. The mass and energy release into the containment will heat up the containment and increase containment pressure. Water spilled in the containment and steam condensed on the containment walls and other containment structures will end up in the IRWST which is located at the bottom of the containment. It will replenish the tank's water that was removed by the ECCS, thus water will be available for long term cooling as long as it is needed, and GDC 38 is satisfied in regard to water availability. In order to provide heat removal from the containment and provide subcooled water for continued safety injection, the LHSI water taken from the IRWST is circulated through a heat exchanger cooled by the CCWS.

As the coolant temperature in the reactor core decreases, the reactivity of the core increases. Furthermore, GDC 27 requires the assumption of one stuck control rod for postulated accident scenarios. The ECCS must compensate for these reactivity changes (GDC 27). This is accomplished by borating the water stored in the accumulators and by borating the water in the IRWST. The accumulators are filled with 1,800 mg/Kg (1,800 ppm) borated water. The IRWST provides the necessary inventory of borated water for each of the above mentioned accidents. It contains a minimum amount of 1,892 m^3 (66,800 ft^3) of borated water at 1,800 mg/Kg (1,800 ppm) boron concentration. The water in the tank is monitored for its level, temperature, and boron concentration.

In addition, the EBS is available for providing additional boration. The EBS is a safety-related system capable of injecting a concentrated boric acid solution into the RCS against RCS pressure. The system consists of two trains, each with a high pressure positive displacement pump and an EBS tank. During normal operation, the pumps are in standby mode; they must be started manually. The staff's review of the EBS system is contained in Section 9.3.4 of this report.

Boron precipitation in the reactor vessel is prevented by continual pumping of water through the core. The flow prevents boron concentration increases in the water remaining in the reactor vessel. Furthermore, safety injection can be switched from cold leg injection to hot leg injection to further mitigate boron precipitation and steaming from the break.

Some of the SBLOCAs do not initiate safety injection immediately, because the reactor coolant loss can be made up by the chemical and volume control system. Eventually, the loss of coolant results in a decrease in RCS pressure and pressurizer level. Low RCS pressure triggers a reactor trip. The reactor trip signal also trips the turbine and closes the main feed water high-load lines. A safety injection signal is actuated on low-low pressurizer pressure.

The safety injection signal starts the MHSI and LHSI pumps. If offsite power is available, the low-load lines feed the steam generators. If offsite power is unavailable, the safety injection signal starts the emergency feedwater system. The secondary system will start to cool down, which causes the primary systems to cool down and decrease in pressure. As pressure decreases, the MHSI pumps will start to deliver. Eventually, RCS pressure falls below the accumulator's pressure and possibly below the LHSI pressure. The core may partially uncover before the rate of ECCS water addition exceeds the flow out of the break and the core then recovers. The operator can initiate other systems to continue cooldown, depressurize the RCS, and bring the reactor to a safe-shutdown condition.

The pressurizer of the U.S. EPR has three relief lines, each with a PSRV. The PSRVs also serve as safety valves, thus there are no block valves downstream of the PSRVs. The PSRVs are normally closed. The spurious opening of a PSRV is considered an anticipated operational occurrence. An inadvertent opening of a PSRV event is similar to a SBLOCA on the hot side of the RCS. It is different from an SBLOCA in that the pressurizer level initially increases, and the RCPs continue to run until a RCP trip signal is generated on the combination of a safety injection signal and low pressure differential across the pumps. The safety injection signal is generated on detection of low-low pressurizer pressure and automatically starts the MHSI and LHSI pumps.

Main Steam Line Break Accident

In the event of a main steam line rupture, the MHSI system primary functions are to provide borated water to the RCS to control the reactivity and coolant inventory during the cooldown. MHSI initial injection depends on the size of the main steam line break; for instance, MSHI injection would occur approximately 12 minutes into the event for a large break when the RCS cold-leg pressures drop below the MHSI pump shutoff head. The safety analysis demonstrates that the heat transfer from the SG primary side to the secondary side varies significantly during the event. Initially, before the MHSI system begins to inject, the heat transfer increases due to an increase of feedwater flow into the SG. Later in the event, the heat transfer decreases when the secondary side of the SG tubes experience a dry out. Due to the changes in heat transfer capabilities, the primary side under goes a series of reactivity, power, and pressure changes which eventually causes the pressurizer to reach the low-low pressurizer pressure setpoint which results in the actuation of MHSI. The safety analysis conservatively assumes that offsite power remains available to operate the reactor coolant pumps, thus increasing heat transfer through the SGs. The event ends when the operator terminates the emergency feedwater flow to the affected SG, after which the RCS gradually reheats and reactor shutdown margin increases. This event is evaluated in detail in Section 15.1.5 of this report.

Steam Generator Tube Rupture Accident

The postulated accident is a double-ended rupture of one SG tube. Coolant from the RCS begins to enter the secondary system. The RCS begins to depressurize, and the pressurizer level decreases. In the meantime, pressure, inventory and radioactivity increase in the affected SG. A reactor trip occurs either on low pressurizer pressure or high SG pressure. Alternately, the operator trips the reactor. The increasing SG pressure opens the main steam relief train, which discharges steam to the atmosphere. The RCS pressure continues to decrease. At the low-low pressurizer set point, the safety injection system is actuated, which starts the MHSI and LHSI pumps. Eventually, the affected SG is isolated, terminating the radiological release. Coolant loss is terminated when the RCS pressure reaches the pressure of the affected SG. The MHSI flow restores RCS inventory, leading to a controlled state.

RCS Loop Level Decrease

During reactor shutdown or mid loop operation, the LHSI pumps operate in the RHR mode. The pumps are taking suction from the hot legs of the RCS, recirculating the water through the LHSI heat exchangers, and injecting into the cold legs of the RCS. In the event of spurious drainage from the RCS or a SBLOCA, the LHSI pumps are not available for safety injection. In this case, the MHSI pumps provide reactor coolant makeup. A line connects the MHSI discharge line back to the IRWST to provide for minimum flow. This line branches into two flow lines, the smaller one for pump minimum flow protection and the larger one for reducing the MHSI discharge head. To compensate for the reduced pressure and flow credited for this event, the large MHSI minimum flow line opens prior to injection to reduce the MHSI discharge head in order to maintain the pressure of the RCS near nominal operating pressures during the event and avoid a sharp increase in RCS pressure. Typically, the RCS pressure stays below a gauge pressure of 3,999 kPa (580 psig) during this event.

Summary

As discussed above, the staff finds that the description provided in FSAR Tier 2, Section 6.3 includes a complete discussion of the ECCS functional design basis for the U.S. EPR. The system is designed to perform each specified ECCS safety function. The staff finds that the applicant considered all postulated accident scenarios where ECCS involvement in accident mitigation is credited, and designed the system to perform its intended function under the conditions that are assumed to exist. The FSAR Tier 2, Section 6.3 description states that the ECCS meets all functional design requirements.

<u>System Design</u>

The ECCS has four redundant safety injection trains. Each train consists of an accumulator, an MHSI pump, an LHSI pump, a heat exchanger and related piping and valves. The ECCS takes suction from the IRWST and delivers it to the RCS. A single suction line connects each train to the IRWST, and then branches into the MHSI suction line and the LHSI suction line. The heat exchanger is placed in the LHSI suction line. Normal safety injection is delivered into the cold legs of the RCS through a single nozzle for each train. The accumulator injection line, MHSI injection line, and the LHSI injection line join upstream of the nozzle. The LHSI pump also has a suction line from the RCS hot leg for residual heat removal. This line can be re-aligned for LHSI into the hot leg. Cross-connects between Trains 1 and 2, and Trains 3 and 4, allow individual trains to be removed from service for maintenance.

Each of the trains has its own independent power supply connected to an onsite and an offsite power system housed in a separate Safeguard Building adequately ventilated to accommodate post-accident radioactive conditions as further described in FSAR Tier 2, Section 9.4.5, "Safeguard Building Controlled-Area Ventilation System." As described in Section 8.2 of this report, each of the power system trains can individually power its respective ECCS train to its full capacity, satisfying the requirements of GDC 17. In addition, each of the trains is also powered from a separate emergency diesel generator. Each of the trains is housed in a separate seismic Category I structure. As described in Section 3.1.1 of this report, the buildings protect against damage from other natural phenomena (flood, weather, missiles, tsunami) as required by GDC 2. All components of the ECCS exposed to the borated water coolant are fabricated of austenitic stainless steel.

The functions of the ECCS are automatic, and are initiated by a safety injection actuation signal. No operator action is necessary. The water source is the IRWST, which is located at the bottom of the containment and is continually replenished with water spilled and steam condensed in the containment. No realignment of the system is necessary to initiate safety injection. Later in the accident, to prevent boron precipitation in the reactor vessel and to mitigate steaming from the break, safety injection is switched to the hot leg injection mode. This occurs one to three hours into the event and it is done manually. Operation of the ECCS is performed from the main control room for all operating conditions. Operation can also be performed from the remote shutdown station.

The design basis of the ECCS is presented in FSAR Tier 2, Section 6.3.1, "Design Bases." The staff compared the specific design specifications with the acceptance criteria given in SRP Section 6.3. The staff finds that the U.S. EPR design specifications meet the SRP acceptance criteria with a few exceptions described below, where additional information is needed to establish compliance with GDC 35, and conformance to the SRP acceptance criteria. The exceptions are discussed below under the headings: Net Positive Suction Head, and Piping and Valves.

Accumulators

Each accumulator is an austenitic stainless steel tank, approximately 9.14 m (30 ft) high, located in the containment and filled with borated water and nitrogen gas. A minimum gauge pressure of 4,399 kPa (638 psig) in the accumulator is needed for operation and a minimum liquid volume of 35 m^3 (1,236 ft^3). The minimum boron concentration is 1,700 mg/Kg (1,700 ppm), and the maximum and minimum operating temperatures are 50 °C (122 °F) and 15 °C (59 °F), respectively.

During normal operation, the accumulator isolation valves are open. Only isolation check valves separate the accumulators from the RCS. Thus, when RCS pressure drops below the accumulator pressure, the check valves open, and the borated water is forced from the accumulator into the RCS by the nitrogen gas pressure. The accumulators are connected to the minimum flow line of the MHSI pumps to facilitate filling the accumulators.

The accumulator is designed for a gauge pressure of 5,516 kPa (800 psig). Each accumulator is equipped with a safety relief valve that prevents a pressure rise of more than 10 percent over the design pressure. The existence and operation of the accumulator safety relief valves were not explicitly discussed in the FSAR (in Tier 1 or Tier 2) for the ECCS. Therefore, in RAI 212, Question 06.03-5, the staff requested that the applicant explain why the safety valves shown in FSAR Tier 2, Figure 6.3-2, "Safety Injection/Residual Heat Removal System Train (Typical)," were not shown in FSAR Tier 1, Figure 2.2.3-1, "Safety Injection System and Residual Heat Removal System Functional Arrangement," and also requested that the applicant provide a discussion of the overpressure protection of the accumulators. In a May 13, 2009, response to RAI 212, Question 06.03-5, the applicant stated that safety valves are not shown in Tier 1, Figure 2.2.3-1, because, in accordance with FSAR Tier 2, Section 14.3.2, "Tier 1, Chapter 2, System Based Design Descriptions and ITAAC," features provided solely for protection do not need to be included in the Tier 1 material. The applicant explained that each accumulator's safety valve is designed to prevent a rise of more than 10 percent above its design pressure, per the ASME B&PV Code, Section III, Article NC-7311(b), "Relieving Capacity of Pressure Relief Devices." The safety valves prevent the accumulators from being over pressurized during filling by the MHSI pumps. In addition, the applicant stated that flow restrictors are present in the accumulator filling lines to reduce flow from the MHSI line into the accumulators. Since each accumulator has a safety valve that prevents significant overpressure in the accumulator in accordance with ASME Section III, NC-7311(b), the staff finds the applicant's

May 13, 2009, response to RAI 212, Question 06.03-5 is acceptable. Therefore, the staff considers RAI 212, Question 06.03-5 resolved.

Safety Injection Pumps

The MHSI and LHSI pumps are horizontally mounted centrifugal pumps. The motors of the MHSI pumps are cooled by the CCWS. The LHSI pump motors and seals in Trains 2 and 3 are also cooled by CCWS, while the pumps in Trains 1 and 4 are cooled by the safety chilled water system.

The nominal flow rate of the MHSI pumps is 2,271 L/m (600 gpm) at a nominal dynamic head of 689 m (2,260 ft) that is equivalent to a gauge pressure of 6,757 kPa (980 psig). The minimum net positive suction head (NPSH) of the pumps at maximum flow rate is 3.05 m (10 ft). Each of the MHSI pumps has its separate minimum flow line. The minimum flow line branches into two flow lines, a relatively small line for minimum flow protection and a larger one for reducing the MHSI discharge head. The smaller line is sufficient to provide adequate circulation to prevent overheating of the pumps. The larger minimum flow line is sized to reduce the MHSI injection head, in order to maintain the pressure of the RCS near nominal operating pressure and avoid a sharp increase in RCS pressure, if safety injection is needed during shutdown or midloop operation. The line is sufficient to compensate for the system pressure and makeup flow rate during shutdown.

The LHSI pump nominal flow rate and nominal flow head are 8,328 L/m (2,200 gpm) and 146 m (480 ft), which is equivalent to a gauge pressure of 1,434 kPa (208 psig). The minimum NPSH of the pumps at maximum flow rate is 2.5 m (8.2 ft), or a gauge pressure of 24.5 kPa (3.6 psig). Each of the pumps is equipped with a separate minimum flow line sufficient to prevent overheating of the pump. Having an individual minimum flow line for each pump precludes pump to pump interaction during minimum flow operation, thus preventing dead-heading (running a pump with little or no flow, which typically occurs when the system resistance at the pump discharge is higher than the pump discharge pressure) one or more of the pumps. Furthermore, failure of a motor operated valve in a minimum flow line would affect only one of the ECCS pumps.

The NPSH available for each pump type during the course of LOCAs is discussed in the following section.

Net Positive Suction Head

In order to show compliance with GDC 35, the applicant must demonstrate that the ECCS pumps will perform their intended functions during postulated accidents, among other things. The ECCS should be designed so that sufficient NPSH margin is provided to the system pumps assuming the maximum expected temperature of the pumped fluid and no credit for containment pressurization during the accident. Additional guidelines for evaluating the adequacy of pump performance and the availability of the sump for recirculation cooling following a LOCA are presented in RG 1.82, “Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident,” November 2003. The staff’s evaluation of whether the inlet design of the containment sump suction screen, assures that the containment sumps provide a reliable, long-term recirculation cooling capability, ECCS pump performance will be adversely affected by post-LOCA conditions impacting the sumps, and system operation and performance is consistent with the guidance of RG 1.82, is contained in Section 6.2.2 of this report.

When safety injection flow is switched to the hot leg mode, the path to cold leg injection remains open. There is also a minimum flow line open to the IRWST. Consequently, the safety injection flow will split three ways. In order for the ECCS to (1) prevent boron precipitation in the reactor vessel, and (2) reduce steam release into the containment, at least a specified fraction of the ECCS flow must reach the hot legs. The staff determined that the applicant should establish a criterion for the flow split, revise the Design Basis in FSAR Tier 2, Section 6.3.1, and incorporate a requirement to verify the flow splits as designed into the Safety Injection System ITAAC and into the startup testing program.

In RAI 212, Question 06.03-11, the staff requested that the applicant address the three-way flow split issue. In a May 13, 2009, response to RAI 212, Question 06.03-11, the applicant did not acknowledge the safety significance of hot leg injection that was already established in AREVA Technical Report ANP-10299P. The May 13, 2009, response to RAI 212, Question 06.03-11 quoted FSAR Tier 2, Section 14.2.12.2.4, which did not address this issue. Furthermore, the applicant did not acknowledge the need for an ITAAC. However, in a March 24, 2010, response to RAI 212, Question 06.03-11, the applicant adequately addressed the staff's concerns by revising FSAR Tier 1, Table 2.2.3-3, "Safety Injection System and Residual Heat Removal System ITAAC," to include two new ITAAC steps. These steps will confirm that the safety injection pump flow is delivered to the RCS within the maximum elapsed times allowed with and without off site power available, and that the flow rate for each LHSI pump to its respective RCS hot leg is within the credited minimum and maximum flow rates. The staff finds that the changes adequately address the concerns that were identified. The staff confirmed that Revision 2 of the FSAR, dated August 31, 2010, contains the changes committed to in the RAI response. As stated in Section 6.2.1.3 of this report, the applicant has stated that the flow split will provide 75 percent of flow to the hot leg during the most limiting postulated large break LOCA event. The staff notes that an ITAAC has been included, as discussed in Section 6.2.1.3 of this report, to verify the minimum amount of flow split. Therefore, the staff finds that the applicant has adequately addressed this issue and, therefore, the staff considers RAI 212, Question 06.03-11 resolved.

FSAR Tier 2, Section 6.3.3.3, "NPSH Evaluation," states that an evaluation performed for the MHSI and LHSI pumps demonstrates sufficient NPSH is available during postulated DBAs. Assumptions used for the evaluation are:

- Conservatively calculated IRWST temperature (higher temperature means lower NPSH)
- Maximum pump flow head-capacity curves
- Maximum system resistances
- Debris laden sump screen resistance
- Reduced IRWST level to account for liquid hold up in the containment
- IRWST liquid is at the saturation pressure corresponding to the peak calculated IRWST temperature
- The MHSI and LHSI pumps operate simultaneously

The details and the results of the evaluation were not presented. Features of the IRWST, including sump screen design, debris accumulation on the screens, and sump screen flow resistance, are discussed in FSAR Tier 2, Section 6.3.2.2.2, "System Components." For further details on post-accident debris accumulation and conformance with the recommendations of RG 1.82, the applicant referenced AREVA Technical Report ANP-10293, "U.S. EPR Design Features to Address Generic Safety Issue (GSI)-191," Revision 0, February 2008 (this information was unchanged in ANP-10293, Revision 2, November 2010). ANP-10293, Section 3.2.1, "NPSH Assessment," presents NPSH assessment results in a summary form. There is no mention of how the results were obtained, how the minimum NPSH was determined, or what the justification is for the assumed 1.524 m (5 ft) strainer head loss. The report also does not mention whether the assumptions discussed in U.S. EPR FSAR Tier 2, Section 6.3.3.3, were used. In RAI 212, Question 06.03-6, the staff requested that the applicant provide additional information on the NPSH calculations. In a May 13, 2009, response to RAI 212, Question 06.03-6, the applicant provided details of the NPSH calculations, confirmed use of the assumptions in FSAR Tier 2, Section 6.3.3.3; and attempted to justify use of the saturation pressure corresponding to the peak IRWST temperature and use of a bounding calculation instead of the transient analysis recommended in RG 1.82. The applicant also discussed in the response input to a simplified bounding calculation without providing results. The staff notes that calculations reviewed during a February 15 and 16, 2011, audit appeared to indicate that the assumptions made in the response to RAI 212, Question 06.03-6 may no longer be valid. Therefore, in follow-up RAI 416, Question 06.03-15, the staff requested that the applicant provide a complete evaluation of NPSH using appropriate assumptions. **RAI 416, Question 06.03-15 is being tracked as an open item.**

The calculations cited by the applicant do not follow the recommendations of RG 1.82. There are two major deviations: use of the saturation pressure and use of a bounding analysis. The acceptance criterion on containment pressure calls for use of the containment pressure that existed prior to the accident. The staff's concern that is if proper operation of the ECCS system depends upon maintaining the containment pressure above a specified minimum, then too low an internal pressure could significantly affect the ability of the system to accomplish its safety function by causing pump cavitation.

Specifically, RG 1.82 (and RG 1.1) states that ECCS pump performance should be independent of the calculated increases in containment pressure caused by postulated LOCAs, and RG 1.82 also states that sufficient available NPSH should be provided to system pumps assuming no increase in containment pressure from that present prior to the postulated LOCA and assuming the maximum expected temperature of the pumped fluid. For NPSH analyses, the U.S. EPR design assumes the containment pressure is equal to the saturation pressure, which corresponds to the fluid (IRWST) temperature. Because the IRWST water temperature exceeds 100 ^{o}C (212 °F), the corresponding saturation pressure exceeds the pressure initially present in containment. Pressure above the pressure initially present in containment is termed containment accident pressure (CAP). In other words, CAP is the pressure developed in containment during a postulated accident. Therefore, because the U.S. EPR uses containment pressure greater than what is initially present in the containment, the U.S. EPR design is considered to use CAP in evaluating the NPSH for pumps that perform emergency core cooling and containment heat removal functions. The staff determined that the use of CAP deviates from staff guidance and justification needs to be provided. In RAI 416, Question 06.03-15, the staff requested that the applicant provide a complete evaluation of NPSH using appropriate assumptions, including consideration of CAP. **RAI 416, Question 06.03-15 is being tracked as an open item.**

RG 1.82 also recommends that calculation of available NPSH should be performed as a function of time until it is clear that the available NPSH will not decrease further. The applicant elected to replace the time dependent calculation with a bounding calculation. Input data for a revised bounding calculation was presented, without providing the results of the calculation. However, it was stated that the NPSH margin obtained from the bounding calculation was less than that obtained from the time dependent calculation. No justification was provided for the assumed IRWST water level. The pressure loss used to account for debris accumulation on the sump strainers was not given and was not justified. Use of the worst pump performance curve (pump operating at the minimum curve) was not justified. Only the cold leg injection mode was considered. The applicant did not discuss the effect of air ingestion on NPSH. Depending on the minimum water level in the IRWST and the hydraulic characteristics of the sump, air ingestion can significantly reduce the NPSH margin. The NPSH margin should be corrected for air ingestion. Thus, there is no assurance that the calculated NPSH margin is bounding.

The staff determined that the applicant's NPSH calculation did not consider all the contributing factors, and that the applicant's May 15, 2009, response to RAI 212, Question 06.03-6, was unacceptable. The applicant failed to adequately demonstrate that sufficient NPSH is available at the suctions of the ECCS pumps. In follow-up RAI 416 Question 06.03-15, the staff requested that the applicant provide additional information to address the unresolved NPSH considerations listed above. **RAI 416, Question 06.03-15 is being tracked as an open item.**

Piping and Valves

The piping arrangement of the ECCS and location of ECCS valves is shown in FSAR Tier 2 Figure 6.3-1, "Safety Injection System Overview." The pipes, valves, and fittings are austenitic stainless steel. The ECCS piping is protected from over pressurization by safety relief valves placed in each pump's injection line. The relief valves are designed to limit system over pressurization to 110 percent of design pressure. ECCS leakage is detected and monitored in containment, as well as in the Safeguards Buildings. Leakage inside containment is monitored by the reactor coolant pressure boundary leakage detection system. This system is described in FSAR Tier 2, Section 5.2.5, "RCPB Leakage Detection." Each safeguards building has sump level indication to detect leakage. FSAR Tier 2, Section 6.3 states that control is exercised through operating procedures.

In RAI 32, Questions 06.03-1 and 06.03-2, the staff requested that the applicant clarify why some of the valves shown in FSAR Tier 2, Figure 6.3-1 were not included in the ITAAC program (FSAR Tier 1, Table 2.2.3-1, "SIS/RHRS Equipment Mechanical Design," and Figure 2.2.3-1, "Safety Injection System and Residual Heat Removal System Functional Arrangement"). In an July 29, 2008, response to RAI 32, Questions 06.03-1 and 06.03-2, the applicant stated that the three valves identified in the RAIs were a manually-operated maintenance isolation valve, a three-way isolation valve used to isolate the safety injection system (SIS) suction line from the IRWST, and a check valve that do not have a safety-related active function and could, therefore, be omitted from the ITAAC program per the guidance in SRP Section 14.3. Since the valves do not have an active safety function, and the preoperational testing of the ECCS system will verify that the valves are correctly installed and do not obstruct the flow path in the ECCS suction lines, the staff finds the applicant's July 29, 2008, response to RAI 32, Questions 06.03-1 and 06.03-2 acceptable..

FSAR Tier 2, Section 6.3 states that the suction piping from the IRWST to the MHSI and LHSI pumps is designed to be self venting to preclude void formation. The LHSI suction piping from the RCS hot legs is designed similarly. Having the suction lines continuously vented if the entire ECCS is flooded prevents suction cavitation of the pumps and occurrence of water hammer in the piping, thereby satisfying GDC 4 requirements, with respect to those phenomena. The staff noted that gas accumulation can cause water hammer, gas binding in pumps, and inadvertent relief valve actuation that may damage pumps, valves, piping, and supports and may lead to loss of system operability. Therefore, in RAI 310, Question 06.03-13, the staff requested that the applicant confirm the current features of the U.S. EPR ECCS that allow the implementation of SR 3.5.2.2 during all modes of plant operation to demonstrate compliance with GDC 4. In a November 20, 2009, response to RAI 310, Question 06.03-13, the applicant described high point vents in the SIS/RHRS lines, which will be vented on a 31-day basis in accordance with FSAR Tier 2, Chapter 16, SR 3.5.2.2 during all modes of plant operation. The applicant states that these vents will reduce or eliminate the effects of gas intrusion in regard to NPSH and water hammer issues. The staff will confirm this when the open item discussed below is resolved.

To address concerns on potential gas accumulation in the ECCS, the staff issued RAI 310, Question 06.03-12, to request that the applicant identify and evaluate the potential pathways for gas intrusion into the ECCS (CS/RHR) system, describe design features in the U.S. EPR that prevent or control gas accumulation to ensure CS/RHR system operability, identify any means of detecting unacceptable levels of gas accumulation, and describe the ITAAC test conditions for the CS/RHR pumps. In a November 20, 2009, response to RAI 310, Question 06.03-12, the applicant identified potential pathways for gas intrusion in the SIS/ RHRS and design features that would prevent or control gas accumulation to acceptable levels. The staff finds that the applicant's responses to RAI 310, Questions 06.03-12 and 06.03-13 satisfy two of the three conditions of the proposed staff guidance in DC/COL-ISG-019, "Proposed Interim Staff Guidance - Review of Evaluation to Address Gas Accumulation Issues in Safety Related Systems." Specifically, the staff finds that the applicant has addressed the conditions of potential gas accumulation locations and intrusion mechanisms, and surveillance and venting procedures. The condition that was not discussed involves confirming that ITAAC exist that require the applicant to compare the as-built plant configuration to the P&IDs and isometric drawings to confirm all potential gas accumulation areas have been properly identified and that appropriate preventative measures are in place. Since the applicant has not provided ITAAC to confirm that the as-built plant configuration is consistent with the P&IDs and isometric drawings and properly identify that all potential gas accumulation areas, the staff determined that the applicant's November 20, 2009, response to RAI 310, Question 06.03-12 was unacceptable. The staff issued RAI 480, Question 06.03-17 to request that the applicant provide an ITAAC that satisfies the guidance in ISG-019. **RAI 480, Question 06.03-17 is being tracked as an open item.**

The ECCS is isolated from the RCS by two isolation check valves in series, i.e., the RCS isolation check valves and the ECCS isolation check valves in the MHSI and LHSI pump injection lines. The section of the ECCS piping that is located outside the containment can be isolated from the containment in the event of a break in the piping in accordance with 10 CFR 50.34(f)(2)(xxvi).

In order to provide protection against intersystem LOCAs, ECCS piping from (and including) the first isolation check valves in the injection lines to the RCS are designed to the RCS design pressure. The remaining portion of the ECCS piping has ultimate rupture strength greater than the RCS operating pressure. ECCS design features addressing intersystem LOCAs are discussed in FSAR Tier 2, Section 5.4.7, "Residual Heat Removal System," and evaluated by the staff in Section 5.4 of this report.

A given train's accumulator injection line, MHSI pump injection line, and the LHSI pump injection line join together into a common header to share an injection nozzle into the RCS. In the event of a large break LOCA, the safety injection actuation signal initiates both the MHSI and the LSHI systems. Also, the accumulators will start to discharge as soon as the RCS pressure drops below the accumulators' pressure. All three components of the ECCS are injecting at the same time through a common injection line segment and a common injection nozzle. The merging flows interfere with one another and introduce flow resistance and uncertainties in calculation of the safety injection flow. LOCA calculations performed to show compliance with the ECCS acceptance criteria use minimum safety injection flow, while some of the containment pressure calculations discussed in FSAR Tier 2, Section 6.2.1.3, "Mass and Energy Release Analyses for Postulated Loss of Coolant Accidents," use maximum safety injection flow. Information was not presented on how the maximum and minimum safety injection flows are calculated, and how the common injection line and nozzle affect the safety injection flow calculations. Therefore, in RAI 212, Question 06.03-7, the staff requested that the applicant address this issue.

In a May 13, 2009, response to RAI 212, Question 06.03-7, the applicant referred to FATHOM™ calculations and stated that the minimum safety injection flow is calculated using maximum system flow resistance and minimum pump curves. Similarly, maximum safety injection flow calculations utilize minimum system flow resistance and maximum pump curves. The applicant also indicated that the sample calculations requested by the staff are available for NRC inspection upon request. The staff finds the applicant's May 13, 2009, response to RAI 212, Question 06.03-7 acceptable, because the applicant used a computer code in wide use in the industry and with which the staff is familiar. In the staff's experience, the FATHOM™ code yields reasonable results for steady state calculation applications, and the applicant used this code with conservative inputs for performing their maximum and minimum safety injection flow calculations. In addition, the FATHOM™ code models the mixing effects identified above. Therefore, the staff considers RAI 212, Question 06.03-7, resolved.

Except for the RCS isolation check valves, all control valves of the ECCS are motor operated valves. Valve position indication is provided in the control room. Operation of the ECCS is performed from the main control room for all operating conditions. Operation can also be performed from the remote shut-down station. Accordingly, the RCS isolation check valves have a passive function and active failure of the control valves during normal operation will be identified through FSAR Chapter 16, SR 3.5.2.1 and SR 3.5.2.4. One set of valves, the downstream isolation valves for the accumulators, is secured in the open position (breakers racked out) during normal operation to protect against active failures, since the four accumulators are all assumed to be available in the safety analysis and maintenance is prohibited during power operation. Based on the above, the staff finds that, with exception of the open items identified above, the RCS and ECCS piping interface and valve system meets the acceptance criteria defined in GDC 35.

Heat Exchangers

The heat exchangers of the ECCS for core heat removal and containment cooling are placed in the LHSI trains. They are 8.23 m (27 ft) long, horizontally mounted U-tube type heat exchangers. The heat exchanger tubes, which are exposed to the borated water of the ECCS, are austenitic stainless steel. They are cooled by the CCWS on the secondary side. In the LOCA injection mode, they are designed for a LHSI flow rate of 150 kg/s (330 lbm/s), with the minimum flow recirculation line closed. The CCWS flow rate for Trains 1 and 4 is 376 kg/s (829 lbm/s) and for Trains 2 and 3 is 276 kg/s (608 lbm/s). Performance of the heat exchangers is discussed in FSAR Tier 2, Section 5.4.7.3, "Performance Evaluation." The staff's review of heat exchanger performance is contained in Section 5.4.7.3 of this report. Appropriateness of the fouling factors used in the evaluation is reviewed in Section 6.2.2, "Containment Heat Removal Systems," of this report.

Instrumentation and Control

The ECCS trains are monitored and controlled from the main control room. Valve positions and flow conditions in the system are displayed in the control room. Actuation of the system is automatic under accident conditions. A safety injection signal will open the motor operated valves in the injection path and will start the MHSI pumps and the LHSI pumps. No operator action is necessary. To control long term cooling, all important parameters (ECCS flow and temperature, containment pressure) are available to the operators in the control room. Abnormal operating conditions of the ECCS are alarmed in the control room to facilitate corrective action by the operator. Examples of abnormal conditions are: high bearing oil temperature, high motor winding temperature, or high motor air temperature, or loss of suction head.

There are various signals that initiate safety injection. LBLOCAs, SBLOCAs, and steam generator tube rupture events cause a safety injection signal on low-low pressurizer pressure. When reactor shutdown and cooldown is in progress, safety injection is initiated on a low difference signal between the hot leg pressure and hot leg saturation pressure. During reactor shutdown or mid loop operation, the MHSI pumps start automatically on loss of RCS level. The various safety injection actuation signals are sufficient to initiate emergency core cooling in a timely manner for each postulated accident type. Thus, the consequences of the events are mitigated and kept within regulatory limits.

For each train of the ECCS, the process variables measured by the instrumentation and control system are derived from independent sources. Similarly, the instrumentation and control system signals corresponding to a given train are powered from the normal and emergency power supplies that power the equipment in the train and that satisfy the requirements for redundancy and independence as described in Section 8.3 of this report, thus eliminating common failures. The staff's detailed review of the instrumentation and control is discussed and documented in Chapter 7, "Instrumentation and Controls," of this report.

In-Containment Refueling Water Storage Tank

The IRWST is an open pool at the bottom of the containment. It serves as a water source for both refueling and safety injection operations. In the event of design-basis events, it provides the necessary inventory of borated water for operation of the ECCS and provides sufficient water depth (static pressure head) to the suction of the ECCS pumps. FSAR Tier 2, Section 6.3 states that the IRWST is a concrete structure with a 0.99 cm (0.39 in.) thick austenitic stainless steel liner. The borated water volume is maintained between 1,894 m^3 (66,886 ft^3) and 1,982 m^3 (70,008 ft^3). The approximate water depth is 3.75 m (12.3 ft). Boron concentration can vary between 1,700 mg/kg (1,700 ppm) and 1,900 mg/kg (1,900 ppm.) The minimum operating temperature is 15 °C (59 °F), while the maximum operating temperature is 50 °C (122 ºF).

The IRWST is monitored for its water level, temperature, and boron concentration. Should any of the variables deviate from the specified range, corrective action must be taken within 8 hrs.

The IRWST has three debris entrainment prevention features:

- Trash racks and weirs above the heavy floor openings
- Retaining baskets below each heavy floor opening
- Strainers above each sump

These features are described and discussed in FSAR Tier 2, Section 6.3.2.2.2. Furthermore, the applicant submitted Technical Report ANP-10293, which provides additional information on these design features and on the impact of post-accident debris accumulation on ECCS performance. The report also summarizes the performance evaluation and component test program, and discusses how the design features of the IRWST meet the regulatory positions of RG 1.82. The Technical Report and the debris entrainment design features of the IRWST, including the sump strainer head loss due to debris, are reviewed in this report under Section 6.2.2.

The IRWST is connected by pipes to the molten core spreading area that is located at a lower elevation. Valves in these pipes are closed during normal operation and also under accident conditions. In the event of a severe accident, molten core material that reaches the spreading area melts the actuation device and opens the flooding valves. Single failure of the flooding valves was not addressed in FSAR Tier 2, Section 6.3. In RAI 212, Question 06.03-9, the staff requested that the applicant explain why this failure was not considered and what would be the consequence of an inadvertent opening of a flooding valve during a design basis accident. In a July 16, 2009, response to RAI 212, Question 06.03-9, the applicant stated that a design modification made in response to RAI 212, Question 06.02.02-23, precludes an inadvertent opening of a flooding valve from having a negative impact on the performance of the MHSI and LHSI pumps. The applicant stated that the original design called for two normally open isolation valves per passive flooding line connecting the IRWST to the spreading area, which is located in a compartment separate from the IRWST. In order to address the safety related function of protecting the IRWST water inventory and to provide sufficient NPSH for the safety injection system and residual heat removal pumps, the applicant made the following changes to the design:

1. One of the motorized isolation valves in each of the passive flooding lines was removed.

2. The normal operating position of the remaining motorized isolation valves was changed from “normal open” to “normal closed” position.

3. The motorized isolation in each remaining passive flooding line is powered from separate divisions (Division 1 and Division 4) and backed by 12-hour uninterruptible power supply. The valves are deactivated during normal operation with their electrical supply breakers open to remove electrical power.

Based on the above changes, the isolation valves are normally closed and disconnected from the power system to maintain a condition in which a single failure cannot result in the loss of a safety function. Accordingly, no single failure can result in the loss of the safety function to maintain the water level of the IRWST at an appropriate level for emergency core cooling system pump NPSH, and the staff finds the applicant’s response acceptable. Therefore, the staff considers RAI 212, Question 06.3-09, Part a resolved.

Another emergency core cooling function of the IRWST is to provide sufficient water depth to prevent cavitation of the safety injection pumps during DBA conditions. Water level in the tank varies during accidents. Sufficient NPSH needs to be available throughout the course of these events. The staff evaluates protection against loss of NPSH above under the “Net Positive Suction Head” heading.

The staff noted that FSAR Tier 2, Chapter 16, “Technical Specifications,” TS 3.5.4, “In-Containment Refueling Water Storage Tank (IRWST) – Operating” requires correction of the IRWST’s water temperature and boron concentration within eight hours. In RAI 212, Question 06.03-8 requested that the applicant clarify how that would be done. Also, the staff requested that the applicant explain why the TS did not specify a completion time for restoring the IWRST water level. In a May 13, 2009, response to RAI 212, Question 06.03-8, the applicant stated that the IRWST boron concentration and level are restored by makeup from the reactor boron water makeup system. Operating procedures provide for the control and restoration of IRWST water parameters. The applicant further noted that an FSAR change made in response to RAI 101, Question 16-95, incorporates the restoration time for the IRWST water level into the TS. The staff confirmed that Revision 1 of the U.S. EPR FSAR, dated May 29, 2009, contains the changes committed to in the response to RAI 212, Question 06.03-8. Accordingly, the staff finds that the applicant adequately addressed this issue and, therefore, the staff considers RAI 212, Question 06.03-8 resolved.

Testing, Inspection and Qualification

The staff notes that ECCS testing requirements are specified in GDC 37. The staff's review confirms that the design of the ECCS permits online testing of individual trains and components to assess their operational status and availability. There are four parallel trains each corresponding to an RCS loop. At any time, including during power operation, one of the trains can be taken out of service for maintenance, consistent with TS. As explained in Section 15.6.5 of this report, the remaining trains, even with an assumed single failure, are sufficient to fulfill all safety functions of the ECCS. The accessibility incorporated into the design allows complete testing of components when plant conditions allow, such as during outages.

Initial testing of the ECCS verifies that the as-designed and as-constructed system functions as credited in the safety analysis. Initial testing is discussed in FSAR Tier 2, Section 14.2. The tests applicable to the ECCS in FSAR Tier 2, Section 14.2 are:

- Section 14.2.12.2.2 (Test No. 014), Medium Head Safety Injection
- Section 14.2.12.2.3 (Test No. 015), Safety Injection Accumulator
- Section 14.2.12.2.4 (Test No. 016), Residual Heat Removal
- Section 14.2.12.13.15 (Test No. 175), Pre-Core Safety Injection Check Valve Test
- Section 14.2.12.13.17 (Test No. 177), Pre-Core Safety Injection Initiated at Hot Zero Power (HZP)

In each of the above tests, the acceptance criteria involve whether the system functions as described in the design basis of FSAR Tier 2, Section 6.3. The staff reviewed these tests and finds the testing to be sufficient and adequate to demonstrate proper operation of the ECCS in accordance with its design-basis objectives with one exception. As discussed under the "Net Positive Suction Head" heading above, a certain fraction of the emergency core cooling (ECC) flow must reach the hot legs during hot leg injection. Otherwise, the ECCS cannot fulfill some of its safety functions. FSAR Tier 2, Section 6.3, should be revised to establish an additional portion of the design basis for the ECCS that specifies the minimum fraction of ECC to reach the hot legs during hot leg injection under accident conditions. As noted in Section 6.2.1.3 of this report, the applicant has stated that the flow split will provide 75% of the ECC flow to the hot leg during the most limiting postulated large break LOCA event and included ITAAC to verify this proportion of flow split. In a November 20, 2009, response to RAI 310, Question 06.03-12, the applicant included a revision to FSAR Tier 2, Section 14.2.12.2.4 (Test No. 16), to describe the NPSH test for the LPSI pumps at minimum suction level. The staff confirmed that Revision

2 of the U.S. EPR FSAR, dated August 31, 2010, contains the changes committed to in the RAI response. Accordingly, the staff finds that the applicant has adequately addressed this issue and, therefore, the staff considers RAI 212, Question 06.03-12 resolved. Periodic in-service testing confirms the continuing capability of the system as presented in FSAR Tier 2, Section 3.9.6, "Functional Design, Qualification, and In-service Testing Programs for Pumps, Valves, and Dynamic Restraints." It confirms the structural and leak tight integrity of components, and assurance of the operability of pumps and valves. The staff's review of structural and leak tight integrity of components, and assurance of the operability of pumps and valves is contained in Section 3.9.6 of this report.

Safety-related motor operated valve testing is addressed in FSAR Tier 2, Section 3.9.6. Fluid paths are isolated and are instrumented to accommodate maintenance and testing of the valves. Dynamic testing of the ECCS pumps is facilitated by minimum flow lines. Separate minimum flow lines are provided for each pump, eliminating concerns of pump loss due to sharing of minimum flow lines. The staff's review of the adequacy of the in-service testing program for pumps and valves is contained in Sections 3.9.6 and 6.6 of this report.

FSAR Tier 2, Section 6.3.1, "Design Bases," states that the U.S. EPR ECCS is designed to permit periodic inspection of important components and piping to verify the integrity and capability of the system in accordance with the requirements of GDC 36. Surveillance requirements for the system are described in FSAR Tier 2, Chapter 16. Safety-related motor operated valve surveillance is addressed in FSAR Tier 2, Section 3.9.6, and evaluated by the staff in Section 3.9.6 of this report. The staff's review of the surveillance program is presented in Chapter 16 of this report.

Some active components of the system, typically motor operated valves, are located inside the Containment Building. These components will be exposed to a harsh environment (high temperature steam air mixture at high pressure) in the event of a RCS break or a steam line break within containment. The ECCS must perform its intended function under these conditions. The electrical components of the system located inside containment must be qualified to operate in the expected environment in accordance with 10 CFR 50.49, "Environmental qualification of electric equipment important to safety for nuclear power plants." The ECCS valves inside containment are located above the maximum floor flooding level which protects the valve motor operators from submersion following an accident. Environmental qualification of the valves is addressed in FSAR Tier 2, Section 3.11, "Environmental Qualification of Mechanical and Electrical Equipment," and it is reviewed under the corresponding section of this report.

GDC 2 requires that systems important to safety, like the ECCS, be designed to withstand the effects of natural phenomena including earthquakes. Furthermore, RG 1.29 assigns Seismic Category I designation to the ECCS, meaning that the system is designed to remain functional after a safe-shutdown earthquake. The staff confirmed that seismic Category I is specified for the ECCS. The system and its components are designed to Seismic Category I requirements. The most severe ground motion spectra postulated as design basis information for the ECCS, among other SSCs, is set forth in FSAR Tier 2, Section 2.5.2, and whether the ECCS design will withstand the SSE based on these spectra is evaluated in Section 3.7.1 of this report. In view of the evaluation in that section, the requirement of GDC 2 is met. Also, each of the four independent trains of the ECCS is housed in separate reinforced concrete structures. The structures are Seismic Category I, thus protecting the system from potential earthquake damage, as well as other natural phenomena and external hazards. The structures are described in FSAR Tier 2, Section 3.8.4, "Other Seismic Category I Structures." Equipment seismic qualification is addressed in FSAR Tier 2, Section 3.10, “Seismic and Dynamic Qualification of Mechanical and Electrical Equipment.” FSAR Tier 2, Section 3.3, “Wind and Tornado Loadings,” and Section 3.4, “Water Level (Flood) Design,” discuss protection against other natural phenomena. These areas are evaluated by the staff in the corresponding sections of this report.

System Reliability

FSAR Tier 2, Section 6.3 states that the ECCS consists of four independent trains housed in four separate Seismic Category I structures. The process variables, such as RCS pressure and pressurizer level, used in the instrumentation and control system, including the safety injection actuation signal, are derived from an independent source for each of the four trains. They are independently powered from the same normal and emergency sources that power the associated motive equipment of the train. The process variables for the I&C, such as RCS pressure and pressurizer level, derive their input from independent sources. The design of the SIS I&C, including its quality, redundancy, and protection against the effects of single failure, is evaluated in Section 7.3 of this report..

The applicant conducted a detailed failure mode and effects analysis for the ECCS. The analysis is presented in FSAR Tier 2, Table 6.3-7, “Safety Injection System Failure Modes and Effects Analysis.” GDC 35 requires that the ECCS accomplish its safety functions assuming a single failure. During design basis accidents, opening of a valve could potentially deplete the IRWST level below the level needed to supply NPSH for the safety injection pumps. The applicant’s failure mode and effects analysis did not address potential failure of the flooding valves leading to the molten core spreading area. However, in a July 16, 2009, response to RAI 212, Question 06.03-9, the applicant noted that a design modification, made in the July 16, 2009, response to RAI 212,Question 06.02.02-23, which is discussed above in connection with the latter RAI, precludes the inadvertent opening of a flooding valve from having a negative impact on the performance of the MHSI and LHSI pumps. For the same reasons set forth in the staff evaluation of RAI 212, Question 06.02.02-23 above, the staff finds the response acceptable and considers RAI 212, Question 06.03-9, Part b resolved.

In RAI 212, Question 06.03-10, the staff requested that the applicant clarify the consequence of losing an EDG in alternate feed mode in conjunction with a SIS/RHRS train out for maintenance. In a May 13, 2009, response to RAI 212, Question 06.03-10, the applicant explained that the EDG in alternate feed mode provides power to one in-service SIS/RHRS train, as well as power to selected equipment of the train undergoing maintenance. Therefore, the loss of the EDG on alternate feed mode will result in the loss of interruptible power to two SIS/RHRS trains, one of which is already out of service for maintenance. Mission success criteria are satisfied for the remaining two SIS/RHRS trains. Since the EDGs are single failure proof as described in Section 8.4.2, power will be available to these two SIS/RHRS trains. Accordingly, the staff finds the applicant’s response acceptable. The staff notes that the most limiting single active failure for the SIS, which is assumed to occur at the onset of the design basis LOCA event, is the complete loss of one train. The staff reviewed the failure mode and effects analysis for the SIS, which includes electrical, mechanical, and instrumentation and control failure modes, and finds that no single active failure will affect more than one train of the ECCS since in each case of a mechanical, electrical, or instrumentation and control single failure, the SIS can successfully perform its mission criteria. The redundancy incorporated into the system design allows the SIS to fulfill its safety function in spite of such failures, as further addressed in Section 15.6.5 of this report. Therefore, the staff considers RAI 212, Question 06.03-10 resolved.

In addition to the single active failure evaluation, the applicant also examined the potential consequences of single passive failures during the long term cooling phase. The worst single passive failure is the loss of one coolant supply path. Guard pipes on the sump suction lines upstream of the three-way isolation valves limit the consequences of a suction line failure and prevent draining of the IRWST. Accordingly, the other trains are not affected.

Performance Evaluation

While the design of the ECCS is reviewed in this section of the report, the ECCS performance evaluation under accident conditions is presented FSAR Tier 2, Chapter 15, "Transient and Accident Analyses," and evaluated in the corresponding sections of this report. The limiting cases for ECCS performance evaluation are the LOCAs, including both SBLOCAs and LBLOCAs.

FSAR Tier 2, Section 15.6.5.1, "Large Break Loss of Coolant Accident," addresses the LBLOCA accident analysis and the corresponding ECCS performance evaluation. A spectrum of large breaks were analyzed starting from 10 percent of the cold leg pipe cross-sectional area (approximately 0.0465 m^2 (0.5 ft^2)) up to double ended split breaks and double ended guillotine breaks. A best estimate evaluation model was used for the analyses. Uncertainties were incorporated using a non-parametric statistical approach. The calculated peak cladding temperature is 833 °C (1,531 °F); the total cladding oxidation at the peak cladding temperature location is 0.83 percent. Hydrogen generation was below one percent, and coolable core geometry was maintained. The staff's evaluation of ECCS performance in case of an LBLOCA is contained in Section 15.6.5 of this report.

Analysis of SBLOCAs is presented in FSAR Tier 2, Section 15.6.5.2, "Small Break Loss of Coolant Accident." Various break sizes and break locations were considered starting with a 1.27 cm (0.5 in.) diameter break up to a 24.33 cm (9.58 in.) diameter break. The small break analysis encompasses breaks of charging and letdown lines, drain lines, instrumentation lines, and the opening of relief and safety valves. A conservative evaluation model was used for the analysis together with 10 CFR Part 50, Appendix K, "ECCS Evaluation Models," assumptions. The limiting break is a 16.51 cm (6.5 in.) diameter break with the assumption of loss of offsite power. The calculated peak cladding temperature is 892 °C (1,638 °F), total cladding oxidation is 0.38 percent, coolable geometry of the core is maintained, and design conditions of the RCS pressure boundary are not exceeded. Thus, the applicant's calculations show that the requirements of 10 CFR 50.46 are met. ECCS performance in the events of SBLOCAs is satisfactory. The staff's evaluation of ECCS performance in case of an SBLOCA is contained in Section 15.6.5 of this report

Inspection, Tests, Analysis, Acceptance Criteria

The ITAAC associated with the ECCS and the IRWST are given in FSAR Tier 1, Section 2.2.3 and Section 2.2.2, respectively. FSAR Tier 1, Table 2.2.3-3 specifies: (1) The design commitments of the ECCS; (2) the inspections, tests and analyses to be performed; and (3) the acceptance criteria to ensure that the ECCS is built as designed. Equivalent information for the IRWST is contained in FSAR Tier 1, Table 2.2.2-3. The staff's review indicates that the ITAAC information presented in FSAR Tier 1, Sections 2.2.2 and 2.2.3 is adequate except for the noncondensible gas purging tests identified above. With exception of this issue, the staff determined that the information presented in FSAR Tier 2, Section 6.3, is properly reflected in FSAR Tier 1, Sections 2.2.2 and 2.2.3. In RAI 480, Question 06.03-17, the staff requested that the applicant provide an ITAAC that satisfies the guidance in ISG-019 with regard to noncondensible gas purging tests. **RAI 480, Question 06.03-17 is being tracked as an open item.**

The staff notes that during hot leg injection, the ECC flow is split between hot legs, cold legs, and the IRWST. The staff notes that a certain fraction of the flow must reach the hot legs to maintain core cooling. As discussed above, the safety injection system ITAAC was provided to specify water delivery to the hot legs in the hot leg injection mode.

Technical Specifications

Technical Specifications relating to the ECCS are presented in FSAR Tier 2, Chapter 16, Section 3.5. Both the required actions and surveillance requirements were reviewed together with the completion times allotted for corrective action and surveillance frequencies. As part of the review, the staff identified that while there is a surveillance requirement in SR 3.5.4 and SR 3.5.6, "In-Containment Refueling Water Storage Tank (IRWST) – Shutdown, MODE 5," for the borated water volume of the IRWST, no specific action is required for restoring the water volume if it is found to be outside of the specified range. This issue was addressed by the staff in RAI 101, Question 16-95. In a December 12, 2008, response to RAI 101, Question 16-95, the applicant introduced a number of markups to the FSAR. One of these markups established an 8-hour completion time for restoring the IRWST water volume if it is found to be outside the specified range. Since the allowed completion time requires that prompt action will be taken to restore the tank to an operable condition as stated in FSAR Tier 2, Chapter 16, B 3.5.4, the staff finds the proposed markup to be acceptable. The staff confirmed that Revision 1 of the U.S. EPR FSAR, dated May 29, 2009, contains the changes committed to in the RAI response. Accordingly, the staff finds that the applicant has adequately addressed this issue and, therefore, the staff considers RAI 101, Question 16-95 resolved. The staff finds the specific TS requirements acceptable.

6.3.5 Combined License Information Items

Table 6.3.5-1 provides an ECCS related COL information item and description from FSAR Tier 2, Table 1.8-2, "U.S. EPR Combined License Information Items":

Table 6.3.5-1 U.S. EPR Combined License Information Items

Item No.	Description	FSAR Tier 2 Section
6.3-1	A COL applicant that references the U.S. EPR design certification will describe the containment cleanliness program which limits debris within containment.	6.3.2.2.2

The staff finds the above listing to be complete. Also, the list adequately describes actions necessary for the COL applicant or holder. No additional COL information items need to be included in FSAR Tier 2, Table 1.8-2, for ECCS consideration.

6.3.6 Conclusions

The staff evaluated the proposed design of the ECCS for the U.S. EPR as described primarily in FSAR Tier 2, Section 6.3, and reviewed proposed design requirements that pertain to the ECCS as specified in FSAR Tier 1, Sections 2.2.2 and 2.2.3. Other FSAR Tier 2 information was also evaluated with respect to ECCS considerations, such as the proposed Technical Specification requirements provided in Chapter 16, and initial test program considerations described in FSAR Tier 2, Section 14.2. The staff's evaluation was performed in accordance with the guidance provided in SRP Section 6.3 and included review of piping and instrumentation diagrams, equipment layout drawings, failure modes and effects analyses, and functional design specifications for essential components. The staff's evaluation included review of ECCS design considerations, such as piping and component arrangements and locations; actuation and control features; normal and post-accident configurations; electrical power supplies; and leakage detection, isolation, and atmospheric release considerations. The staff evaluated the design criteria and design bases that were proposed by the applicant for the ECCS, and the manner in which the design conforms to these criteria and bases and satisfies applicable NRC requirements such as the GDC and TMI Action Plan Items.

The ECCS is actuated to mitigate the consequences of an accident upon receipt of a safety injection signal. Each of the four ECCS trains includes LHSI and MHSI pumps, an accumulator, and piping, valves and controls necessary to provide makeup water from the IRWST and cool the core. Each train individually is capable of providing adequate core cooling, and is powered from separate safety-related electrical divisions that receive emergency power from their respective EDG sets. Because there is an ECCS train for each of four RCS loops, the ECCS safety function is assured with one train out for maintenance, one train inoperative due to a postulated single failure, and one train discharging its flow through the break.

Except for the open items identified above, and for the reasons set forth above, the staff finds that the design of the ECCS is acceptable and meets the requirements of GDC 2, GDC 4, GDC 27, GDC 35, GDC 36, GDC 37, and GDC 38; 10 CFR 50.34(f)(1)(vi); 10 CFR 50.34(f)(2)(xi), (xviii), and (xxvi); and 10 CFR 50.46 and 10 CFR 52.47(a)(8).

The staff also found that Technical Specifications and ITAAC that are proposed for the ECCS are acceptable. The ITAAC ensures that critical ECCS design details are satisfied by the as-built plant. Therefore, with the exception of the open items identified above, the staff finds that the proposed ITAAC are acceptable and adequate to ensure that the plant will be built in accordance with the design specifications, thereby satisfying the requirements specified by 10 CFR 52.47(b)(1). The staff also finds that the Technical Specifications are adequate to ensure that the plant will be operated in accordance with the design specifications, and that the ECCS will remain capable of adequately cooling the core during plant operation consistent with accident analysis assumptions and during shutdown conditions consistent with the ECCS design basis.

6.4 Habitability Systems

6.4.1 Introduction

The U.S. EPR main control room habitability systems (CRHS) are designed to protect control room operators from the effects of accidental releases of radioactive and toxic gases. Additionally, habitability systems provide the necessary support for Control Room Envelope (CRE) which includes the technical support center (TSC). The major review interfaces are the control room ventilation system and its associated filtration trains, and Emergency Planning for the TSC. The review of the control room ventilation system is performed under the CRACS and its subsystem CREF, in accordance with NUREG-0800, Section 9.4.1, "Control Room Area Ventilation System." The CREF (Iodine Filtration) Train Subsystem and iodine removal efficiencies of the CREF discussed in Section 6.5.1 of this report. Review of ITAAC requirements for emergency planning, including requirements for the TSC are in accordance with NUREG-0800, Section 14.3.10.

6.4.2 Summary of Application

FSAR Tier 1: FSAR Tier 1, Section 2.6, "HVAC Systems," contains Section 2.6.1, "Main Control Room Air Conditioning System." The CRACS supplies air to the CRE. The major equipment and the associated locations and design data for the CRACS are shown in FSAR Tier 1, Tables 2.6.1-1, "Main Control Room Air Conditioning System Equipment Mechanical Design," and 2.6.1-2 "Main Control Room Air Conditioning System Equipment I&C and Electrical." The CRACS ITAAC are shown in FSAR Tier 1, Revision 2, Table 2.6.1-3, "Main Control Room Air Conditioning System ITAAC."

FSAR Tier 2: The MCR habitability systems are designed to protect the plant operators in the CRE from the effects of accidental releases of radioactive and toxic gases. The CRE includes the control room and TSC and allows control room operators to remain in the CRE to operate the plant safely under normal conditions and to maintain the plant in a safe state under accident conditions.

ITAAC: The CRHS-related ITAAC are CRACS ITAAC, which are addressed in FSAR Tier 1, Revision 2, Section 2.6., FSAR Tier 1, Revision 2, Table 2.6.1-3, and emergency planning ITAAC as they relate to the TSC. FSAR Tier 2, Table 1.8-2, COL item 14.3-1 states that ITAAC for emergency planning will be provided by the COL applicant.

Technical Specifications: The Technical Specifications for CRHS are FSAR Tier 2, Revision 2, Chapter 16, Section 3.7.10, "Control Room Emergency Filtration (CREF)"; Section 3.7.11, "Control Room Air Conditioning System (CRACS)"; and Section 5.5.17, "Control Room Envelope Habitability Program."

6.4.3 Regulatory Basis

The relevant requirements of NRC regulations for this area of review, and the associated acceptance criteria, are given in NUREG-0800, Section 6.4, "Control Room Habitability System," and are summarized below. Review interfaces with other NUREG-0800 sections can also be found in NUREG-0800, Section 6.4.

1. GDC 4, as it relates to SSCs important to safety being designed to accommodate the effects of and to be compatible with the environmental conditions associated with postulated accidents.

2. 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 cool-down of the remaining unit(s).

3. GDC 19, "Control Room," as it relates to providing a control room actions can be taken to maintain the nuclear power unit in a safe condition under accident conditions and providing radiation protection adequate to permit access to and occupancy of the control room and accident condition.

4. 10 CFR 50.34(f)(2)(xxviii), “Contents of applications; technical information,” as it relates to evaluations of potential pathways for radioactivity and radiation and associated design provisions to preclude certain control room habitability problems.

5. 10 CFR 52.47(b)(1), “Contents of applications; technical information,” 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 NRC regulations.

Acceptance criteria adequate to meet the above requirements include:

1. RG 1.196, “Control Room Habitability at Light-Water Nuclear Power Reactors,” and RG 1.197, “Demonstrating Control Room Envelope Integrity at Nuclear Power Reactors,” provide acceptable guidance for meeting control room habitability requirements.

2. RG 1.183, “Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors.”

3. 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.”

4. ASME Code AG-1, “Code on Nuclear Air and Gas Treatment.”

5. RG 1.78, “Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release.”

6. TMI Action Plan Item III.D.3.4 (NUREG-0737), regarding protection against the effects of toxic substance releases, either onsite or offsite.

7. GSI, 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.”

8. GSI, Item B-66, “Control Room Infiltration Measurements.”

9. GSI-83, “Control Room Habitability (Revision 3)”

The NUREG-0800 acceptance criteria are also based on conformance to NUREG-0696, “Criteria for Emergency Operations Facilities for Nuclear Power Reactors,” for guidance in establishing the habitability criteria for the TSC portion of the CRE.

6.4.4 Technical Evaluation

Review of the CRHS in the FSAR was performed in accordance with SRP Section 6.4, Section III, “Review Procedures.” The CRHS is composed of the control room envelope along with the control room ventilation system, which is called the control room air conditioning system, with its subsystem control room emergency filtration system. The results and conclusions reached are as follows:

Control Room Envelope

The CRE was reviewed to determine if it included those facilities discussed in the acceptance criteria of the Control Room Emergency Zone in NUREG-0800, Section 6.4. The CRE includes the MCR, the TSC, an instrumentation and controls room, two computer rooms, a shift supervisor's office, an operator's washroom (sanitary facilities), a kitchen, and interconnecting corridors (refer to FSAR Tier 2, Section 6.4.2.1, "Definition of Control Room Envelope," FSAR Tier 2, Figure 6.4-1, "Control Room Envelope Plan View 1," and FSAR Tier 2, Figure 9.4.1-3, "Control Room Envelope Air Supply and Recirculation Subsystem"). As stated in FSAR Tier 2, Sections 6.4.4, "Design Evaluations," and 6.4.5, "Testing and inspection," the CRE conforms to the guidelines of RG 1.196, January 2007, "Control Room Habitability at Light Water Nuclear Power Reactors," and RG 1.197, May 2003, "Demonstrating Control Room Envelope Integrity at Nuclear Power Reactors."

NUREG-0800, Section 6.4, Acceptance Criteria 1 recommends the followng;

> The control room emergency zone should include the following:
> A. Instrumentation and controls necessary for a safe shutdown of the plant, i.e., the control room, including the critical document reference file;
> B. Computer room, if it is used as an integral part of the emergency response plan;
> C. Shift supervisor's office; and
> D. Operator washroom and the kitchen.
> E. The control room emergency zone should conform to the guidelines of Regulatory Guide 1.196, May 2003, "Control Room Habitability at Light Water Nuclear Power Reactors," and Regulatory Guide (RG) 1.197, May 2003, "Demonstrating Control Room Envelope Integrity at Nuclear Power Reactors."

As such the CRE meets NUREG-0800, Section 6.4, Acceptance Criteria 1A through 1E, and is therefore acceptable to the staff. Additionally, the control room envelope and the MCR ventilation system were reviewed for the acceptance criteria of NUREG-0800, Section 6.4, by reviewing the results of other SRP reviews. The CRACS/CREF was reviewed as a ventilation system using SRP Section 9.4.1, and the filters in CREF were reviewed using NUREG-0800, Section 6.5.1.

Ventilation System Criteria

Isolation Dampers

As discussed in Section 9.4.1 of this report, the U.S. EPR CRE isolation dampers are low leakage dampers that comply with the component design criteria for dampers contained in ASME-AG-1-2003, which the staff finds acceptable.

Single Failure Criterion

The staff reviewed the CRACS/CREF system for single failure vulnerability in conjunction with Section 9.4.1 of this report and finds it acceptable.

Based on the above, the staff finds that the U.S. EPR design meets SRP Section 6.4, Acceptance Criteria 2A and 2B for ventilation system criteria in NUREG-0800, Section 6.4, and is therefore acceptable.

Occupancy Limitations

The CRACS has an isolate mode that isolates the Division 2 and 3 air intakes and recirculates the CRE atmosphere. NUREG-0800, Section 6.4, Subsection III , states that, "air inside a 2830 m^3 (100,000 cubic foot) control room would support five persons for at least six days. Thus, CO_2 buildup in an isolated emergency zone is not normally considered a limiting problem." The CRE volume is 5,663 m^3 (200,000 ft^3), which is well above criteria in NUREG-0800, Section 6.4, Subsection III used to evaluate Control Room Personnel capacity and the potential for CO_2, buildup in the control room operators. Therefore, the staff concludes that the CRE design is sufficient to remove concern for the buildup of CO_2 for 10 persons for at least 6 days. Based on the above, the staff finds the U.S. EPR design meets the Acceptance Criterion 2 for ventilation system criteria in NUREG-0800, Section 6.4, and is therefore acceptable.

Type of Pressurization System and CRE Zone

The staff reviewed the U.S. EPR design against NUREG-0800, Acceptance Criterion 3, "Pressurization Systems," and related regulatory guidance.

The staff determined that the U.S. EPR CRACS is a ventilation system of the type described in NUREG-0800, Section 6.4, Subsection III, Paragraph 3A, subparagraph iii for radiological sources and a "Type iv" (NUREG-0800 Section 6.4, Subsection III.3.E.iv) system for toxic gas sources. In RAI 462, Question 06.04-7(a), the staff requested that the applicant describe if or how the standard design, as described in FSAR Tier 2, Section 6.4, complies with SRP Section 6.4 guidance for a "Type iii" (NUREG-0800 Section 6.4, Subsection III.3.E.iii) system. The staff also requested that the applicant add a paragraph to FSAR Tier 2, Section 6.4, that details this compliance. The CRACS design has positive pressure (supplied by 28.32 m^3/min. (1,000 cfm) of make up outside air), has recirculation air that gets filtered, and has dual outside air inlets. The ventilated release point contributor to the control room dose is from an elevated plant stack (release point). In addition, the CRACS has an isolation mode with outside air supply isolated and with filtered recirculation for toxic gas accidents in which the control room is isolated from outside air. RG 1.197 describes low-inleakage control rooms as having leakage less than 2.832 m^3/min (100 cfm). Since the CRE is designed with a DBA inleakage rate of 1.416 m^3/min (50 cfm), the staff finds that the CRE is a low leakage design. This functional design is the optimum of those presented in NUREG-0800, Section 6.4, and is subject to NUREG-0800, Section 6.4, Acceptance Criterion 3B. Revision 2 to FSAR Tier 2, Section 6.4 specifies manual alignment for toxic gas protection. RG 1.78 recommends automatic alignment for toxic gas protection. In RAI 462, Question 06.04-7(b), the staff requested that the applicant clarify the inconsistency between RG 1.78 guidance and the design of the CRE as described in FSAR Tier 2, Section 6.4. **RAI 462, Question 06.04-7 is being tracked as an open item.**

CRE Positive Pressure

When the CREF is in operation, the ventilation system maintains a positive pressure in the CRE of 3.17 mm (0.125 in.) water gauge within the CRE areas with respect to adjacent environmental zones to prevent uncontrolled, unfiltered inleakage during normal and accident conditions as stated in FSAR Tier 2, Section 6.4.2.2. During a radiological accident, exhaust from the kitchen and sanitary rooms is stopped and all other exhaust air is recirculated. During normal operation, air is exhausted from the sanitary rooms and the kitchen area. In RAI 89, Question 06.04-3, the staff requested that the applicant clarify the ability to maintain positive pressure of 3.17 mm (0.125 in.) of water gauge within the CRE areas during normal operating conditions. In a December 15, 2008, response to RAI 89, Question 06.04-3, the applicant proposed to change FSAR Tier 2, Section 9.4.1.2.3, to clarify the positive pressure CRE during normal operation. The change states that CRACS will continue to maintain a minimum positive pressure of 3.17 mm (0.125 in.) of water gauge in the CRE by way of a pressure control damper that will control the amount of outside air that enters the control room. The applicant states that during normal operation, air is exhausted from the sanitary rooms and the kitchen area through a small throttle damper that minimizes the open CRE boundary area. The staff notes that FSAR Tier 1, Revision 2, ITAAC 6.1 shows a new, separate CRE overpressure acceptance criteria for normal lineup. The new AC for CRE overpressure is significantly less than that specified in FSAR Tier 2, Section 6.4, as the normal overpressure design commitment. This design change varies from the response received in the December 15, 2008, response to RAI 89, Question 06.04-3. In follow-up RAI 462, Question 06.04-6, the staff requested that the applicant clarify this change with respect to the existing FSAR Tier 2 CRACS design commitment described in FSAR Tier 2, Sections 6.4.2.2 and 6.4.2.3, and to clarify how the test to verify this very low positive pressure would be performed such that a determination can be made of the adequacy of the design. **RAI 462, Question 06.04-6 is being tracked as an open item.**

Since TS Surveillance Requirement 3.7.10.1 specifies an operational test of each CREF train every 31 days, and TS Surveillance Requirement 3.7.10.4 requires performance of the CRE unfiltered air inleakage testing, which will also test CRE positive pressure, and reflects Technical Specifications Task Force (TSTF)-448, the staff finds that verification of CRE positive pressure meets the guidance of NUREG-0800, Section 6.4 Acceptance Criterion 3B, and with the exception of the noted open item is therefore acceptable.

Atmosphere Filtration Systems

The staff reviewed the U.S. EPR design against NUREG-0800. Acceptance Criterion 4, “Emergency Standby Atmosphere Filtration Systems.”

The atmospheric filtrations systems are described in other sections of the Tier 2 FSAR and are also reviewed under other sections of the NUREG-0800. Specifically, the CREF (Iodine Filtration) Train subsystem is described in FSAR Tier 2, Section 6.5.1, “ESF Filter Systems,” is reviewed against NUREG-0800, Section 6.5.1, and the CREF subsystem of control room ventilation is described in FSAR Tier 2, Section 9.4.1 and reviewed against NUREG-0800. The staff reviewed these Tier 2 FSAR sections and considered the NRC staff safety evaluations contained in Sections 6.5.1 and 9.4.1 of this report. The staff determined that no additional information beyond the documentation contained in Sections 6.5.1 and 9.4.1 of this report is needed. As discussed in these Sections 6.5.1 and 9.4.1 of this report, the staff finds that the CREF conforms to the guidelines of RG 1.52, ASME Code AG-1, NUREG-0800, Section 6.4, Acceptance Criterion 4 for ventilation system criteria, and is therefore acceptable.

Relative Location of Source and Control Room

Radiation Sources

The staff reviewed the U.S. EPR design against NUREG-0800, Section 6.4, Acceptance Criterion 5, “Relative Location of Source and Control Room, Radiation Sources,” as discussed below. In addition, review of the design against TMI Action Plan, Item III.D.3.4 and 10 CFR 50.34(f)(2)(xxviii) are discussed in this section.

NUREG-0800, Section 6.4, Acceptance Criterion 5A, “Radiation Sources,” states that the control room ventilation inlets should be separated from the major potential release points by at least 31 meters (100 ft) laterally and by 16 meters (50 ft) vertically, or be based on dose analyses. As it applies to radiation sources, 10 CFR 50.34(f)(2)(xxviii) requires that the applicant 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. This requirement is identified as TMI Action Plan Item III.D.3.4.

If high radioactivity is detected in the CRE outside air supply duct, the CRE normal air supply is automatically isolated, and the GDC 19 habitability requirements are met by a CREF train. The CREF (iodine filtration) train subsystem provides emergency ventilation and pressurization for the CRHA. The CRACS has a pressurization mode with its two divisional air intakes on the roofs of Safeguard Buildings Division 2 and 3.

The staff reviewed the two CRACS divisional air intake locations relative to radioactivity release points in an accident. The intakes are physically separated and reasonably located away from the elevated vent stack and the Main Steam Relief (MSR) exhausts. The radiological dispersion factors (χ/Q, FSAR Tier 2, Table 15.0-51, “MCR Composite χ/Q and Filter-Bypass Fractions LOCA Releases at the Main Stack Base”) will always be greater for the Division 3 air intake for radiological accidents dominated by a release from the vent stack due to its shorter distance from the stack to the intake.

The analysis of the radiological consequences of design-basis accidents in the control room is discussed in FSAR Tier 2, Section 15.0.3. FSAR Tier 2, Section 15.0.3.4.1 discusses the details of the analysis assumptions on control room envelope unfiltered inleakage and ventilation system operation. FSAR Tier 2, Table 15.0-18, “Summary of MCR/TSC Characteristics,” gives a summary of the main control room characteristics used in the DBA control room dose analyses. The impact on control room habitability is evaluated for each of the design-basis accidents analyzed in FSAR Tier 2, Section 15.0.3. The limiting DBA was determined to be the main steam line break case with assumed fuel rod clad failure or fuel melting, with a resulting control room TEDE of 0.045 Sv (4.5 rem). The shielding and dose calculations in FSAR Tier 2, Section 12.2, “Radiation Sources,” and the DBA control room dose calculations in FSAR Tier 2, Section 15.0.3 include the contribution from direct radiation shine from the radioactive material in the main control room charcoal filtration system. The staff reviewed the Division 2 and 3 air intakes and the assumed unfiltered inleakage intakes relative to the distance to the stack, the MSR exhausts, and other DBA release locations. The control room dose analyses in FSAR Tier 2, Section 15.0.3 used short-term atmospheric dispersion factors for the control room that appropriately accounted for the relative locations of DBA releases to the control room intake and assumed inleakage location. More detail on the staff’s review of the control room atmospheric dispersion factors can be found in Section 2.3.4 of this report. The staff finds that the U.S. EPR design conforms to RG 1.183 for control room habitability analyses because the release and control room receptor locations for the U.S. EPR plant have been specifically modeled in the analyses. More detail on the staff’s review of the control room DBA radiological consequences analyses is given in Section 15.0.3 of this report.

The staff has noted that COL Information Item 6.4-4 requires a COL applicant that references the U.S. EPR design certification to confirm that the radiation exposure of main control room occupants resulting from a design basis accident at a nearby unit on a muli-unit site is bounded by the radiation exposure from the postulated design basis accidents analyzed for the U.S. EPR or confirm that the limits of GDC-19 are otherwise met.

Based on the above, the staff finds that the U.S. EPR design is satisfactory and complies with 10 CFR 50.34(f)(2)(xxviii) as it applies to radiation sources, and NUREG-0800, Section 6.4, Criterion 5A is met for the U.S. EPR design.

Toxic Gases

The staff reviewed the U.S. EPR design against NUREG-0800, Section 6.4, Acceptance Criterion 5, “Relative Location of Source and Control Room- Toxic Gases,” and Acceptance Criterion, “Toxic Gas Hazards,” as discussed below.

The staff reviewed the U.S. EPR design against NUREG-0800, Section 6.4, Acceptance Criterion 5B, “Toxic Gases.” This criterion states that the minimum distance between the toxic gas source and the control room will depend on the amount and the type of the gas in question, and other site-specific parameters. NUREG-0800, Section 6.4, Acceptance Criterion 5B is met by demonstrating conformance with the guidance of RG 1.78, as described below.

As it relates to toxic gases, TMI Action Plan, Item III.D.3.4 recommends that control room operators be adequately protected against the effects of the accidental release of toxic and radioactive gases such that the nuclear power plant can be safely operated or shut down under DBA conditions.

The CRACS has an isolation mode for dealing with toxic chemical accidents or smoke, which generally cannot be filtered by the intake filters in the pressurization mode. If toxic gas is detected by a toxic gas sensor in the CRE outside air supply duct, the CREF (iodine filtration) units on both intakes are automatically placed in the filtered alignment. The outside air intake (at the inlet location where the toxic gas is detected) will be closed by the control room operator. The GDC 19 habitability requirements are met by a CREF train. The CREF (iodine filtration) train subsystem provides emergency ventilation and pressurization for the CRHA.

The potential sources of toxic or otherwise potentially hazardous gases from nearby industrial, transportation, and military facilities are addressed in FSAR Tier 2, Sections 2.2, “Nearby Industrial, Transportation, and Military Facilities,” and 2.3, “Meteorology.” The capability for the CRACS to go into the isolation or recirculation mode (no inlet air) is provided in the design of CRACS and is the applicable mode for hazardous chemical accidents and toxic gas releases. The staff notes that the applicant commits to the provisions for donning self-contained breathing apparatus (SCBA) within a short period of time.

FSAR Tier 2, Section 6.4 originally referred to control room design for protection from hazardous chemicals and toxic gas sensors as conceptual design information to be addressed by the COL FSAR for these site-specific potential hazards. The staff, however, determined that the applicant’s use of conceptual design information for the control room protection does not meet GDC 19 and NUREG-0800, Sections 6.4 and 9.4.1. In RAI 89, Question 06.04-4, the staff requested that the applicant identify the necessary infromation and provide specific COL information items. In an October 31, 2008, response to RAI 89, Question 06.04-4, the applicant removed the designation of toxic gas evaluations as conceptual design basis information and designated toxic gas evaluations as a specific COL information item. The applicant provided changed pages in FSAR Tier 2, Sections 1.8, 6.4, 9.4.1, 14.2 and Chapter 16. COL Item 6.4-1 was revised to state:

> A COL applicant that references the U.S. EPR design certification will identify the type(s) of Seismic Category I Class IE toxic gas sensors (i.e., the toxic chemical(s) of concern) necessary for control room operator protection.

The staff reviewed the October 31, 2008, response to RAI 89, Question 06.04-4, and determined that more information was necessary in order to make a regulatory finding.
In follow-up RAI 462, Question 06.04-7, the staff requested that the applicant clarify if or how

the standard design, as described in FSAR Tier 2, Section 6.4, conforms to the guidance in RG 6.4, clarify the inconsistency between the guidance in RG 1.78 and the design of the CRE as described in FSAR Tier 2, Section 6.4, and clarify the changes made to the FSAR Tier 2, Sections 9.5.1.2.1, 14.2, and FSAR Tier 2, Chapter 16. **RAI 462, Question 06.04-7 is being tracked as an open item.**

COL item 6.4-3 directs the applicant to evaluate the results of toxic chemical accidents from Section 2.2.3 of the site-specific FSAR. With the exception of the open items noted above, the staff concludes that the applicant has met the TMI Action Plan, Item III.D.3.4 guidance as it applies to toxic gases and NUREG-0800, Section 6.4, Acceptance Criteria 5B and 7 are met for the U.S. EPR design in view of COL Item 6.4-3 of the FSAR Tier 2, Table 1.8-2. COL Item 6.4-3 requires that the COL applicant identify potential site-specific toxic or hazardous materials that may affect control room habitability and address the recommendations of TMI Action Plan, Item III.D.3.4. In addition, COL item 6.4-1 requires that a COL applicant identify the types of safety-related sensors that are necessary for MCR operator protection, and thus provides assurance that SSCs necessary to detect and alert operators of a toxic gas event will be incorporated into the site-specific design.

Radiation Hazards

The staff reviewed the U.S. EPR design against NUREG-0800, Section 6.4, Acceptance Criterion 6, "Radiation Hazards." NUREG-0800, Section 6.4, Acceptance Criterion 6B states that applicants for design certifications shall meet the requirements of GDC 19. Compliance with GDC-19 is discussed below. Radiological protection of the TSC is also addressed in this section. Control room operators as well as TSC occupants are protected from radiation sources by a combination of shielding and distance.

GDC 19

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

Radiation Shielding

The MCR and TSC are protected from radioactivity inside containment by over 1.83 m (6 ft) of concrete. The MCR and TSC are protected from the shine due to the radioactive cloud that could pass over the Safeguard Building Division 2 and 3. The roof slab above the CRE is over 0.61 m (2 ft) thick; additionally, the floor between the CRACS ventilation filters and the is MCR/TSC is 0.51 m (20 in.) thick as specified in FSAR Tier 2, Section 14.0.3.4.1. Likewise, the double walled external design of the Safeguard Building provides radiation shielding. The floor below the CRE contains batteries and switchgear, so the CRE has more than two floors of concrete between it and mechanical equipment that may contain radioactivity (e.g., safety injection pump rooms). Accordingly, this design meets the radiation shielding concerns expressed in NUREG-0800, Section 6.4.

The CRE is well shielded with its enclosure inside the SB. 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 CREF supply air

2. the unfiltered inleakage from all other sources

CRE Unfiltered Inleakage

The CRHS (CRE/CRACS/CREF) have a design value for CRE unfiltered inleakage in the pressurization mode of 1.42 m^3/m (50 cfm) according to FSAR Tier 2, Section 15.0.3.4.1. Unfiltered inleakage addresses the concern that the 3.17 mm (0.125 in.) of water gauge CRE positive pressure alone will not assure limiting CRE inleakage. The staff notes that this design value is a small inleakage value and may be less than the accuracy of measurement for CRE Habitability Testing (tracer gas testing). However, the measurement uncertainties need not be added to measured inleakage values when they are below 2.83 m^3/m (100 cfm) as stated in RG 1.197, Section 1.4, Test Results and Uncertainty, as follows:

> Test uncertainty may be an issue when reporting inleakage test results for pressurized CREs with low inleakage. For such CREs, inleakage is usually determined by calculating the total inleakage and subtracting from the total the makeup flow to the CRE (typically the flow rate through the CRE's emergency filtration unit). When the difference between these two numbers is small (usually an indication that the CRE's inleakage is small), the uncertainty may approach the nominal value for inleakage. Adding the uncertainty to the nominal results of the test for inleakage could result in some facilities that do not meet GDC-19 because of the added uncertainty in the analyses.
>
> If the CRE has been demonstrated to have low inleakage, the uncertainty may be an artifact of the calculations and not representative of the CRE's integrity. There may be a limited number of actions that may be taken to improve the leakage characteristics of low-inleakage CREs. Therefore, the staff has concluded that it is optional to include the uncertainty for facilities that demonstrate a CRE inleakage less than 100 cfm. The option will allow the significance of the uncertainty to be addressed on a plant-specific basis.

The staff finds acceptable the 1.42 m^3/m (50 cfm) unfiltered CRE inleakage assumption, which includes 0.283 m^3/m (10 cfm) for ingress/egress, used in the radiological and toxic gas evaluations in FSAR Tier 2, Chapter 15.

The question of unfiltered inleakage entering the CRACS via the space between the carbon filter and the CREF fan was reviewed against to NUREG-0800, Section 6.4, Subsection III, Paragraph 3.C. The physical arrangement of the duct plenum configuration to limit the potential for inleakage through the surface area between the outlet of the carbon filter and the inlet to the CREF fan is important, but depends on detailed CREF duct layout which is not provided. However, the unfiltered control room inleakage that is allowed is specified and controlled by testing. Periodic testing of CRE unfiltered inleakage and CRE positive pressure will be in accordance with the CRE habitability program, which references RG 1.197 and the Ventilation Filter Testing Program, which references RG 1.52 as to test frequency. In FSAR Tier 2, Section 6.4.2.3, "Leak-tightness", the applicant states that the "CRE boundary limits leakage from adjacent environmental zones to a maximum of 50 cfm unfiltered-inleakage." The staff issued RAI 462, Question 06.04-8, to request that the applicant modify the FSAR to clearly state either the CRE design limit for boundary inleakage, or the CRE unfiltered inleakage ASTM E-741 test acceptance criteria. **RAI 462, Question 06.04-8 is being tracked as an open item.**

GSI 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 Chapter 16, TS Section 5.5.17 on the Control Room Envelope Habitability Program. Therefore, the staff considers GSI Item B-66 satisfied.

With the exception of the open item identified above, the staff finds that NUREG-0800, Section 6.4, Acceptance Criterion 6 is met as it pertains to shielding for the U.S. EPR because the applicant has demonstrated protection from radiation hazards via distance and shielding using the above assumptions as discussed below.

Input Parameters to the Radiological Dose Analysis

The values presented in FSAR Tier 2, Section 15.0.3.4.1, “Main Control Room/Technical Support Center Modeling,” and shown in FSAR Tier 2, Figure 15.0-4, “Nuclear Steam Supply System Power Levels Assumed in the Accident Analysis,” were reviewed by the staff. The staff compared the input to the Chapter 15 dose analysis to the characteristics of the design and finds that the values of CRE volume, carbon filter efficiency, HEPA filter efficiency, outside recirculation, and total air flow, roof slab thickness, and the MCR charcoal filter shine to the CRE below were satisfactorily modeled in the dose analysis. The CRE post accident filtration model properly considers the outside intake, recirculation, total and unfiltered inleakage, and total air flow in cfm for the purposes of the MCR dose analysis.

Iodine Protection Factor and Finite Cloud Correction

FSAR Tier 2, Section 15.0.3.4.1, “Finite-Cloud Correction,” states that the Murphy-Campe model (taken from K. G. Murphy and K. W. Campe, “Nuclear Power Plant Control Room Ventilation System Design for Meeting General Design Criterion 19,” Proceedings of 13th AEC Air Cleaning Conference, Atomic Energy Commission, August 1974) was used for the LOCA control room analysis. Although FSAR Tier 2, Section 15.0.3.4.1 discusses the correction for the difference between the finite cloud geometry in the control room and the semi-infinite cloud assumption, which is the basis for the deep dose equivalent (DDE) dose conversion factors used by the applicant, it was not clear if the analyses also used the iodine protection factor methodology given in the Murphy-Campe paper. As stated in RG 1.183, Paragraph 4.2.3, the Murphy-Campe iodine protection factor (IPF) methodology may not be adequately conservative. FSAR Tier 2, Section 15.0.3 references the RADTRAD computer code. RG 1.183 states that the RADTRAD computer code incorporates suitable methodologies to provide conservative estimates of exposure to control room personnel. RADTRAD is referenced in FSAR Tier 2, Section 15.0.3.11.2 and FSAR Tier 2, Tables 15.0-50 and 52, but only with regard to modeling natural deposition in the primary containment leakage pathway. There was insufficient information to determine if RADTRAD or another transport model was used, or if the Murphy-Campe IPF methodology was used with proper rigor as specified in NUREG-0800, Section 6.4, Review Procedure, Paragraph 3.D, and RG 1.183, Paragraph 4.2.3. Therefore, in RAI 89, Question 06.01-1, the staff requested that the applicant describe the use of the Murphy-Campe model in the LOCA control room dose analysis. In an October 2, 2008, response to RAI 89, Question 06.04-1, the applicant stated the LOCA control room dose calculation was performed using the Polestar Applied Technology, Inc., proprietary STARDOSE computer code. The applicant stated that STARDOSE and RADTRAD are similar with respect to the control room modeling and comparative calculations were done with RADTRAD to confirm the STARDOSE control room dose results. The applicant stated that although both STARDOSE and RADTRAD employ the Murphy-Campe finite-cloud correction endorsed by RG 1.183, Paragraph 4.2.7, neither code makes use of the Murphy-Campe IPF methodology discouraged by RG 1.183, Paragraph 4.2.3. A rigorous alternative methodology is employed by both STARDOSE and RADTRAD to integrate the rate of change of activity in the control room over time. Since the applicant does not use the Murphy-Campe IPF methodology, the staff finds this response satisfactory and agrees with the applicant that no further detail is needed for the FSAR. Therefore, the staff considers RAI 89, Question 06.04-1 resolved.

Radiation Protection

As stated in FSAR Tier 2, Section 15.0.3, Table 15.0-53, “Radiological Consequences of U.S. EPR Design Basis Accidents (rem TEDE),”the CRHS limits the control room dose to 0.04 Sv (4.0 rem) TEDE in the loss-of-coolant accident, which is less than the 0.05 Sv (5 rem) TEDE limit specified in the NUREG-0800 Acceptance Criteria. The applicant also calculated projected control room doses for each of the hypothetical DBAs analyzed in FSAR Tier 2, Chapter 15.0.3. FSAR, Tier 2, Table 15.0-12 gives the calculated control room doses for each of the DBAs. The bounding DBA for dose in the control room is the main steam line break with either fuel rod cladding failure or fuel overheat, which results in a TEDE of 0.045 Sv (4.5 rem). The projected dose for each DBA is less than the GDC 19 criterion of 0.05 Sv (5 rem) TEDE for main control room dose. This adequately protects control room envelope occupants during postulated accident conditions, and therefore acceptable to the staff. Because the U.S. EPR design includes the TSC entirely within the CRE, the radiological consequences of DBAs in the TSC is also shown by the above to meet the TSC dose acceptance criterion of 0.005 Sv (5 rem) TEDE. The staff’s review of the DBA radiological consequences analyses, including the control room and technical support center radiological habitability, is discussed in Section 15.0.3 of this report.

Therefore, the staff finds that GDC-19 radiological requirements are met as they apply to the CRE, which includes the MCR and the TSC.

GDC 4 - Dynamic Effects

GDC 4 requires that structures, systems, and components important to safety are designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant accidents. The staff evaluated whether the CRACS meets the requirements of GDC 4.

Confined Area Releases

The location of the Division 2 and 3 CRACS air intakes is elevated and removed from pressure containing tanks and equipment. A rupture of a main steam line in the vicinity of the Safeguard Buildings Division 2 and 3 would not severely impact the CRE, which is contained inside massive concrete structures surrounding the MCR and TSC.

FSAR Tier 2, Section 6.4 states that the control room envelope along with the MCR air conditioning system (and its CREF subsystem) is designed to protect the occupants of the envelope from the effects of postulated accidents. CRHS components are contained within the SB, a seismically qualified Category Class I building. As described in FSAR Tier 2, Table 3.2.2-1, the control room habitability systems are designed as Seismic Category I, as discussed in FSAR Tier 1, Section 2.6.1, and are designed to be compatible with the environmental conditions associated with postulated accidents. As discussed in Section 9.4.1 of this report, the staff finds that the CRE and CRACS meet the requirements of GDC 4.

GDC 5 – Shared Systems and Components important to Safety

The staff reviewed the design of the CRHS to ensure that the relevant requirements of GDC 5 are met.

GDC 5 governs the sharing of structures, systems and components important to safety among nuclear power plant units in order to ensure such sharing will not significantly impair their ability to perform their safety functions. The U.S. EPR design is a single-unit station, and the requirements of GDC 5 are not applicable to the single-unit design.

The staff also reviewed the U.S. EPR design based on the regulatory guidance listed below as related to MCR ESF filter system design, testing and maintenance.

GSI Task Action Plan Item B-36 called for 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. As discussed in Sections 6.5 and 9.4.1 of this report, the applicant addresses GSI Item B-36 by conforming to the guidance in RG 1.52, for the safety-related CREF system design, testing and maintenance; therefore, the staff finds this acceptable.

GSI-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). GSI-83 recommends increased training of NRC and licensee personnel in inspection and testing of control room habitability systems.

Generic Letter (GL) 2003-01 reemphasized this concern. The applicant has included COL Information Item 6.4-2, which requires that the COL applicant address procedures and training for control room habitability. The CRE Habitability Program in FSAR Tier 2, Chapter 16, Section 5.5.17 requires the program to include elements to establish the CRE boundary test method and frequency. Accordingly, the staff finds that GSI-83 has been adequately addressed.

The staff reviewed the CRE TSC in accordance with NUREG-0696 guidance. FSAR Tier 2, Chapter 13.3, "Emergency Planning," states that space suitable for a TSC, which demonstrates compliance with the design provisions for staffing levels consistent with NUREG-0654, Revision 1 of NUREG-0654/FEMA-REP-1, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants (Criteria for Emergency Planning in an Early Site Permit Application)," is provided in the U.S. EPR standard design, and is located in the integrated operations area adjacent to the MCR. NUREG-0654, Criterion H.1 states that, "each licensee shall establish a TSC in accordance with NUREG-0696, Revision 1." Therefore, the staff finds that the U.S. EPR proposed TSC location in the CRE will comply with NUREG-0696.

The staff reviewed FSAR Tier 2, Sections 9.4.1 and 6.4.1 in order to determine if the proposed CRE TSC location meets the guidance of NUREG-0696 as they apply to TSC habitability and ventilation. These are NUREG-0696, Sections 2.4, "Size," 2.5, "Structure," and 2.6, "Habitabiltiy."

TSC Size

NUREG-0696, Section 2.4 provides guidance on TSC size and, in part, states that “the TSC working space shall be sized for a minimum of 25 persons, including 20 persons designated by the licensee and five NRC personnel. This minimum size shall be increased if the maximum staffing level specified by the licensee’s emergency plan exceeds 20 persons.” In RAI 24, Question 13-03-03, the staff requested that the applicant provide details on how the proposed TSC meets this guidance. In a June 27, 2008, response to RAI 24, Question 13.03-03, the applicant stated that the TSC area that is within integrated operations area of at least 174.2 m^2 (1875 ft^2) is allocated to the TSC, and thus, the TSC is large enough to provide space for 25 personnel at 7 m^2 (75ft^2) per person.

The staff reviewed the response and noted that the information provided in the response is design basis information, necessary to evaluate the proposed TSC in the standard design, and needs to be added to the FSAR. In RAI 462, Question 06.04-5, the staff requested that the applicant include the information in the response to RAI 24, Question 13.03-3 be added to the FSAR Tier 2, Section 6.4, along with a more explicit statement of compliance with NUREG-0696 as it applies to TSC habitability and TSC size. **RAI 462, Question 06.04-5 is being tracked as an open item.**

TSC Structure

NUREG-0696, Section 2.5 provides guidance on the TSC structure that states that the TSC complex must be able to withstand the most adverse conditions reasonably expected during the design life of the plant. Since the proposed TSC shares the same building as the control room, it is protected from earthquakes, high winds and floods in the same manner as the CRE, and is protected from dynamic effects in the same manner. Therefore the staff finds that the design of the TSC conforms to the guidance of NUREG-0696, Section 2.5 and NUREG-0800, Sections 6.4 and 9.4.1 and, therefore, complies with the requirements of GDC 2 and GDC 4.

TSC Habitability

NUREG-0696, Section 2.6 provides guidance on TSC habitability, stating that the TSC shall have the same radiological habitability as the control room under accident conditions and that TSC personnel shall be protected from radiological hazards, including direct radiation and airborne radioactivity from inplant sources under accident conditions, to the same degree as control room personnel. NUREG-0696, Section 2.6 also states that applicable criteria are specified in GDC 19 and NUREG-0800, Section 6.4

Regarding the TSC ventilation system, NUREG-0696, Section 2.6 guidance states that the TSC ventilation system shall function in a manner comparable to the control room ventilation system and that a TSC ventilation system that includes HEPA and charcoal filters is needed, as a minimum.

Since the proposed TSC is incorporated into the CRE, it shares the same ventilation systems and is subject to the same radiological protection as the main control room, thus the above conclusions with respect to the radiological consequences in the control room also apply to the TSC, and the TSC design complies with NUREG-0800, Section 6.4 and satisfies the dose criterion in GDC 19 as they apply to the proposed TSC location. Therefore, the TSC meets the guidance of NUREG-0696 section 2.6.

Based on the above discussions, the staff concludes that the TSC area of the U.S. EPR CRE complies with the requirements GDC 2, GDC 4, and GDC 19. Additionally, the U.S. EPR design conforms to the guidelines of NUREG-0696, Sections 2.4, 2.5, and 2.6 as they apply to TSC ventilation and habitability

ITAAC

Proposed ITAAC for the CRACS are given in FSAR Tier 1, Table 2.6.1-3. The control room habitability and CRACS related ITAAC are addressed in FSAR Tier 1, Section 2.6.1 and FSAR Tier 1, Table 2.6.1-3. The 3.17 mm (0.125 in.) of water positive pressure and the 1.42 m^3/m (50 cfm) unfiltered inleakage are verified in the tests specified. Additionally, CRACS ventilation testing review is addressed in the staff's safety evaluation related to FSAR Tier 2, Section 9.4.1.

By way of background, the staff noted that FSAR Tier 2, Section 6.4.5 states that air quality testing is performed in accordance to ASHRAE 62, which is not an air quality testing procedure. Air quality testing demonstrates that design air flows and carbon and HEPA filter performance criteria are met on a periodic basis. In RAI 89, Question 06.04-02, the staff requested that the applicant provide a commitment to an ASHRAE air quality testing standard. In a December 15, 2008, response to RAI 89, Question 06.04-2, the applicant proposed changes to FSAR Tier 2, Sections 6.4.5 and 6.4.7 to remove reference to ASHRAE 62 and to commit to air quality testing in accordance with ASHRAE 52.2-1999, "Methods for Testing General Air Cleaning Devices for Removal Efficiency by Particulate Size," and ASME N510-1989 (R1995), "Testing of Nuclear Air-Treatment Systems." Both of these standards have undergone a recent revision as follows: ASHRAE 52.2-2007 and ASME N510-2007. The applicant is consistent with referencing these earlier revisions in FSAR Tier 2, Section 9.4, "Air Conditioning, Heating, Cooling and Ventilation Systems." The staff finds this commitment and the proposed changes to FSAR Tier 2, Sections 6.4.5 and 6.4.7 acceptable. The staff confirmed that FSAR Tier 2, Revision 2, Sections 6.4.5 and 6.4.7 incorporated the proposed revision and, therefore, considers RAI 89, Question 06.04-02 resolved.

With regard to the TSC, the applicant has included COL Item 14.3-1 which requires a COL applicant that references the U.S. EPR design certification to address ITAAC for emergency planning, physical security, and site-specific portions of the facility that are not included in the FSAR Tier 1 ITAAC associated with the certified design (10 CFR 52.80(a)). NRC guidance for emergency planning ITAAC includes guidance for ITAAC that verify compliance with NUREG-0696 with regard to TSC size, location relative to the control room, TSC habitability, and equipment. Compliance with guidance of the remaining areas of NUREG-0696 is discussed Section 13.3 of this report.

Based on the above, the staff finds that these specified ITAAC are sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the U.S. EPR has been constructed in accordance with the design certification as required by 10 CFR 52.47(b)(1).

Technical Specifications

Technical Specifications in FSAR Tier 2, Chapter 16, Sections 3.7.10 and 3.7.11 address CREF and CRACS requirements during all operational modes including during movement of irradiated fuel assemblies. FSAR Tier 2, Chapter 16, provides for formal periodic surveillance testing of the control room emergency filtration system (FSAR Tier 2, Chapter 16, Section 3.7.10) and MCR air conditioning system (FSAR Tier 2, Chapter 16, Section 3.7.11). FSAR Tier 2, Chapter 16, Section 5.5.17 requires an Administrative Control Program be established for a Control Room Envelope Habitability Program. This is implemented to ensure that CRE habitability is maintained. As described in Chapter 16 of this report, the proposed Technical Specifications for the U.S. EPR conform to NUREG-0800, Section 16 and NUREG-1431, "Standard Technical Specifications (STS) for Westinghouse Plants," and are therefore acceptable.

Initial Plant Test Program

Initial plant testing requirements given for the CRACS in FSAR Tier 2, Section 14.2, "Initial Plant Test Program," are Main Control Room Air Conditioning System (Test #082). No additional preoperational testing is necessary. The staff reviewed these tests and finds them to be an acceptable means to verify the system will perform as stated in FSAR Tier 2, Section 9.4.1.

6.4.5 Combined License Information Items

Table 6.4-1 provides a list of habitability systems related COL information item numbers and descriptions from FSAR Tier 2, Table 1.8-2:

Table 6.4-1 U.S. EPR Combined License Information Items

Item No.	Description	FSAR Tier 2 Section
6.4-1	A COL applicant that references the U.S. EPR design certification will identify the type(s) of Seismic Category I Class IE toxic gas sensors (i.e., the toxic chemical(s) of concern) necessary for control room operator protection.	6.4.2.4
6.4-2	A COL applicant that references the U.S. EPR design certification will provide written emergency planning and procedures in the event of a radiological or a hazardous chemical release within or near the plant, and will provide training of control room personnel.	6.4.3.2
6.4-3	A COL applicant that references the U.S. EPR design certification will evaluate the results of the toxic chemical accidents from Section 2.2.3 and address their impact on control room habitability in accordance with RG 1.78.	6.4.4
6.4-4	A COL applicant that references the U.S. EPR design certification will confirm that the radiation exposure of main control room occupants resulting from a design basis accident at a nearby unit on a multi-unit site is bounded by the radiation exposure from the postulated design basis accidents analyzed for the U.S. EPR; or confirm that the limits of GDC 19 are met.	6.4.4
14.3-1	A COL applicant that references the U.S. EPR design certification will provide ITAAC for emergency planning, physical security, and site specific portions of the facility that are not included in the Tier 1 ITAAC associated with the certified design (10 CFR 52.80(a))	13.3

The staff determined the above listing to be complete. Also, the list adequately describes actions necessary for the COL applicant or holder. No additional COL information items need to be included in FSAR Tier 2, Table 1.8-2 for control room habitability considerations.

6.4.6 Conclusions

The staff's review was based on NUREG-0800, Section 6.4, and it addresses plant-specific control room habitability data as discussed in FSAR Tier 2, Section 6.4 and the control room air conditioning system as discussed in FSAR Tier 1, Section 2.6.1.

Except for the open and confirmatory items identified above, the staff concludes that the U.S. EPR CHRS complies with GDC 4, GDC 19, and 10 CFR 50.34(f)(2)(xxviii). Because the U.S. EPR design is a single unit, GDC 5 is not applicable. Conformance to the guidelines of RG 1.78 is addressed by a COL information item. The ITAAC and Technical Specification requirements will ensure that the CRHS, including the CRACS, and the CREF can be properly inspected, tested, and operated in accordance with FSAR requirements, and therefore the staff finds that the U.S. EPR CRHS design complies with 10 CFR 52.47(b)(1). The above conclusions also apply to the proposed TSC location as they apply to TSC radiological habitability.

6.5 Fission Product Removal and Control Systems

The U.S. EPR fission product removal control systems (FPCS) are designed to prevent or limit the release of fission products following a postulated DBA or fuel handling accident. These systems include the ESF filter systems, primary and secondary containment structures and systems, and containment isolation system.

6.5.1 ESF Filter Systems

The ESF filter systems consist of filter assemblies, heaters, fans, dampers, and ductwork. They remove particulate and gaseous radioactive material from the atmosphere. Four ESF filter systems work in conjunction with the five ventilation systems given below:

- Main control room air conditioning system
- Annulus ventilation system
- Safeguard building controlled area ventilation system (SBVS)
- Fuel building (FB) ventilation system (FBVS)
- Containment building ventilation system (CBVS) for the low-flow purge exhaust subsystem

6.5.1.1 *Introduction*

The function of the U.S. EPR ESF filter systems (referred to as ESF Atmospheric Cleanup Systems in NUREG-0800) is to mitigate the consequences of postulated accidents by removing released radioactive material from the ventilated spaces serviced by the systems.

These systems include primary systems (e.g., in-containment recirculation) and secondary systems (e.g., annulus and emergency or post-accident air-cleaning systems) for the Fuel Handling Building, control room, Shield Building, and areas containing ESF components. There are provisions to preclude temperatures in the ESF filter adsorber section from exceeding design limits. Ventilation systems are aligned to ESF filter systems to support plant operations and accident mitigation. ESF filters in the CBVS low-flow purge exhaust subsystem are aligned during purging operations. The FBVS aligns to the SBVS ESF filters in response to high radiation in the RB or a containment isolation signal. Portions of the FBVS and CBVS are aligned to the ESF filters of the SBVS during the movement of irradiated fuel assemblies.

6.5.1.2 *Summary of Application*

FSAR Tier 1: There are no FSAR Tier 1 entries specifically for the ESF filter systems. The ESF filter systems are discussed in FSAR Tier 1, Section 2.6, "HVAC Systems." The sections for the ventilation systems that have ESF filtration capability provide the descriptions and piping and instrumentation diagrams of these ventilation systems, along with design bases and safety evaluations.

FSAR Tier 2: The applicant has provided a system description of the ESF filter systems in FSAR Tier 2, Section 6.5.1, "ESF Filter Systems," summarized here, in part, as follows:

FSAR Tier 2, Section 6.5.1.2, "System Design," states that the ESF filter systems are designed to limit the release of fission products to the environment and to limit radiation dose to the personnel in the MCR. The applicant indicates that the EFS filter systems meet 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," design and performance recommendations with provisions to filter air, remove moisture, and utilize charcoal adsorption to remove iodine. The ESF filter systems remove particulate and gaseous radioactive material from the atmosphere that could be released to the environment (GDC 41). The ESF filter systems are designed to permit periodic inspection and periodic pressure and functional testing (GDC 42, "Inspection of Containment Atmosphere Cleanup Systems," GDC 43). These systems, although not credited in the radiological analyses, also provide protection during fuel handling (GDC 61, "Fuel Storage and Handling and Radioactivity Control").

ITAAC: There are no specific ITAAC for ESF filter systems. The inspections, tests, analyses, and acceptance criteria associated with the ESF filter system design features are part of the following FSAR Tier 1 sections for systems in which ESF filter systems are provided: FSAR Tier 1, Sections 2.6.1, "Main Control Room Air Conditioning System," 2.6.3, "Annulus Ventilation System," 2.6.4, "Fuel Building Ventilation System," 2.6.6, "Safeguards Building Controlled-Area Ventilation System," and 2.6.8 "Containment Building Ventilation System."

Technical Specifications: There are no specific TS for ESF filter systems. Technical Specifications for filtration trains are part of the following TS for systems in which ESF filter systems are provided: TS 3.6.6, "Shield Building"; TS 3.6.7, "Annulus Ventilation System (AVS)"; TS 3.7.10, "Control Room Emergency Filtration (CREF)"; TS 3.7.11, "Control Room Air Conditioning System (CRACS)"; and TS 3.7.12, "Safeguard Building Controlled Area Ventilation System (SBVS)."

6.5.1.3 *Regulatory Basis*

The relevant requirements of NRC regulations for the ESF filter systems, and the associated acceptance criteria, are given in NUREG-0800, Section 6.5.1, "ESF Atmospheric Cleanup Systems," and are summarized below. Review interfaces with other SRP sections also can be found in NUREG-0800, Section 6.5.1.

1. GDC 19, “Control Room,” as it relates to maintaining the control room in a safe condition under accident conditions, including LOCAs.

2. GDC 41, “Containment Atmosphere Cleanup,” as it relates to providing systems to control the release of fission products to the environment and to control the concentration of hydrogen, oxygen, and other substances in containment following postulated accidents.

3. GDC 42, “Inspection of Containment Atmosphere Cleanup Systems” as it relates to designing containment ESF atmosphere cleanup systems to permit inspection.

4. GDC 43, “Testing of Containment Atmosphere Cleanup Systems,” as it relates to designing containment ESF atmosphere cleanup systems to permit pressure and functional testing.

5. GDC 61, “Fuel Storage and Handling and Radioactivity Control,” as it relates to design of systems for radioactivity control under normal and postulated accident conditions.

6. GDC 64, “Monitoring Radioactivity Releases,” as it relates to monitoring releases of radioactivity from normal operations, including anticipated operational occurrences, and from postulated accidents.

7. 10 CFR 52.47(b)(1), which requires that a U.S. EPR 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 NRC regulations.

Acceptance criteria adequate to meet the above requirements include:

- RG 1.52 as it relates to the design, inspection, and testing of the ESF filter systems.

RG 1.52 endorses criteria of ASME AG-1-1997, “Code on Nuclear Air and Gas Treatment Systems,” ASME N509-1989, “Nuclear Power Plant Air-Cleaning Units and Components,” and ASME N510-1989, “Testing of Nuclear Air Treatment Systems,” to be acceptable for design, construction, acceptance testing, and quality assurance of ESF atmospheric cleanup systems to adequately protect public health and safety.

6.5.1.4 *Technical Evaluation*

Review of the ESF filter systems in the FSAR was performed in accordance with SRP Section 6.5.1, Revision 3, Section III, March 2007, “Review Procedures.” The applicant’s design description, applicant’s evaluation of the design versus requirements, and results and conclusions reached in this review are as follows:

Four ESF filter systems work in conjunction with the five ventilation systems given below:

- Main control room air conditioning system: (CRACS), Control Room Emergency Filtration (CREF) (iodine filtration) train subsystem

- Annulus ventilation system: AVS Accident Trains

- Safeguard building controlled area ventilation system: Accident Iodine Exhaust filtration trains

- Fuel building ventilation system: Accident Iodine Exhaust filtration trains

- Containment building ventilation systems: Low-Flow Purge Exhaust Subsystem

Each ESF filter system consists of two independent trains. Each train has an activated charcoal carbon adsorber with motorized dampers, electric heater, pre-filter, and inlet and outlet HEPA filters. A booster fan and isolation dampers are included to provide the flow to the ventilation stack for the discharge of filtered air. Each ESF filter train is powered by an emergency bus that can be powered by a diesel generator.

Ventilation systems are aligned to ESF filter systems to support plant operations and accident mitigation. ESF filters in the CBVS low-flow purge exhaust subsystem are aligned during purging operations. The FBVS aligns to the SBVS ESF filters in response to high radiation in the RB or a containment isolation signal. Portions of the FBVS and CBVS are aligned to the ESF filters of the SBVS during the movement of irradiated fuel assemblies. Therefore, neither the FBVS nor the CBVS is an ESF system required to control the release of fission products after a design-basis accident.

NUREG-0800, Section 6.5.1 Section II states that conformance with the regulatory positions of RG1.52 as to the design, testing, and maintenance of ESF filter systems constitutes acceptable bases for satisfying the requirements of GDC 19, GDC 41, GDC 42, GDC 43, GDC 62, GDC 64, and 10 CFR 52.47(b)(1).

The applicant states that the U.S. EPR complies with GDC 19, GDC 41, GDC 42, GDC 43, GDC 61, and GDC 64 and follows the guidance of RG 1.52. In FSAR Tier 2, Section 6.5.1.1, the applicant stated that the ESF filter systems are designed to meet the design and performance guidance of RG 1.52 Revision 3, ASME N509 and ASME N510. RG 1.52 endorses and the applicant commits to the criteria of ASME AG-1-1997, ASME N509-1989, and ASME N510-1989 as acceptable for design, construction, acceptance testing, and quality assurance of ESF atmospheric cleanup systems to adequately protect public health and safety.

The U.S. EPR design-basis accident radiological consequences analyses in FSAR Tier 2, Chapter 15, "Transient and Accident Analyses," were reviewed and it was verified that accident dose sources originate in the Containment, Annulus, Safeguard Building, and Fuel Building, which have atmospheres that are treated by the ESF filter system. Additionally, the most limiting accident dose (compared to offsite dose) is the MCR dose. The ventilation supply and recirculation flow to the MCR is also treated by the ESF filter system.

A detailed comparison of the ESF filter systems in the FSAR was made to the acceptance criteria of NUREG-0800, Section 6.5.1, Section II, Acceptance Criteria as follows:

Control Room Dose (GDC 19)

The U.S. EPR design relies on the CRACS as the system to provide adequate protection against radiation and hazardous chemical releases in order to permit access and occupancy of the control room under accident conditions.

As described in FSAR section 9.4.1 the ESF filter system component in the CRACS, is aligned automatically upon receipt of an ESF actuation signal, including safety injection, or detection of high radiation levels. In addition, these ESF filtration systems can be manually aligned.

The applicant used the design-basis accident radiological consequences analysis guidance in RG 1.183, "Alternative Radiological Source Terms for Evaluating Design-Basis Accidents at Nuclear Power Plants," to develop the U.S. EPR DBA control room dose analyses to show compliance with GDC 19, as it relates to maintaining the control room in a safe condition with adequate radiation protection under accident conditions. The DBA radiological consequences analyses show that the ESF filter system within the CRACS limits the control room dose to 0.04 Sv (4 rem) TEDE for the LOCA and 0.045 Sv (4.5 rem) TEDE for the MSLB. These estimated doses are compared to the control room dose criterion in GDC 19 of 0.05 Sv (5 rem) TEDE. The control room dose for each of the other analyzed DBAs was shown to be less than 0.05 Sv (5 rem) TEDE in FSAR Tier 2, Table 15.0-12, "Radiological Consequences of U.S. EPR Design Basis Accidents (rem TEDE)." Detailed review of the DBA control room dose analyses is discussed in Section 15.0.3 of this report. This review finds that the control room design and ESF filter system meet the radiological habitability requirements of GDC 19 by limiting the post-DBA dose in the control room for all applicable DBAs to less than 0.05 Sv (5 rem) TEDE.

Review of GDC 19 as it relates to compliance with RG 1.52 as to the design, testing, and maintenance of the CRACS (with the exception of the CRACS ESF filter system components), and control room habitability is addressed in Sections 9.4.1 and 6.4 of this report respectively. Review of GDC 19 as it relates to compliance with RG 1.52 as to the design, testing, and maintenance of the CRACS ESF filter system components is discussed below.

Systems to Control Fission Product Release after an Accident (GDC 41)

As described in the respective FSAR sections on ESF filter systems, ESF filter systems in the CRACS, AVS, and SBVS are aligned automatically with their associated ventilation systems upon receipt of an ESF actuation signal, including safety injection, or detection of high radiation levels. In addition, these ESF filtration systems can be manually aligned. These capabilities are addressed in FSAR Tier 2, Sections 7.3.1.2.16, "Main Control Room Air Conditioning System Isolation and Filtering"; 6.4.3, "System Operational Procedures"; and 6.4.5, "Testing and Inspection," for the CRACS; Section 6.2.3.3, "Safety Evaluation," for the AVS; Section 9.4.5.2.3, "System Operation," for the SBVS. GDC 41 is addressed in the respective sections of this report:

The review of the AVS is contained in Section 6.2.3 of this report. The review of the CRACS is contained in Sections 9.4.1 and 6.4 of this report, and the review of the SBVS is contained in Section 9.4.5 of this report.

The U.S. EPR low-flow purge exhaust subsystem is normally not in operation and is aligned manually. It provides automatic isolation of containment atmosphere by quick closure of containment isolation valves. The CB low flow exhaust can also be aligned to the SBVS Accident Iodine Exhaust filter trains in an emergency for redundancy. Compliance with GDC 41 is reviewed in Section 9.4.7 of this report.

For the reasons discussed in the sections of this report identified above, the staff finds that the HVAC systems containing ESF filter systems identified above meet the requirements of GDC 41 as they apply to these systems (with the exception of their ESF filter system components).

The staff's review concerning GDC 41 for these systems as it relates to compliance with RG 1.52 as to the design, testing, and maintenance of ESF filter systems is discussed below.

GDC 42 and GDC 43

FSAR Tier 2, Sections 6.5.1.1, "Design Bases," and 6.5.1.3, "Design Evaluation," state that the ESF filter systems are designed to permit periodic inspection and periodic pressure and functional testing. The ESF filter systems permit appropriate periodic inspection of components such as filters and ducts to verify the capability and integrity of the systems.

Review of GDC 42 and GDC 43 as they relate to compliance with RG 1.52 as to the design, testing, and maintenance of ESF filter systems is discussed below.

GDC 61

FSAR Tier 2, Sections 6.5.1.1 states that the ESF filter systems are designed to control radioactive releases under both normal and postulated accident conditions in compliance with GDC 61. Although not credited in the radiological analyses, the ESF filter systems provide protection during fuel handling. Review of GDC 61 as it relates to compliance with RG1.52 as to the design, testing, and maintenance of ESF filter systems is discussed below.

GDC 64

The FSAR addresses the design bases for monitoring releases of radioactivity from normal operations, including anticipated operational occurrences, and from postulated accidents in FSAR Tier 2, Sections 6.5.1.1, "Design Bases" for ESF filter systems and FSAR Tier 2, Section 12.3.4, "Area Radiation and Airborne Radioactivity Monitoring Instrumentation." Radiation instruments for controlling airborne radioactivity releases via the plant stack are addressed in Section 11.5, "Process and Effluent Radiological Monitoring and Sampling Systems" of the FSAR.

Review of GDC 64 as it relates to compliance with RG1.52 as to the design, testing, and maintenance of ESF filter systems is discussed below.

Conformance to RG 1.52

FSAR Tier 2, Section 6.5.1.1, "Design Basis," states that the ESF filter systems are designed to meet the guidance in RG 1.52. NUREG-0800, Section 6.5.1 states that conformance to the guidance in RG 1.52 constitutes acceptable bases for satisfying the requirements of GDC 19, GDC 41, GDC 42, GDC 43, GDC 62, GDC 64, and 10 CFR 52.47(b)(1). The following review was performed to verify conformance to RG 1.52.

The staff reviewed the provisions of the ESF filter systems in FSAR Tier 2, Section 6.5.1.2, "System Design." The review was conducted to determine if the guidance of RG 1.52 to prefilter air, remove moisture, preheat air, adsorb iodine, and post-filter air was met.

FSAR Tier 2, Section 6.5.1.3 states that the ESF filter systems are redundant, designed to Seismic Category I, and are powered by an emergency bus that is backed up by an emergency diesel generator. The structural ability of the filters to operate after a design-basis accident is addressed by their safety-related status and demonstrated by their seismic design. These are specified in FSAR Tier 2, Table 3.2.2-1, "Classification Summary." The staff reviewed these specifications and found them acceptable.

The staff reviewed the provisions of the ESF filter system components in FSAR Tier 2, Section 6.5.1.2.2, “Component design.” The review was conducted to determine if the design, construction and testing of ESF filter system components meet the guidance of RG 1.52 and NUREG-0800.

Generally, FSAR Tier 2, Section 6.5.1.1, “Design Basis,” states that the ESF filter systems are designed, constructed, qualified, and factory tested to ASME AG-1. The staff finds that this meets the RG 1.52 guidance, since RG 1.52 specifies ASME AG-1. Therefore the staff finds that the design, construction and testing of ESF filter system components , such as demisters, heaters, prefilters, HEPA filters, mounting frames, filter housings, adsorbent, fans, ductwork and dampers meet the guidance of NUREG-0800. Section 6.5.1. Staff review of specific areas described in RG 1.52 and NUREG-0800, Section 6.5.1 are discussed below.

As stated in FSAR Tier 2, Section 6.5.1.2.2, the components, including heaters, HEPA filters, and carbon adsorbers meet the standards of ASME AG-1-2003.

In RAI 312, Question 09.04.02-3, the staff requested that the applicant justify the use of ASME AG-1-2003 referenced in FSAR Tier 2, Section 6.5.4, in lieu of ASME AG-1-1997, which is the version of this code endorsed by RG 1.52. The staff also requested the applicant justify the use of ASME N509 (2002) referenced in FSAR Tier 2, Section 6.5.4, in lieu of ASME N509 (1989), which is the version of this code endorsed by RG 1.52. In an October 21, 2009, response, the applicant provided a reconciliation of the two versions of the code as they relate to the U.S. EPR design. The applicant also provided marked up correction to FSAR Tier 2, Table 1.9-2, which notes that the U.S. EPR standard design takes exception to RG 1.52 and 1.140 as noted above.

Based on a review of the applicant’s response and proposed FSAR corrections, the staff has determined that additional information is necessary. In RAI 462, Question 06.05.01-5, the staff requested that the applicant provide sufficient information to justify the use of ASME AG-1-2003 and ASME N509 (2002). **RAI 462, Question 06.05.01-5 is being tracked as an open item.**

In RAI 135, Question 09.04.05-1(6), the staff requested that the applicant clarify that the design of the Fuel Building Ventilation System incorporates RG 1.52 guidance as it relates to provisions to address boiling in the spent fuel pool. In a February 27, 2009, response to RAI 135, Question 09.04.05-1(6), the applicant stated that spent fuel pool (SFP) boiling need not be considered in the design of the U.S. EPR FBVS based on the following justification:

- The U.S. EPR fuel pool cooling system is designed as safety-related and Seismic Category 1. The FPCS is also capable of removing the maximum SFP heat load following a single failure. Therefore, as stated in NUREG-0800, Section 9.1.3, “Spent Fuel Pool Cooling and Cleanup System” (Revision 2-March 2007), the FBVS need not be designed for SFP boiling considerations.

In addition, the applicant responded to RAI 135, Question 09.04.05-13 where the applicant addressed ESF filter system design features to address moisture removal for other operational events as discussed below.

Provisions to Prefilter Air and Remove Moisture

In a February 27, 2009, response to RAI 135, Question 09.04.05-1(13) the applicant addressed moisture removal for the SBVS during other operational events. The staff also requested that the applicant clarify the ESF filter system design as it relates to RG 1.52, Position C.3, Paragraph 3.1, which recommends installation of a moisture separator prior to the heater to remove entrained water droplets from the inlet air stream, thereby protecting downstream HEPA filters and iodine adsorbers. In a February 27, 2009, response to RAI 135, Question 09.04.05-1(13), the applicant updated the FSAR to include a description of the moisture separator in the design of the ESF filter systems. The applicant clarified that the SBVS Iodine Filtration Trains incorporate moisture separators in the prefilter design, and the applicant clarified that the SBVS moisture separator meets the standards of RG 1.52, ASME N509, and ASME AG-1.

FSAR Tier 2, Section 12.3.6.5.6 states that HVAC air handling equipment is provided with drains. The drainage fluid flows through hard piping to collection tanks for processing.

The staff reviewed the February 27, 2009, response to RAI 135, Question 09.04.05-1(13) and confirmed the proposed FSAR revisions are incorporated in FSAR Revision 2. The staff issued follow-up RAI 462, Question 06.05.01-3, which requested that the applicant clearly describe the ESF filter system moisture separators and associated drains, provide additional descriptions of the moisture separators in the FSAR, and clarify that drains from the moisture separators are routed to a drain system that will handle contaminated liquid. **RAI 462, Question 06.05.01-3 is being tracked as an open item.**

Carbon Adsorbers

FSAR Tier 2, Section 6.5.1.2.2 specifies the use of activated charcoal and a design system efficiency for removal of iodine and organic iodides. This removal efficiency is met using 4-inch deep carbon beds with a laboratory decontamination efficiency of 99 percent fission products removed, as tested per American Society for Testing and Materials (ASTM) D3803, “Standard Test Method for Nuclear-Grade Activated Carbon.” The staff finds that this specification conforms to RG 1.52, Table 1. The applicant committed that carbon adsorbers are constructed, qualified, and tested in accordance with ASME AG-1, which meets the guidance in RG 1.52. As stated in FSAR Tier 2, Section 6.5.1.2.2, the applicant stated that the fabrication of the charcoal tray and screen involves all-welded construction to preclude potential loss of charcoal from adsorber cells. Since the tray and screen design uses welded construction as recommended by NRC Bulletin 80-03, the staff finds that the design conforms to the guidance in NRC Bulletin 80-03 regarding preclusion of charcoal loss.

Other than a stated commitment to meet the guidance in RG 1.52, the staff determined that there was insufficient detailed information in the FSAR to demonstrate conformance to the provisions of RG 1.52. Therefore, in RAI 233, Question 06.05.01-1, the staff requested that the applicant provide details on how the carbon adsorber conforms to the following specific guidance stated in the RG:

1. The design average atmosphere residence time should be 0.25 seconds per 5.08 cm (2 in.) in thickness of adsorbent bed.

2. The maximum charcoal loading for the adsorbent train should be below 2.5 mg/gm (≤2.5 mg of total iodine per gram of charcoal) ($2.5x10^{-9}$ lb/lb).

3. The design percent of the impregnant carbon (activated carbon impregnated with TEDA or KI) should be no more than five percent of the total carbon.

4. The maximum component temperature in the adsorber section with normal flow conditions is not specified. The iodine loading post-accident radioactivity-induced heat in the adsorbent should not exceed that design temperature.

5. Under conditions of a failed fan post-LOCA, the charcoal temperature rise resulting from radioactivity-induced heat in the adsorbent should be below the 329 °C (625 °F) charcoal ignition temperature. A water deluge from a fire protection sprinkler may be necessary to control this temperature rise.

In July 10, 2009, and September 1, 2009, responses to RAI 233, Question 06.05.01-1, the applicant stated that, in accordance with RG 1.52, the design of the charcoal filters includes the following specifications:

1. The design of the carbon adsorber will provide for an average atmospheric residence time of 0.25 seconds per two inches of adsorbent bed thickness in accordance with the provisions of ASME AG-1-2003 (Section FD for Type II adsorber cells and Section FE for Type III adsorber cells).

2. The maximum charcoal loading for the carbon adsorbent trains is below 2.5 mg/gm (≤2.5 mg of total iodine per gram of charcoal):

    a. Annulus exhaust filtration system (KLB): 0.08 mg/gm

    b. Safeguard Building exhaust filtration system (KLC): 0.94 mg/gm

    c. Control room emergency filtration subsystem of the CRACS: 2.0x10E-06 mg/gm

    d. These analytical results show that the calculated values are below the maximum charcoal loading for the carbon adsorbers per RG 1.52.

3. No more than 5 percent of impregnant (50 mg of impregnant per gram of carbon) will be used.

4. The maximum component temperature in the adsorber section with normal air flow during normal operation is 115 °F. The maximum component temperature deep inside the adsorber section with the fan shut down and the charcoal adsorption unit isolated post-loss of coolant accident (LOCA) is 148 °F.

5. Water deluge from a fire protection sprinkler is not used to control adsorber temperature post-LOCA.

    The maximum post-accident radioactivity-induced heat is in the annulus exhaust filtration system (KLB). The heat gain (beta plus gamma radiation) is 560 watts over a mission time of 30 days. The maximum localized temperature deep inside the KLB adsorber unit (unit inlet and outlet dampers closed) under conditions of a failed fan post-LOCA with a temperature rise resulting from radioactivity-induced heat in the adsorbent is 148 °F, with a corresponding maximum bed surface temperature of 126 °F (see the Response to Question 06.05.01-1, Part 4). This is well below the 625 °F charcoal-ignition temperature referenced in this question.

    For fire protection control of the engineered safety feature (ESF) carbon adsorbers in a fan shutdown condition post-LOCA alignment, the adsorber unit is isolated (stop the airflow and close all isolation dampers) to allow the carbon adsorber to cool."

The staff has reviewed the response to RAI 233, Question 06.05.01-1 and determined that additional clarification of the charcoal filter design as described in the FSAR is necessary. Although the information was provided in the RAI response, the applicant did not propose to revise the FSAR to add this design basis information. Since the staff used this information to verify compliance with applicable regulations, the staff considers the information provided in the response to RAI 233, Question 06.05.01-1, to be design-basis information that must be incorporated into FSAR Tier 2, Section 6.5.1.2.2. Therefore, the staff requested the FSAR be revised to include the design information provided in the RAI response. **RAI 462, Question 06.05.01-2 is being tracked as an open item.**

High Efficiency Particulate Air Filter

For the ESF filter systems, the applicant specifies in FSAR Tier 2, Section 6.5.1.2.2 that the High Efficiency Particulate Air (HEPA) filter units are designed, constructed, qualified, and factory tested to ASME AG-1 standards. The efficiency for removal of particulates by the HEPA filter resulting from a DBA is 99.97 as factory tested. The applicant also stated that the HEPA filters are periodically tested to ASME N510 standards once installed to demonstrate an efficiency of 99.95 percent. The staff finds that this conforms to RG 1.52 Regulatory Position 4.4 and 6.3 as they apply to HEPA filter factory and in-place testing respectively. Since NUREG-0800, Section 6.5.1 states that relevant aspects of the applicable regulatory requirements are met by use of the regulatory positions of RG 1.52, the staff finds that the HEPA filter design and testing conform to the guidance of NUREG-0800, Section 6.5.1.

Other than a stated commitment to meet the guidance in RG 1.52, the staff determined that there was insufficient detailed information to verify how the specific RG 1.52 guidance is met for the HEPA filters to have sufficient design margin to accommodate fission product loading during an accident without restricting flow rate. Therefore, in RAI 233, Question 06.05.01-1, the staff requested that the applicant provide additional information on HEPA filter design. In a September 1, 2009, response to RAI 233, Question 06.05.01-1, the applicant stated:

In accordance with RG 1.52, the design of the HEPA filters includes the following specification: "The efficiency for removal of particulates by the HEPA filter resulting from a design basis accident shall be 99.97/99.95 percent tested to ASME N510."

The maximum mass loading of the HEPA filters is:

- Annulus exhaust filtration system (KLB): 822 mg
- Safeguard Building exhaust filtration system (KLC): 631 mg
- Control Room Emergency Filtration subsystem of the main control room air conditioning system: 0.0070 mg

The staff reviewed the design description of the ESF subsystem of the AVS, CRACS, SBVS and Containment Ventilation System, and verified that each of the ESF carbon filter units will have at least one HEPA filter installed. A single HEPA filter, with a standard size of .61 m x .61m x .3m (24 in.x24 in.x12 in.), is rated for 708 ls (1,500 cfm) of air flow and has a dust loading capacity of 1,140 g. A clean HEPA filter (rated for 1,500 cfm) has an initial pressure drop of 0.24 cm (1.3 in.) of water gauge. A dirty HEPA filter (with a full dust loading of 1,140 grams) has a pressure drop across the HEPA filter of approximately 0.560 cm (3 in.) of water gauge. The actual dust spot loading of the worst-case ESF adsorber unit (i.e., KLB) is 822 mg. Therefore, the increase in pressure drop between the clean and dirty conditions for the worst case ESF adsorber unit (KLB) is within the normal expected pressure increase for a typical filtered exhaust fan design.

The staff has reviewed the response to RAI 233, Question 06.05.01-1 and determined that additional clarification of the ESF HEPA filter design as described in the FSAR is necessary. Although the information was provided in the RAI response, the applicant did not propose to revise the FSAR to add this design basis information. Since the staff used this information to verify compliance to applicable regulations, the staff considers the information provided in the response to RAI 233, Question 06.05.01-1, to be design-basis information that must be incorporated into FSAR Tier 2, Section 6.5.1.2.2. Therefore, the staff requested that the applicant revise the FSAR to include the design information provided in the RAI response. **RAI 462, Question 06.05.01-2 is being tracked as an open item.**

Air Flow Rate: FSAR Tier 2, Section 6.5.1.3, provides the following ESF filter systems capacities:

- AVS: ≥ 30.0 m^3/min and ≤ 36.7 m^3/min (nominal 34.0 m^3/min)
- (≥ 1,060 cfm and ≤ 1,295 cfm (nominal 1,200 cfm))
- SBVS: ≥ 61.2 m^3/min and ≤ 74.8 m^3/min (nominal 68.0 m^3/min)
- (≥2,160 cfm and ≤2,640 cfm (nominal 2,400 cfm))
- CRACS: ≥ 102.0 m^3/min and ≤ 124.6 m^3/min (nominal 113.3 m^3/min)
- (≥3,600 cfm and ≤4,400 cfm (nominal 4,000 cfm))
- CBVS: ≥ 76.5 m^3/min and ≤ 93.4 m^3/min (nominal 85.0 m^3/min)
- (≥2,700 cfm and ≤3,300 cfm (nominal 3,000 cfm))

The staff finds that these flow capacities are specified to accommodate the ventilation flow credited in the safety analysis and meet the RG 1.52 upper limit of 850 m^3/min (30,000 cfm) for each atmospheric cleanup system.

Instrumentation Requirements

FSAR Tier 2, Section 6.5.1.3 states that the ESF filter systems have instrumentation that conforms to RG 1.52, Table 6.5.1-1. The automatic operation and continuous indication of system parameters in FSAR Tier 2, Section 6.5.1.5, "Instrumentation Requirements," and FSAR Tier 2, Table 9.4.1-1, "Minimum Instrumentation, Indication, and Alarm Features for ESF Filter Systems," are claimed by the applicant to demonstrate that the guidance of RG 1.52 and ASME N509-1989 is met. In RAI 462, Question 06.05.01-4, the staff requested that the applicant provide additional information on the instrumentation for the ESF filter systems. **RAI 462, Question 06.05.01-4 is being tracked as an open item.**

Post-LOCA Iodine Removal by Filter Systems

NUREG-0800, Section 6.5.1 ,Section III, Paragraph 3I provides guidance for staff review of specific deviations from RG 1.52 guidance. FSAR Tier 2, Section 15.0.3, shows that a radioiodine decontamination factor of greater than 10 for the credited filtration systems is needed so that the calculated offsite and control room doses meet the regulatory dose criteria given in 10 CFR 52.47(a)(2) and GDC 19, respectively. Therefore, in accordance with NUREG-0800, Section 6.5.1, Part III.I.iii, the ESF filter systems should meet the guidance of RG 1.52; however, deviation from Regulatory Positions C.3.2 and C.3.4 of RG 1.52, Revision 3 is allowed. In RAI 462, Question 06.05.01-4, the staff requested that the applicant provide additional information on the iodine removal by filter systems used in the U.S. EPR design, in order to determine if the U.S. EPR design deviates from these regulatory positions. **RAI 462, Question 06.05.01-4 is being tracked as an open item.**

Based on review of the system descriptions in the respective ventilation sections of FSAR Tier 2, Section 6.5.1, the staff finds that, with exception of the open items identified above, RG 1.52 Regulatory Position C.3.2 is met for the CRACS, CREF train subsystem, AVS accident filtration trains, SBVS accident iodine exhaust filtration trains, and the low flow purge exhaust subsystem.

The staff finds that RG 1.52 Regulatory Position C.3.4 is met for the U.S. EPR ESF filter systems since all ESF components are designed to Seismic Category 1 as described in the FSAR and discussed above.

Therefore, the staff finds that the NUREG-0800, Section 6.5.1 guidance on RG 1.52 deviations is not applicable for the U.S. EPR ESF filter systems, because the U.S. EPR does not deviate from these RG 1.52 Regulatory Positions.

Operation > 90 Hours Per Year

NUREG-0800, Section 6.5.1, Paragraph 4 calls for verification that the level of use does not impair the capability of the ESF filter systems to perform its intended function in the event of a LOCA. Based on review of the design basis for the ESF filter systems as described in FSAR Tier 2, Section 6.5.1.1 and the design basis sections of the related HVAC system sections in the FSAR, the staff finds that the ESF filter systems are designed to operate during accident conditions only. The ESF filter systems have the capability to be periodically inspected, to undergo routine testing, and to monitor component performance during system operation. The staff finds that these capabilities will allow detection of impairments that would impact accident capabilities. Because the ESF filter systems are not designed to function for normal atmosphere cleanup, the staff finds that the level of use of these systems is low, and would not likely impair their ability to perform their intended functions.

Release Point Monitors

The ESF filter systems all release through the U.S. EPR vent stack with an elevated release point. The staff finds that the single release point for all systems is monitored for radiological releases. Therefore, the staff finds that the applicant has adequately provided for monitoring all radioactive releases from the ESF atmosphere cleanup systems, and that NUREG-0800, Section 6.5.1, Part III, Paragraph 4 is met. The staff review of the radiation instrumentation is contained in Chapter 11, “Radioactive Waste Management,” of this report.

Based on the above discussion, the staff finds that the U.S. EPR ESF filter systems conform to the guidance of NUREG-0800, Section 6.5.1 in regard to monitoring of release, therefore, the design conforms to the guidance of RG 1.52 on that function.

ITAAC

The staff reviewed the proposed ITAAC for the ESF filter systems. The applicant proposed ITAAC requirements in FSAR Tier 1, Sections 2.6.1 (CRACS), 2.6.3 (AVS), 2.6.4 (FBVS), 2.6.6 (SBVS), and 2.6.8 (CBVS). The staff finds that sufficient information has been provided to satisfy NUREG-0800, Section 14.3 and NUREG-0800, Section 14.3.7.

The staff finds that the laboratory and in-place testing for airflow distribution to the HEPA filters, DOP testing of the HEPA filter sections, and bypass leakage testing of the activated carbon adsorber section, as well as halide testing of carbon filters, is addressed by the Ventilation Filter Testing Program (VFTP) as described in U.S. EPR Technical Specifications and is discussed below.

In RAI 462, Question 06.05.01-4, the staff requested that the applicant revise the FSAR, Tier 1 ITAAC tables to list the minimum inventory of alarms, displays and controls for ESF filter system instrumentation associated with the SBVS, AVS and the CBVS. **RAI 462, Question 06.05.01-4 is being tracked as an open item.**

With the exception of the open item identified above, the staff finds that the FSAR Tier 1 ITAAC tables adequately address verification of the functional arrangement, physical separation, and seismic qualification of the ESF filter systems. The design, fabrication, inspection of ESF filter systems are in accordance with ASME AG-1 code criteria. With the exception of the open item, FSAR Tier 1 ITAAC tables verify the minimum inventory of ESF filter system alarms, displays, and controls. Automatic actuation of the ESF filter systems is also verified. Based on this review and with the exception of the open item identified above, the staff has concluded that the ESF filter systems in a plant that incorporates the design certification will be built and will operate in accordance with NRC regulations.

Initial Plant Test Program

The staff review of FSAR Tier 2, Section 14.2, “Initial Plant Test Program,” and FSAR Tier 2, Section 14.2.12.8, “HVAC Systems,” is discussed in Chapter 14, “Initial Test Program and ITAAC-Design Certification,” of this report.

Technical Specifications: The specific TS for ESF filter systems that are part of the Ventilation System Technical Specifications: TS 3.7.10 (CREF), TS 3.7.11 (CRACS), and TS 3.7.12 (SBVS) were reviewed. Routine testing and inspection of ESF filter systems are conducted under the ventilation filter testing program described in TS 5.5.10. As stated in this section, testing will be in accordance with RG 1.52, Revision 3, ASME N510-1989, and ASME AG--1-2003. Laboratory testing of samples of activated carbon absorber material is performed in accordance with ASTM D3803. The review for technical specifications has been coordinated with the primary review of SRP Section 16.0. The staff finds that these TS follow the guidance of NUREG-1431, Revision 3.1. Therefore, the staff concludes that the TS meet the in-place testing criteria guidance contained in RG 1.52 and are adequate to demonstrate the capabilities of the ESF filter systems.

6.5.1.5 *Combined License Information Items*

No applicable items were identified in the FSAR. No additional COL information items need to be included in FSAR Tier 2, Table 1.8-2 for ESF filter systems consideration.

6.5.1.6 *Conclusions*

This review addresses the four ESF filter systems used by the five ventilation systems discussed in FSAR Tier 2, Section 6.5.1 and their mitigation of control room and offsite dose. The staff finds that the applicant has used RG 1.183 to calculate the DBA radiological releases and expected CRACS inlet conditions that would result from accidents in the proposed FSAR. With the exception of the open items identified above, the staff finds that the proposed design and operation of the ESF filter systems will provide adequate fission product removal in post-accident environments. Following the guidance in SRP Section 6.5.1, acceptability of the U.S. EPR ESF filter systems was determined by reviewing the plant-specific data against the criteria stated in the RG 1.52. SRP Section 6.5.1 states that conformance to the requirements of the provisions of RG 1.52 constitutes acceptable bases for satisfying the requirements of GDC 19, GDC 41, GDC 42, GDC 43, GDC 61, GDC 64, and 10 CFR 52.47(b)(1). Therefore, with exception of the open items identified above, the staff finds that the ESF filter systems used in the U.S. EPR for atmosphere cleanup meet the requirements of GDC 19, GDC 41, GDC 42, GDC 43, GDC 61, and GDC 64; and 10 CFR 52.47(b)(1).

6.5.2 Containment Spray Systems

An automatically actuated containment spray system is not required to mitigate the consequences of any design-basis accident and is, therefore, not credited in the design-basis containment or radiological analyses. However, a manually initiated containment dome spray system is part of the SAHRS and is used to reduce pressure and remove fission products from the containment atmosphere under severe accident conditions. The SAHRS is described in FSAR Tier 2, Section 19.2.3.3, "Severe Accident Mitigation Features." The staff's review of the SAHRS is documented in Chapter 19, "Severe Accidents," of this report.

6.5.3 Fission Product Control Systems

The release of fission products following a postulated design-basis accident is mitigated by several U.S. EPR design features. The major features that provide this function include primary and secondary containment structures, ventilation systems, and post-accident chemical addition to the fluid in the containment. This section provides the evaluation of those features that prevent or limit the release of fission products from primary containment. The DBA radiological consequences analyses and assumed sequence of events that demonstrate the effectiveness of these fission product removal and control systems in maintaining radioactivity releases within regulatory limits are presented in FSAR Tier 2, Section 15.0.3.

Fission product control systems or structures are those systems or structures that contain or process fission products not removed from the containment atmosphere and the ventilated spaces serviced by other systems. The Seismic Category I safety-related RCB and RSB provide the primary means of containing fission products that have breached the RCS. Primary containment is provided by the RCB. Secondary containment is provided by the RSB. Additional fission product control within the RCB is provided by natural deposition and passive[1] chemical addition of trisodium phosphate dodecahydrate (TSP-C) to the IRWST. In the event of a severe accident, fission products can also be scrubbed from the primary containment atmosphere by the non-safety-related SAHRS. The SAHRS is not credited as a fission product

[1] Chemical addition is achieved without forced injection by the ECCS. The TSP-C is dissolved into the IRWST during the LOCA recirculation phase as water flows over the TSP-C baskets.

removal system in the design-basis accident radiological consequences analyses; however, the system does provide defense-in-depth capability to remove fission products from the primary containment atmosphere.

6.5.3.1 *Introduction*

The purpose of this section is to evaluate the U.S. EPR systems that are designed to ensure that radiological releases during normal and accident conditions are below the reference dose values used in the evaluation of plant design features with respect to postulated reactor accidents established in 10 CFR 50.34(a)(1)(ii), "Contents of applications; technical information," and 10 CFR 52.47(a)(2). The system and component design criteria for fission product control systems are outlined in RG 1.52, Regulatory Positions C.1, C.2, and C.3.

6.5.3.2 *Summary of Application*

FPCSs are described in multiple FSAR sections for the primary and secondary containment structures and ventilation systems, and for gaseous elemental iodine (I_2) control. These sections are given below and have been reviewed with respect to design features associated with FPCSs.

FSAR Tier 1: There are no FSAR Tier 1 entries specific to the FPCSs.

FSAR Tier 2: The U.S. EPR FPCS includes containment and containment ventilation systems that process exhaust air during normal operation and accident conditions. Leakage from primary containment penetrations and seals is captured within the secondary containment. The annulus area between the primary and secondary containment is maintained at a slight vacuum by the annulus ventilation system during normal and accident conditions. A leak-off system collects leakage from airlock door seals, ventilation system containment isolation valves, and fuel transfer tube isolation valves. The leak-off system is routed to the annulus. The leak-off system forms part of the secondary containment bypass leakage barrier. The AVS and leak-off systems process containment leakage and are addressed in FSAR Tier 2, Section 6.2.3, "Secondary Containment Functional Design."

Penetrations and seals that terminate outside the secondary containment (outside the annulus) exit the RSB structure into either the Fuel Building or one of the four Safeguard Buildings. Ventilation systems for these buildings and the ESF filter systems are addressed in FSAR Tier 2, Sections 9.4.1, "Main Control Room Air Conditioning System," through 9.4.5, "Safeguard Building Controlled-Area Ventilation System."

The primary mechanism to limit release of fission products that are produced following a DBA is provided by the containment structures. A description of the primary and secondary containment structures, acceptable leakage criteria, and design features used for fission product control is provided in the following FSAR Tier 2 Sections:

- 3.8.1.1, "Description of the Containment"
- 3.8.4.1.1, "Reactor Shield Building and Annulus"
- 6.2.1, "Containment Functional Design"
- 6.2.3, "Secondary Containment Functional Design (includes AVS)"

- 6.2.4, "Containment Isolation System"
- 6.2.6, "Containment Leakage Testing"
- 6.3.2.2.2, "System Components," Trisodium Phosphate Dodecahydrate discussion under the "In-Containment Refueling Water Storage Tank," subheading
- 6.5.3.1, "Primary Containment"
- 6.5.3.2, "Secondary Containment"
- 9.4.7, "Containment Building Ventilation"

ITAAC: ITAAC for the containment structures are in FSAR Tier 1, Table 2.1.1-7, "Nuclear Island Inspections, Tests, Analyses, and Acceptance Criteria."

Technical Specifications: Technical Specifications are described in FSAR Tier 2, Chapter 16. The following Technical Specification sections are applicable for FPCS:

- TS 3.6.1, "Containment"
- TS 3.6.6, "Shield Building"
- TS 3.6.7, "Annulus Ventilation System (AVS)"
- TS 3.6.8, "pH Adjustment"

6.5.3.3 *Regulatory Basis*

The relevant requirements of NRC regulations for this area of review, and the associated acceptance criteria, are given in NUREG-0800, Section 6.5.3, "Fission Product Control Systems and Structures," and are summarized below. Review interfaces with other SRP sections also can be found in NUREG-0800, Section 6.5.3.

1. GDC 41 as it relates to the containment atmosphere cleanup system being designed to control fission product releases to the environment following postulated accidents.
2. GDC 42 as it relates to the containment atmosphere cleanup system being designed to permit periodic inspections.
3. GDC 43 as it relates to the containment atmosphere cleanup system being designed to permit appropriate functional testing.
4. 10 CFR 50.34(a)(1)(ii) and 10 CFR 52.47(a)(2), as they relate to ensuring that nuclear power plant radiological releases during normal and accident conditions are below the reference dose values used in the evaluation of plant design features with respect to postulated reactor accidents.
5. 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 has been constructed and will operate in

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

Acceptance criteria adequate to meet the above requirements include:

1. RG 1.52, Regulatory Positions C.1, C.2, and C.3, as they relate to system and component design criteria for fission product control systems.

2. RG 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors."

3. RG 1.206, "Combined License Applications for Nuclear Power Plants (LWR Edition)."

6.5.3.4 *Technical Evaluation*

The technical evaluation is based on review of the primary and secondary containment structures and associated systems designed to mitigate the release of fission products. This review includes FSAR Tier 2 sections identified in Section 6.5.3.2 above. The design features used for FPCS include structural barriers provided by the primary and secondary containment, filtration by ventilation systems, natural deposition, and chemical addition. The design basis of each of these FPCS features was compared against acceptance criteria provided in SRP Section 6.5.3, RG 1.52, RG 1.183, and RG 1.206. Airborne radioactivity released into the containment during a DBA LOCA is bounding over that released during a rod ejection accident, which is the only other DBA evaluated for offsite and control room dose that includes radioactive material release to the containment. Therefore, the LOCA analysis was reviewed as the bounding case for determining acceptable FPCS design. Specific acceptance criteria applicable to each design feature for determination of an acceptable FPCS design are given in SRP Section 6.5.3 and include the following:

Primary Containment

Primary containment design leakage rates for which credit is given should not be less than 0.1 percent per day.

Secondary Containment

To be classified as a secondary containment for the purpose of fission product control, a structure or structures should completely surround the primary containment, and at minimum should be held at a pressure of 6.35 mm (0.25 in.) water gauge (0.47 mm Hg or 62.2 Pa), below adjacent regions, under all wind conditions up to the wind speed at which diffusion becomes great enough to ensure site boundary exposures less than those calculated for the design-basis accidents even if exfiltration occurs.

Other Fission Product Control Systems

Fission product retention credit may be taken by the applicant for other systems (e.g., filtration and adsorption units as evaluated in SRP Section 6.5.1 and described in RG 1.52).

6.5.3.4.1 Primary Containment

The Reactor Building structure consists of the Reactor Containment Building and the Reactor Shield Building. Together, these two structures provide a principle barrier against a release of radiation and radioactive materials to the environment. The RCB provides a barrier to prevent the release of airborne radioactivity and provides radiation shielding for the reactor core and associated support systems inside containment. This is achieved by lining the interior of the RCB, including the basemat, with a leak-tight, 6.35 mm (0.25 in.) thick carbon steel liner. In addition, both the IRWST and containment sumps are lined with 6.35 mm (0.25 in.) thick stainless steel for corrosion protection of the underlying carbon steel liner. The liner is coated with approved sealants for application inside primary containment and is periodically inspected for degradation. The containment is designed for the maximum pressure and temperature resulting from the release of stored energy during a design-basis LOCA or MSLB within the containment. The RCB structure provides an essentially leak tight enclosure following a DBA, with an internal design pressure of 427 kPa (62 psig) and a design temperature of 170 °C (338 °F).

The maximum allowable RCB leakage rate (La) is limited to 0.25 percent of the RCB air mass per day with building pressure at the containment design testing pressure (Pa) of 379 kPa (55 psig), as given in U.S. EPR Technical Specification 5.5.15, "Containment Leakage Rate Testing Program." This satisfies SRP Section 6.5.3 Acceptance Criterion 1 that primary containment design leakage rates for which credit is given should not be less than 0.1 percent per day due to difficulties in measuring very low leakage rates. The calculated peak internal pressure following a DBA LOCA is 359 kPa (52 psig). Therefore, margin is available in the maximum allowed leakage rate applied to the accident analysis described in FSAR Tier 2, Chapter 15, "Transient and Accident Analyses." Actual containment leakage resulting from a DBA LOCA will not result in RCB design leakage rates greater than allowable release rates. Therefore, dose calculations based on the allowable primary containment leakage rate are conservative and acceptable.

ITAAC: FSAR Tier 1, Table 2.1.1-7 specifies the inspections, tests, analyses, and associated acceptance criteria for the primary containment structure.

Technical Specifications: Primary containment operability is dependent on maintaining a leak tight barrier as defined by FSAR Tier 2, Chapter 16, TS 3.6, "Containment Systems." Details of containment leak rate testing and the inspection and functional testing of primary containment are in FSAR Tier 2, Section 6.2.6 and Chapter 16, TS 5.5.15. Design information on containment isolation is provided in FSAR Tier 2, Section 6.2.4. Valves in the containment building ventilation system that are associated with containment purging operations close within 5 seconds of the initiation signal, which is within the time assumed in the DBA radiological consequences analyses in FSAR Tier 2, Section 15.0.3. Primary containment isolation satisfies SRP Section 6.5.3 Acceptance Criterion 1, which states that containment isolation methods and times must be such that the calculated radiological doses resulting from the escape of radioactive material prior to and following isolation after a LOCA do not exceed the applicable dose criteria of 10 CFR 52.47(a)(2) and GDC 19.

6.5.3.4.1.1 *Natural Deposition*

The U.S. EPR DBA radiological consequences analyses take credit for natural deposition inside primary containment during a DBA LOCA and control rod ejection accident (REA). The radiological evaluations are based on the guidance in SRP Section 15.0.3, "Design Basis Accident Radiological Consequence Analyses for Advanced Light Water Reactors," and RG 1.183. Detailed review of natural deposition as modeled in the U.S. EPR DBA dose analyses is discussed in Section 15.0.3 of this report.

The U.S. EPR primary containment is designed to contain the energy released from the RCS in the event of a LOCA or REA. The mass and energy released into the containment is from the reactor coolant. Coolant released from the primary system causes an increase in containment steam mass, which in turn increases containment pressure and temperature. This rise is limited by steam cooling and condensing on contact with colder surfaces. The passive heat sink inside the primary containment consists of all painted and unpainted concrete and metal surfaces that are exposed to the primary containment atmosphere. The specific passive heat sinks considered in the containment pressure-temperature analysis and their parameters are given in FSAR Tier 2, Table 6.2.1-5, "Containment Heat Sink Inventory." A conservative minimum heat sink surface area was considered.

As described above, water vapor condenses on contact with the internal surfaces during a LOCA or REA. Natural processes that result in deposition of fission products on the surfaces inside containment are credited for fission product removal from the containment atmosphere in the DBA LOCA and REA radiological consequences analyses. Reduction in airborne radioactivity in the containment by natural deposition within the containment is acceptable per RG 1.183, Appendix A, Section 3.2. Natural deposition of radioactive particulates and elemental iodine on surfaces within containment is addressed in FSAR Tier 2, Section 15.0.3.11, "Loss of Coolant Accident." The U.S. EPR deposition model is based on natural deposition models in NUREG/CR-6604, "RADTRAD: A Simplified Model for Radionuclide Transport and Removal and Dose Estimation," NRC, December 1997, and Supplements (Supplement 1, June 1999, and Supplement 2, October 2002).

The amount of natural deposition of particulates is based on a combination of the Powers and Henry models in the RADTRAD software, as shown in FSAR Tier 2, Table 15.0-52, "Effective Natural Deposition Decontamination Coefficients." Particulate removal was assumed to go on indefinitely, although the removal rate decreases with time. The deposition removal constants for particulates were conservatively applied to elemental iodines but were limited to a total decontamination factor of 100. These assumptions on particulate and elemental iodine removal by natural processes conform to the guidance in RG 1.183. Detailed discussion of the staff's review and acceptance of the modeling of aerosol and iodine removal in containment for the LOCA and REA is included in Section 15.0.3.4.10 of this report in conjunction with the DBA radiological consequences review.

As described, the U.S. EPR crediting of natural deposition as a means of fission product control is in accordance with the guidance in RG 1.183 and SRP Section 15.0.3, and is therefore acceptable.

6.5.3.4.1.2 *Trisodium Phosphate Dodecahydrate (TSP-C) Addition*

Following a DBA LOCA and subsequent large release of radioactive materials into the containment, the pH of ECCS recirculation inventory inside containment is automatically adjusted to greater than or equal to 7.0 to enhance iodine retention in the IRWST water. Trisodium phosphate dodecahydrate is passively added to the IRWST during the ECCS recirculation phase of LOCA mitigation. This chemical addition buffers effects of ECCS injected boric acid and acids produced in the post-LOCA environment (nitric acid from the irradiation of water and air and hydrochloric acid from irradiation and decomposition of electric cable insulation). Controlling pH to greater than or equal to 7.0 significantly reduces formation of elemental iodine in the containment water, which reduces the production of organic iodine and the total airborne iodine in the containment. This pH adjustment is also provided to prevent stress corrosion cracking (SCC) of safety-related containment components during long term cooling. The determination of the quantity of TSP-C credited to control pH is based on the RG 1.183 alternative source term methodology. The LOCA radiological consequences analysis takes credit for iodine retention in the IRWST.

As described above, the use of TSP-C as a means of controlling pH and retaining gaseous iodine in solution for fission product control is acceptable. Additional information on water chemistry control is provided in FSAR Tier 2, Section 15.0.3.12, “Postaccident Reactor Building Water Chemistry Control.” The staff’s evaluation is given in Section 15.0.3 of this report.

ITAAC: FSAR Tier 1, Table 2.2.2-3, “IRWST Inspections, Tests, Analyses, and Acceptance Criteria,” specifies the inspections, tests, analyses, and acceptance criteria for the IRWST.

Technical Specifications: The pH adjustment satisfies Criterion 3 of 10 CFR 50.36(d)(2)(ii). TS 3.6.8 ensures that minimum TSP-C is available so that the gaseous iodine contribution to the LOCA dose analysis remains valid.

6.5.3.4.1.3 *Containment Building Ventilation System*

The CBVS controls the building pressurization to reduce spreading of contamination and provides filtration to reduce radioactive contamination inside the equipment compartment. It also provides purge capability and a minimal air change rate to the RCB. Containment purging is possible during power operation using the low flow purge subsystem of the CBVS. During low flow purging operations, the CBVS exhaust is aligned to ESF filters to filter potential radioactive releases. The CBVS low flow purge exhaust can also be directed to the safeguard building controlled area ventilation system ESF iodine filtration trains. The CBVS maintains a negative pressure in the CB relative to the environment when the CBVS low flow purge subsystem is operating. Upon receipt of a containment isolation signal, the containment purge path is isolated within 5 seconds. The DBA LOCA and REA dose analyses conservatively assume isolation after 10 seconds.

The containment purge subsystem satisfies TMI Requirements (10 CFR 50.34(f)(2)(xv)) and Generic Issues (NUREG-0933) to provide capability for a containment purging/venting design that minimizes containment purge time. Additional information on the CBVS is provided in FSAR Tier 2, Section 9.4.7.2, “System Description.”

The CBVS provides the following safety-related functions:

- Upon receipt of a containment isolation signal, the CBVS provides automatic isolation of the containment atmosphere by quick closure of the system containment isolation valves. The containment isolation valves are automatically closed within 5 seconds upon receipt of a containment isolation signal, in accordance with NUREG-0800, BTP 6-4, Revision 3, “Containment Purging During Normal Plant Operations,” to maintain the integrity of the containment boundary and to limit the potential release of radioactive material.

- The CBVS low flow purge exhausts to iodine filtration trains. The containment purge subsystem is designed in accordance with ASME AG-1-2003 and RG 1.52 for atmospheric cleanup.

The LOCA radiological consequences sequence of events is based on the guidance in RG 1.183. The LOCA is assumed to occur coincident with a loss of offsite power (LOOP) and the sequence of events is given in FSAR Tier 2, Table 15.0-49, “LOCA Radiological Sequence of Events,” which includes the time-phase releases as described in RG 1.183, Table 4.

The CBVS release from the RCB following a DBA LOCA to the environment is based on the following conservative assumptions:

- The containment is assumed to be in the CBVS low flow purge mode at the start of the accident. The purge flow is directed to the main stack and is conservatively assumed to be isolated within 10 seconds. Exhaust filtration is not credited.

- After purge-flow termination, airborne contaminated leakage from the primary containment is based on the proposed limit of 0.25 percent per day for the first 24 hours, and then a 50 percent reduction in this leakage value is applied after 24 hours as allowed by RG 1.183. No credit is taken for dose reduction due to holdup outside primary containment.

- For the initial 305 seconds of the accident, the pressures in the Annulus, Safeguard, and Fuel Buildings (FSAR Tier 2, Table 6.2.3-2, “Secondary Containment Response Analysis”) are being reduced (drawn down). Therefore, all of the primary containment leakage is assumed to be released to the environment unfiltered, at the closest point to the MCR intakes, adjacent to the SG 3 silencer.

- After drawdown is completed, primary containment leakage is directed through the dedicated ESF system with 99 percent efficiency filtration credited for all iodine species and is modeled as being released at the base of the main stack.

Based upon the above description, the staff concludes that the U.S. EPR CBVS satisfies regulatory requirements for fission product control by containing and processing airborne contamination prior to release to the environment under DBA conditions. Further discussion and conclusions of the staff’s review of the CBVS as an ESF atmosphere cleanup system is given in Section 6.5.1 of this report.

ITAAC: FSAR Tier 1, Table 2.6.8-4, “Containment Ventilation System Inspections, Tests, Analyses, and Acceptance Criteria,” specifies the inspections, tests, analyses, and acceptance criteria for the CBVS.

Technical Specifications: The CBVS is not an engineered safety feature and has no safety-related function except for containment isolation and low-flow purge. Containment low flow purge isolation is part of containment isolation valve testing in accordance with FSAR Tier 2, Chapter 16, TS 3.6.3 and SR 3.6.3.2.

6.5.3.4.2 Secondary Containment

Additional fission product control is provided within the annulus between the RSB and the RCB. When operating post-DBA as described in FSAR Section 6.2.3, in the areas within annulus that contain containment penetration, the AVS, forms a secondary containment. Per SRP Section 6.5.3, Acceptance Criterion 2, to be classified as a secondary containment for the purpose of fission product control, a structure or structures should completely surround the primary containment, and should be held at a pressure of at least 6.35 mm (0.25 in.) water gauge below adjacent regions, under all wind conditions up to the wind speed at which diffusion becomes great enough to ensure site boundary exposures will be less than those calculated for the design-basis accidents even if exfiltration occurs. The RSB is a 70 m (230 ft) high cylindrical structure with a dome roof 57 m (186 ft) in diameter. The RSB completely encloses the RCB, creating an annular region approximately 1.8 m (5 ft 11 in.) in width, which is maintained at a negative pressure by the AVS discussed below. The RSB is surrounded by SBs 1, 2, 3, 4, and by the FB, which are Seismic Category I safety-related structures. The RSB structure is described in FSAR Tier 2, Section 3.8.4.1.1.

The staff's review of the applicant's information raised several questions regarding secondary containment provisions. Therefore, in RAI 233, Question 06.05.03-1, the staff requested that the applicant provide the assumed leakage rates into the annulus, programs to test secondary containment inleakage, mixing fractions to be applied to the annulus area, and maximum wind speeds at which negative pressures can be maintained.

In July 10, 2009, and September 1, 2009, responses to RAI 233, Question 06.05.03-1, the applicant stated:

- The inleakage from the surroundings, including the Safeguard Buildings and Fuel Building, is 0.2 percent of the total primary containment volume per day. This is conservative as the secondary containment (Shield Building) wall is approximately 1.5 ft thicker than the primary containment wall, is not subject to the expansion during a loss of coolant accident that the primary containment wall may experience, and includes the total containment volume as opposed to the containment free volume.

- 10 CFR Part 50, Appendix J, Paragraph IV.B, 'Special Testing Requirements,' specifies in part that other structures of multiple barrier containments (such as secondary containments/shield buildings for pressurized water reactors) shall be subject to individual tests in accordance with the procedures specified in the Technical Specifications, or associated Bases. U.S. EPR Technical Specification 3.6.6, 'Shield Building,' specifies testing to fulfill this requirement.

- No mixing is assumed to occur within the annulus volume.

- The U.S. EPR Shield Building (secondary containment) is a tightly-fitted, axisymmetric, reinforced concrete structure completely surrounding the Reactor Building (primary containment) with no penetrations exposed to the environment. The Shield Building is further surrounded by the Nuclear Island (NI) buildings. Such a structure is not subject to the wind- and buoyancy-driven exchanges with the environment of the kind envisioned by RG 1.183, Regulatory Position 4.3. The NI configuration is shown in FSAR Tier 2, Figure 1.2-1 − 3-Dimensional Conceptual Configuration of U.S. EPR Buildings and Figure 1.2-2 − U.S. EPR Cutaway.

The applicant's use of 0.2 percent of the total primary containment volume is acceptable to the staff given the shield building is a tightly-fitted, axisymmetric, reinforced concrete structure and none of the penetration are exposed to the environment. Therefore, the staff finds the applicant's responses acceptable and considers RAI 233, Question 06.05.03-1 resolved.

The annulus area provides access for personnel to inspect the outside of the RCB, and to route piping, ventilation ducts, electrical cables, and other items. As discussed in FSAR Tier 2, Section 6.2.6.5, "Special Testing Requirements and Bypass Leakage," the RCB and RSB contain penetrations for mechanical, electrical, and instrumentation systems, as well as air locks and hatches. These penetrations provide a flow path by which atmosphere from the containment may leak when pressurized during accident conditions. Two categories of leakage paths are evaluated. The first category includes penetrations and seals that terminate within the secondary containment volume. The second category includes penetrations and seals that terminate outside the secondary containment.

The first category of leakage is processed by the AVS. The AVS maintains a slight vacuum in the secondary containment during normal and accident conditions. This capability is verified via TS Surveillances: SR 3.6.6.1, SR 3.6.7.1, SR 3.6.7.2, and SR 3.6.7.3. This provides assurance that the AVS can effectively draw on the air in the Safeguard and Fuel Building areas in order to direct any primary containment leakage to the ESF filters. A leak-off system collects leakage from airlock door seals, ventilation system containment isolation valves, and fuel transfer tube isolation valves.

The AVS and the leak-off system are discussed in FSAR Tier 2, Section 6.2.3. The leak-off system is part of the secondary containment bypass leakage barrier and is leak rate tested in accordance with 10 CFR Part 50, Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors."

The second category of leakage is considered bypass leakage. Penetrations and seals that exit the annulus region of the RSB structure terminate in either the Fuel Building or one of the four Safeguard Buildings. The U.S. EPR design has no primary or secondary containment penetrations or seals that terminate directly to the environment. Leakage through penetrations and seals that terminate in the secondary containment do not become bypass leakage during normal or accident operation modes, because airborne leakage is processed by the AVS before release.

The AVS controls and removes fission products that leak from the primary containment following a DBA. The AVS maintains the annulus at a slightly negative pressure to prevent leakage from the annulus through the RSB to the environment. The discharge of the AVS is automatically aligned through its ESF filter trains by a containment isolation actuation signal for the DBA. The ventilation systems for the Fuel Building and Safeguard Buildings are provided with the SBVS ESF filter system to process secondary containment bypass leakage. The annulus bypass leakage into the Fuel Building and Safeguard Buildings is automatically processed through the SBVS ESF filter trains by a containment isolation actuation signal for the DBA. The ESF filtration trains are discussed in FSAR Tier 2, Section 6.5.1. The staff's review of the AVS, FBVS, and SBVS as ESF atmosphere cleanup systems is discussed in Section 6.5.1 of this report. As discussed in Section 6.5.1 of this report, the staff has requested further information regarding the structures credited to perform secondary containment functions.

Secondary containment response to a DBA LOCA is tabulated in FSAR Tier 2, Table 6.2.3-2.

ITAAC: FSAR Tier 1, Table 2.1.1-7 specifies the inspections, tests, analyses, and associated acceptance criteria for the RCB containment structure.

Technical Specifications: Secondary containment operability is dependent on maintaining a negative pressure in the annulus and RSB integrity as defined by TS 3.6.6. Establishment of this pressure is confirmed by SR 3.6.6.3, which demonstrates that the annulus can be drawn down to a negative pressure ≥6.36 mm (0.25 in.) water gauge using one AVS train. The staff noted that the maximum nominal AVS fan flow was stated to be 33.3 m^3/min (1,177 cfm) in FSAR Tier 2, Table 6.2.3-1. However, FSAR Tier 2, Chapter 16, TS Bases for SR 3.6.6.3 and SR 3.6.6.4 stated the AVS fan flow to satisfy the surveillance requirement must be ≤37.4 m^3/min (≤1,320 cfm). Therefore, in RAI 89, Question 06.02.03-1, the staff requested that the applicant explain which AVS train flow rate was correct. In an October 31, 2008, response to RAI 89, Question 06.02.03-1, the applicant clarified that the AVS fan design air flow is 1.7 - 33.3 m^3/min (60 - 1,177 cfm), as stated in FSAR Tier 2, Table 6.2.3-1. The FSAR Tier 2, Chapter 16, SR 3.6.6.4 and TS 5.5.10 are based on a nominal AVS fan flow of 1,177 cfm. The values given in TS 5.5.10 for AVS are ≥30.0 m^3/min and ≤36.7 m^3/min (≥1,060 cfm and ≤1,295 cfm), which are ± 10 percent of the nominal AVS fan flow. The applicant stated that the AVS fan flow rate stated in FSAR Tier 2, Chapter 16 TS Bases for SR 3.6.6.3 and SR 3.6.6.4 will be changed from ≤37.4 m^3/min (≤1,320 cfm) to ≤36.7 m^3/min (≤1,295 cfm). The proposed Technical Specifications for the U.S. EPR are in conformance with SRP Section 16 and NUREG-1431, "Standard Technical Specifications for Westinghouse Plants." Based on review of the RAI response and the proposed clarification, the staff finds that the proposed technical specifications would adequately demonstrate secondary containment operability as it applies to the capability of establishing and maintaining a negative pressure. In order to ensure that the proposed changes to the TS are made, **RAI 89, Question 06.02.03-1 is being tracked as a confirmatory item.**

6.5.3.5 *Combined License Information Items*

No applicable items were identified in the FSAR.

6.5.3.6 *Conclusions*

Several plant features serve to reduce or limit the release of fission products during normal operation and following a postulated accident. These systems include the containment structures, ventilation systems, and ESF filter systems. The U.S. EPR DBA radiological consequences analyses take credit for fission product removal and control systems and structures. The U.S. EPR DBA dose analyses also incorporate credit for passive fission product removal in the primary containment, in accordance with RG 1.183 guidance on DBA radiological consequences analyses using an alternative source term. The FSAR discusses the performance capability of each system used for fission product control, including operation following a design-basis accident.

For the reasons described above, the staff finds the FPCS design meets the guidance of RG 1.52, Regulatory Positions C.1, C.2, and C.3, meets the recommendations in SRP Section 6.5.3 and is acceptable. Accordingly, the U.S. EPR FPCS design is in compliance with the applicable GDC of 10 CFR Part 50, Appendix A and satisfies the applicable requirements of 10 CFR 52.47. In addition, the staff reviewed the ITAAC and Technical Specifications for the FCPS and as described above finds that the FPCS design can be properly inspected, tested, and operated in accordance with regulatory requirements.

6.6 Inservice Inspection of ASME Class 2 and 3 Components

6.6.1 Introduction

Inservice inspection (ISI) programs are based on the requirements of 10 CFR 50.55a, which requires that ASME Code Class components meet the applicable inspection requirements set forth in Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," of the ASME B&PV, hereinafter, "the ASME Code." ISI includes preservice inspection (PSI) prior to initial plant startup. The details of the ISI program will be evaluated by the staff as part of its review of an applicant's COL application.

6.6.2 Summary of Application

FSAR Tier 1: There are no FSAR Tier 1 entries for this area of review. The system-based descriptions of FSAR Tier 1, Chapter 2, "System Based Design Descriptions and ITAAC," address ASME design-related Code requirements for system components.

FSAR Tier 2: The applicant has provided an FSAR Tier 2 description of its ISI program for ASME Class 2 and 3 components in Section 6.6, summarized here, in part, as follows:

ASME Code Class 2 and 3 systems, which correspond to RG 1.26, "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants," Quality Group B and C, respectively, are designed to permit periodic inspection and testing of important components and piping to assure the integrity and capability of the systems. The code of record for PSI and ISI at the design certification stage is the 2004 Edition of the ASME Code (no addenda). ASME Code Class 2 pressure retaining components and their welded attachments are pressure tested and are inspected by visual, surface and volumetric examination techniques. Class 2 components that meet specified criteria are exempt from the volumetric examination requirements of the ASME Code. ASME Code Class 3 pressure retaining components and their welded attachments are pressure tested and are inspected by visual examination techniques. ASME Code Class 3 components that meet specified criteria are exempt from the VT-1 visual examination requirements of the ASME Code. Systems and components were designed for ease of inspection and full ASME Code compliance. No exemptions to or relief from code requirements are requested, or code cases invoked, for ASME Code Class 2 or 3 PSI or ISI requirements at the design certification stage.

The application addresses:

- Examination techniques and procedures (e.g., visual, liquid penetrate, magnetic particle, eddy current, ultrasonic, radiography)
- Inspection intervals
- Examination categories and requirements
- Evaluation of examination results
- System pressure tests
- Augmented ISI to protect against high-energy piping failures

In each of these areas, the application references the applicable ASME Code requirements. A COL applicant that references the U.S. EPR design certification will identify the implementation milestones for the site-specific ASME Code Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," PSI and ISI program for ASME Code Class 2 and 3 components, in compliance with the requirements of 10 CFR 50.55a(g). The program will identify the applicable edition and addenda of the ASME Code Section XI and will identify additional relief requests and alternatives to ASME Code requirements. (See Combined License Information Item No. 6.6-1)

ITAAC: There are no ITAAC items for this area of review.

Technical Specifications: There are no Technical Specifications for this area of review. However, TS 5.5.8, “Steam Generator (SG) Program,” describes the steam generator program which uses the ISI program to define the “as-found” condition of SG tubes to determine which SG tubes require plugging and to establish the SG tube inspection intervals.

6.6.3 Regulatory Basis

The relevant requirements of NRC regulations for this area of review, and the associated acceptance criteria, are specified in NUREG-0800, Section 6.6, “Inservice Inspection and Testing of Class 2 and 3 Components,” and are summarized below. Review interfaces with other SRP sections also can be found in NUREG-0800, Section 6.6.

1. GDC 36, “Inspection of Emergency Core Cooling System,” as it pertains to designing the emergency core cooling system to permit appropriate periodic inspection of important components, such as spray rings in the reactor pressure vessel.

2. GDC 37, “Testing of Emergency Core Cooling System,” as it pertains to designing the emergency core cooling system to permit appropriate testing to assure structural integrity, leak tightness, and the operability of the system.

3. GDC 39, “Inspection of Containment Heat Removal System,” as it pertains to designing the containment heat removal system to permit inspection of important components, such as the torus and spray nozzles to assure the integrity and capability of the system.

4. GDC 40, “Testing of Containment Heat Removal System,” as it pertains to designing the containment heat removal system to permit appropriate pressure and functional testing.

5. GDC 42, “Inspection of Containment Atmosphere Cleanup Systems,” as it pertains to designing the containment atmospheric clean up system to permit appropriate inspection of components such as filter frames and ducts.

6. GDC 43, “Testing of Containment Atmosphere Cleanup Systems,” as it pertains to designing the containment atmospheric clean up system to permit appropriate periodic pressure and functional testing to assure structural integrity of components and the operability and performance of active components of the system, such as fans, filters, and dampers.

7. GDC 45, “Inspection of Cooling Water Systems,” as it pertains to designing the cooling water system to permit appropriate periodic inspection of important components, such as heat exchangers.

8. GDC 46, “Testing of Cooling Water Systems,” found in 10 CFR Part 50, Appendix A, as it pertains to designing the cooling water system to permit appropriate periodic pressure and functional testing to assure structural and leak-tight integrity of its components.

9. 10 CFR 50.55a, “Codes and Standards,” as it pertains to specification of the preservice and periodic inspection and testing requirements of the ASME Code for Class 2 and 3 systems and components.

6.6.4 Technical Evaluation

The staff reviewed FSAR Tier 2, Section 6.6, “Inservice Inspection of ASME Code Class 2 and 3 Components,” in accordance with SRP Section 6.6. The ASME Code of record (edition) for the design of the U.S. EPR is the 2004 edition of the ASME B&PV, as stated in FSAR Tier 2, Section 5.2.1.1, “Compliance with 10 CFR 50.55a.”

6.6.4.1 *Components Subject to Inspection*

The SRP states that the applicant’s definition of ASME Code Class 2 and 3 components and systems subject to an ISI program is acceptable if it is in agreement with the NRC quality group classification system or the definitions in ASME Code, Section III, Article NCA-2000. In addition, exceptions to the testing of specified components under ASME Code, Section XI requirements will be identified and properly justified.

FSAR Tier 2, Section 6.6.1, “Components Subject to Examination,” states that ASME Code, Section III, Subsubarticle NCA-2130 presents the construction requirements for ASME Code Class 2 and 3 components, and ASME Code, Section XI defines the PSI and ISI examination requirements. The design follows the ASME Code Section III as required by 10 CFR 50.55a. The ASME Code Class 2 and 3 components subject to inspection are in agreement with definitions acceptable to the staff in ASME Code, Section III, Article NCA-2000. In addition, no exceptions to testing requirements were identified. The components subject to inspection meet the acceptance criteria of the SRP, and are therefore acceptable.

6.6.4.2 *Accessibility*

The SRP acceptance criteria states that the design and arrangement of ASME Code Class 2 and 3 systems should include allowances for adequate clearances to conduct the examinations specified in ASME Code, Section XI, Articles IWC-2000 and IWD-2000 at the frequency specified. The design and arrangement of system components are acceptable if adequate clearance is provided in accordance with ASME Code, Section XI, Subarticle IWA-1500. Based on the SRP acceptance criteria, special design consideration should be given to those systems that are intended to be examined during normal reactor operation. In RAI 81, Question 06.06-1, the staff requested that the applicant provide additional information in the FSAR to include consideration of plant operational configurations if radiography is to be used as a supplement to obtain 100 percent volumetric examination of welds susceptible to primary water stress-corrosion cracking (PWSCC) such as austenitic and dissimilar metal welds. The staff concern was that the design may not in all cases enable the performance of two-sided access for ultrasonic examination during plant operation.

In a November 3, 2008, response to RAI 81, Question 06.06-1, the applicant stated that FSAR Tier 2, Section 6.6.2, “Accessibility,” states that the U.S. EPR design provides ready access to systems, structures, and components to accommodate comprehensive inspections using currently available equipment and techniques. It also states that “the components and welds requiring ISI have design features that allow ready inspection, including …two-sided access….” Therefore, any design changes of U.S. EPR components by the COL applicant that would change inspection accessibility, including configuration changes that would preclude draining affected piping when required for inspection, must include a discussion of the departure from FSAR Tier 2, Section 6.6.2. The staff concludes from the applicant’s Novermber 3, 2008, response to RAI 81, Question 06.06-1, that sufficient consideration of ISI is included in the design to assure that ASME Code Class 2 and 3 components can be adequately examined in accordance with 10 CFR 50.55a(g)(3)(ii). In addition, any departure from the design is, in fact, the responsibility of the COL applicant/holder to address in its application. ASME Code Class 2 and 3 components are designed in accordance with 10 CFR 50.55a and ASME Code, Section XI, Subarticle IWA-1500, meeting the SRP acceptance criteria, and are therefore acceptable.

6.6.4.3 *Examination Categories and Methods*

The examination categories and methods specified in the FSAR are acceptable if they agree with the criteria in ASME Code, Section XI, Article IWB-2000, "Examination and Inspection." Every area subject to examination which falls within one or more of the examination categories in Article IWB-2000 must be examined, at least to the extent specified. The requirements of Article IWB-2000 also list the methods of examination for the components and parts of the pressure-retaining boundary.

The applicant's examination techniques and procedures used for preservice inspection or ISI of the system are acceptable if they meet the following criteria:

- The methods, techniques, and procedures for visual, surface, or volumetric examination are in accordance with ASME Code, Section XI, Article IWA-2000.

- Alternative examination methods, combinations of methods, or newly developed techniques to those above are acceptable provided that the results are equivalent or superior, in accordance with 10 CFR 50.55a(a)(3).

- The methods, procedures, and requirements regarding qualification of personnel performing ultrasonic examination reflect the requirements provided in ASME Code, Section XI, Division 1, Appendix VII, "Qualification of Nondestructive Examination Personnel for Ultrasonic Examination."

- The performance demonstration for ultrasonic examination procedures, equipment, and personnel used to detect and size flaws reflects the requirements provided in Appendix VIII, "Performance Demonstration for Ultrasonic Examination Systems," to Division 1 of ASME Code, Section XI.

FSAR Tier 2, Section 6.6.3, "Examination Techniques and Procedures," discusses examination techniques, categories, and methods. The dye penetrant test (PT) method or the magnetic particle method is used for surface examinations. Radiographic, ultrasonic, or eddy current techniques (manual or remote) are used for volumetric examinations. The visual, surface, and volumetric examination techniques and procedures agree with the requirements of ASME Code, Section XI, Articles IWA-2000, IWC-2000, and IWD-2000. The examination acceptance standards are in accordance with ASME Code, Section XI, Articles IWC-3000 and IWD-3000.

FSAR Tier 2, Section 5.2.1.1, indicates that the baseline ASME Code used for the evaluation done in support of the safety analysis report and the design certification is the 2004 Edition of the ASME Code, Section XI. This edition and addenda will be used by non-destructive examination (NDE) personnel for ultrasonic examination and the implementation of Appendix VIII for performance demonstration for ultrasonic examination of reactor pressure boundary piping, reactor vessel (RV) welds, and RV head bolts.

Because the examination methods and categories applied to ASME Code Class 2 and 3 components will comply with the SRP acceptance criteria and the requirements of ASME Code, Section XI, the staff finds examination categories and methods for the U.S. EPR design of ASME Code Class 2 and 3 components acceptable.

6.6.4.4 *Inspection Intervals*

The required examinations and pressure tests must be completed during each 10-year interval of service, hereinafter designated as the "inspection interval." In addition, the scheduling of the program must comply with the provisions of ASME Code, Section XI, Articles IWA-2000, IWC-2000, and IWD-2000, as related to inspection intervals of the ASME Code, Section XI. FSAR Tier 2, Section 6.6.4, "Inspection Intervals," discusses inspection intervals. ASME Code, Section XI, Subarticles IWA-2400, IWC-2400, and IWD-2400 define inspection intervals. The inspection intervals specified for the U.S. EPR components conform to the definitions in ASME Code, Section XI, and therefore are acceptable. However, the applicant states that it is not necessary that the inspection intervals for the ASME Code Class 1 portions of the ISI Program conform to the same inspection programs as those for the ASME Code Class 2 and 3 inspections. Similarly, FSAR Tier 2, Section 6.6.4 states that the inspection intervals do not need to be the same for ASME Code Class 2 and 3 components. The staff recognizes that an interval may be extended by up to a year, but in practice, that extension has applied to all components within the ISI program. The staff expressed concerns that the applicant's interpretation of the regulations and the ASME Code is not consistent with the implementation of inspection intervals used in operating reactors. Therefore, in RAI 81, Question 06.06-2, the staff requested that the applicant provide clarification.

In a November 3, 2008, response to RAI 81, Question 06.06-2, the applicant stated that the regulations do not preclude a licensee from applying different inspection intervals for the ASME Code Class 1, 2, and 3 components. The applicant also quoted ASME Code, Section XI, Paragraph IWA-2430, which states: "It is not required that the inspection intervals of ASME Code, Section XI, IWB, IWC, IWD, IWE, and IWF conform to the same inspection program." The staff does not fully agree with the applicant's interpretation of both 10 CFR 50.55a and IWA-2430. However, the guidance for designating ISI program changes does not belong in a design certification document, because the issue is not design related, but rather, additional details of an operational program and is the responsibility of the COL applicant or holder. In follow-up RAI 208, Question 05.02.04-6, the staff requested that the applicant remove these statements from FSAR Tier 2, Sections 5.2.4.1.3, "Inspection Intervals," and 6.6.4. In a May 8, 2009, response to RAI 208, Question 05.02.04-06, the applicant indicated that it would remove the statement, "It is not necessary that the inspection intervals be the same for the IWC (Class 2) and the IWD (Class 3) portions of the ISI program." The staff concludes that the proposed changes to FSAR Tier 2, Section 6.6.4, as shown in the FSAR mark-up, will bring the FSAR into compliance with ASME Code, Section XI in this respect, and is therefore acceptable. **RAI 208, Question 05.02.04-6 is being tracked as a confirmatory item.**

6.6.4.5 *Evaluation of Examination Results*

The SRP states that the standards for examination evaluation in the program for flaw evaluation are acceptable if in agreement with the requirements of ASME Code, Section XI, Article IWC-3000, and IWD-3000, "Acceptance Standards." The SRP also states that the proposed program regarding repairs of unacceptable indications or replacement of components containing unacceptable indications is acceptable if it is in agreement with the requirements of ASME Code, Section XI, Article IWA-4000, "Repair/Replacement Activities." The criteria that establish the need for repair or replacement are described in ASME Code, Section XI, Article IWB-3000, "Acceptance Standards."

FSAR Tier 2, Section 6.6.6, “Evaluation of Examination Results,” discusses the evaluation of examination results. Examination results are evaluated according to ASME Code, Section XI, IWA-3000, IWC-3000, and IWD-3000, with flaw indications being evaluated according to Table IWC-3410-1. Repair procedures, if required, are evaluated according to ASME Code, Section XI, IWA-4000. Based on this method of evaluating examination results, and the use of the appropriate ASME Code rules for repair, the applicant’s evaluation of examination results for U.S. EPR components meets the SRP acceptance criteria, and is therefore acceptable.

6.6.4.6 *System Pressure Tests*

The pressure-retaining ASME Code Class 1 component leakage and hydrostatic pressure test program is acceptable if the program is in accordance with the requirements of the ASME Code, Section XI, Articles IWC-5000 and IWD-5000. FSAR Tier 2, Section 6.6.7, “System Pressure Tests,” states that ASME Code Class 2 and 3 systems and components are pressure tested in accordance with ASME Code, Section XI, Articles IWC-5000, IWD-5000, and IWA-5000 (general requirements) of the ASME Code. Based on the applicant’s use of the appropriate sections of the ASME Code that meet the SRP acceptance criteria for system pressure tests, the staff concludes that the applicant’s methodology is acceptable.

6.6.4.7 *Augmented ISI to Protect Against Postulated Piping Failures*

The SRP states that the augmented ISI program for high-energy fluid system piping between containment isolation valves is acceptable if:

- Access is provided in order to enable the performance of ISI examinations.
- During each inspection interval, 100 percent of the circumferential and longitudinal welds are examined with the boundary of the piping.
- Inspection ports are provided if access is restrained due to guard pipes.
- The areas subject to examination should be defined in accordance with Article IWC-2000, Examination Category C-F for Class 2 piping welds.

FSAR Tier 2, Section 6.6.8, “Augmented ISI to Protect against Postulated Piping Failures,” states that access provisions are incorporated into the U.S. EPR design to allow access for personnel and equipment to inspect the affected welds. Hand holes, ports, or removable sections of guard pipe are provided as necessary to enable examination. One-hundred percent volumetric examination of the circumferential and longitudinal welds during the inspection interval is performed in accordance with the requirements of ASME Code, Section XI, IWC-2000 for Examination Category C-F welds. Accordingly, the staff concludes the U.S. EPR design incorporates the access provisions, correct ASME Code methodology and frequency in accordance with the SRP acceptance criteria for augmented ISI to protect against postulated piping failures, and is therefore acceptable.

6.6.4.8 *Code Exemptions, Relief Requests, and Code Cases*

FSAR Tier 2, Section 6.6 states that no exemptions to or relief from code required examinations are requested, or code cases invoked, for ASME Code Class 2 or 3 preservice or inservice inspection requirements for the U.S. EPR design. The staff concludes that the SRP acceptance criteria are not applicable. Any exemptions, relief requests to the ASME Code and ASME Code Cases invoked are the responsibility of the COL applicant.

6.6.4.9 *General Design Criteria*

GDC 36, GDC 37, GDC 39, GDC 40, GDC 42, GDC 43, GDC 45, and GDC 46 require that the respective safety systems addressed by these criteria be designed such that they permit periodic inspection pressure testing and functional testing of system components and piping. Compliance with these GDC ensures that the design of the safety systems will allow accessibility of important components so that periodic inspections can be performed that detect degradation, leakage, signs of mechanical or structural distress caused by aging, and fatigue or corrosion, prior to jeopardizing the ability of the systems to perform their intended safety function.

As stated under Section 6.6.4.2 of this report, the staff concluded that accessibility by design enabled the performance of inservice examinations and pressure testing, thereby meeting ASME Code, Section XI, Subarticle IWA-1500 and 10 CFR 50.55a(g)(3)(ii). The staff concludes that the respective systems are designed to permit periodic inspection and testing, meeting the requirements of GDC 36, GDC 37, GDC 39, GDC 40, GDC 42, GDC 43, GDC 45, and GDC 46, and are therefore acceptable.

6.6.5 Combined License Information Items

Table 6.6-1 provides a list of inservice inspection of ASME Code Class 2 and ASME Code Class 3 components related COL information item numbers and descriptions from FSAR Tier 2, Table 1.8-2:

Table 6.6-1 U.S. EPR Combined License Information Items

Item No.	Description	FSAR Tier 2 Section
6.6-1	A COL applicant that references the U.S. EPR design certification will identify the implementation milestones for the site-specific ASME Section XI preservice and inservice inspection program for Class 2 and Class 3 components, consistent with the requirements of 10 CFR 50.55a(g). The program will identify the applicable edition and addenda of the ASME Code Section XI, and will identify additional relief requests and alternatives to Code requirements.	6.6

The staff determines the above listing to be complete. Also, the list adequately describes actions necessary for the COL applicant or holder. No additional COL information items need to be included in FSAR Tier 2, Table 1.8-2 for inservice inspection of Class 2 and 3 components consideration.

6.6.6 Conclusions

To ensure that no deleterious defects develop during service, ASME Code Class 2 system components, selected welds and weld heat-affected zones are inspected prior to reactor startup and periodically throughout the life of the plant. In addition, ASME Code Class 2 and 3 systems receive visual inspections while the systems are pressurized in order to detect leakage, signs of mechanical or structural distress, and corrosion.

The applicant has stated that the ISI program complies with the rules published in 10 CFR 50.55a and Section XI of the ASME Code, 2004 Edition. The final ISI program is required to meet the latest ASME Code, Section XI Edition/Addenda incorporated by reference 12 months before the date scheduled for initial loading of fuel. The ISI program will consist of preservice and inservice inspection plans. The staff concludes that the description of the inservice inspection program is acceptable and meets the inspection and pressure testing requirements of GDC 36, GDC 37, GDC 39, GDC 40, GDC 42, GDC 43, GDC 45, and GDC 46, and 10 CFR 50.55a. This conclusion is based on the applicant's meeting the requirements of the ASME B&PV Code, Section XI as reviewed by the staff, and determined to be appropriate for this application. The inservice testing of pumps, valves, and dynamic restraints is further discussed in Section 3.9.6 of this report.

6.7 Main Steam Isolation Valve Leakage Control System (BWR)

The MSIV leakage control system is not applicable to the U.S. EPR.

6.8 Extra Borating System

The review of the U.S. EPR extra borating system (EBS) is contained in Section 9.3.4 of this report.