6.0 Engineered Safety Features

This chapter of the United States - Advanced Pressurized-Water Reactor (US-APWR) Design Certification (DC) Safety Evaluation Report (SER) describes the U.S. Nuclear Regulatory Commission (NRC) staff's (hereinafter referred to as the staff) review (and the results thereof) of Chapter 6, "Engineered Safety Features," (ESFs) of the US-APWR Design Control Document (DCD), Revision 3, and supporting documentation, submitted by Mitsubishi Heavy Industries, Ltd. (MHI), hereinafter referred to as the applicant. ESFs mitigate the consequences of postulated accidents (PAs). Furthermore, ESFs protect the public health and safety in the unlikely event of an accidental release of radioactive fission products from the reactor coolant system (RCS).

ESFs are designed to act automatically to limit, control, and terminate unplanned events, while maintaining the radiation exposure to the public well below the applicable regulatory limits and guidelines. The following are the ESFs of the US-APWR:

- Containment system (Section 6.2 below)
- Emergency core cooling system (ECCS) (Section 6.3 below)
- Habitability system (Section 6.4 below)
- Fission product removal and control system (Section 6.5 below)

This chapter also addresses the ESF materials used in constructing and fabricating ESF components and systems, as well as inservice inspection (ISI) and inservice testing (IST) programs to address regular and periodic examinations, tests, and inspections of pressure retaining components and supports required by NRC regulations.

6.1 Engineered Safety Features Materials

6.1.1 Metallic Materials

6.1.1.1 Introduction

ESF materials 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. Also, processes for welding, non-destructive examination, and cleaning of ESF systems must be controlled to ensure initial quality and prevent deterioration. ESF systems materials reviewed in this section include those used in the ECCS, Residual Heat Removal System (RHRS), Emergency Feedwater System (EFWS), pre-stressed concrete containment vessel (PCCV), Containment Isolation System (CIS) and Containment Spray System (CSS).

6.1.1.2 Summary of Application

DCD Tier 1: The Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC) associated with Tier 2 Section 6.1.1, "Metallic Materials," are given in Tier 1 for various ESF systems including Section 2.4.4 for ECCS, Section 2.4.5 for RHRS, Section 2.7.1.11 for EFWS, Section 2.11.1 for the PCCV, Section 2.11.2 for the CIS and Section 2.11.3 for the CSS.

The aforementioned ITAAC sections address materials by verifying compliance with the applicable articles of Section III of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (the ASME Code) that include requirements for materials.

DCD Tier 2: The applicant has provided a DCD Tier 2 description in DCD, Tier 2, Section 6.1.1, summarized here in part as follows:

> Materials specifications and grades for pressure-retaining materials used in ESF systems are listed in DCD Tier 2, Table 6.1-1, “Principle Engineered Safety Feature Pressure Retaining Material Specifications.” ESF construction materials that would be exposed to core coolant and containment spray solutions in the event of a design-basis accident (DBA) are listed in Table 6.1-2, “Principle Engineered Safety Features Materials Exposed to Core Coolant and Containment Spray.”
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> ESF materials meet the applicable requirements of ASME Code, Section III. ASME materials Code Cases used for ESF components comply with Regulatory Guide (RG) 1.84 “Design, Fabrication, and Materials Code Case Acceptability, ASME Section III.”
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> All ESF components in contact with core coolants and containment spray solutions are either fabricated from stainless steel or ferritic materials clad with stainless steel. Austenitic stainless steel base materials are solution-annealed to prevent stress corrosion cracking. In addition, the guidelines in RG 1.44, “Control of the Use of Sensitized Stainless Steel,” are followed during the manufacture and construction of ESF components and structures.
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> Fracture toughness properties of ferritic materials used in ESF components meet the requirements of ASME Code, Section III, Subarticles NC/ND/NE-2300. Materials, fabrication and processing of ESF components follow the guidelines of RG 1.31, “Control of Ferrite Content in Stainless Steel Weld Metal,” RG 1.36, “Nonmetallic Thermal Insulation for Austenitic Stainless Steel,” RG 1.37, “Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants,” and RG 1.50, “Control of Preheat Temperature for Welding Low-Alloy Steel.” Welder qualification for areas of limited accessibility follows the guidance provided in RG 1.71, “Welder Qualification for Areas of Limited Accessibility.”
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> The materials used in ESF systems are selected for compatibility with core coolant and containment spray solutions as described in ASME Code, Section III, Articles NC-2160 and NC-3120. Nickel-chromium-iron alloy used in ESF systems is limited to Alloy 690 and its associated weld filler materials which have been shown to be highly resistant to stress corrosion cracking.
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> The control of welding, heat treatment, welder qualification, and contamination protection for ESF ferritic and austenitic stainless steel material fabrication described in DCD Section 5.2.3, “Reactor Coolant Pressure Boundary Materials,” is also applicable to ESF components.
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> Controls are instituted to maintain the chemistry of the reactor coolant and the borated water in the refueling water storage pit (RWSP). The RWSP water pH value is maintained at approximately 4.3. Crystalline sodium tetraborate decahydrate (NaTB) spray additive is stored in containment and is used to raise the pH of the RWSP water from 4.3 to, at least 7.0, post loss of coolant accident (LOCA). This follows the guidance

of NRC Branch Technical Position (BTP) MTEB-6.1 in Chapter 6 of NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, LWR [light-water reactor] Edition,” (the SRP) to minimize chloride-induced stress corrosion cracking of austenitic stainless steels. Chlorides and fluorides, which promote intergranular stress-corrosion cracking (IGSCC), are managed, such that their concentrations are below 0.15 parts per million (ppm). The pH of the ESF fluids is controlled during a DBA using NaTB as a buffering agent.

ITAAC: The ITAAC associated with DCD Tier 2, Section 6.1.1, “Metallic Materials,” are given in Tier 1 for various ESF systems including Section 2.4.4 for ECCS, Section 2.4.5 for RHRS, Section 2.7.1.11 for EFWS, Section 2.11.1 for the PCCV, Section 2.11.2 for the CIS and Section 2.11.3 for the CSS.

Technical Specifications (TS): There are no TS applicable to ESF metallic materials; however, related TS information can be found in Tier 2 Chapter 16, “Technical Specifications,” Section 3.5.5 for pH adjustment.

Topical Reports: There are no topical reports submitted by the applicant for this area of review.

Technical Reports: There are no technical reports submitted by the applicant for this area of review.

US-APWR Interface Issues identified in the DCD: There are no US-APWR interface issues for this area of review.

Site Interface Requirements Identified in the DCD: There are no site interface requirements for this area of review.

Cross cutting Requirements (Three Mile Island [TMI], Unresolved Safety Issue [USI]/Generic Safety Issue [GSI], Operating Experience [Op Ex]): Op Ex has demonstrated that certain nickel chromium iron alloys are susceptible to stress corrosion cracking. When necessary, nickel chromium iron alloys used in the fabrication of ESF components in the US-APWR design is limited to Alloy 690. Alloy 690 was shown to have a high resistance to stress corrosion cracking.

Regulatory Treatment of Non Safety System (RTNSS): There is no RTNSS for this area of review.

Title 10 of the *Code of Federal Regulations* (10 CFR) Part 20.1406: There are no 10 CFR 20.1406 requirements for this area of review.

Conceptual Design Information (CDI): There is no CDI for this area of review.

6.1.1.3 Regulatory Basis

The relevant requirements of the Commission’s regulations for this area of review, and the associated acceptance criteria, are given in Section 6.1.1, “Engineered Safety Features,” of NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, LWR [light-water reactor] Edition,” (the SRP) and are summarized below. Review interfaces with other SRP sections can be found in Section 6.1.1 of NUREG-0800.

1. General Design Criterion (GDC) 1, “Quality standards and records,” of Appendix A, “General Design Criteria for Nuclear Power Plants,” to 10 CFR Part 50 as it relates 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 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 nonbrittle 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. Criterion IX, “Control of Special Processes,” and Criterion XIII, “Handling, Storage, and Shipping,” of Appendix B, Quality Assurance (QA) Criteria for Nuclear Power Plants and Fuel Reprocessing Plants,” to 10 CFR Part 50 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.

8. 10 CFR 52.47(b)(1), which requires that a DC application include 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 DC, the provisions of the Atomic Energy Act of 1954, and NRC regulations.

The acceptance criteria adequate to meet the above regulatory requirements include:

1. RG 1.7, “Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident.”

2. RG 1.31, “Control of Ferrite Content in Stainless Steel Weld Metal.”

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

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

5. RG 1.44, “Control of the Use of Sensitized Steel.”

6. RG 1.50, “Control of Preheat Temperature for Welding Low-Alloy Steel.”

7. RG 1.71, “Welder Qualification for Areas of Limited Accessibility.”

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. To satisfy these requirements, 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 Boiler and Pressure Vessel Code or RG 1.84, “Design, Fabrication, and Materials Code Case Acceptability, ASME Section III.” Section III references applicable portions of ASME Code Section II, Parts A, B C and D.

US-APWR DCD Tier 2, Table 6.1-1, “Principle Engineered Safety Features Pressure Retaining Material Specifications,” and Table 6.1-2, “Principle Engineered Safety Features Materials Exposed to Core Coolant and Containment Spray,” list materials used in ESF systems. Pressure retaining components are either fabricated from austenitic stainless steel or clad with austenitic stainless steel. RCPB materials are discussed in DCD Section 5.2.3, “Reactor Coolant Pressure Boundary Materials.” The staff reviewed the materials specifications and grades, including weld filler materials, listed in the aforementioned tables and verified that the materials listed meet ASME Code Section III requirements and are, therefore, acceptable for use in the US-APWR design.

The EFWS components materials specifications and grades are listed in DCD Table 10.4.9-7, “Principle Emergency Feedwater System Materials.” The staff reviewed the materials specifications and grades, including weld filler materials, for EFWS components and found them acceptable because they meet ASME Code Section III requirements. However, the staff noticed that the applicant failed to list stainless steel weld filler materials specifications and classifications used to fabricate EFWS components. In Request for Additional Information **(RAI) 612-4828, Question 06.01.01-21**, the staff requested that the applicant modify Table 10.4.9-7 to include stainless steel weld filler material specifications and classifications used to fabricate the EFWS. In its response to RAI 612-4828, Question 06.01.01-21, dated August 25, 2010, the applicant provided a proposed revision to Table 10.4.9-7 that includes stainless steel weld filler material specifications and classifications and also adds an additional stainless steel pipe specification and grade. The staff reviewed the proposed modifications to the table and determined that the materials listed comply with ASME Code Section III and are, therefore, acceptable. The staff will verify that the appropriate changes are made to DCD Table 10.4.9-7 in DCD Revision 4. **RAI 612-4828, Question 06.01.01-21 is being tracked as a Confirmatory Item.**

6.1.1.4.2 Austenitic Stainless Steels

The US-APWR 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, of rapidly propagating failure, and gross rupture; and (3) the QA requirements of Appendix B to 10 CFR Part 50, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants." These requirements may be met 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 control of 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. DCD Subsection 6.1.1.2.2, "Controls for Austenitic Stainless Steel," states that the delta ferrite content in stainless steel weld metal will be controlled in accordance with the recommendations of RG 1.31. In addition, DCD Table 1.9.1-1, "US-APWR Conformance with Division 1 Regulatory Guides," indicates that the applicant does not take any exception to the guidance provided in RG 1.31.

RG 1.37 describes QA requirements for cleaning of fluid systems and associated components of water-cooled nuclear power plants. The applicant's standards provide for control of tools used in abrasive work operations such as grinding. Tools used in abrasive work operations on austenitic stainless steel, such as grinding or wire brushing, do not contain and are not contaminated with ferritic carbon steel or other materials that could contribute to intergranular cracking or stress-corrosion cracking (SCC). The applicant follows the guidance in RG 1.37 which is further discussed and evaluated in Section 6.1.1.4.3, "Ferritic Steel Welding," of this report. In addition, DCD Table 1.9.1-1 indicates that the applicant does not take any exception to the guidance provided in RG 1.37.

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, DCD Section 6.1.1.1, "Materials Selection and Fabrication," states that the requirements of RG 1.44 are followed during the manufacture and construction of ESF components and structures. Austenitic stainless steel base materials used in ESF components are solution annealed to prevent sensitization and stress corrosion cracking. Also, Table 1.9.1-1 indicates that the applicant does not take any exception to the guidance provided in RG 1.44. DCD Table 6.1-1 lists standard grades of austenitic stainless steels as well as low carbon austenitic stainless. Standard grades of austenitic stainless steels have a carbon content of ≤ 0.08 percent. Low carbon grades have a carbon content of ≤ 0.03 percent. Section 6.1.1.1 states that when standard grades of austenitic stainless steels are used for pressure retaining applications, materials will have a limited carbon content not exceeding 0.05 percent heat analysis and 0.06 percent product analysis. RG 1.44, Section 4, recommends that low carbon grade (≤ 0.03 percent) austenitic stainless steels be used unless the controlled concentration of dissolved oxygen has a limiting value of 0.10 ppm to prevent IGSCC. The standard dissolved oxygen content for reactor coolant water in the US-APWR is ≤ 0.005 ppm. Although the standard dissolved oxygen content for reactor coolant water in the US-APWR is far below the limit of 0.10 ppm recommended in RG 1.44, stagnant flow can cause isolated areas to have high dissolved oxygen content in excess of 0.10 ppm which could make standard grades of stainless steels susceptible to SCC. To address potential stagnant flow, the applicant stated in DCD Section 6.1.1.1 that “During the detailed design, MHI will determine if there are local areas where flow stagnation may be present resulting in dissolved oxygen content greater than 0.10 ppm in piping and components that have a normal operating temperature above 200°F (93 °C). For piping and components where the above conditions exist, stainless steel with a carbon content less than or equal to 0.03 percent will be used.” The staff finds this acceptable because the applicant will ensure that local areas where the dissolved oxygen content may be above 0.10 ppm will use low carbon stainless steel thus meeting the intent of RG 1.44.

DCD Section 6.1.1.1 states that cold-worked austenitic stainless steel is not used for pressure boundary applications. The DCD further states that if such material is used for other applications when there is no proven alternative available, cold work is controlled, measured and documented during each fabrication process. In addition, cold-worked austenitic stainless steels have a maximum 0.2 percent offset yield strength of 620 MPa (90,000 psi) to reduce the probability of stress-corrosion cracking in ESF systems. The staff finds this acceptable because it meets the guidelines listed in SRP 6.1.1, “Engineered Safety Features Materials,” for the use of cold worked austenitic material to prevent stress corrosion cracking in these materials.

6.1.1.4.3 Ferritic Steel Welding

To meet the requirements of GDC 1 related to general QA and codes and standards, Criterion IX of Appendix B to 10 CFR Part 50 for control of special processes, and 10 CFR 50.55a, for quality standards and records, the amount of minimum specified preheat must 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. For areas of limited accessibility to perform welding, RG 1.71 provides guidance acceptable to the staff to ensure the quality of welds in areas of limited accessibility.

The amount of specified preheat for the welding of carbon steel and low-alloy steel should be in accordance with ASME Code, Section III, Appendix D, Article D-1000. Appendix D is supplemented by positions described in RG 1.50 for the control of preheat temperatures for low alloy steels. The applicant stated in DCD Section 6.1.1.1 that the minimum preheat temperatures used for welding carbon and low alloy steels in ESF system will meet the guidelines listed in ASME Code Section III, Appendix D, Article D-1000. The applicant further stated in DCD Section 6.1.1.1 that the recommendations of RG 1.50 are applied during welding fabrication. The staff finds this acceptable because the applicant will conform to staff guidance for the preheating of carbon steel and low-alloy steel materials used in ESF Systems.

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. DCD Section 6.1.1.1 states that the applicant follows the guidance provided in RG 1.71. In addition, DCD Table 1.9.1-1 indicates that the applicant does not take any exception to the guidance provided in RG 1.71 as it relates to DCD Section 6.1.1.1. On that basis the staff finds that welding in areas of limited accessibility will be properly qualified following the guidance provided in RG 1.71.

DCD Subsection 6.1.1.2.1, “Compatibility of Construction Materials with Core Cooling Coolants and Containment Sprays,” states, in part: “The materials used in the fabrication of the ESF components are corrosion resistant in normal operation and the post-LOCA environment. General corrosion is negligible with the exception of low-alloy and carbon steels.” The staff requested, in **RAI 544-4267, Question 06.01.01-15**, that the applicant discuss the corrosion allowance for low-alloy and carbon steel materials and state the technical basis for the corrosion allowance for low-alloy and carbon steels used in ESF systems to ensure that it is sufficient for the design life of the plant. In its response to RAI 544-4267, Question 06.01.01-15, dated April 21, 2010, the applicant stated that the materials of ESF components, with the exception of the containment vessel liner, in contact with coolants are either fabricated from or clad with austenitic stainless steel. The applicant also stated that the carbon steels used such as the containment liner are coated and therefore general corrosion is negligible. The staff finds this acceptable because the applicant has adequately addressed the potential for general corrosion in ESF systems. **RAI 544-4267, Question 06.01.01-15, is therefore resolved and closed**.

6.1.1.4.4 Dissimilar Metal Welds

In **RAI 544-4267, Question 06.01.01-18**, the staff requested that the applicant provide additional details regarding dissimilar metals welds (DMWs) in ESF systems. In its response to RAI 544-4267, Question 06.01.01-18, dated April 21, 2010, the applicant stated that the only DMWs in the ESF systems are used to connect the carbon steel accumulator and the stainless steel outlet piping with Alloy 52/152 weld filler material. The maximum normal operating temperature of the accumulator is 120°F (49 °). Given the low operating temperature, the staff determined that it is highly unlikely that these welds will be susceptible to SCC and controls or precautions for welding, beyond ASME Code requirements, are not necessary. **RAI 544-4267, Question 06.01.01-18, is therefore resolved and closed**.

6.1.1.4.5 Composition and Compatibility of ESF Fluids

To meet the requirements of GDC 4, 14, 35 and 41, the composition of containment spray and core cooling water should be controlled to ensure a minimum pH of 7.0, as addressed in 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 Concentration in Containment.”

In order to reduce the probability of SCC of austenitic stainless steel components, containment and core coolants should be maintained at a pH level of at least 7.0. DCD Section 6.1.1.2, “Composition and Compatibility of Core Cooling Coolants and Containment Sprays,” states that the pH of ESF fluids is controlled during a DBA using NaTB as a buffering agent which results in a post-LOCA pH of at least 7.0. This is consistent with BTP 6-1 and therefore acceptable to the staff. DCD Subsection 6.1.1.2.1 refers the reader to Section 6.3.1.3, “Containment pH Control,” Subsection 6.3.2.2.5, “NaTB Baskets and NaTB Basket Containers,” and Table 6.3-5, “Safety Injection System Design Parameters,” for information regarding boric acid in the RWSP water and NaTB in the containment (these sections provide details on the location and quantity of NaTB, the boric acid concentration, and the initial and final pH). In addition, the staff performed a confirmatory calculation of the RWSP pH following a LOCA that confirmed that the pH will be adjusted to a value greater than 7.0. The acceptability of the applicant’s calculation of the post-accident pH is addressed and details of the staff’s confirmatory calculation are located in SER Section 6.5.2, “Containment Spray System,” and Section 15.0.3, “Design Basis Accident Radiological Consequence Analyses for Advanced Light Water Reactors.”

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. DCD Subsection 6.1.1.2.1, states that aluminum and zinc are materials that would yield hydrogen gas by corrosion from the emergency cooling or containment sprays solution in the containment, and its use is limited as much as practical. Although the DCD does not specify the quantities of aluminum and zinc, this is not necessary because the design of the hydrogen igniter system, which controls hydrogen, is based on the capability to ensure that containment integrity is maintained when the hydrogen ignition system is functional, assuming hydrogen generated from 100 percent fuel cladding-coolant reaction (10 CFR 50.34(f)(3)(v)(A)(1) and 10 CFR 50.44(c)(5)), as discussed in DCD Section 19.2.3.3.2, "Hydrogen Generation and Control." The hydrogen generated from the aluminum and zinc materials in containment would be insignificant compared to the hydrogen produced by the cladding-coolant reaction. Therefore, the staff finds that it is not necessary for the applicant to quantify the amount of aluminum and zinc in the containment.

DCD Section 6.1.1.2 indicates that the chemistry of the borated reactor coolant and the borated water in the RWSP is controlled such that the concentrations of chloride and fluoride are below 0.15 ppm. Section 6.1.1.2 also states that during periods of high temperatures, dissolved oxygen concentrations remain below 0.10 ppm. Section 6.1.1.2 further states that the controls on water chemistry include the chemical and volume control system (CVCS) and the spent fuel pit cooling and purification system (SFPCS), and that details on these control systems are provided in Chapter 9, "Auxiliary Systems," Section 9.3.4, "Chemical and Volume Control System," for the CVCS and in Section 9.1.3, "Spent Fuel Pit Cooling and Purification System," for the SFPCS. The staff notes that for refueling water storage tanks, the EPRI Pressurized-Water Reactor (PWR) Primary Water Chemistry Guidelines (EPRI Guidelines) recommend sampling several other parameters in addition to chloride and fluoride, such as sulfate. Also, DCD Subsection 6.1.1.2.1, states that the water quality requirements for the RCS and RWSP are described in Chapter 9, Section 9.3.4. However, Section 9.3.4 only describes the RCS water quality requirements. In **RAI 379-2756, Question 06.01.01-10**, the staff requested that the applicant address the above inconsistencies. In its response to **RAI 379-2756, Question 06.01.01-10**, dated July 10, 2009, the applicant provided the impurity limits for the RWSP. The staff reviewed the impurity limits and finds that they are consistent with the recommended limits of the EPRI Guidelines for refueling water storage tanks, with respect to chloride, fluoride, and sulfate, which are limited to less than or equal to 0.15 ppm. The applicant also provided a standard value, intended to be representative of normal operating conditions, for chloride, fluoride, and sulfate of less than or equal to 0.05 ppm. The applicant also provided limits for boron of 4000-4200 ppm which are consistent with the TS (SR 3.5.4.3). Additionally, the applicant provided a limit of 1 ppm for turbidity and a standard value recommended for analysis of less than or equal to 0.5 ppm for silica. The staff finds the silica recommended standard value acceptable because it is more stringent that the RCS silica standard value recommended for analysis.

The EPRI Guidelines do not provide a recommended value for turbidity or total suspended solids. However, turbidity is typically measured in nephelometric turbidity units (NTU) rather than ppm. Ppm would generally be an appropriate unit for total suspended solids. The EPRI Guidelines allow either turbidity or total suspended solids to be measured for refueling water storage tanks. The staff requested, in **RAI 487-3939, Question 06.01.01-11**, additional information to clarify whether the applicant intended to specify measurement of total suspended solids rather than turbidity and to request that the applicant modify their proposed Table 6.1-3, "Water Chemistry Specifications of the RWSP," accordingly. In its response to **RAI 487-3939, Question 06.01.01-11**, dated December 3, 2009, the applicant provided a proposed revision to Table 6.1-3 replacing the limit on turbidity with a limit on total suspended solids of 0.35 ppm, which is consistent with the total suspended solids limit for the RCS. The staff finds that the proposed change is acceptable and **RAI 487-3939, Question 06.01.01-11, is therefore resolved and closed**.

Regarding sodium, the EPRI guidelines recommend sampling for sodium in the refueling water storage tank if there is a possible mechanism whereby sodium could contaminate the refueling water storage tank. In its response to **RAI 379-2756, Question 06.01.01-10**, dated July 10, 2009, the applicant provided the following explanation for the absence of a limiting value for sodium:

> In US-APWR, NaTB is used at letdown line of spray system, therefore ingress of Na dose [sic] not occur. This specification is almost consistent with EPRI Guidelines except Na.

Since the meaning of the applicant's statement that "NaTB is used at letdown line of spray system," was unclear; the staff requested additional information in **RAI 487-3939, Question 06.01.01-12,** to clarify this statement. In its response to **RAI 487-3939, Question 06.01.01-12**, dated December 3, 2009, the applicant clarified that the NaTB is contained in baskets on the containment maintenance platform and thus, does not have the potential to contaminate the RWSP unless the CSS is inadvertently actuated. Therefore, the applicant does not consider it necessary to routinely sample the RWSP for sodium. The applicant further indicated in the RAI response that the refueling water storage system (RWS), which purifies the RWSP water, can be cross connected to the SFPCS to enable removal of dissolved impurities and solids from the RWSP in the event of contamination by sodium. The staff agrees that routine sampling of sodium is unnecessary because contamination by sodium during normal operation is unlikely, and the system is designed for corrective action if contamination occurs. The staff therefore finds the response to **RAI 487-3939, Question 06.01.01-12,** acceptable. Combined with the supplemental information in the responses to **RAI 487-3939, Question 06.01.01-11,** and **RAI 487-3939, Question 06.01.01-12**, the staff finds the response to **RAI 379-2756, Question 06.01.01-10**, acceptable because the applicant described water chemistry impurity limits that are consistent with the EPRI Guidelines, which the staff considers acceptable guidance for meeting GDC 14. Therefore, **RAI 379-2756, Question 06.01.01-10, and RAI 487-3939, Questions 06.01.01-11 and 12, are resolved and closed.**

6.1.1.4.6 Component and System Cleaning

RG 1.37 describes QA requirements 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 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 Appendix B to 10 CFR Part 50, subject to the regulatory positions listed in RG 1.37.

DCD Subsection 6.1.1.2.2, "Controls for Austenitic Stainless Steel," states that the process of cleaning of materials and components, cleanliness control, and preoperational flushing for systems that contain austenitic stainless steel components follows RG 1.37 and the QA program (QAP) complies with the provisions and recommendations provided by ASME NQA-1-1994, Part II. The staff finds this acceptable because the applicant complies with RG 1.37 and ASME NQA-1-1994. Additional information regarding the applicant’s compliance with ASME NQA-1-1994 is discussed in SER Section 17.5, “Quality Assurance Program Description.”

6.1.1.4.7 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 nonmetallic 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 SCC 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 nonmetallic insulation materials.

DCD Subsection 6.1.1.2.1 states that non-metallic insulation is controlled in accordance with RG 1.36 to control leachable concentrations of chlorides, fluorides, sodium compounds and silicates. In addition, DCD Table 1.9.1-1 indicates that the applicant does not take any exception to the guidance provided in RG 1.36 as it relates to DCD Section 6.1.1. The staff therefore finds this acceptable because the applicant complies with staff guidance provided in RG 1.36 to control leachable contaminants in nonmetallic insulation materials.

6.1.1.4.8 ITAAC

10 CFR 52.47(b)(1) requires that a DC application include the proposed ITAAC that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the DC is built and will operate in accordance with the DC, the provisions of the Atomic Energy Act, and the NRC's regulations. The staff evaluation of ITAAC associated with ESF systems can be found in Section 14.3 of this report, “Inspections, Tests, Analyses, and Acceptance Criteria.”

6.1.1.5 Combined License Information Items

There are no COL information items for this area of review identified in DCD Section 6.1.1 or Table 1.8-2, “Compilation of All Combined License Applicant Items for Chapters 1 - 19.”

6.1.1.6 Conclusions

With the exception of the above confirmatory item, the staff concludes, based on the information provided by the applicant, that the US-APWR ESF materials specifications, and controls on fabrication and compatibility of materials with ESF fluids are acceptable and meet the relevant requirements of GDCs 1, 4, 14, 31, 35, and 41 of Appendix A to 10 CFR Part 50; Appendix B to 10 CFR Part 50; and 10 CFR 50.55a.

With respect to control of ESF systems water chemistry, the staff finds the applicant has specified appropriate limits that support compliance with GDC 4 as it relates to assuring compatibility of the system materials with the normal operating and post-accident environment, and GDC 14 as it relates to preventing corrosion-related failures of the RCPB. The basis for this finding is the consistency of the water chemistry limits with those recommended by the EPRI PWR Primary Water Chemistry Guidelines, which constitutes guidance for water chemistry acceptable to the staff. The staff also finds the ESF water chemistry meets BTP 6.1 since a buffer has been provided to ensure a post accident pH of 7.0 or greater, thus supporting compliance with GDC 4 as it relates to compatibility of the materials with the post-accident environment. The staff also finds that adequate controls on hydrogen generation are provided by limiting the use of materials that produce hydrogen as a corrosion byproduct thus supporting compliance with GDC 41.

6.1.2 Organic Materials

6.1.2.1 Introduction

This section specifies the requirements for protective coating systems (paints) used inside containment and how those requirements are met. Radiation and chemical effects are considered in determining the stability of materials and the potential formation of decomposition products under DBA conditions. The protective coatings and their applications are subject to the requirements of Appendix B of 10 CFR Part 50.

6.1.2.2 Summary of Application

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

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

> With the notable exception of coatings and electrical insulation, organic materials (e.g., wood, plastics, lubricants, asphalt) are not freely available in containment. A corrosion inhibiting primer is typically applied as a base coating over the steel plate lining of the containment vessel, as well as to structural steel support members. A scuff-resistant top coat (e.g., epoxy) is then applied for durability and decontamination considerations. The operating surfaces of components (i.e., valve handwheels, operating handles) are typically factory-coated for mechanical durability and resistance to the containment operating environment. These coatings may be dry-powder- or water-reduced materials. With rare and minor exceptions (e.g., protective coatings on trim pieces, faceplates, and covers) coatings used inside containment are applied in accordance with RG 1.54, "Service Level I, II, and III Protective Coatings Applied to Nuclear Power Plants," and meet the applicable environmental qualifications described in Chapter 3, Section 3.11, "Environmental Qualification of Mechanical and Electrical Equipment."

ITAAC: The ITAAC pertaining to coatings are found in Tier 1 of the DCD, Revision 3, Table 2.4.4-5, "Emergency Core Cooling System Inspections, Tests, Analyses, and Acceptance Criteria," (Sheet 6 of 10).

TS: There are no TS for this area of review.

Topical Reports: There are no topical reports for this area of review.

Technical Reports: There are no technical reports submitted by the applicant for this area of review.

US-APWR Interface Issues identified in the DCD: There are no US-APWR interface issues for this area of review.

Site Interface Requirements identified in the DCD: There are no site interface requirements for this area of review.

Cross-cutting Requirements (TMI, USI/GSI, Op Ex): None for this area of review.

RTNSS: There is no RTNSS for this area of review.

10 CFR 20.1406: There are no 10 CFR 20.1406 requirements for this area of review.

CDI: There is no CDI for this area of review.

6.1.2.3 Regulatory Basis

The relevant requirements of the Commission's regulations for this area of review, and the associated acceptance criteria, are given in Section 6.1.2, "Protective Coating Systems (Paints) - Organic Materials," of NUREG-0800, the SRP, and are summarized below. Review interfaces with other SRP sections can be found in Section 6.1.2 of NUREG-0800.

1. Appendix B to 10 CFR Part 50 as it relates to the QA requirements for the design, fabrication and construction of safety-related SSCs.

2. 10 CFR 52.47(b)(1), which requires that a DC application include 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 DC, the provisions of the Atomic Energy Act of 1954, and the NRC's regulations.

Acceptance criteria adequate to meet the above requirements include:

1. RG 1.54, "Service Level I, II, and III Protective Coatings Applied to Nuclear Power Plants."

2. ASTM D5144-00, "Standard Guide for Use of Protective Coating Standards in Nuclear Power Plants."

3. 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

In the DCD, Revision 1, the applicant did not provide any substantial information regarding what coatings should be specified for various surfaces. Neither did they discuss Service Level I, II, or III distinctions for different surfaces. Finally, they did not specify any distinct combined operating license (COL) action items specifically related to coatings selection and integrity.

In its response to **RAI 45-876, Question 06.02.02-1**, dated August 26, 2008, and **RAI 85-1445, Question 06.02.02-10**, dated November 12, 2008, the applicant stated that only DBA-qualified epoxy coatings (as defined by standards ASTM D5144 and ASTM D3912) will be used in containment, and that no inorganic zinc coatings should be present. The applicant indicated that this applies to both top coat and primer. The first cited standard (ASTM D5144) is referred to by RG 1.54 as a "…top-level ASTM standard that incorporates by reference other key ASTM standards…" In these same responses, the applicant described a minor change to the DCD which mentions organic materials in containment, but did not address the detailed coatings requirements mentioned in the RAI responses. However, in its response to **RAI 354-2585, Question 06.02.02-31**, dated July 7, 2009, the applicant proposed changes to the DCD (Sections 6.2.2.3, "Design Evaluation," and Table 1.9.1-1) which specifically commit the COL holder to a coatings monitoring program consistent with RG 1.54, Revision 1. Since the proposed changes in the responses to **RAI 354-2585, Question 06.02.02-31,** clarify the applicant's commitment for the coatings program to meet RG 1.54, thereby conforming to the regulatory guidance of SRP 6.1.2, the staff finds the response acceptable with respect to coatings programmatic aspects (for the full evaluation of this RAI question, refer to Section 6.2.2, "Containment Heat Removal Systems," of this SER). The staff confirmed the change to Section 6.2.2.3 is incorporated in Revision 3 of the DCD.

In its response to **RAI 263-2072, Question 06.02.02-12**, dated March 31, 2009, the applicant presented a modification to Table 2.4.4-5 of the Tier 1 document that includes ITAAC required to verify all coatings in containment are DBA qualified. This response was expanded upon in the response to **RAI 348-2587, Question 14.03.11-39**, dated June 11, 2009, which revised the proposed ITAAC to verify all coatings are consistent with the ECC/CS suction strainer debris generation, debris transport and downstream effects evaluations, in addition to being DBA qualified. The staff confirmed that this change was incorporated in Revision 2 of the DCD. The staff finds that the responses to **RAI 263-2072, Question 06.02.02-12,** and **RAI 348-2587, Question 14.03.11-39,** support the conclusion that the coatings program will follow the recommendations of SRP 6.1.2 since the proposed ITAAC will confirm the coatings are DBA qualified. Although SRP 6.1.2 and RG 1.54 do not recommend that all coatings be DBA qualified, if they are qualified, it can be concluded that the coatings conform to the guidance of RG 1.54. The full evaluation of the acceptability **RAI 263-2072, Question 06.02.02-12,** and **RAI 348-2587, Question 14.03.11-39,** is found in Sections 6.2.2 and 14.03.11 of this SER.

Also, in its response to **RAI 365-2774, Question 06.01.02-1**, dated June 12, 2009, the applicant presented a COL item (to be included in future revisions of the DCD) which requires the COL applicant to identify the implementation milestones for the coatings program. The staff confirmed that the revised COL Information Item 6.1(7) is incorporated in Revision 2 of the DCD.

Also in its the amended response to **RAI 365-2774, Question 06.01.02-1**, dated June 12, 2009, the applicant proposed changes to the DCD which discuss the Service Level I, II, and III coatings, and states that the definitions for these levels are exactly those described in RG 1.54. It is stated that the Service Level I coatings will satisfy ASME NQA-1-1994, ASTM D3843-00 and 10 CFR 50 Appendix B, Criterion IX. This change in the DCD does not describe the quality and type of coatings themselves, but rather the Coatings Program, including procurement, application, maintenance, and QA. The staff confirmed these changes were incorporated in Revision 2 of the DCD. The staff finds the response to **RAI 365-2774, Question 06.01.02-1,** acceptable because the applicant has provided sufficient details of the coatings programmatic requirements in the DCD, and has included a COL item that will ensure the COL applicant identifies the coatings program implementation milestone, which will facilitate NRC inspection of the COL holder's coatings program. Therefore, **RAI 365-2774, Question 06.01.02-1, is resolved and closed.**

In summary, the applicant has committed through RAI responses to coatings that satisfy RG 1.54, ASTM D5144, and ASTM D3912, which constitute the SRP Acceptance Criteria. The applicant has proposed changes to the DCD which commit to Service Level I, II, and III definitions from RG 1.54, and to a coatings program that will satisfy RG 1.54. The applicant has also proposed revisions to normal DCD text (Sections 6.2.2.3 and 6.1.3, "Combined License Information") to require a coatings monitoring program consistent with RG 1.54, and a COL Information Item [6.1.3, COL Information Item 6.1(7)] that requires the COL holder to define the implementation milestones for this program. The applicant has also added an ITAAC to verify all coatings are consistent with the ECC/CS suction strainer debris generation, debris transport and downstream effects evaluations, in addition to being DBA qualified.

6.1.2.5 Combined License Information Items

The following is a list of item numbers and descriptions from Table 1.8-2 of the DCD.

Table 6.1.2-1
US-APWR Combined License Information Items

Item No.	Description	Section
6.1(7)	The COL Applicant is responsible for identifying the implementation milestones for the coatings program"	6.2.2.3

COL information items needed, but not listed in Table 1.8-2 of the DCD: None

6.1.2.6 Conclusions

The applicant has stated that all coatings in containment will be consistent with SRP Acceptance Criteria (RG 1.54, ASTM D5144, and ASTM D3912), and that the COL applicant will verify this in Tier 1 of the ITAAC. The applicant has also modified the DCD to clarify how the recommended coating program conforms to RG 1.54. Hence, the staff concludes that the requirements of 10CFR 50 and 10CFR52.47(b)(1) will be satisfied.

6.2 Containment Systems

This section describes the physical attributes of the reactor containment and how these physical attributes address and satisfy the containment functional design requirements. This section also describes the following ESF systems directly associated with containment:

- Containment structure (vessel), including subcompartments.
- Containment spray system.
- Containment isolation system.
- Containment hydrogen monitoring and control system.

For each of these systems and structures, this section describes the design bases, the design features, and the evaluations of the acceptability of the design. For some systems (such as the containment structure), the design evaluation is conducted in conjunction with analyses of postulated accidents (documented in Chapter 15, “Transient and Safety Analyses”), which can release material and energy into the containment, resulting in increased pressure and temperatures inside the containment vessel. This section describes the detailed assessments of the mass and energy releases associated with these postulated accidents.

6.2.1 Containment Functional Design

The various containment aspects to be reviewed in accordance with this section of NUREG-0800 have been separately reviewed under other NUREG-0800 sections, as follows:

1. Pressurized water reactor (PWR) dry containments, including sub atmospheric containments (Section 6.2.1.1.A).

2. Subcompartment analysis (Section 6.2.1.2).

3. Mass and energy release analysis for postulated loss of coolant accidents (Section 6.2.1.3).

4. Mass and energy release analysis for postulated secondary system pipe ruptures (Section 6.2.1.4).

5. Minimum containment pressure analysis for ECCS performance capability studies (Section 6.2.1.5).

Areas related to the evaluation of the containment functional capability are treated in other sections; e.g., Containment Heat Removal (Section 6.2.2), Containment Isolation System (Section 6.2.4), Combustible Gas Control (Section 6.2.5), and Containment Leakage Testing (Section 6.2.6). In addition, the evaluation of the secondary containment functional design capability is reviewed under Section 6.2.3 of NUREG-0800.

6.2.1.1 Containment Structure

6.2.1.1.1 Introduction

The containment encloses the reactor system and is the final barrier against the release of significant amounts of radioactive fission products in the event of an accident. The containment structure must be capable of withstanding, without loss of function, the pressure and temperature conditions resulting from postulated loss-of-coolant, steam line, or feedwater line break accidents. The containment structure must also maintain functional integrity in the long term following a postulated accident, i.e., it must remain a low leakage barrier against the release of fission products for as long as postulated accident conditions require. The design and sizing of containment systems are largely based on the pressure and temperature conditions, which result from release of the reactor coolant in the event of a LOCA. The containment design basis includes the effects of stored energy in the RCS, decay energy, and energy from other sources such as the secondary system, and metal-water reactions including the recombination of hydrogen and oxygen. The containment system is not required to be a complete and independent safeguard against a LOCA by itself, but functions to contain any fission products released while the ECCS cools the reactor core.

6.2.1.1.2 Summary of Application

DCD Tier 1: The Tier 1 information associated with this evaluation is provided in Tier 1, Section 2.2.1.2, "Prestressed Concrete Containment Vessel (PCCV)," Tier 1, Section 2.2.1.3, "Containment Internal Structure," and Tier 1, Section 2.11, "Containment Systems," that provides a table with the ITAAC (Table 2.11.1-2, "Containment Vessel Inspections, Tests, Analyses, and Acceptance Criteria").

DCD Tier 2: The Tier 2 information associated with this evaluation is provided in Tier 2, Section 6.2.1, "Containment Functional Design." A summary of the technical information is as follows.

> The US-APWR containment is a PWR dry containment with some differences when compared to conventional dry containments. The water source for the ECCS and spray systems is from the RWSP, which is located inside the containment (at the bottom). This configuration has the advantage of not requiring switching the ECCS water source from a tank to the containment sump when the water in the tank is depleted and water recirculation starts. In the US-APWR, when water spills from the ECCS or from the sprays onto the containment floor, this water drains to the RWSP and is recirculated automatically without the need to switch the water source from an initial water tank to the containment sump. The containment has spray systems and fan coolers; however, only the spray systems are safety systems required to remove heat and/or fission products from the containment atmosphere.
>
> The US-APWR dry containment consists of a prestressed concrete structure with a cylindrical wall, hemispherical dome, and reinforced concrete foundation slab. It is 69.04 m (226.5 feet) high, with an inner diameter of 45.42 m (149 feet). The lateral walls are 1.32 m (4 feet, 4 inches) thick, and the dome is 1.12 m (3 feet, 8 inches) thick. The inner surface of the containment is covered with a 6.35 mm (0.25 inch) welded steel plate liner. The containment completely encloses the reactor and RCS and is designed to be leak-tight to prevent uncontrolled release of radioactive material to the environment. The primary containment encloses the steam supply system which includes the reactor vessel, steam generators (SGs), reactor coolant pumps (RCPs), pressurizer, and associated piping. Additionally, the containment houses mechanical support components, electrical support components, and heating, ventilation, and air conditioning support components. All lines (piping) that penetrate the containment are provided with isolation features that automatically close when required.
>
> The application describes that the containment is designed to be compatible with all environmental effects experienced during normal operations that include, but are not limited to, containment temperature, pressure, humidity, presence of fluids (e.g., equipment lubricants and borated reactor coolant), and other assorted environmental effects of reactor operation, testing, and maintenance.
>
> The application further states that the containment is also designed to accommodate conditions during and following postulated accidents, such as the design basis LOCA. These conditions include elevated temperature, pressure and humidity. Other conditions include radioactive fission products, sodium tetraborate decahydrate (NaTB), and borated water. The application identifies that the peak pressure for the most severe postulated accident does not exceed the containment internal design pressure, 68 pounds per square inch gauge (psig) or 570 kilo Pascal (kPa).

The application also states that the containment function is applicable to hot shutdown conditions, when the postulated accident could cause a release of radioactive material in the containment and an increase in containment pressure and temperature. For Mode 1 or Mode 2, the containment analysis identifies that the energy sources including reactor coolant fluid and metal energy, SG fluid and metal energy, core stored energy, and decay heat are much larger than that in the Mode 3 and 4 shutdown condition.

The application describes containment functional design, design data in Tables 6.2.1-1 through 6.2.1-14.

ITAAC: The ITAAC associated with this evaluation are provided in DCD Tier 1, Section 2.11.1, "Containment Vessel," and Table 2.11.1-2.

Technical Specifications: The TS associated with DCD Tier 2, Section 6.2.1 are provided in DCD Tier 2, Chapter 16, "Technical Specifications," Sections 3.6, "Containment Systems."

Topical Reports: Mitsubishi Heavy Industries, Ltd., Topical Report MUAP-07012-P-A, "LOCA Mass and Energy Release Analysis Code Applicability Report for US-APWR," Revision 2 and MUAP-07012-NP-A, Revision 2, issued June 2009.

Technical Reports:

- Mitsubishi Heavy Industries, Ltd., Technical Report MUAP-07031-P, "Subcompartment Analyses for US-APWR Design Confirmation," Revision 1, and MUAP-07031-NP, Revision 1, issued October, 2009.

- Mitsubishi Heavy Industries, Ltd., Technical Report MUAP -08001-P, "US-APWR Sump Strainer Performance," Revision 2, issued December 2008, and Revision 7, issued August 2012.

US-APWR Interface Issues identified in the DCD: There are no US-APWR interface issues for this area of review.

Site Interface Requirements identified in the DCD: There are no site interface requirements for this area of review.

Cross-cutting Requirements (TMI, USI/GSI, Op Ex): GSI 191, "Assessment of Debris Accumulation on PWR Sump Performance," closure activities (ML120820125).

RTNSS: There is no RTNSS for this area of review.

10 CFR 20.1406: There are no 10 CFR 20.1406 requirements for this area of review.

CDI: There is no CDI for this area of review.

6.2.1.1.3 Regulatory Basis

The relevant requirements of the Commission's regulations for this area of review, and the associated acceptance criteria, are given in Section 6.2.1.1.A, "PWR Dry Containments, Including Sub-atmospheric Containments," of NUREG-0800, the SRP, and are summarized below. Review interfaces with other SRP sections can be found in Section 6.2.1 of NUREG-0800.

1. GDC 13, “Instrumentation and control,” requires instrumentation to be provided to monitor variables and systems over their anticipated ranges for normal operation and for accident conditions as appropriate to assure adequate safety. This means that instrumentation must be capable of operating in the post-accident environment in order to monitor the containment atmosphere pressure and temperature, and the sump water level and temperature following an accident. It must have adequate range, accuracy, and response to assure that the above parameters can be tracked and recorded throughout the course of an accident. Meeting this requirement helps ensure that the containment precludes the release of radioactivity to the environment.

2. GDC 16, “Containment design,” as it relates to the 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. The primary reactor containment must protect against the uncontrolled release of radioactivity to the environment by preserving containment under the dynamic conditions imposed by LOCAs. This means that the containment must be designed as an essentially leak tight barrier that will withstand the most extreme accident conditions for the duration of any postulated accident.

3. GDC 38, “Containment heat removal,” as it relates to the heat removal system functions to rapidly reduce the containment pressure and temperature following any LOCA and maintain them at acceptably low levels. The containment heat removal system supports the containment function by minimizing the duration and intensity of the pressure and temperature increase following a LOCA, lessening the challenge to containment integrity. Meeting GDC 38 helps ensure that the containment can fulfill its role as the final barrier against the release of radioactivity to the environment.

4. GDC 50, “Containment design basis,” as it relates to the containment structure and its internal compartments being designed to accommodate the calculated pressure and temperature conditions resulting from any LOCA without exceeding the design leakage rate, including sufficient margin. In order to satisfy both this requirement and GDC 38, the LOCA analysis should be based on the assumption of loss of offsite power (LOOP) and the most severe single failure in the emergency power system (i.e., a diesel generator failure), the containment heat removal systems (i.e., a fan, pump, or valve failure), or the core cooling systems (i.e., a pump or valve failure). The selection of the single failure must result in the highest calculated containment pressure. In addition to satisfying the requirements of GDC 38 and GDC 50 with respect to containment heat removal capability and design margin, the containment response analysis for postulated secondary system pipe ruptures should be based on the most severe single active failure in the containment heat removal systems (i.e., a fan, pump, or valve failure) or the secondary system isolation provisions (i.e., main steam isolation valve failure or feedwater line isolation valve failure). The analysis must be based on a spectrum of pipe break sizes and reactor power levels. Accident conditions selected should result in the highest calculated containment pressure or temperature depending on the purpose of the analysis.

5. GDC 64, “Monitoring radioactivity releases,” requires that the containment atmosphere be monitored for the release of radioactivity from normal operations, anticipated

operational occurrences (AOOs), and accidents. Meeting this requirement allows operators to ensure that containment meets its safety function of preventing a release of radioactivity to the environment.

6. 10 CFR 52.47(b)(1) requires that the DC application contain proposed ITAAC that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and acceptance criteria met, a plant that incorporates the certified design is built and will operate in accordance with the DC, the provisions of the Atomic Energy Act of 1954, and the NRC's regulations.

Acceptance criteria adequate to meet the above regulatory requirements:

1. RG 1.82, "Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident," Revision 3, issued November 2003.

2. In order to satisfy the requirements of GDC 38 and GDC 50 to ensure the functional capability of the containment heat removal systems and containment structure, provisions must also be made to protect the containment structure against possible damage from external pressure conditions that may result, for example, from inadvertent operation of containment heat removal systems. The external design pressure of the containment must provide adequate margin above the maximum expected external pressure to account for uncertainties in the analysis of the postulated events.

6.2.1.1.4 Technical Evaluation

The staff reviewed the calculated temperatures and pressures of the primary containment to a spectrum of LOCAs and main steam line breaks (MSLBs), and completed the review of the calculated minimum containment pressure for LOCA analyses.

The applicant has examined several break sizes and locations to determine the breaks that most severely challenge the containment limits. In DCD Subsection 6.2.1.1.1, "Design Bases," the applicant states that DCD Tier 2, Table 6.2.1-1, "Summary of Calculated Containment Temperature and Pressure Results for the Worst Case of Postulated Piping Failure Scenarios," "…summarizes containment temperature and pressure, for a broad range of postulated breaks,…." yet this table only has one break included. This inconsistency was questioned by the staff in **RAI 126-1558, Question 06.02.01-2**. In its response to RAI 126-1558, Question 06.02.01-2, dated January 29, 2009, the applicant revised the title of DCD Tier 2, Table 6.2.1-1 by replacing "a broad range" with "the worst case." The staff accepted this revision. Therefore, **RAI 126-1558, Question 06.02.01-2, is resolved and closed.**

Per SRP 6.2.1.1.A, LOCA analyses should be performed using the assumption of LOOP with the most severe single failure in the emergency power system, in the containment heat removal systems, or in the core cooling systems to evaluate the highest containment pressure and temperature. The applicant has determined that the single failure condition related to containment pressure and temperature limits is the failure of one of the four emergency power sources. In addition, the applicant has assumed that one other emergency power source is out of service, so that only two emergency power sources are assumed to be available.

A unique feature of the US-APWR is that the RWSP is located at the bottom of the containment. It is a horseshoe-shaped pool along the containment perimeter. The location of the RWSP inside containment is the primary difference between existing PWR dry containment designs and the US-APWR design. The open end of the RWSP horseshoe is where the reactor coolant drain tank, the reactor coolant drain pumps and the containment sumps are located. The RWSP is the source of the borated-water for emergency core cooling water for the safety injection system (SIS) and for the CSS. During SIS and CSS operation, water returns to the RWSP through various drainage paths. Its location inside the containment eliminates the need for "switch-over" of the ECCS suction from an external source to the containment recirculation sump. The staff issued **RAI 331-935, Question 06.02.01-8,** requesting information regarding drain path characteristics, the split between retained and returned water, the timing of the returned water, etc., and how these values were determined. In its response to RAI 331-935, Question 06.02.01-8, dated May 26, 2009, the applicant provided the requested information as reflected in Section 3.7.1, "Hold-up Volumes," of MHI Technical Report MUAP-08001-P, "US-APWR Sump Strainer Performance," Revision 2. The staff accepts the response. Therefore, **RAI 331-935, Question 06.02.01-8, is resolved and closed**.

In order to effectively delay the transfer of containment debris to the reactor core after a LOCA, the applicant made certain containment internal design changes to reroute the recirculation water flow paths for RCS break flow and Refueling Cavity/Containment Vessel (RC/CV) spray water flow to the RWSP. These design changes, made as a part of the GSI-191, "Assessment of Debris Accumulation on PWR Sump Performance," closure activities (ML120820125), reduced the free containment volume; reorganized the sub-compartments; and modified their capacities and the interconnecting flow paths. The staff audited these changes during the April 16 - 17, 2012, audit (ML12187A758), and issued **RAI 923-6420, Question 06.02.01-21,** to assess the impact of the containment internal design changes on the containment functional design DBA analyses performed in DCD Tier 2, Section 6.2.1. In its response, dated July 31, 2012, the applicant showed the results of the revised analyses updated for the design changes. The analyses include the maximum containment pressure and temperature calculations for a spectrum of LOCA and MSLB events in Section 6.2.1.1; containment subcompartment analysis in Section 6.2.1.2, "Containment Subcompartments"; mass and energy release calculations for LOCA and secondary system piping rupture in Sections 6.2.1.3, "Mass and Energy Release Analysis for Postulated Loss-of-Coolant Accidents," and 6.2.1.4, "Mass and Energy Release Analysis for Postulated Secondary-System Pipe Ruptures Inside Containment," respectively; and the evaluation of the minimum containment pressure for the ECCS performance, in Section 6.2.1.5, "Minimum Containment Pressure Analysis for Performance Capability Studies of the Emergency Core Cooling System." The applicant's response to **RAI 923-6420, Question 06.02.01-21** was expected to show that the GSI-191 related design changes are conservative with respect to the containment functional design, and have no adverse impact on the limiting containment and the sub-compartment pressure/temperature response analyses for LOCA, MSLB, mass and energy release; and the containment minimum pressure. The response to RAI 923-6420, Question 06.02.01-21, dated July 31, 2012, was partly acceptable for the safety analyses in Sections 6.2.1.1, 6.2.1.3, and 6.2.1.5. However, the staff found the applicant's response lacking necessary details on the safety analyses in Sections 6.2.1.2, "Containment Subcompartments," and 6.2.1.4, "Mass and Energy Release Calculations for Secondary System Pipe Ruptures." The applicant's response on the Sections 6.2.1.2 and 6.2.1.4 safety analyses is currently under staff's evaluation. **RAI 923-6420, Question 06.02.01-21 is being tracked as an Open Item.**

Technical Report MUAP-08001-P, Revision 7, issued August 2012, assesses the US-APWR design with respect to RG 1.82, "Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident," Revision 3, issued November 2003. As explained earlier, the applicant made certain containment internal design changes to reroute the recirculation water flow paths in order to delay the RCS break flow and RC/CV spray water flow to the RWSP.

These design changes modified the drain paths, the split between retained and returned water, and the timing of the returned water. These changes also reorganized the containment sub-compartments and modified their capacities and the interconnecting flow paths. **RAI 923-6420, Question 06.02.01-21,** was issued to assess the impact of the changes on the containment functional design DBA analyses performed in DCD Tier 2, Section 6.2.1. The applicant's response is under the staff's evaluation, and as stated above, **RAI 923-6420, Question 06.02.01-21, is being tracked as an Open Item.**

DCD Tier1, Section 6.2.1.1, "Containment Structure," describes the containment design pressure and temperature limits. The US-APWR containment design pressure is 570 kPa (68 psig), and its test pressure is 641 kPa (78.2 psig). The limiting calculated peak pressure occurs for a double-ended guillotine break (DEGB) in the RCS cold-leg at the pump suction with a value of 498 kPa (57.5 psig). Revised calculations, submitted in response to **RAI 126-1558, Question 06.02.01-6**, dated April 21, 2009, resulted in a slightly higher maximum peak pressure of 511.6 kPa (59.5 psig). The containment is designed for a negative pressure of -26.9 kPa differential (-3.9 psid). The limiting event for the negative pressure is an inadvertent initiation of the containment spray system, which causes a -26.5 kPad (-3.84 psid) negative pressure differential in the containment. Containment maximum design temperature is 149°C (300°F). The limiting peak containment atmosphere temperature occurs for a double ended steam line break, producing a maximum containment atmospheric temperature of 179°C (355°F) and a maximum containment liner temperature of 134°C (273°F). The maximum temperature of the containment liner is below the maximum containment design temperature. A summary of these results is presented below in SER Table 6.2-1, "Summary of Design Values, Limiting Events and Results."

Table 6.2-1 Summary of Design Values, Limiting Events and Results

Containment design limit	**Containment design criterion**		**Limiting event**	**Limiting condition**	
Maximum pressure	570 kPa	68 psig	Cold leg break	511.6 kPa	59.5 psig
Max. negative pressure differential	26.9 kPad	3.9 psid	Inadvertent CSS activation	26.5 kPad	3.84 psid
Max. containment temperature	149°C	300°F	DEG steam line break	179°C 134°C	355°F (containment atmosphere) 273°F (containment liner)

6.2.1.1.4.1 Containment Analytical Model

The applicant used the lumped-parameter option of the GOTHIC code (Reference 1) to calculate the short- and long-term containment pressure and temperature responses. The code solves the conservation equations for mass, momentum and energy for multi-component, multi-phase flow in lumped parameter and/or multi-dimensional geometries. The code is a non-equilibrium, non-homogeneous transient code, and can account for thermal non-equilibrium conditions with different velocities between the phases. It tracks multiple fluid fields, including steam/gas, liquid, and liquid droplets, and uses constitutive models for interface mass, energy, and momentum transfer. It is capable of including up to eight non-condensable gases in the steam/gas mixture; these are defined by the user. In addition, GOTHIC can model heat transfer between the fluid and solid surfaces as well as heat transfer within the solid structures and equipment located in the containment.

Solid structures within containment are modeled using thermal conductors, which are one dimensional slabs that interface thermally with the fluid in containment. A general heat transfer model is used between the thermal conductors and the steam/gas mixture or liquid (no conductor/droplet heat transfer is possible in GOTHIC). Mass sources and sinks can be specified for control volumes and thermal boundary conditions can be applied to a thermal conductor surface to act as thermal interfaces for the solid structures.

The applicant used a single volume to model the containment. The RWSP is modeled as a pool at the bottom of containment, with "appropriate assumptions on the heat and mass transfer at the pool." Details of these assumptions and sensitivities to these assumptions are not presented in the DCD. The staff issued **RAI 126-1558, Question 06.02.01-3,** requesting details of the heat transfer models between the containment atmosphere and the RWSP water pool used. In its response, dated March 19, 2009, the applicant included a sensitivity study on the effect of isolating the pool and the atmosphere, but the details of the heat transfer models between the pool and the atmosphere were not provided. In its response to supplemental **RAI 587-4689, Question 06.02.01.01.A-1**, dated June 7, 2010, the applicant referenced the DCD statement that the mass and heat transfer between the RWSP water pool and the rest of the containment is set to zero. This assumption is conservative because during the initial phases of an accident, the RWSP water is colder than the containment atmosphere. By suppressing heat and mass transfer from the containment atmosphere to the RWSP water, the resulting calculated pressure and temperature in the containment atmosphere will be higher and therefore, conservative. This statement addresses the staff concerns, and therefore, **RAI 126-1558, Question 06.02.01-3, and RAI 587-4689, Question 06.02.01.01.A-1, are resolved and closed**.

During the refill phase of the postulated LOCA, no details are given in the DCD Subsection 6.2.1.1.3.4, "Description of Containment Analyses," on how the accumulator nitrogen injection is modeled. The staff issued **RAI 6-256, Question 06.02.01.02-1,** requesting clarification on how the nitrogen injection from the accumulators is modeled. The following information was requested: the assumptions used in modeling the accumulator activation during the postulated DBA LOCA, the number of accumulators used, accumulator gas volume, and total amount of nitrogen used in DBA analysis. In its response to RAI 6-256, Question 06.02.01.02-1, dated June 27, 2008, the applicant included the following new paragraph to be incorporated into the next revision of the DCD:

> The non-condensable cover gas (nitrogen) in all accumulators is assumed to be released directly to the containment using the boundary conditions in the GOTHIC evaluation model. Total mass of the released nitrogen is calculated on the assumption that the accumulator is depressurized from the initial pressure to atmospheric pressure. The nitrogen temperature is assumed 120ºF, which is the maximum operating temperature, although the nitrogen temperature decreases with nitrogen gas expansion as the water is being injected."

This paragraph describes the assumptions used, and that all (four) accumulators were considered. The amount of nitrogen released by each accumulator is [ ] for all four. The applicant's response to **RAI 6-256, Question 06.02.01.02-1**, dated June 27, 2008 is acceptable and **RAI 6-256, Question 06.02.01.02-1, is resolved and closed**.

The applicant used the GOTHIC code to model the primary system during the post-reflood stage of the transient after the core has been recovered as well as the containment response. Prior to this, GOTHIC uses time dependent mass and energy release rates calculated by the SATAN and WREFLOOD codes as input for LOCAs and by the MARVEL-M code as input for MSLB accidents. The details of the LOCA mass and energy release calculations are described in Reference 2.

Justification for the single node containment model is made by the applicant via three arguments. The first is that this modeling technique has been used for previous licensing calculations, and has been accepted by the NRC in the past. Since the US-APWR containment is not significantly different than existing containment designs, a single node should be sufficient for this application as well. One implication of using a single node model is the assumption that everything is well mixed throughout containment. The second argument compares experimental results from the Carolinas-Virginia Tube Reactor experiment (CVTR) with calculations performed using both lumped parameter analysis as well as 3D models for the containment. Results from the experiment indicated that detailed nodal models implied well mixed conditions in the upper portions of the experiment where steam was injected, and lower levels of mixing in the lower portions of the experiment (below an operating deck). Additional experiments cited (i.e., Marviken and Heissdampfreactor (HDR)) also show significant variations within containment. However, comparison between lumped and subdivided GOTHIC models for the CVTR tests indicate that the predicted peak pressure and temperature form the lumped analysis are more conservative than those of subdivided analysis. Finally, before containment sprays are activated, the major mechanism for energy removal is convection and condensation on containment structures. The applicant notes that increased condensation in steam-rich regions more than compensate for the reduced exposure to containment structures during this period. After the containment sprays are activated, the containment is assumed to be well mixed.

In order to help ensure conservative results, the minimum containment free volume was assumed and no heat and mass transfer was assumed between the RWSP and the containment atmosphere.

Containment structures were modeled in GOTHIC using one dimensional heat structures ("conductors"). Smaller structures of similar material composition were lumped together. However, GOTHIC performs a one dimensional conduction calculation for the resulting structures, subdividing the structures into multiple nodes to numerically determine the energy release versus time. The method of lumping the thicknesses is therefore only approximate, and no discussion is provided regarding the impact of this approximation. The applicant uses the DIRECT heat transfer option in the Diffusion Layer Model for condensation for all of the containment heat sinks modeled in GOTHIC. The applicant states that this model has been validated and documented in the GOTHIC Qualification Report (Reference 3), and has been accepted for use in previous licensing analysis. This issue was addressed in **RAI 126-1558, Question 06.02.01-4**. In its March 19, 2009 response to RAI 126-1558, Question 06.02.01-4, the applicant performed a sensitivity study, with finer mesh subdivision in some structures, and the results show no significant differences. The staff finds the applicant's response acceptable. **RAI 126-1558, Question 06.02.01-4, is resolved and closed**.

Additional information is required to determine the sensitivity of containment response to variations in droplet diameter. Drag coefficients in GOTHIC include the influence of droplet to droplet interactions, while heat transfer rates were validated from evaporation data. The applicant concludes that since evaporation rates are reasonably predicted by GOTHIC models, condensation rates should be as well. The staff believes that additional justification needs to be made regarding condensation coefficients used, and an evaluation of sensitivity to this parameter needs to be performed. This issue was addressed in **RAI 126-1558, Question 06.02.01-5**. In its response to this RAI, dated March 19, 2009, the applicant provided the heat transfer coefficients employed and results of a sensitivity study with different droplet sizes (1.016-mm, nominal, 0.508-mm, and 2.032-mm or 0.04, 0.02 and 0.08-in, drop diameters). The results yielded lower pressures for smaller droplets, but the differences are not significant. The peak containment pressure was [ ] for the largest drop (2.032 mm or 0.08 in) and [ ] for the smallest drop (0.508 mm or 0.02 in). The staff finds the RAI response acceptable. **RAI 126-1558, Question 06.02.01-5, is resolved and closed**.

Containment analysis performed by the applicant covered a spectrum of LOCA and MSLB accidents. The applicant states that "the effects of uncertainties and tolerances have been selected to produce conservatively high containment internal pressure." Four classes of LOCA events were analyzed:

1) Double-ended pump suction guillotine break (Coefficient of Discharge (CD) = 1.0).
2) Double-ended pump suction guillotine break (CD = 0.6).
3) 0.28 m^2 (3 ft^2) pump suction split break (CD = 1.0).
4) Double-ended hot leg guillotine break (CD = 1.0).

Containment initial conditions were chosen to maximize containment peak pressure for the LOCA and to maximize containment peak temperature for the MSLB. These are shown in SER Table 6.2-2, "Containment Initial Conditions," and are consistent with those outlined in NUREG-0800, Section 6.2.1.2, "Subcompartment Analysis," with the exception of the maximum containment initial pressure being used for LOCA events. However, the applicant has stated that higher initial pressure gives higher air partial pressure and larger heat capacity in the containment atmosphere, which results in higher pressure and lower temperature during the postulated LOCA accident, thus making the maximum initial pressure condition limiting.

Table 6.2-2 Containment Initial Conditions

Parameter	Value	Setting for Conservatism
Reactor Power (MW)	4451x1.02	Max (102 percent)
Containment Pressure LOCA	115 kPa (2 psig)	Max
Containment Pressure MSLB	101 kPa (0 psig)	Min
Containment Temperature	49°C (120°F)	Max
Containment Relative Humidity (%)	0	Min

6.2.1.1.4.2 Analysis Results

The applicant has performed a series of calculations looking at a spectrum of breaks including sensitivity of results to variations in safety injection and accumulator assumptions. The LOCA calculations assumed a LOOP, and the loss of one emergency generator (the limiting single failure in the emergency power system) while one additional emergency generator is out of service. This leaves two out of the four emergency generators unavailable – thus, two of the four trains of the containment spray system will still be available. Since each train has 50 percent capacity, two trains will provide 100 percent capacity. The conclusion from the results is that the double-ended pump suction guillotine break (CD = 1.0) is limiting for containment maximum pressure when minimum emergency core cooling system flow conditions are assumed, and accumulator conditions are assumed that minimize steam condensation. Results indicate a maximum containment pressure of 498 kPa (57.5 psig), and a maximum containment atmosphere temperature of 139°C (282°F). Table 6.2-1 of the SER reports the maximum calculated pressure. The calculations were continued up to 24 hours, the containment pressure was reduced to 163 kPa (9 psig) which is less than 50 percent of the peak value. These calculations were revised and the revised values are provided in the following paragraphs.

The applicant has completed calculations up to 86,400 seconds (24 hours) for just one of the cases evaluated. In **RAI 126-1558, Question 06.02.01-6**, the staff requested long term calculations (24 h) for all the accidents and cases analyzed. In its response to RAI 126-1558, Question 06.02.01-6, dated April 21, 2009, the applicant provided the requested long term calculations, with the following model modifications:

(1) employed the methodology of report MUAP-07012-P, Revision 2, "LOCA Mass and Energy Release Analysis Code Applicability Report for US-APWR," Reference 2, and

(2) subcooled spillage during reflood is released as continuous liquid flow to conservatively minimize steam condensation in containment.

As a result of these model modifications, the resulting peak containment pressure and temperatures for the LOCA cases are higher. The new calculated peak pressure is 511.6 kPa (59.5 psig) and the new calculated peak temperature is 140°C (284°F). The pressure at 24 hours was calculated as 177.9 kPa (11.1 psig).

The staff agrees that the following NUREG-0800, Section 6.2.1.1.A, SRP acceptance criteria have been met for this event:

(1) The containment design pressure (570 kPa [68 psig]) provides at least a 10 percent margin above the calculated peak containment pressure following the limiting DBA,

(2) The containment pressure is reduced to less than 50 percent of the peak calculated DBA pressure within 24 hours after the postulated accident, and

(3) The peak containment temperature is less than the design temperature (149°C [300°F]).

The long term calculations demonstrated the containment pressure at 24 hours to be below 50 percent of the calculated peak pressure, as required by GDC 38 and SRP 6.2.1.1.A. The staff reviewed and accepted the revised calculations. **RAI 126-1558, Question 06.02.01-6, is resolved and closed**.

A spectrum of MSLB events was also analyzed to investigate their challenge to the containment acceptance criteria. Similar assumptions and initial conditions were used for the MSLBs as for the LOCAs. The limiting MSLB was found to be a double ended guillotine break with a CD =1, at a reactor power level of 102 percent - Case 1 of DCD Table 6.2.1-8, "Description and Summary Results For Evaluations of Various Pipe Sizes and Break Locations for Postulated Secondary Steam System Piping Failures (includes Plant Power Levels)," for the maximum containment temperature, and Case 5 of DCD Table 6.2.1-8, for the maximum containment pressure. For these analyses, offsite power was assumed to be available, however, one containment heat removal system was assumed to have failed (single failure), and one was assumed to be out of service. The peak containment pressure calculated for Case 1 was 433.7 kPa (48.2 psig) and the peak atmospheric temperature was 179.4°C (355°F). Case 5 results in the highest pressure, 437.8 kPa (48.8 psig), but a lower atmospheric temperature, 175°C (347°F), than Case 1. The peak pressures clearly meet the peak pressure acceptance criteria, however, only atmospheric temperatures are provided in the DCD rather than containment wall temperatures. Additionally, the calculated temperatures are over the 149°C (300°F) containment temperature limit. Calculations completed by the applicant in its response to **RAI 113-786, Question 06.02.01.04-2**, dated January 15, 2009, demonstrated that the containment walls do not exceed the 149°C (300°F) temperature limit. The maximum calculated temperature for the containment liner is 134°C (273°F). The staff finds the applicant's response acceptable. **RAI 113-786, Question 06.02.01.04-2, is resolved and closed**.

The NRC staff has completed confirmatory calculations with the computer code MELCOR (Reference 4). Two transients have been modeled: a LOCA with the DEPSG break, Case 1 of DCD Tier 2, Table 6.2.1-6, "Summary of LOCA Transients Evaluated," and a MSLB, Case 5 of Table 6.2.1-8 of the DCD. The containment was modeled with either one or six nodes. MELCOR calculated temperatures and pressures for the one-node containment model were in good agreement with the reported DCD values, using also a one-node containment model.

The applicant performed an external pressure analysis to calculate the containment maximum negative pressure differential in the event of an inadvertent spray actuation. For this analysis the applicant assumed that the containment has an initial air temperature of 49°C (120°F) and the spray temperature of 0°C (32°F). The containment was assumed to be at an initial pressure of 99.3 kPa (-0.3 psig), and at a relative humidity of 100 percent. These conditions are representative of the low end of values that can be anticipated under normal operating conditions. The combination of a low initial pressure, a high initial temperature and humidity and a low spray water temperature will result in a very low (minimum) containment pressure. The applicant calculated the resultant pressure when the air cooled down to the spray temperature. The resulting minimum containment pressure was determined to be 74.8 kPa (-3.8 psig), with a differential pressure across containment of 26.5 kPad (3.84 psid) which is lower than the design differential pressure of 26.9 kPad (3.9 psid), meeting this acceptance criteria. Table 6.2-1 of the SER reports this calculated maximum negative pressure differential.

6.2.1.1.5 Combined License Information Items

There are no COL information items identified for this section in DCD Tier 2, Table 1.8-2.

6.2.1.1.6 Conclusions

The staff has found that GDC 16 has been met since in the US-APWR DCD, the applicant has shown that the calculated containment pressure is always 10 percent below the containment design pressure, for all limiting breaks. Additionally, the applicant has also met GDC 38. The applicant has shown that containment pressure is reduced to less than 50 percent of the peak calculated pressure for the design basis LOCA within 24 hours of the postulated accident initiation.

GDC 50 has been satisfied because the applicant has assumed a LOOP, and has performed parametric evaluations on single failures in the emergency power system (LOCA), and single active failures (secondary system analysis) to determine the most limiting accident scenarios. In order to satisfy the requirements of GDC 38 and 50 to ensure the functional capability of the containment heat removal systems and containment structure from external pressure conditions, the applicant has performed a limiting analysis in order to comply with this requirement. The applicant has complied with all requirements established in SRP Section 6.2.1.

GDC 13 requirements on instrumentation and control for the containment and associated systems are reviewed and findings are made in Section 7.3, “Engineered Safety Feature Systems,” of this SER. Similarly, compliance to the GDC 64 requirements on monitoring radioactivity releases is covered under Sections 11.3, “Gaseous Waste Management System,” and 11.5, “Process Effluent Radiation Monitoring and Sampling Systems,” of this SER

6.2.1.2 Containment Sub-compartments

6.2.1.2.1 Introduction

The application presented information concerning the determination of the design differential pressure values for containment subcompartments. A subcompartment is defined as any fully or partially enclosed volume within the primary containment that houses high-energy piping and would limit the flow of fluid to the main containment volume in the event of a pipe rupture within the volume.

This internal compartment is described in GDC 50, of Appendix A to 10 CFR Part 50. A short-term pressure pulse would exist inside a containment subcompartment following a pipe rupture within the volume. The resulting pressure transient would produce a pressure differential across the walls of the subcompartment, which would generally reach a maximum value within the first second after blowdown begins. The magnitude of this maximum value is a function of several parameters, which include blowdown mass and energy release rates, subcompartment volume, vent area, and vent flow behavior. A transient differential pressure response analysis should be provided for each subcompartment or group of subcompartments that meets the above definition.

6.2.1.2.2 Summary of Application

DCD Tier 1: The Tier 1 information associated with this evaluation is provided in DCD Tier 1, Section 2.2.1.3, “Containment Internal Structure.”

DCD Tier 2: The Tier 2 information associated with this evaluation of the sub-compartment analysis is provided in DCD Tier 2, Section 6.2.1.2, and is summarized in part as follows.

> The application describes that several reactor system components are located within subcompartments in the containment vessel. Further, high-energy lines are routed inside the subcompartments, such as the branch lines from the reactor coolant piping, feedwater piping, and SG blowdown lines. The applicant evaluated the maximum differential pressure that the containment compartment walls would be subjected to as a result of the most limiting postulated line break within a particular compartment. This DCD section is very short and results of this analysis are not provided in the DCD but in the separate Technical Report MUAP-07031-P, “Subcompartment Analyses for US-APWR Design Confirmation,” Revision 1 (Reference 5). DCD Tier 2, Table 1.6-2, “Material Referenced as Technical Reports,” as supplemented by Reference 9, incorporates-by-reference Technical Report MUAP-07031-P, Revision 1. The staff opened **Confirmatory Item (CI)-SRP06.02.01.02-1** to verify that the final design

document incorporates the associated supplemental DCD information provided by Reference 9.

Several compartments within the US-APWR reactor containment house reactor system components, as well as high energy lines. They include:

1) Reactor cavity.
2) SG compartments.
3) Pressurizer sub-compartment.
4) Pressurizer surge piping room (underneath pressurizer compartment).
5) Pressurizer spray valve room (south side of pressurizer compartment).
6) Regenerative heat exchanger room (northwest side of pressurizer compartment).
7) Letdown heat exchanger room (south side of pressurizer compartment).

The first three of these components were analyzed by the applicant using the GOTHIC code to determine compartment maximum pressure. The pressurizer surge piping room was not analyzed based on leak-before-break (LBB) approach. The last three of the above compartments were not analyzed because the applicant concluded that vent paths from these compartments were large in comparison to potential pipe break sizes within these compartments.

The applicant assumed initial atmospheric conditions within the compartments to be air only (i.e., relative humidity equal to zero) at the maximum allowable temperature, and at minimum absolute pressure.

The application describes that analyses are performed to calculate the peak differential pressure of applicable subcompartment walls for the most severe specified pipe rupture. The application also describes the modeling techniques used in performing the calculations. Further, the application states that the US-APWR design does not rely on piping restraints to limit the break area of potential high-energy piping failures within these subcompartments. Calculation results for each of the analyzed subcompartments indicated that the peak differential pressures for each subcompartment were below the structural design differential pressures. SER Table 6.2-3, “Design and Peak Calculated Pressures for Subcompartment Analysis,” depicts the design and peak differential pressures for each subcompartment. The details of these calculations are not provided in the DCD. They were taken from Technical Report MUAP-07031-P (Reference 5) .

ITAAC: The ITAAC associated with DCD Tier 2, Section 6.2.1.2 are provided in DCD Tier 1, Section 2.11.1.

TS: The TS associated with DCD Tier 2, Section 6.2.1.2 are provided in DCD Tier 2, Chapter 16, Section 3.6.

Topical Reports: There are no topical reports submitted by the applicant for this area of review.

Technical Reports: Mitsubishi Heavy Industries, Ltd., Technical Report MUAP-07031-P, “Subcompartment Analyses for US-APWR Design Confirmation,” Revision 1, and MUAP-07031-NP, Revision 1, issued October 2009.

US-APWR Interface Issues identified in the DCD: There are no US-APWR interface issues for this area of review.

Site Interface Requirements Identified in the DCD: There are no site interface requirements for this area of review.

Cross cutting Requirements (TMI, USI/ GSI, Op Ex): None for this area of review.

RTNSS: There is no RTNSS for this area of review.

10 CFR 20.1406: There are no 10 CFR 20.1406 requirements for this area of review.

CDI: There is no CDI for this area of review.

6.2.1.2.3 Regulatory Basis

The relevant requirements of the Commission's regulations for this area of review, and the associated acceptance criteria, are given in Section 6.2.1.2, "Subcompartment Analysis," of NUREG-0800, the SRP, and are summarized below. Review interfaces with other SRP sections can be found in Section 6.2.1.2 of NUREG-0800.

1. GDC 4, "Environmental and dynamic effects design bases," requires that the containment internal compartments shall be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, which may result from equipment failures and from events and conditions outside the nuclear power unit. These include those environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including LOCAs. GDC 4 allows the dynamic effects associated with postulated pipe breaks in nuclear power units to be excluded from the design basis when analysis reviewed and approved by the Commission demonstrates that the probability of fluid system piping rupture is extremely low under conditions consistent with the design basis for the piping. Thus, GDC 4 allows the use of LBB to eliminate dynamic effects of the break as discussed above.

2. GDC 50, "Containment design basis," requires that the design of the containment and internal compartments ensure that the reactor containment structure, including access openings, penetrations, and the containment heat removal system are designed so that the containment structure and its internal compartments can accommodate the calculated pressure and temperature conditions resulting from any LOCA without exceeding the design leakage rate.

3. 10 CFR 52.47(b)(1) requires that the DC application contain proposed ITAAC that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and acceptance criteria met, a plant that incorporates the certified design is built and will operate in accordance with the DC, the provisions of the Atomic Energy Act of 1954, and the NRC's regulations.

Acceptance criteria adequate to meet the above requirements include:

1. GDC 4 allows the dynamic effects associated with postulated pipe ruptures in nuclear power units to be excluded from the design basis when analyses reviewed and approved

by the Commission demonstrate that the probability of fluid system piping rupture is extremely low under conditions consistent with the design basis for the piping.

2. GDC 50 requires that the containment structure and associated heat removal system be designed to accommodate, without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions resulting from a LOCA. Because a LOCA is the most severe challenge expected, the design of the containment and its subcompartments must be able to withstand such an accident. Providing sufficient design margin will assure that the design can withstand all postulated accidents regardless of unanticipated factors. Meeting the requirements of GDC 50 will help to maintain the structural integrity of containment subcompartments and protect the containment structure and systems from the effects of a subcompartment high-energy line break.

6.2.1.2.4 Technical Evaluation

The staff reviewed this section of the DCD and the supporting Technical Report MUAP-07031-P (Revision 1), "Subcompartment Analyses for US-APWR Design Confirmation" (Reference 5). Technical Report MUAP-07031, discusses the adequacy of the GOTHIC code use for licensing analysis through comparison to selected data. The data used are the Battelle-Frankfurt Test Facility and the HDR Full Scale Containment Experiments. The report was reviewed by the NRC staff and **RAI 111-932, Questions 06.02.01.02-2 through -14,** were issued to address different shortcomings in the report. **All the RAI question responses were acceptable, and RAI 111-932, Questions 06.02.01.02-2 through -14 are resolved and closed**.

The applicant applied LBB considerations to the primary system coolant piping and to the pressurizer surge line. SRP, Section 6.2.1.2, states "Although LBB technology allows applicants to eliminate consideration of local dynamic effects of postulated pipe ruptures in the design basis of structures, systems and components (SSC), the staff will continue to require consideration of the global effects of postulated pipe ruptures for the design of subcompartment enclosures because the global effects provide a convenient and conservative design envelope."

The application of the LBB criterion is consistent with that described in Reference 6, which states that when LBB is approved for a particular piping system, applicants can only exclude local dynamic effects from:

- Missiles.
- Pipe whipping.
- Pipe break reaction force.
- Discharging fluids.

and upon NRC approval, the applicant can:

- Remove jet impingement barriers or shields.
- Remove pipe whip restraints.
- Re-design pipe connected components.
- Disregard jet impingement forces on adjacent components, decompression waves within the intact portion of the piping system, and dynamic or nonstatic pressurization in cavities, subcompartments, and compartments.

In the US-APWR DCD the applicant states that when piping is classified as leak-before-break, "It is not necessary to analyze the dynamic effects of a postulated pipe rupture, including pipe whip, jet impingement loads, and sub-compartment pressurization." Although, the last of these three items seems to be in conflict with the SRP, the applicant's position is consistent with the memo discussed above (Reference 6). This approach was also approved during the AP-1000 application review, i.e., as in its review the staff stated "Application of LBB to the containment subcompartment analysis allows the postulated rupture of large pipes to be precluded from the spectrum of postulated breaks." The applicant interpretation of LBB requirements is consistent with this statement and the staff finds the applicant's approach acceptable.

Seven separate compartments within the US-APWR containment were identified by the applicant. The reactor cavity consists of a cylindrical narrow gap between the reactor vessel and the concrete primary shield wall, the space under the reactor vessel, and the reactor cavity access tunnel. Four, 100 mm (4-inch) diameter direct vessel injection (DVI) lines are connected to the reactor vessel within this cavity and are used to establish the limiting pipe break within this cavity. It has several vent paths including a path through the pipe sleeves penetrating the primary shield wall and leading to the SG compartment which is open to containment. In addition, a vent path exists in the gap between the reactor vessel and the primary shield wall to the pressurizer surge pipe room and then to the SG compartment.

The four SG compartments are formed by secondary shield walls surrounding the SGs and are open at the top to containment atmosphere. The SG compartment has a volume of approximately 1,250 m^3 (44,000 ft^3), and encloses several high-energy lines including feedwater piping, SG blowdown lines, branch lines from the reactor coolant piping, etc. The applicant states that SG compartment worst case analysis assumes a 150 mm (6-inch) diameter break of the pressurizer spray line or a 400 mm (16-inch) diameter feedwater pipe break. The feedwater pipe break is analyzed in report MUAP-07031, Section 7.3, "Steam Generator Subcompartment." Table 6.2-4 of this report, "Postulated Breaks and Compartment Design Pressures," lists the postulated breaks that have been analyzed by the applicant. However, the pressurizer spray line break is not one of the breaks analyzed in the DCD or in Technical Report MUAP-07031. This issue was addressed in the review of Technical Report MUAP-07031, as well and **RAI 111-932, Question 06.02.01.02-6,** was issued requesting clarification. In its response to RAI 111-932, Question 06.02.01.02-6, dated January 16, 2009, the applicant stated that the analysis of the pressurizer line break was not performed, but instead breaks at the Residual Heat Removal (RHR) pump outlet and inlet lines (which are 200 mm and 250 mm or 8-in and 10-in in diameter respectively) were analyzed. These breaks are more severe than the pressurizer spray line break and therefore envelope the pressurizer spray line break. The staff agrees that such an approach is conservative and, therefore, the applicant response, dated January 16, 2009, is acceptable. **RAI 111-932, Question 06.02.01.02-6, is resolved and closed**.

The pressurizer compartment houses the pressurizer and has a volume of approximately 570 m^3 (20,000 ft^3). Two personnel accesses are located in the ceiling of this compartment that provide vent paths to containment. In addition, an entrance to the SG compartment is located at the bottom of the pressurizer compartment, providing a third vent path. The worst case pipe break assumes a break in the 200 mm (8-inch) diameter pressurizer relief line located at the top of the pressurizer. The pressurizer surge pipe room is located underneath the pressurizer compartment. LBB considerations are applied to the surge pipe from the pressurizer, and using the applicant's interpretation for LBB discussed above for compartment analysis, a pipe break is not considered for this compartment.

The three additional compartments are not considered in the subcompartment analysis by the applicant because vent paths in these compartments are considered to be large in comparison to the line sizes in the compartments and will not result in significant differential pressure between the compartment and containment. These compartments are the regenerative heat exchanger room, the letdown heat exchanger room, and the pressurizer spray valve room. There was no explicit discussion of the volumes and vent path areas associated with these rooms by the applicant. This issue is addressed in **RAI 111-932, Question 06.02.01.02-4**. In its response to RAI 111-932, Question 06.02.01.02-4, dated February 2, 2009, the applicant stated that with additional analyses for the regenerative heat exchanger room, the regenerative heat exchanger valve room, the letdown heat exchanger room. All the calculated results are well below design pressures. The applicant did not perform a subcompartment analysis for the pressurizer spray valve room because there are no piping postulated breaks inside this room. The applicant’s response, dated February 2, 2009, is acceptable, and **RAI 111-932, Question 06.02.01.02-4, is resolved and closed**.

Mass and energy releases from the breaks were calculated using two methods depending on how close the peak pressure was to a limit. One is based on an assumption of a constant blowdown profile with an appropriate choked flow correlation. The second is similar to the one used in a SBLOCA analysis. Assuming a constant blowdown profile is an acceptable method of calculating mass and energy release rates as documented in SRP, Section 6.2.1.3, “Mass and Energy Release Analysis for Postulated Loss-of-Coolant Accidents (LOCAs).” However the statement in the DCD seems to be inconsistent with that provided in Technical Report MUAP-07031, Section 4.2, “Short Term Mass and Energy Release,” which states that M-RELAP5 is used for mass and energy release rate calculations for primary piping breaks, and the constant blowdown profile technique along with initial mass and energy release conditions by M-RELAP5 for secondary piping breaks. This issue was addressed in **RAI 111-932, Question 06.02.01.02-7,** which requested clarification from the applicant. In its response to RAI 111-932, Question 06.02.01.02-7, dated February 2, 2009, the applicant stated that with a modified methodology that is more conservative. The modified methodology and the revised results are included in the Revision 1 of MUAP-07031. The response to this RAI, dated February 2, 2009, is acceptable, and **RAI 111-932, Question 06.02.01.02-7, is resolved and closed**. Section 6.2.1.4, “Mass and Energy Release Analysis for Postulated Secondary System Pipe Ruptures,” of this SER discusses the containment response analysis for the postulated secondary system pipe ruptures.

Initial containment atmospheric conditions for the GOTHIC calculations were chosen consistent with those prescribed in SRP, Section 6.2.1.2, i.e., air at the maximum allowable temperature, minimum absolute pressure, and zero-percent relative humidity. Initial temperatures vary between 49°C and 60°C (120°F – 140°F) with all the initial pressures being 101 kPa (0 psig). The compartment flow calculations have been performed using homogeneous, equilibrium conditions consistent with SRP, Section 6.2.1.2, Acceptance Criterion 3. Each compartment was modeled separately by the applicant, with the compartment nodalization chosen to minimize pressure gradients within each node. The applicant also performed limited nodalization studies to verify their nodalization scheme. This issue was addressed in **RAI 111-932, Question 06.02.01.02-8,** which requested clarification from the applicant.

In its response to RAI 111-932, Question 06.02.01.02-8, dated February 2, 2009, the applicant provided modified methodology and modified calculations as Appendices A and B. The applicant’s response, dated February 2, 2009, is acceptable, and **RAI 111-932, Question 06.02.01.02-8, is resolved and closed**.

Node selection in the cavity compartment was performed by the applicant in accordance with Section 3.2.2.1 of NUREG-0609 (Reference 8). The applicant completed several sensitivity studies reported in Reference 5. A nodalization sensitivity studies was performed for the pressurizer compartment using two nodalizations. Time step sensitivity studies were also completed to justify the time step used in the calculations. Additional sensitivity studies included vent loss coefficients, inertia terms, different options for running GOTHIC and comparison with another code, as described in Sections 5 and 6 of Reference 8.

SER Tables 6.2-3 and 6.2-4 summarize the evaluation of the postulated breaks that were analyzed and the associated compartments. These results are from Technical Report MUAP-07031 (Reference 5). For all compartments, the postulated breaks envelop other line breaks that could be postulated to rupture. This was evaluated by developing a list of all high-energy lines within each compartment. For each compartment, all lines except those covered by LBB considerations were grouped according to the pressure and temperature of fluid in the line. Qualitative evaluation was performed to eliminate some lines from further consideration (mass and energy available, etc.) and a detailed pipe break simulation was performed for the largest diameter line remaining in each group for each compartment. In **RAI 111-932, Question 06.02.01.02-5,** the staff requested additional information regarding a complete list of lines consider during the selection process and not included in the original Table 3-1, "Subcompartment and Postulated Break Line Condition," of MUAP-07031-P (Reference 5). In its response to RAI 111-932, Question 06.02.01.02-5, dated January 16, 2009, the applicant provided the revised Table 3-1 as requested. This table is also presented as DCD Tier 2 Table 6.2.1-17, "Subcompartment and Postulated Break Line Condition." **RAI 111-932, Question 06.02.01.02-5, is resolved and closed.**

Table 6.2-3
Design and Peak Calculated Pressures for Subcompartment Analysis
(Data taken from MUAP-07031-P, Revision 1, Reference 5)

Compartment	Design diff. pressure		Peak calculated diff. press	
	psid	kPad	psid	kPad
Reactor cavity	39/14	269/96.5	7.56/4.91	52.1/33.9
SGcompartment	18/7/14	124/48.3/96.5	3.16/4.15/13.81	21.8/28.6/95.2
Pressurizer	14	96.5	6.92	47.7

Note: multiple numbers indicate multiple locations in the compartment

Table 6.2-4 Postulated Breaks and Compartment Design Pressures

Compartment	**Design diff. pressure**		**Postulated break**
	Psid	kPad	
Reactor cavity	39/14	269/96.5	DVI line (100 mm[4-in])
SG compartment	18 7 14	124 48.3 96.5	RHR pump inlet (250 mm [10-in]) RHR pump outlet (200 mm [8-in]) Feedwater line (400 mm [16-in])
Pressurizer compartment	14	96.5	Spray line (150 mm [6-inch]) Relief line (200 mm [8-inch])

Note: multiple numbers indicate multiple locations in the compartment

A separate GOTHIC model was prepared for each compartment and calculations performed for each break location. The GOTHIC model includes multiple nodes for each compartment, vent paths from the compartment, with the nodalization selected so that node boundaries are at the location of flow obstructions. Details of the models are presented in Reference 5. The selected nodalization, vent flow paths, and the specified distribution of mass and energy release were expected to lead to a maximum subcompartment pressure. The options chosen in the GOTHIC input include homogenous equilibrium model (HEM), 100 percent water entrainment, and homogeneous equilibrium vent choking model. The model disallows deposition of droplets in volumes. Vent area changes due to insulation collapsing are not considered. No new vents were employed after the time of pipe rupture. All of these assumptions are consistent with acceptance criteria specified in SRP, Section 6.2.1.2. The results, summarized in Table 6.2-4 and detailed in Technical Report MUAP-07031-P (Revision 1) (Reference 5).

The staff has completed confirmatory calculations of the sub-compartment analyses using the computer code COMPARE (Reference 7). A DEGB of a feedwater line in the SG sub-compartment and a DVI line break in the annular region of the cavity were simulated with the code COMPARE. The results were comparable to the results reported by the applicant in the DCD.

6.2.1.2.5 Combined License Information Items

There are no COL information items identified for this section in DCD Tier 2, Table 1.8-2.

6.2.1.2.6 Conclusions

As a result of the open item the staff is unable to finalize its conclusions on Section 6.2.1.2 related to containment sub-compartments, in accordance with NRC regulations.

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

6.2.1.3.1 Introduction

The analyses of the mass and energy release are reviewed to assure that the data used to evaluate the containment and subcompartment functional design are acceptable for that purpose. Specific to the review are the following:

1. Energy sources that are available for release to the containment.

2. Mass and energy release rate calculations for the initial blowdown phase of the accident.

3. Mass and energy release rate calculations for the SG energy available with respect to the core reflood and post-reflood phases.

6.2.1.3.2 Summary of Application

DCD Tier 1: The Tier 1 information associated with this evaluation is provided in DCD Tier 1, Section 2.4.4, “Emergency Core Cooling System (ECCS).”

DCD Tier 2: The Tier 2 information associated with this evaluation is provided in DCD Tier 2, Section 6.2.1.3, and is summarized in part as follows:

The application presented information concerning a postulated LOCA transient. The application identifies that a LOCA is typically divided into the following four phases:

1. Blowdown phase - which includes the period from accident initiation (when the reactor is operated at full power) to the time that the RCS pressure reaches equilibrium with containment.

2. Refill phase - the period when the lower plenum is being filled by ECCS injection water up to the bottom of the core. This period is conservatively ignored to maximize the release rate to the containment in the evaluation model described later.

3. Core reflood phase - begins when the water from the lower plenum enters the core and ends when the core is completely quenched.

4. Long-term cooling phase - describes the period after the core has been quenched and energy is released to the containment via reactor coolant by the RCS metal, core decay heat, and the SGs.

Multiple codes are used by the applicant to determine the mass and energy release rates due to a LOCA. Details of the SATAN-VI(M1.0), WREFLOOD(M1.0), and GOTHIC computer codes and evaluation models used to calculate mass and energy release rates are presented in the report of Reference 2, “LOCA Mass and Energy Release Analysis Code Applicability Report for US-APWR.” No model details are given in this section of the DCD. The applicant methodology used in developing the LOCA mass and energy release rates was reviewed and approved by the staff (Memorandum from Christopher Jackson to Larry Burkhart, issued June 30, 2008).

The applicant presented four tables of mass and energy blowdown for four cases: the DEPSG break, the double ended hot leg guillotine (DEHLG) break, the reflood phase of the DEPSG break, and the long-term cooling phase of the DEPSG break. Three additional tables are presented, one with elevations, flow areas and hydraulic diameters, another one with the safety injection flow rates, and the last one with the stored energy inside the primary system to be released during a LOCA.

ITAAC: The ITAAC associated with DCD Tier 2, Section 6.2.1.3 are provided in DCD Tier 1, Section 2.11.1.

TS: The TS associated with DCD Tier 2, Section 6.2.1.3, are provided in DCD Tier 2, Chapter 16, Section 3.5.1, “Accumulators,” Section 3.5.2, “Safety Injection System (SIS) - Operating,” Section 3.5.3, “Safety Injection System (SIS) - Shutdown,” Section 3.5.4, “Refueling Water Storage Pit (RWSP),” Section 3.7.2, “Main Steam Isolation Valves (MSIVs),” Section 3.7.7, “Component Cooling Water (CCW) System,” Section 3.7.8, “Essential Service Water System (ESWS),” and Section 3.8.1, “AC Sources - Operating.”

Topical Reports: Mitsubishi Heavy Industries, Ltd., Topical Report MUAP-07012-P-A, “LOCA Mass and Energy Release Analysis Code Applicability Report for US-APWR,” Revision 2 and MUAP-07012-NP, Revision 2, issued June 2009.

Technical Reports:

- Mitsubishi Heavy Industries, Ltd., Technical Report MUAP-07031-P, "Subcompartment Analyses for US-APWR Design Confirmation," Revision 1 and MUAP-07031-NP, Revision 1, issued October, 2009.

US-APWR Interface Issues identified in the DCD: There are no US-APWR interface issues for this area of review.

Site Interface Requirements identified in the DCD: There are no site interface requirements for this area of review.

Cross-cutting Requirements (TMI, USI/GSI, Op Ex): None for this area of review.

RTNSS: There is no RTNSS for this area of review.

10 CFR 20.1406: There are no 10 CFR 20.1406 requirements for this area of review.

CDI: There is no CDI for this area of review.

6.2.1.3.3 Regulatory Basis

The relevant requirements of the Commission's regulations for this area of review, and the associated acceptance criteria, are given in Section 6.2.1.3, "Mass and Energy Release Analysis for Postulated LOCAs," of NUREG-0800, the SRP, and are summarized below. Review interfaces with other SRP sections can be found in Section 6.2.1.3 of NUREG-0800.

1. GDC 50 requires that the containment structure and its internal compartments be designed to accommodate the calculated pressure and temperature conditions resulting from any LOCA accident without exceeding the design leakage rate, including sufficient margin. In order to satisfy this requirement, the LOCA analysis should be based on the assumption of LOOP and the most severe single failure in the emergency power system (i.e., a diesel generator failure), the containment heat removal systems (i.e., a fan, pump, or valve failure), or the core cooling systems (i.e., a pump or valve failure). The selection of the single failure must result in the highest calculated containment pressure. The analysis must be based on a spectrum of pipe break sizes and reactor power levels. Accident conditions selected should result in the highest calculated containment pressure or temperature depending on the purpose of the analysis.

2. 10 CFR Part 50, Appendix K, "ECCS Evaluation Models," requires that all sources of energy be considered in analyzing a LOCA event. These include but are not limited to: reactor power, decay heat, stored energy in the core, stored energy in the RCS metal, including the reactor vessel internals, metal-water reaction energy, and stored energy in the secondary system, including the SG tubing and secondary water. Calculations of energy release must be done in general accordance with the requirements of 10 CFR Part 50, Appendix K, paragraph 1.A. Additional conservatism must be included to maximize the energy release to containment during the blowdown and reflood phases of a LOCA event. The requirements of paragraph 1.B in 10 CFR Part 50, Appendix K, concerning prediction of clad swelling and rupture should not be considered in order to maximize the energy available for release from the core.

3. 10 CFR 52.47(b)(1) requires that the DC application contain proposed ITAAC that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and acceptance criteria met, a plant that incorporates the certified design is built and will operate in accordance with the DC, the provisions of the Atomic Energy Act of 1954, and the NRC's regulations. ITAAC are addressed in Section 14.3.11 of this report.

Acceptance criteria adequate to meet the above requirements include:

1. Sources of Energy

   The sources of stored and generated energy that should be considered in analyses of LOCAs include: reactor power; decay heat; stored energy in the core; stored energy in the RCS metal, including the reactor vessel and reactor vessel internals; metal-water reaction energy; and stored energy in the secondary system (PWR plants only), including the SG tubing and secondary water.

2. Break and Size Location

   The staff's review of the applicant's choice of break locations and types is discussed in Section 3.6.2, "Determination of Rupture Locations and Dynamic Effects Associated with the Postulated Rupture of Piping," of NUREG-0800.

   a. Of the several breaks postulated on the basis of the sources of energy listed in "1.", above, the break selected as the reference case for subcompartment analysis should yield the highest mass and energy release rates, consistent with the criteria for establishing the break location and area.
   b. Containment design basis calculations should be performed for a spectrum of possible pipe break sizes and locations to assure that the worst case has been identified.

3. Calculations

   In general, calculations of the mass and energy release rates for a LOCA should be performed in a manner that conservatively establishes the containment internal design pressure (i.e., maximizes the post-accident containment pressure and the containment subcompartment response). The criteria given below for each phase of the accident indicate the conservatism that should exist.

   a. Subcompartment Analysis.
   b. Initial Blowdown Phase Containment Design Basis.
   c. PWR Core Reflood Phase (Cold Leg Breaks Only).
   d. PWR Post-Reflood Phase.
   e. PWR Decay Heat Phase.

6.2.1.3.4 Technical Evaluation

The applicant developed tables of mass and energy releases for LOCAs. The containment mass and energy release analysis used a combination of codes depending on the phase of the transient. SATAN-VI (M1.0) was used for the blowdown phase of the transient. This phase includes the period from break initiation to the time that the RCS pressure reaches equilibrium with containment. During this phase, as high pressure, high temperature water is released from the break, it flashes into a mixture of steam and liquid entering the containment. The primary system depressurizes reaching saturation pressure at the water temperature. The primary then depressurizes more slowly as the heat within the system is removed by boiling, and additional coolant inventory is lost through the break. Once primary system pressure equalizes with containment pressure, this phase is assumed to end. At this point in time, the liquid level within the core is below the top of the heated length. The refill phase of the transient begins at the end of blowdown and continues until the ECCS is able to refill the primary vessel lower plenum. The applicant has chosen to neglect this phase of the transient for their LOCA analysis which should maximize the release rate to the containment using their evaluation methods. The core reflood phase of the transient was analyzed using the WREFLOOD (M1.0) code. This is the period beginning when water in the lower plenum enters the core and ends when the core is completely quenched. This phase starts with part of the core operating at temperatures above the minimum film boiling point, and is characterized by a quench front moving from a point below the top of the heated length, until the core is completely covered with water. During this period, a steam/water mixture is exiting the break as liquid is added to quench the core. The GOTHIC code is used for the long-term cooling phase of the transient after the core is quenched, and energy is released to containment from sensible heat in the cooling system structures, the core decay heat, and stored energy in the SGs. This cooling phase corresponds to the “Post-Reflood Phase” and the “Decay Heat Phase” of Acceptance Criterion 1.C of NUREG-0800, Section 6.2.1.3. The decay heat employed is based on ANSI/ANS-5.1-1979 model (ANS-1979 model).

For subcompartment analysis, the applicant notes in DCD Section 6.2.1.2, “Containment Subcompartments,” that they have used two different techniques to determine mass and energy release, (1) the assumption of a constant blowdown profile (acceptable according to NUREG-0800, Section 6.2.1.2), and (2) an analytical approach with a computer code and volume noding of the piping system similar to those of small-break (SB) LOCA analysis. The second approach employs the M-RELAP code as noted in MUAP-07031-P, Revision 1 (Reference 5). No details of the mass and energy release calculation techniques used in the subcompartment analysis were given in Section 6.2.1.3 of the DCD, in the subcompartment Section 6.2.1.2 of the DCD, or in the subcompartment analysis report, MUAP 07031-P(Revision 1). In the review of Technical Report MUAP-07031, the staff requested clarification of the methods the applicant used to determine mass and energy release rates for the subcompartment analysis in **RAI 111-932, Question 06.02.01.02-7**. In its response to RAI 111-932, Question 06.02.01.02-7, the applicant described a modified methodology that results in more conservative mass and energy releases for secondary piping breaks. Values for both the feedwater pump side and the SG side mass and energy releases were provided. The response to **RAI 111-932, Question 06.02.01.02-7,** is acceptable and **RAI 111-932, Question 06.02.01.02-7**, is **resolved and closed.**

A spectrum of break sizes was evaluated for containment integrity analysis. These included both primary system hot- and cold-leg breaks. Three break locations were evaluated, the hot leg between the reactor vessel and the SG, the cold leg at the RCP discharge, and the cold leg at the RCP suction. In addition, a range of break sizes was investigated in the cold leg; from a double-ended guillotine break to a break 0.28 m^2 (3 ft^2) in effective flow cross sectional area. The double-ended guillotine break of the cold leg at the RCP suction (DEPSG) was found to be most limiting. It resulted in the maximum containment pressure and temperature of all the LOCAs analyzed (Section 6.2.1.1). The hot-leg break (DEHLG) was less severe as the energy released by the SG is minimized due to fluid from the core bypassing the hot SG and exiting the break. The cold leg break (DECLG) at the RCP discharge is also less limiting than the pump suction break (DEPSG) because in this case, the flow resistance between the core outlet and the break location is maximized and the amount of energy released from the broken-loop SG is less than for the pump suction break. The staff review of the applicant's choice of break locations and types is covered in Section 3.6.2 of this SER.

The subcompartment energy releases and break sensitivity studies for subcompartment analysis are presented in MUAP-07031-P, Revision 1 (Reference 5). The break sensitivity studies satisfy SRP Section 6.2.1.3 acceptance criteria stemming from GDC 50 and 10 CFR Part 50, Appendix K, requirements to evaluate the most severe break location for containment analysis. All the available energy sources were accounted for in the analysis in accordance with Appendix K to 10 CFR Part 50. They include: energy stored in the reactor vessel and reactor vessel internals, energy stored in the secondary system and secondary water, decay heat, sensible heat of the core, reactor coolant and metal, SG fluid and metal energy, accumulators, refueling water storage pit, and metal-water reaction.

Conservative assumptions were also made in calculating the available energy as follows:

- margin in volume of +3 percent,
- allowance for calorimetric error of +2 percent core power,
- maximum core stored energy considering burn-up and uncertainty in fuel temperature,
- margin in core stored energy of +20 percent,
- maximum expected operating temperature,
- allowance in initial temperature to account for instrument error and dead band of +4F,
- allowance in RCS pressure uncertainty of +30 psi,
- maximum SG mass inventory,
- metal-water reaction from one percent of the zirconium in the active core cladding
- decay heat maximized, based on ANSI/ANS-5.1-1979 model (ANS-1979 model).

In its response to **RAI 331-935, Question 06.02.01-16**, dated May 26, 2009, the applicant provided additional justification regarding the conservatism of these assumptions. In particular, the assumptions of 1-percent zirconium oxidation is conservative given a conclusion in Chapter 15, "Transient and Accident Analyses", that the clad temperature never reaches the level that would cause any significant metal-water reaction. The RAI response is acceptable and **RAI 331-935, Question 06.02.01-16, is resolved and closed**.

These energy sources and conservative assumptions agree with recommendations of SRP Section 6.2.1.3 to maximize the calculated temperature and pressure. They are in accordance with the requirements of 10 CFR Part 50, Appendix K, paragraph I.A.

6.2.1.3.5 Combined License Information Items

There are no COL information items identified for this section in DCD Tier 2, Table 1.8-2.

6.2.1.3.6 Conclusions

GDC 50 requires that the containment structure and its internal compartments be designed to accommodate the calculated pressure and temperature conditions resulting from any LOCA accident including sufficient margin. In order to satisfy this requirement, the LOCA analysis should be based on the assumption of LOOP and the most severe single failure in the emergency power system. In its analysis, the applicant has assumed a LOOP, and has performed parametric evaluations on single failures in the emergency power system to determine the most limiting LOCA scenarios. The applicant satisfies the requirements established in GDC 50 for using the most severe single failure in its analysis. The calculations of mass and energy released during the postulated LOCAs also follow the requirements of 10 CFR 50 Appendix K in accordance with Section 6.2.1.3 of NUREG-0800.

6.2.1.3.6.1 Conclusions for Sections 6.2.1.1 through 6.2.1.3

The results shown and methodology used in Sections 6.2.1.1, 6.2.1.2 and 6.2.1.3 of the US-APWR DCD meet the requirements established by the acceptance criteria presented in the appropriate sections of NUREG-0800.

Confirmatory analyses performed by the staff, agree with the applicant's conclusions regarding the acceptability of the containment design. These analyses include containment integrity and compartment verification calculations, as well as confirmation of the mass and energy release rates used in the applicant calculations. They showed that the mass and energy release rates for LOCA were calculated to maximize the post-accident containment pressure and the containment subcompartment response.

6.2.1.3.7 References

1. GOTHIC Containment Analysis Package User Manual, Version 7.2a(QA), NAI 890702, Revision17, Numerical Applications Inc., Richland, WA, January 2006.

2. Mitsubishi Heavy Industries, Ltd., "LOCA Mass and Energy Release Analysis Code Applicability Report for US-APWR," MUAP-07012-P-A Revision 2 (Proprietary) and MUAP07012-NP Revision 2 (non-Proprietary), June 2009.

3. GOTHIC Containment Analysis Package Qualification Report, Version 7.2a(QA), NAI 8907 09, Revision 9, Numerical Applications Inc., Richland, WA, January 2006.

4. MELCOR Computer Code, Version 1.8.6, NUREG/CR-6119, SAND2005-5713, Revision 3, September 2005.

5. Mitsubishi Heavy Industries, Ltd., "Subcompartment Analyses for US-APWR Design Confirmation," MUAP-07031-P(R1), proprietary and MUAP-07031-NP(R1), non-proprietarily, October, 2009.

6. Memo to John A. Grobe, Associate Director for Engineering and Safety Systems, Office of Nuclear Reactor Regulation, from Michele G. Evans Director, Division of Component Integrity, Office of Nuclear Reactor Regulation with Subject "Division of Component Integrity Response to Yellow Ticket No 020060239, Differing Professional Opinion (DPO) Panel Response to DPO2006-003" (ML071500582).

7. R. G. Guido et al, "COMPARE-MOD1 Code Addendum," NUREG/CR-1185 Addendum

1, LA-7199-MS, June 1980.

8. US. Nuclear Regulatory Commission, NUREG-0609, “Asymmetric Blowdown Loads on PWR Primary Systems, Resolution of Generic Task Action Plan A-2” January 1981.

9. Mitsubishi Heavy Industries, Ltd., US-APWR DCD Revision 3 Tracking Report, MUAP-11021(Revision 1), December 2011 (ML12030A217).

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

6.2.1.4.1 Introduction

Secondary pipe ruptures inside the reactor containment may result in significant releases of high-energy water that can generate high pressure and temperatures inside the containment. Mass and energy releases for both, MSLB and main feedwater line break (MFLB), are considered in this section. The main objective of this section is to provide conservative input (mass and energy releases) for the calculation of the pressures and temperatures in the containment after secondary pipe ruptures.

6.2.1.4.2 Summary of Application

DCD Tier 1: There is no DCD Tier 1 (Reference 1) information provided for this area of review.

DCD Tier 2: The Tier 2 information associated with this evaluation is provided in DCD Tier 2, Section 6.2.1.4, and is summarized in part as follows:

> This section describes the sequence of events and effects of transient phenomena, the evaluation model, the input parameters and input conditions, and the evaluation results for the postulated secondary system pipe ruptures. Feedwater pipe ruptures are not considered. The applicant analyzed a total of nine MSLBs inside the containment. The results are provided in tables and figures that were discussed in SER Section 6.2.1.1.4, “Technical Evaluation.” Case 1 resulted in the highest containment temperature, 179ºC (355ºF), and Case 5 in the highest containment pressure, 437.8 kPa (48.8 psig). Case 8 also resulted in the same high containment temperature as Case 1, 179ºC (355ºF).
>
> The mass and energy release analysis performed on the nuclear steam supply system (NSSS) is separate from the containment response analysis. Different sets of assumptions regarding single failures and availability of offsite power may be made for these two analyses for the purpose of assuring that the analyzed containment response bounds combinations of plant operating conditions, break characteristics, and pertinent combinations of assumed failures.
>
> The application describes the mass and energy evaluation model for a postulated steam piping failure (MSLB). The model is based on the MARVEL-M plant transient analysis code. The application also describes the input parameters used in the MARVEL-M analysis.

ITAAC: The ITAAC associated with DCD Tier 2, Section 6.2.1.4 are provided in DCD Tier 1, Section 2.11.1.

TS: The TS associated with DCD Tier 2, Section 6.2.1.4 are provided in DCD Tier 2, Chapter 16, Section 3.6.1, “Containment,” Section 3.6.6, “Containment Spray System,” and Section 3.8.1, “AC Sources - Operating.”

Topical Reports: Mitsubishi Heavy Industries, Ltd, Topical Report MUAP-07010-P, “Non-LOCA Methodology,” Revision 1, and MUAP-07010-NP, Revision 1, issued October 2010.

Technical Reports: There are no technical reports submitted by the applicant for this area of review.

US-APWR Interface Issues identified in the DCD: There are no US-APWR interface issues for this area of review.

Site Interface Requirements Identified in the DCD: There are no site interface requirements for this area of review.

Cross cutting Requirements (TMI, USI/ GSI, Op Ex): None for this area of review.

RTNSS: There is no RTNSS for this area of review.

10 CFR 20.1406: There are no 10 CFR 20.1406 requirements for this area of review.

CDI: There is no CDI for this area of review.

6.2.1.4.3 Regulatory Basis

The relevant requirements of the Commission’s regulations and the associated acceptance criteria are given in Section 6.2.1.4, “Mass and Energy Release Analysis for Postulated Secondary System Pipe Ruptures,” of NUREG-0800, the SRP, and are summarized below. Review interfaces with other SRP sections can be found in Section 6.2.1.4 of NUREG-0800.

1. GDC 50, as it relates to providing sufficient conservatism in the mass and energy release analysis for postulated PWR secondary system pipe ruptures 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 LOCA.

2. 10 CFR 52.47(b)(1), which requires that a DC application include 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 DC, the provisions of the Atomic Energy Act of 1954, and the NRC's regulations.

Acceptance criteria, as stated in SRP Section 6.2.1.4, adequate to meet the above requirements include:

1. Sources of Energy - 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 SG’s metal, including the vessel tubing, feedwater line, and steamline; stored energy in the

water contained within the affected SG; stored energy in the feedwater transferred to the affected SG before closure of the isolation valves in the feedwater line; stored energy in the steam from the unaffected SG(s) before the closure of the isolation valves in the SG crossover lines; and energy transferred from the primary coolant to the water in the affected SG during blowdown.

2. Mass and Energy Release Rate - In general, 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 (i.e., the post-accident containment pressure and temperature are maximized). The following criteria indicate the degree of conservatism that is desired:

    a. Mass release rates should be calculated using the Moody model for saturated conditions or a model that is demonstrated to be equally conservative.

    b. Calculations of heat transfer to the water in the affected SG should be based on nucleate boiling heat transfer.

    c. Calculations of mass release should consider the water in the affected SG and feedwater line, feedwater transferred to the affected SG before the closure of the isolation valves in the feedwater lines, steam in the affected SG, and steam coming from the unaffected SG(s) as the secondary system is being depressurized before the closure of the isolation valves in the SG crossover lines.

    d. If liquid entrainment is assumed in the steamline breaks, experimental data should support the predictions of the liquid entrainment model. The effect on the entrained liquid of steam separators located upstream from the break should be taken into account. A spectrum of steamline breaks should be analyzed, beginning with the double-ended break and decreasing in area until no entrainment is calculated to occur. This will allow selection of the maximum release case.

    e. If no liquid entrainment is assumed, a spectrum of the steamline breaks should be analyzed beginning with the double-ended break and decreasing in area until it has been demonstrated that the maximum release rate has been considered.

    f. Feedwater flow to the affected SG should be calculated considering the diversion of flow from the other SGs, feedwater flashing, and increased feedwater pump flow caused by the reduction in SG pressure. An acceptable method for computing feedwater flow is to assume all feedwater travels to the affected SG at the pump runout rate before isolation. After isolation, the unisolated feedwater mass should be added to the affected SG.

Operator action to terminate auxiliary feedwater flow will be reviewed under Section 10.4.9, “Auxiliary Feedwater System (PWR),” of NUREG-0800.

Any general-purpose thermal-hydraulics computer codes that the responsible reviewing organization for the subject application finds acceptable may be used to compute mass and energy releases from steam and feedwater line break accidents.

3. Single-Failure Analyses

   a. Steam and feedwater line break analyses should assume a single active failure in the steam or feedwater line isolation provisions or feedwater pumps to maximize the containment peak pressure and temperature. For the assumed failure of a safety-grade steam or feedwater line isolation valve, operation of non-safety-grade equipment may be relied upon as a backup to the safety-grade equipment. In this event, the reviewer will confer with the responsible organizations for Sections 3.2.1, "Seismic Classification," 3.2.2, "System Quality Group Classification," 3.6.2, "Determination of Rupture Locations and Dynamic Effects Associated with the Postulated Rupture of Piping," and 10.4.9 of NUREG-0800 to ensure a consistent staff position regarding the acceptability of the design criteria for the non-safety-grade equipment.

   b. The areas of review are sources of energy, mass and energy release rates, single failure analyses, and ITAAC.

   c. This review interfaces with Sections 6.2.1 (containment design), 3.2.1, 3.2.2 (steam and feed-water line isolation valves), 3.6.2 (postulated pipe break locations and sizes), and Section 10.4.9, "Emergency Feedwater System," (operator actions for valves).

   d. The ITAAC and its regulatory criteria for DCs is in 10 CFR 52.47(b)(1) and for COL applications in 10 CFR 52.80(a). The ITAAC review is presented in Section 14.3.11.

4. Additional criteria are in RG 1.206, "Combined License Applications for Nuclear Power Plants (LWR Edition)."

6.2.1.4.4 Technical Evaluation

MSLBs release significant amounts of high-energy steam that will result in high containment temperatures and pressures. Main feedwater pipe ruptures were not considered by the applicant because, according to the applicant, these pipe ruptures do not increase the containment temperature or pressure significantly.

There is a total of nine calculations completed by the applicant. The calculations cover a spectrum of power levels (0, 25 percent, 50 percent, 75 percent and 100 percent), break sizes (DEGB with 0.13 m^2 [1.4 ft^2] break area and two split breaks with break areas of 0.15 m^2 [1.65 ft^2] and 0.16 m^2 [1.71 ft^2]) and failure conditions. The results of these nine MSLB calculations are provided as tables and graphs that were presented in the previous DCD Subsection 6.2.1.1.3.4 of DCD Tier 2, together with the LOCA results.

The tables with the description of the MSLB cases and with the results are:

   1. Table 6.2.1-8, "Description and Summary Results for Evaluations of Various Pipe Sizes and Break Locations for Postulated Secondary Steam System Piping Failures (includes Plant Power Levels)," (two pages), that summarizes the pressure and temperature results for the nine cases analyzed,

2. Table 6.2.1-15, "Selected Key Events for the Secondary Steam System Piping Failure Transient Case 5 - Highest Containment Pressure", that describes key events for Case 5, "Highest Containment Pressure,"

3. Table 6.2.1-16, "Selected Key Events for the Secondary Steam System Piping Failure Transient Case 1 - Highest Containment Temperature," that describes key events for Case1, "Highest Containment Temperature,"

4. Table 6.2.1-25, "Description for Evaluations of Various Pipe Sizes and Break Locations for the Secondary Steam System Piping Failures (Includes Plant Power Levels)," describing the nine cases analyzed,

5. Table 6.2.1-26 (5 pages), "Mass and Energy Release Data for the Secondary Steam System Piping Failure Case 5 - Highest Containment Pressure," with the mass and energy release for Case 5, the case that results in the highest containment pressure, and

6. Table 6.2.1-27 (4 pages), "Mass and Energy Release Data for the Secondary Steam System Piping Failure Case 1 - Highest Containment Temperature," with the mass and energy release for Case 1, the case that results in the highest temperature.

The figures showing the MSLB results are Figures 6.2.1-39 through 6.2.1-65, a total of 27 figures, three figures for each of the nine transients, showing containment pressure, containment temperature, and RWSP water temperature as a function of time for the nine transients analyzed.

The maximum pressure calculated for the worst MSLBs is 437.8 kPa (48.8 psig), which is below the containment design pressure of 570 kPa (68 psig) and also below the maximum pressure calculated for the worst LOCA, 511.6 kPa (59.5 psig).

The maximum temperature calculated for the worst MSLB is 179ºC (355ºF), which is higher than the maximum temperature calculated for the worst LOCA, 140ºC (284ºF) and also above the containment design temperature of 149ºC (300ºF). This higher than design temperature was addressed in SER Section 6.2.1.1.4 and will be addressed later in this Technical Evaluation.

The codes employed in these calculations are MARVEL-M (Reference 3), for calculating the mass and energy release rates into containment, and GOTHIC (Reference 4), for containment pressure and temperature calculations. Both, the MARVEL-M and the GOTHIC computer codes have been reviewed by the staff and they are appropriate for these calculations. The sources of energy, both in the primary and in the secondary systems (including feedwater lines, steam lines, water and steam in the SG, etc) that are available for release in the MSLBs, were not provided in this section of the DCD. **RAI 110-784, Question 06.02.01-1,** was issued requesting the applicant for these sources of energy. In its response to RAI 110-784, Question 06.02.01-1, dated December 22, 2008, the applicant provided the sources of energy. They include: stored energy in the affected SG's metal (tubing, feedwater line and steamline), stored energy in the water of the affected SG, stored energy in the feedwater transferred to the affected SG until the closure of the isolation valves, stored energy in the steam of the other (unaffected) SGs until the closure of the isolation valves in the SG crossover lines and energy transferred from the primary coolant to the water in the affected SG during blowdown. The staff finds the RAI response, dated December 22, 2008, acceptable and **RAI 110-784, Question 06.02.01-1, is resolved and closed**. There are two tables in the DCD Tier 2, Table 6.2.1-12, "Distribution of Energy at Selected Locations within Containment for the Worst-Case Postulated DEPSG Break," and Table 6.2.1-14, "Distribution of Energy at Selected Locations within Containment for the Worst-Case Postulated DEHLG Break," which provide the energy inventories for the LOCAs.

The assumptions and input employed for these calculations are conservative to maximize the mass and energy releases. Dry steam blowdown is assumed. Feedwater flashing is assumed for the main feedwater line. The Moody model for f(L/D) = 0 is used for saturated conditions (Reference 5). Feedwater and emergency feedwater flows are considered at maximum values. Feedwater flow to the affected SG is increased due to the reduction in SG pressure. The maximum main feedwater flow is based on the assumption that the SG is at atmospheric pressure for a DEGB. For split breaks, the main feedwater flow is the same as the total steam flow, including the break flow. The break is assumed to be in the loop with the pressurizer. The RCP trip is ignored when power is available. No operator actions are considered.

During the April 16 - 17, 2012, audit, the staff found that the mass and energy release from a postulated MSLB is calculated using a constant RWSP water temperature in the MARVEL-M code, and not the actual water temperature during the transient. The staff issued **RAI 947-6540, Question 06.02.01-23**, to ascertain why using a constant RWSP temperature would be conservative. In its response to RAI 947-6540, Question 06.02.01-23, dated August 1, 2012, the applicant submitted a sensitivity analysis of the RWSP water temperature for the limiting DEGB Case 5 of containment pressure as described in DCD Section 6.2.1.4. The response, dated August 1, 2012, showed that the RWSP water temperature transient has a negligible impact on the mass and energy release. The staff finds the RAI response acceptable, and **RAI 947-6540, Question 06.02.01-23, is resolved and closed**.

All the steam lines are connected to a common header outside the containment. Each steam line has a uni-directional main steam isolation valve in series with a main steam check valve downstream of the containment penetration. Also, each SG has a flow restrictor integral to the SG outlet nozzle with a flow area of 0.13 m^2 (1.4 ft^2). If all the valves were working, only the affected SG will blow down steam to the containment. The analysis assumes failure of the steam line check valves. After a DEGB in a main steam line, the affected SG would blow down through one end of the break and the other SGs would blow down through the other end of the break from the common header. The flow restrictors will limit the amount of steam from each SG and the blow down will terminate when the main steam isolation valves close. These assumptions are consistent with the acceptance criteria.

Tables 6.2.1-26 and 6.2.1-27 provide mass and energy release rates for two cases (Cases 1 and 5). There are some irregularities in the flow, in particular an abrupt flow decrease at around 4.2 s into the transient, and **RAI 112-785, Question 06.02.01.04-1,** was issued asking the applicant to explain adequately the reasons for the flow decrease. In its response to RAI 112-785, Question 06.02.01.04-1, the applicant stated that according to the flow reduction/decrease is a consequence of the flow restrictors in the intact SG lines after the steam in the lines is completely released. The applicant response to **RAI 112-785, Question 06.02.01.04-1**, dated December 22, 2008, was acceptable. **RAI 112-785, Question 06.02.01.04-1, is resolved and closed**.

The Containment Design Temperature is 149ºC (300ºF) per Section 6.2.1.1.1, "Design Basis," also given in Table 6.2.1-2, "Basic Specifications of PCCV." However, the atmospheric calculated temperatures for some transients from Table 6.2.1-8 and Figures 6.2.1-39 – 6.2.1-65 are well over the 149ºC (300ºF) limit, with calculated temperatures between 163ºC and 179ºC (325ºF and 355ºF). The applicant was questioned in **RAI 113-786, Question 06.02.01.04-2,** on the implication of these high temperatures. In its response to RAI 113-786, Question 06.02.01.04-2, dated January 15, 2009, the applicant stated that by calculating the temperatures in the containment walls for the nine cases considered. The maximum calculated temperature inside the containment liner was 134ºC (273ºF), which is below the maximum temperature limit and therefore, acceptable. Although the atmospheric temperatures are above the limit, the containment wall temperatures are not. The response to **RAI 113-786, Question 06.02.01.04-2,** is acceptable. **RAI 113-786, Question 06.02.01.04-2, is resolved and closed.**

Table 6.2.1-25 covers nine MSLB cases. The applicant states that these cases are bounding. The applicant was requested in **RAI 114-787, Question 06.02.01.04-3,** to justify this. In its response to **RAI 114-787, Question 06.02.01.04-3,** dated December 22, 2008, the applicant justified that the nine cases analyzed bound any single failure chosen. The main feedwater isolation valve is assumed to fail, and the analysis maximizes the calculated containment pressure and temperature. The **RAI 114-787, Question 06.02.01.04-3,** response, dated December 22, 2008, is acceptable and **RAI 114-787, Question 06.02.01.04-3, is resolved and closed**.

Concurrent reviews of other DCD Sections (3.2.1, 3.2.2 and 3.6.2 for the valves) are factored into this review. Review of Section 10.4.9, for operator actions is not required because no operator actions are considered.

6.2.1.4.5 Combined License Information Items

There are no COL information items identified for this section in DCD Tier 2, Table 1.8-2.

6.2.1.4.6 Conclusions

As a result of the open item (**RAI 923-6420, Question 06.02.01-21)**, the staff is unable to finalize its conclusions on Section 6.2.1.4 related to mass and energy release analysis for postulated secondary system pipe ruptures, in accordance with NRC regulations.

6.2.1.4.7 References

1. Mitsubishi Heavy Industries, Ltd, "Design Control Document for the US-APWR," Chapter 6, Engineered Safety Features, Section 6.2.1.4, Mass and Energy Release Analysis for Postulated Secondary-System Pipe Ruptures Inside Containment, MUAP-DC006, Revision 3, March 2011.

2. US Nuclear Regulatory Commission, "Standard Review Plan, Chapter 6.2.1.4, Mass and Energy Release for Postulated Secondary System Pipe Ruptures," Revision 3, NUREG-0800, March 2007.

3. Mitsubishi Heavy Industries, Ltd, "Non-LOCA Methodology," MUAP-07010-P Revision 1 (Proprietary), and MUAP-07010-NP Revision 1 (Non-Proprietary), October 2010.

4. Numerical Applications Inc., "GOTHIC Containment Analysis Package User Manual" Version 7.2a(QA), Reports NAI 8907-02, Revision 17, NAI 8907-06, Revision 16, and NAI 8907-09, Revision 9, Richland, WA, January 2007.

5. Moody, F. J., Maximum Flow Rate of Single Component, Two-phase Mixture, Journal of Heat Transfer, Trans. of the ASME, Number1, Feb., 1965.

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

6.2.1.5.1 Introduction

Following a LOCA in a US-APWR plant, the ECCS supplies water to the reactor vessel to reflood, and thereby cool, the reactor core. The core flooding rate is governed by the capability of ECCS water to displace the steam generated in the reactor vessel during the core reflooding period. For US-APWR plants, core flooding rate depends directly on containment pressure (i.e., the core flooding rate increases with increasing containment pressure).

Therefore, as part of the overall evaluation of ECCS performance, the primary reviewer reviews analyses of the minimum containment pressure possible during the time until the core is reflooded following a LOCA to confirm the validity of the containment pressure in ECCS performance capability studies. The review includes assumptions for the operation of engineered safety feature heat removal systems, the effectiveness of structural heat sinks within the containment to remove energy from the containment atmosphere, and such other heat removal processes as steam in the containment mixing with ECCS water spilling from the break in the RCS.

It should be noted that the minimum containment pressure analysis for ECCS performance evaluation differs from the containment functional performance analysis in that the conservatisms and margins are taken in opposite directions; thus, the minimum containment pressure analysis required by the regulations for ECCS performance evaluation is not conservative as to peak containment pressure in a LOCA and cannot be used to determine the containment design basis.

6.2.1.5.2 Summary of Application

DCD Tier 1: There is no Tier 1 information in the DCD (Reference 1) provided in DCD Tier 1, Section 2.2.1.5.

DCD Tier 2: The Tier 2 information associated with this evaluation is provided in DCD Tier 2, Section 6.2.1.5, and is summarized in part as follows

The application describes that the containment pressure and temperature responses, as well as the in-containment RWSP water temperature response, used for the ECCS performance analysis found in Section 15.6.5, “Loss-of-Coolant Accidents Resulting from Spectrum of Postulated Piping Breaks within the Reactor Coolant Pressure Boundary.” The application identifies the analytical modeling used in calculating the time dependent minimum containment backpressure for the ECCS performance evaluation coping with a postulated LOCA. The application identifies the parameters used in the model and includes:

- Mass and Energy Release Data.
- Initial Containment Internal Conditions.
- Containment Volume.

- Active Heat Sinks.
- Steam-water Mixing.
- Passive Heat Sinks.
- Heat Transfer to Passive Heat Sinks.
- Other Parameters.

A double-ended, cold-leg, guillotine (DECLG) break was employed together with conservative assumptions to determine the lowest possible pressure inside the containment. The calculated peak of the minimum containment pressure is 277 kPa (25.5 psig) and the calculated temperature peak is 112ºC (233ºF). The results for this minimum containment pressure are shown in Figure 6.2.1-80 through Figure 6.2.1-83 as containment pressure, containment temperature, RWSP water temperature, and heat transfer coefficients on the heat sinks, respectively.

ITAAC: The ITAAC associated with DCD Tier 2, Section 6.2.1.5 are provided in DCD Tier 1, Section 2.11.1, "Containment Vessel."

TS: The TS associated with DCD Tier 2, Section 6.2.1.5 are provided in DCD Tier 2, Chapter 16, Section 3.6, "Containment Systems."

Topical Reports: Mitsubishi Heavy Industries, Ltd, Topical Report MUAP-07011-P, "Large-Break LOCA Applicability Report for US-APWR," Code WCOBRA/TRAC (M1.0), Revision 1 and MUAP-07011-NP, Revision 1, issued March 2011.

Technical Reports: There are no technical reports submitted by the applicant for this area of review.

US-APWR Interface Issues identified in the DCD: There are no US-APWR interface issues for this area of review.

Site Interface Requirements Identified in the DCD: There are no site interface requirements for this area of review.

Cross cutting Requirements (TMI, USI/ GSI, Op Ex): GSI 191, "Assessment of Debris Accumulation on PWR Sump Performance," closure activities (ML120820125).

RTNSS: There is no RTNSS for this area of review.

10 CFR 20.1406: There are no 10 CFR 20.1406 requirements for this area of review.

CDI: There is no CDI for this area of review.

6.2.1.5.3 Regulatory Basis

The relevant requirements of the Commission's regulations for this area of review, and the associated acceptance criteria, are given in Section 6.2.1.5, "Minimum Containment Pressure Analysis for Emergency Core Cooling System Performance Capability Studies," Reference 2, of NUREG-0800, the SRP, and are summarized below. Review interfaces with other SRP sections can be found in Section 6.2.1.5 of NUREG-0800. The relevant requirements are:

1. 10 CFR 50.46(a)(1)(i), with respect to an acceptable ECCS evaluation model that realistically describes the behavior of the reactor during LOCAs.

2. 10 CFR 50.46(a)(1)(ii), with respect to an ECCS evaluation model developed in compliance with 10 CFR 50, Appendix K, paragraph I.D.2, which requires that the containment pressure used for evaluating cooling effectiveness during reflood and spray cooling not exceed pressure calculated conservatively for that purpose.

3. 10 CFR 52.47(b)(1), which requires that a DC application contain the proposed ITAAC that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests and analyses are performed and the acceptance criteria met, a plant that incorporates the design certification is built and will operate in accordance with the DC, the provision of the Atomic Energy Act of 1954, and the NRC's regulations.

Acceptance criteria adequate to meet the above requirements are also given in SRP BTP 6-2, "Minimum Containment Pressure Model for ECCS Performance Evaluation," Reference 3, and include:

1. 10 CFR 50.46(a)(1)(i) requires plants to have ECCSs, the cooling performance of which is evaluated for the most severe postulated LOCA. 10 CFR 50.46(a)(1)(i) allows the ECCS evaluation to use a realistic model that describes the behavior of the RCS during a LOCA. Containment minimum pressure directly affects ECCS performance. Calculation and analysis of this parameter, therefore, is a part of the ECCS performance evaluation. RG 1.157, "Best-Estimate Calculations of Emergency Core Cooling System Performance," provides specific methods acceptable to the staff for meeting 10 CFR 50.46(a)(1)(i). This regulation assures that, in a LOCA, the ECCS will perform as predicted, meeting limits on maximum peak cladding temperature, maximum cladding oxidation, and maximum hydrogen generation and maintaining a coolable geometry.

2. 10 CFR 50.46(a)(1)(ii) requires plants to have ECCSs, the cooling performance of which is evaluated for the most severe postulated LOCA. As an alternative to the requirements of 10 CFR 50.46(a)(1)(i), 10 CFR 50.46(a)(1)(ii) requires for the ECCS performance evaluation a model based on 10 CFR Part 50, Appendix K, which provides specific calculation methods for evaluating ECCS performance and significant conservatisms to address the uncertainties of post-LOCA plant behavior. Containment minimum pressure directly affects ECCS performance. Calculation and analysis of this parameter, therefore, is part of the ECCS performance evaluation. This regulation assures that, in a LOCA, the ECCS will perform as predicted, limit maximum peak cladding temperature, maximum cladding oxidation, and maximum hydrogen generation; and maintain a coolable geometry.

3. RG 1.157, "Best-Estimate Calculations of Emergency Core Cooling System Performance," Position C.3.12.1, (Reference 4), which describes acceptable containment models for ECCS performance analysis.

6.2.1.5.4 Technical Evaluation

The areas of review are:

Assumptions employed in the ECCS operation including steam and water mixing, spray operation, and structural heat sinks or other heat removal mechanisms employed in order to obtain a conservatively low containment pressure after a LOCA. Also to be reviewed are the input conditions for the model (initial containment internal conditions, initial outside conditions, containment volume, purge supply and exhaust), active heat sinks, passive heat sinks, and heat transfer coefficients.

This review interfaces with Section 6.3, "Emergency Core Cooling Systems," (acceptability of the mass and energy release data).

The ITAAC and their regulatory criteria for DCs are in 10 CFR 52.47(b)(1). The ITAAC review is presented in Section 14.3.11 of this SER.

6.2.1.5.4.1   Minimum Containment Pressure Analysis

Two computer codes were employed in these calculations: GOTHIC (Reference 5), for the pressure and temperature inside the containment, and WCOBRA/TRAC (M1.0) (Reference 6), for the mass and energy releases from the break into containment.

The analytical model of the containment, described in Subsection 6.2.1.5.1, "Analytical Models," of the DCD, employs a single GOTHIC volume in direct contact with the cold RWSP water (the RWSP ceiling was removed) to enhance heat transfer from the containment atmosphere to the water of the RWSP. Both assumptions are conservative and appropriate for this calculation.

DCD, 6.2.1.5.2, "Mass and Energy Release Data" and DCD Table 6.2.1-28, "Break Mass and Energy Flow for the Double-Ended Guillotine Break Resulting in the Minimum Containment Pressures for use in ECCS Evaluation," employ the WCOBRA/TRAC (M1.0) code for a nominal DECLG break. The evaluation models are described in DCD Subsection 15.6.5.3, "Core and System Performance." Steam and liquid water releases are combined as a single mixture to reduce the available energy in the containment. The spillage flow is injected as small droplets to ensure equilibrium with the atmosphere. All of these assumptions are conservative and appropriate.

Inspection of Table 6.2.1-28 values shows some abrupt changes, with mass flow rates decreasing rapidly at 32 sec, end of blowdown flow, then increasing at 44 sec and decreasing again at 46 sec and 56 sec, and with large changes in enthalpy at 30 sec and 34 sec. In **RAI 115-788, Question 06.02.01.05-1**, the staff requested the applicant to provide a revised table correcting the irregular/abrupt changes of the original table. In its response to RAI 115-788, Question 06.02.01.05-1, dated December 25, 2008, the applicant provided a revised table which was acceptable. Therefore, **RAI 115-788, Question 06.02.01.05-1, is resolved and closed**.

In RAI 116-789, Question 06.02.01.05-2, the applicant was asked to explain how the containment calculated back pressure is used as a boundary condition in WCOBRA/TRAC calculations. In its response to **RAI 116-789, Question 06.02.01.05-2**, dated December 25, 2008, the applicant explained that the calculated containment back pressure is provided as an input in the form of a pressure versus time table to the BREAK component of the

WCOBRA/TRAC code. No margin is considered for the calculated back pressure in the procedure. The staff accepts the response and therefore, **RAI 116-789, Question 06.02.01.05-2**, **is resolved and closed**.

The initial containment conditions (DCD Tier 2 Subsection 6.2.1.5.3, "Initial Containment Internal Conditions") employed in this calculation are conservative and appropriate for this calculation in accordance with BTP 6-2. The values are representative of the low end of values anticipated during normal operation. These initial conditions include a minimum pressure of 99.3 kPa (-0.3 psig), a minimum inside temperature of 21ºC (70ºF), a minimum outside temperature of -40ºC (-40 ºF), a minimum RWSP water temperature of 0ºC (32 ºF), a minimum service water temperature of 0ºC (32 ºF), and a maximum relative humidity inside containment of 100 percent.

In **RAI 117-790, Question 06.02.01.05-3**, the staff requested that the applicant provide justification for the minimum temperature employed for the containment interior of 21ºC (70ºF)**.** In its response to RAI 117-790, Question 06.02.01.05-3, dated January 15, 2009, the applicant justified the minimum temperature used as the Fracture Transition Elastic (FTE) temperature, which is defined as the Nil-Ductility Transition Temperature (NDTT) of -12.2ºC (10ºF) plus 33.3ºC (60ºF). The FTE is thus equal to 21ºC (70ºF), which is the temperature used. The staff then issued supplemental **RAI 623-4942, Question 06.02.01-18**, requesting justification of the NDTT used. In its response to RAI 623-4942, Question 06.02.01-18, dated September 29, 2010, the applicant stated that the Reference Nil-Ductility Temperature (RTNGT) for all pressure boundary ferritic Class 1 components is less than -12.2 ºC (10ºF). The relationship between NDTT and RTNDT is established in ASME Code Section III NB-2330. In most cases both temperatures are the same (i.e., NDTT=RTNDT). Therefore, the chosen value for NDTT of -12.2 ºC (10ºF) is justified. The staff concurs. Therefore, **RAI 623-4942, Question 06.02.01-18, is resolved and closed.**

The containment volume used is a maximized value of 80,986 m^2 (2.86×10^6 ft^3) which is appropriate for this calculation in order to obtain the minimum pressure.

6.2.1.5.4.2   Active Heat Sinks

DCD Section 6.2.1.5 and Table 6.2.1-29, "Basic Specifications of ESF used in Minimum Containment Pressure Analysis," list: two SISs, four CSSs with full water flow, a water temperature of 0ºC (32 ºF), five-second delay time, and a small droplet size of 0.1 mm (0.004 in). All of these input conditions are conservative except for the use of only two active safety injection systems (ASIS) instead of the four possible ASIS. Since the source of the water injected is very cold, only 0ºC (32ºF), four ASIS should reduce the pressure and temperature in the containment more than only two ASIS, and therefore, the low number of ASIS (two) is not conservative. Injection of water is conservatively assumed at the lowest possible temperature of 0ºC (32ºF) in order to minimize the resulting containment pressure.

The applicant was asked to justify the use of only two ASIS instead of four in **RAI 118-791, Question 06.02.01.05-4.** In its response to RAI 118-791, Question 06.02.01.05-4, dated December 25, 2008, the applicant stated the two ASIS were used to be consistent with assumptions used in Chapter 15 analysis. The staff concerns with such an approach were submitted in a supplemental **RAI 623-4942, Question 06.02.01-19**, asking the applicant to perform additional analysis using four ASIS. The applicant performed the additional analyses and revised the DCD accordingly. The applicant's response to **RAI 623-4942, Question 06.02.01-19**, dated September 29, 2010, showed a revised calculated minimum pressure peak of 277 kPa (25.5 psig) and a revised calculated minimum temperature peak of 112ºC (233ºF), as now shown in Figures 6.2.1-80, “Containment Pressure vs. Time for Postulated RCS DEGB Transient Employed in Minimum Containment Pressure Analyses for ECCS Performance Evaluations,” and 6.2.1-81,” Containment Atmospheric Temperature vs. Time for Postulated RCS DEGB Transient Employed in Minimum Containment Pressure Analyses for ECCS Performance Evaluations,” respectively, of the DCD. Both calculated values, pressure and temperature, are lower than the previous values when only two ASIS were employed; therefore, the applicant’s RAI response, is acceptable and **RAI 623-4942, Question 06.02.01-19, is resolved and closed.**

6.2.1.5.4.3 Passive Heat Sinks, Table 6.2.1-30 of the DCD

Table 6.2.1-30, “Passive Heat Sinks used in the Minimum Containment Pressure Analysis for ECCS Capability Studies,” was compared with Table 6.2.1-9, “Passive Heat Sinks used in Maximum Pressure Containment Analyses,” (passive heat sinks for nominal breaks), and while the heat transfer areas are always larger in Table 6.2.1-30 (which will increase heat transfer), the thickness of concrete and other materials appeared to be larger as well. For the internal components/walls this is appropriate as the heat sinks are enlarged. However, for the external containment wall, a thicker wall will decrease/delay heat transfer to the outside instead of increasing it. It appears as if the thickness of the containment wall in contact with the exterior cold temperature should be minimized instead of maximized. In **RAI 119-792, Question 06.02.01.05-5**, the staff requested the applicant to address these apparent inconsistencies. In its response to **RAI 119-792, Question 06.02.01.05-5**, dated December 25, 2008, the applicant performed sensitivity calculations with two different thicknesses: 1.32 and 1.37 m (52 in and 53.9 in), for the containment cylinder wall and no effects were found in the results. Therefore, the thicknesses of the exterior walls are sufficiently large that changes (increases) in their values did not affect the results. This RAI response is satisfactory; therefore **RAI 119-792, Question 06.02.01.05-5, is resolved and closed**.

6.2.1.5.4.4 Heat Transfer to Passive Heat Sinks

The heat transfer coefficients employed during the blow-down period are shown in Figure 6.2.1-83, “Condensing Heat Transfer Coefficient on the Typical Structure as a Function of Time for Postulated RCS DEGB Transient Employed in Minimum Containment Pressure Analyses,” with an initial value of about 45 W/m^2-K (8 Btu/ft^2-h-ºF), a peak value of about 4315 W/m^2-K (760 Btu/ft^2-h-ºF) at about 33 sec, and a final value of 227 W/m^2-K (40 Btu/ft^2-h-ºF) after 220 sec (values estimated from Figure 6.2.1-83). The peak value is correctly calculated using the Tagami correlation multiplied by four (equation in British units):

$$h_{peak} = 4x72.5(Q/(Vt))^{0.62} = 290\ (4.5x10^{8}/2.86x10^{6}/33)^{0.62} = 763\ \text{Btu/ft}^{2}\text{-h-ºF}$$

with the blowdown energy released taken from Figure 6.2.1-85, “Containment Energy Distribution Transient for DEHLG Break (C_D = 1.0),” as 4.5x10^8 Btu, the containment volume taken from Section 6.2.1.5.4, “Containment Volume,” as 2.86x10^6 ft^3 and the blowdown time as 33 s estimated from Figure 6.2.1-83.

The value of the Uchida correlation heat transfer coefficient could not be checked because the densities of the steam and gas in the containment are needed for the calculation and its values are not known. The reported value at 400 s (Figure 6.2.1-83) is 227 W/m^2-K (40 Btu/ft^2-h-ºF) which is 1.2x h_{Uchida} , therefore, the h_{Uchida} heat transfer coefficient is:

h_{Uchida} = 227/1.2 = 189 W/m^2-K

h_{Uchida} = 40 /1.2 = 33.33 Btu/ft^2-h-ºF

a value that appears to be reasonable for an air/steam ratio between 2.3 and 3 (according to Table 3 of BTP 6-2).

The use of the Tagami and Uchida correlations conforms to BTP 6-2, but some deviations have been found in its application in the DCD.

The exponent of the Uchida correlation employed in the DCD, Page 6.2-38, is 0.62. This correlation was obtained by fitting the Uchida experimental data by several investigators and different correlations with different exponents have been reported. Values of 0.7 and 0.8 for this exponent are used more widely than the 0.62 value reported in the DCD. The GOTHIC code for example, uses a value of 0.8 for this exponent (Equation 9.32 of the GOTHIC Technical Manual, Reference 5). The effect of this difference is small and does not need further consideration. The applicant has changed, however, the original exponent of 0.62 to 0.8 in Revision 3 of the DCD.

The exponent of the formula for the heat transfer coefficient after blow-down supposedly employed by the applicant on Page 6.2-38 of the DCD is -0.62 instead of the value of -0.025 recommended in BTP 6-2. However, inspection of Figure 6.2.1-83 indicates that the correct value of -0.025 was used. There is disagreement between the formula reported in the page 6.2-38 of the DCD and Figure 6.2.1-83. In **RAI 120-793, Question 06.02.01.05-6,** the staff requested the applicant to explain this discrepancy. In its response to RAI 120-793, Question 06.02.01.05-6, dated December 25, 2008, the applicant acknowledged the typographical error and the correct exponent of -0.025 was used in the next revision of DCD. This response is acceptable. **RAI 120-793, Question 06.02.01.05-6, is resolved and closed**.

Similarly, the transition from the heat transfer coefficient at time zero to the value at the end of blowdown appears to be linear in Figure 6.2.1-83 instead of following the exponential formula that was reported in Section 6.2.1.5.8, “Heat Transfer to Passive Heat Sinks.” Thus, the formula in Section 6.2.1.5.8 of the DCD and the value from time zero to the peak shown in Figure 6.2.1-83 disagreed. In **RAI 121-794, Question 06.02.01.05-7**, the staff requested the applicant to explain this disagreement. In its response to RAI 121-794, Question 06.02.01.05-7, dated December 25, 2008, the applicant acknowledged the typographical error. The correct numerical value of the exponent “1.0” was employed in the formula of the next revision of the DCD. The response is acceptable. Therefore, **RAI 121-794, Question 06.02.01.05-7, is resolved and closed**.

An inspection of Figures 6.2.1-80 and 6.2.1-81, shows that the peak temperature and pressure in the containment occur around 14 sec, but the peak heat transfer coefficient in Figure 6.2.1-83 occurs at 33 sec. Table 6.2.1-28 also shows the end of blowdown at about 33 sec. In **RAI 122-795, Question 06.02.01.05-8**, the staff requested the applicant to explain this discrepancy in the timing of these peaks (14 sec and 33 sec). In its response to RAI 122-795, Question 06.02.01.05-8, dated December 25, 2008, the applicant explained that the peak containment pressure and temperature occur when the heat removal by the containment passive sinks changes from being less than to being more than the energy released into containment. This change occurs at about 14 s, and after this time, energy is removed more rapidly than is released into containment. Consequently, both the containment peak pressure and peak temperature occur at 14 s. On the other hand, the blowdown ends at 33 s and this time coincides with the time of the heat transfer coefficient peak (Tagami correlation). The staff finds the response satisfactory. Therefore, **RAI 122-795, Question 06.02.01.05-8, is resolved and closed**.

6.2.1.5.4.5 Other Parameters

The containment purge was not considered in the calculations.

6.2.1.5.5 Combined License Information Items

There are no COL information items identified for this section in DCD Tier 2, Table 1.8-2. No additional COL information items need to be included for this area of review.

6.2.1.5.6 Conclusions

All the RAIs have been addressed by the applicant in a satisfactory manner. The applicant meets the requirements of 10 CFR 50.46(a)(1)(i) with respect to an acceptable ECCS evaluation model that realistically describes the behavior of the reactor during LOCAs, and the requirements of 10 CFR 50.46(a)(1)(ii) with respect to the calculation of a conservatively low value of the pressure inside the containment. The applicant also complies with RG 1.157, Regulatory Position C 3.12.1, as the post-blowdown containment pressure is calculated using best-estimate conditions, considering the effects of all pressure reduction equipment available in the containment (Active Heat Sinks, Section 6.2.1.5.4.2) and the containment heat sinks (Section 6.2.1.5.4.3, Passive Heat Sinks, and Section 6.2.1.5.4.4, Heat Transfer to Passive Heat Sinks).

The applicant meets the requirements of BTP 6-2 with respect to initial containment conditions, initial outside containment conditions, maximum containment volume, purge supply and exhaust systems, active and passive heat sinks, and heat transfer correlations employed in the calculation of the minimum containment pressure.

6.2.1.5.7 References

1. Mitsubishi Heavy Industries, Ltd, "Design Control Document for the US-APWR," Chapter 6, Engineered Safety Features, Section 6.2.1.5, Minimum Containment Pressure Analysis for Performance Capability Studies of the Emergency Core Cooling System, MUAP-DC006, Revision 3, March 2011.

2. US Nuclear Regulatory Commission, "Standard Review Plan, Chapter 6.2.1.5, Minimum Containment Pressure Analysis for Emergency Core Cooling System Performance Capability Studies," Revision 3, NUREG-0800, March 2007.

3. US Nuclear Regulatory Commission, "Standard Review Plan, BTP 6-2, Minimum Containment Pressure Model for ECCS Performance Evaluation," Revision 3, NUREG-0800, March 2007.

4. US Nuclear Regulatory Commission, Regulatory Guide 1.157, "Best-Estimate Calculations of Emergency Core Cooling System Performance," May 1989.

5. Numerical Applications Inc, “GOTHIC, Containment Analysis Package Technical Manual,” Version 7.2a(QA), NAI 8907-06, Revision 16, Richland, WA, January 2006.

6. Mitsubishi Heavy Industries, Ltd, ”Large Break LOCA Applicability Report for USAPWR,” Code WCOBRA/TRAC (M1.0), Report MUAP-07011 Revision 1 (Proprietary) and MUAP-07011-NP Revision 1 (Non-Proprietary, March 2011)

6.2.2 Containment Heat Removal Systems

6.2.2.1 Introduction

Containment heat removal systems reduce the containment pressure and temperature following a high-energy line break within the containment (either primary or secondary pipe ruptures) and maintain them at acceptably low levels. For the US-APWR, the CSS is designed to function as the containment heat removal system. The US-APWR containment has both sprays and fan coolers but only the CSS is designed as the safety-related active system to remove heat from the containment. The CSS takes suction from the refueling water storage pit (located inside containment), cools the water, and then re-circulates it to the containment. This process provides for long-term containment heat removal. Containment structures also remove heat as passive sinks by storing energy inside them.

A concern regarding long-term recirculation cooling is that a high-energy line break in containment produces debris that together with debris that exists before a line break, called latent debris, could transport to and accumulate on the recirculation sump screens. The debris accumulation on recirculation sump screens could potentially challenge the plant’s capability to provide long-term cooling water to the CSS and ECCS pumps.

As part of the containment heat removal analysis the applicant evaluates the US-APWR’s capability to provide adequate long-term cooling water to the CSS and ECCS pumps in the presence of accident-generated and latent debris within containment. Additional impacts from accident-generated and latent debris discussed in this evaluation include the effect of debris on the flowpaths downstream of the emergency core cooling/containment spray strainer, such as pumps and fuel assemblies.

6.2.2.2 Summary of Application

The DCD information cited in this evaluation section is based on the following documents:

- US-APWR DCD, Revision 3, issued March 2011.
- US-APWR DCD (Revision 3) GSI-191 Tracking Report, (August 2011 Version), issued August 31, 2011 (hereafter referred to as GTR1).
- US-APWR DCD GSI-191 Tracking Report (August 2012 Version), issued August 30, 2012 (hereafter referred to as GTR2).

The Tracking Reports communicate essential information addressing US-APWR GSI-191 issues that the applicant committed to provide as part of the GSI-191 closure activities.

> The tracking reports contain information that is not reflected in DCD Revision 3, but the applicant plans to incorporate into a future DCD revision.
>
> In this safety evaluation section, when DCD information is listed and is supplemented by a tracking report, the tracking report is identified and the tracking report's associated supplemental information is treated as a confirmatory action item.

DCD Tier 1: The Tier 1 information associated with this evaluation is found in Tier 1, Section 2.11.3, "Containment Spray System" and Tier 1, Section 2.4.4, "Emergency Core Cooling System" (related to strainer and refueling water storage pit). Tier 1, Section 2.11.3, states that the CSS is a safety related system, describes the safety related functions, and lists the design features. Tier 1, Section 2.4.4 states that the ECCS is a safety related system, describes the safety related functions, and lists the design features.

DCD Tier 2: The Tier 2 information associated with this evaluation is found in Tier 2, Section 6.2.2, "Containment Heat Removal Systems" and Tier 2, Section 6.3, "Emergency Core Cooling Systems" (related to strainer and the refueling water storage pit). Technical information for the containment heat removal systems is summarized here in part, as follows:

> The containment heat removal systems of the US-APWR address the safety systems in the containment that remove heat, in particular after an accident. The only such system in the US-APWR is the CSS, which is a dual system, removing both heat and fission products (fission product removal is covered in DCD Tier 2, Section 6.5.2, "Containment Spray Systems.")
>
> The CSS and the RHRS share major components, in particular the pumps and the heat exchangers. DCD Tier 2, Section 6.2.2.2, "System Design," indicates that there are four 50 percent capacity trains with four pumps, and four heat exchangers (HXs). Each train has one pump and one HX. Only two trains are needed to provide 100 percent capacity which totals about 0.3786 cubic meters per second (m^3/s) or 0.1893 m^3/s per train (6,000 gallons per minute (gpm), or 3,000 gpm per train).
>
> The containment has also four fan coolers (presented in DCD Tier 2, Chapter 9, "Auxiliary Systems") but these fan coolers are auxiliary systems and "non-safety equipment." Although the fan coolers function to remove heat for normal operations, the fan coolers are not designed to remove heat from the containment under accident conditions. Therefore, the fan coolers are not considered in this section.
>
> DCD Tier 2, Table 1.6-1, "Material Referenced," as supplemented by GTR1, incorporates-by-reference Technical Report MUAP-08001-P, "US-APWR Sump Strainer Performance;" Technical Report MUAP-08011-P, "US-APWR Sump Debris Chemical Effects Test Result;" and Technical Report MUAP-08013-P, "US-APWR Sump Strainer Downstream Effects." The staff opened confirmatory item **CI-SRP06.02.02-1** to verify that the final design document incorporates the associated supplemental DCD information provided by GTR1.
>
> Technical Report MUAP-08001-P, Revision 7, issued August 2012, assesses the US-APWR design with respect to RG 1.82, "Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident," Revision 3, issued November 2003 and Nuclear Energy Institute (NEI) Guidance Report NEI 04-07 "Pressurized Water Reactor Sump Performance Evaluation Methodology," Revision 0, Volume 1, issued December

2004, and the associated NRC safety evaluation (SE), “Safety Evaluation by the Office of Nuclear Reactor Regulation Related to NRC Generic Letter 2004-02,” issued December 2004.

DCD Tier 2, Section 6.2.2.3, “Design Evaluation,” as supplemented by GTR2, indicates that the principal insulation used in the containment is reflective metal insulation (RMI). RMI is used for the reactor vessel, SGs, pressurizer, primary and secondary main and branch lines, and other equipment and piping that require insulation in areas that are potentially subject to high energy line breaks. Additionally, DCD Tier 2, Section 6.2.2.3, states that use of fiber or particulate insulation is eliminated from the zone of influence (ZOI) and that the only fiber source term in containment is from latent debris. The staff opened confirmatory item **CI-SRP06.02.02-2** to verify that the final design document incorporates the associated supplemental DCD information provided by GTR2.

DCD Tier 2, Table 6.2.2-4, “Design Basis Debris,” as supplemented by GTR2, summarizes the design basis debris in containment and indicates that the latent fiber source term is 30 pounds mass (lbm) (13.6 kilograms (kg)). The staff opened confirmatory item **CI-SRP06.02.02-2** to verify that the final design document incorporates the associated supplemental DCD information provided by GTR2.

DCD Tier 2, Section 6.2.2.2.6, “ECC/CS Strainers,” as supplemented by GTR2, indicates that the US-APWR has four independent sets of strainers that are located inside the in-containment RWSP. These strainers are installed at the bottom of the RWSP and are designed to be fully submerged during all postulated events requiring actuation of the Emergency Core Cooling/Containment Spray (ECC/CS) system. The staff opened confirmatory item **CI-SRP06.02.02-2** to verify that the final design document incorporates the associated supplemental DCD information provided by GTR2.

DCD Tier 2, Section 6.2.8, “Combined License Information,” as supplemented by GTR2, establishes COL Information Item 6.2(5) for instituting a containment cleanliness program and COL Information Item 6.2(6) for preparation of administrative procedures to ensure insulation, within the ZOI, and aluminum in containment are consistent with the design basis. The staff opened confirmatory item **CI-SRP06.02.02-2** to verify that the final design document incorporates the associated supplemental DCD information provided by GTR2.

ITAAC: The ITAAC associated with this evaluation are found in Tier 1, Table 2.11.3-5, “Containment Spray System Inspections, Tests, Analyses, and Acceptance Criteria;” and Tier 1, Table 2.4.4-5, “Emergency Core Cooling System Inspections, Tests, Analyses, and Acceptance Criteria.” The ITAAC related to containment systems (e.g., strainers) are evaluated in Section 14.3 of this SE.

TS: The TS associated with Tier 2, Section 6.2.2 and associated Sections of 6.3, “Emergency Core Cooling Systems,” are given in Tier 2, Chapter 16, “Technical Specifications,” Section 3.6.4, “Containment Pressure,” Section 3.6.5, “Containment Air Temperature,” Section 3.6.6, “Containment Spray System,” and Section 3.5, “Emergency Core Cooling System.” Section 6.3 of this evaluation reviews the TS for ECCS (i.e., strainer and RWSP).

Initial plant testing of the CSS and RWSP is discussed in DCD Tier 2, Section 14.2 “Initial Plant Testing Program.” Details are presented in DCD Tier 2, Section 14.2.12.1.58, “Containment

Spray System Preoperational Test," and Section 14.2.12.1.59, "Refueling Water Storage System Preoperational Test."

Topical Reports: There are no topical reports for this area of review.

Technical Reports:

- MHI Technical Report MUAP-08001-P, "US-APWR Sump Strainer Performance," Revision 2, issued May 2009, and Revision 7, issued August 2012.

- MHI Technical Report MUAP-08006-P, "US-APWR Sump Debris Chemical Effects Test Plan," Revision 0.

- MHI Technical Report MUAP-08011-P, "US-APWR Sump Debris Chemical Effects Test Result."

- MHI Technical Report MUAP-08013-P, "US-APWR Sump Strainer Downstream Effects."

US-APWR Interface Issues identified in the DCD: There are no US-APWR interface issues for this area of review.

Site Interface Requirements Identified in the DCD: There are no site interface requirements for this area of review.

Cross cutting Requirements (TMI, USI/ GSI, Op Ex): GSI 191, "Assessment of Debris Accumulation on PWR Sump Performance," closure activities (ML120820125).

RTNSS: There is no RTNSS for this area of review.

10 CFR 20.1406: There are no 10 CFR 20.1406 requirements for this area of review.

CDI: There is no CDI for this area of review.

6.2.2.3 Regulatory Basis

The relevant requirements of the Commission's regulations for this area of review, and the associated acceptance criteria, are given in Section 6.2.2, "Containment Heat Removal Systems," of NUREG-0800, the SRP, and are summarized below. Review interfaces with other SRP sections can be found in Section 6.2.2 of NUREG-0800.

1. GDC 35, "Emergency core cooling," as it relates to providing abundant emergency core cooling to transfer heat from the reactor core following a LOCA.

2. GDC 38, "Containment heat removal," as it relates to the following:

    a. the ability of the containment heat removal system to rapidly reduce the containment pressure and temperature following a LOCA and to maintain these indicators at acceptably low levels,

b. the ability of the containment heat removal system to perform in a manner consistent with the function of other systems,

c. the safety-grade design of the containment heat removal system providing suitable redundancy in components and features and suitable interconnections, leak detection, isolation, and containment capability to ensure that, for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available), the system safety function can be accomplished in the event of a single failure.

3. GDC 39, “Inspection of containment heat removal system,” as it relates to the design of the containment heat removal system to permit periodic inspection of components.

4. GDC 40, “Testing of containment heat removal system,” as it relates to the design of the containment heat removal system to allow periodic testing to ensure system integrity and the operability of the system and active components.

5. 10 CFR 50.46(b)(5), “Long-term cooling,” as it relates to requirements for long-term cooling, including adequate net positive suction head (NPSH) margin in the presence of LOCA-generated and latent debris.

Acceptance criteria adequate to meet the above regulatory requirements include:

1. The containment heat removal systems shall meet the redundancy and power source requirements for an engineered safety feature (i.e., the results of failure modes and effects analyses of each system should ensure that the system is capable of withstanding a single failure without loss of function).

2. With regard to GDC 38 as it relates to the capability of the containment system to accomplish its safety function, the spray system should be designed to accomplish this without pump damage caused by cavitation. A supporting analysis should be presented in sufficient detail to permit the staff to determine the adequacy of the analysis. This analysis should also demonstrate that the available NPSH is greater than or equal to the required NPSH. RG 1.82, Revision 3 describes methods acceptable to the staff for evaluating the NPSH margin. If containment accident pressure is credited in determining available NPSH, an evaluation of the contribution to plant risk from inadequate containment pressure should be made. One acceptable way of making this evaluation is to address the five key principles of risk-informed decision making stated in Section 2 of RG 1.174, “An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis.

3. In evaluating the performance capability of the CSS to satisfy GDC 38, the analyses of its heat removal capability should be based on the following considerations:

   a. The locations of the spray headers relative to the internal structures.

b. The arrangement of the spray nozzles on the spray headers and the expected spray pattern. The spray systems should be designed to ensure that the spray header and nozzle arrangements produce spray patterns which maximize the containment volume covered and minimize overlapping of the sprays.

c. The spray drop size spectrum and mean drop emitted from each type of nozzle as a function of differential pressure across the nozzle.

d. The effect of drop residence time and drop size on the heat removal effectiveness of the spray droplets.

4. In evaluating the heat removal capability of the containment heat removal system to satisfy GDC 38, the potential for surface fouling of the secondary sides of fan cooler, recirculation, and residual heat removal heat exchangers by the cooling water over the life of the plant and the effect of surface fouling on the heat removal capacity of the heat exchangers should be analyzed and the results discussed in the DCD. The analysis will be acceptable if it is shown that provisions such as closed cooling water systems are provided to prevent surface fouling or surface fouling has been accounted for in establishing the heat removal capability of the heat exchangers.

5. To satisfy the requirements of GDC 38 and 10 CFR 50.46(b)(5) regarding the long-term heat removal, the RWSP should be designed to provide a reliable, long-term water source for ECCS pumps. The containment design should allow for the drainage of spray and emergency core cooling water and for recirculation of this water. The design of the protective strainer assemblies is a critical element in ensuring long-term recirculation cooling capability. Therefore, adequate design consideration of RWSP hydraulic performance, evaluation of potential debris generation and associated effects including debris screen blockage, RHR performance under postulated post-LOCA conditions, and impacts of debris penetrating strainers on long-term coolability of the core are necessary. RG 1.82, Revision 3, as modified and supplemented by the NEI Guidance Report (NEI 04-07) and the associated NRC safety evaluation provides guidance for PWR debris evaluations.

6. The review should also be informed by WCAP-16406-NP,"Evaluation of Downstream Sump Debris Effects in Support of GSI-191," Revision 1, dated August 2007, as supplemented by "Final Safety Evaluation for Pressurized Water Reactor Owners Group Topical Report WCAP-16406-P," "Evaluation of Downstream Sump Debris Effects in Support of GSI-191, Revision 1," Revision 0, dated December 20, 2007; the NRC letter dated March 28, 2008, "Revised Guidance for Review of Final Licensee Responses to Generic Letter 2004-02," with enclosures addressing the areas of chemical effects, coatings, and head loss testing; the final safety evaluation by the Office of Nuclear Reactor Regulation on TR WCAP-16530-NP, "Evaluation of Post-Accident Chemical Effects in Containment Sump Fluids to Support GSI-191," dated December 21, 2007; and the NRC letter dated April 6, 2010, "Revised Guidance Regarding Coatings Zone of Influence for Review of Final Licensee Responses to Generic Letter 2004-02."

7. In meeting the requirements of GDC 39 and GDC 40 regarding inspection and testing, the design of containment heat removal systems shall provide for periodic inspection and operability testing of the systems and system components such as pumps, valves, duct pressure- relieving devices, and spray nozzles.

8. To satisfy the system design requirements of GDC 38, instrumentation should be provided to monitor containment heat removal system and system component performance under normal and accident conditions. The instrumentation should be capable of determining whether a system is performing its intended function, or a system train or component is malfunctioning and should be isolated.

9. RG 1.100, Revision 3, “Seismic Qualification of Electrical and Active Mechanical Equipment and Functional Qualification of Active Mechanical Equipment for Nuclear Power Plants,” is applicable to valve design.

10. QME-1-2007, as it relates to valves being designed to operate under normal operating and post-LOCA conditions.

6.2.2.4 Technical Evaluation

The areas of review are:

Single component malfunctions; available NPSH to the ECCS and containment heat removal pumps; heat removal capability of the spray system; heat removal capability of the RHR heat exchangers; potential for surface fouling and flow blockage of RHR heat exchangers; periodic in-service inspection and testing; design of sumps, screens, strainers; and effects of accident generated debris.

This review interfaces with SRP Sections 9.2.2, “Reactor Auxiliary Cooling Water Systems,” and 9.2.3, “Demineralized Water Makeup System,” (secondary cooling systems); Section 7.3, “Engineered Safety Features Systems,” (sensors and actuation instrumentation); Section 3.11, “Environmental Qualification of Mechanical and Electrical Equipment,” (qualification test program); Section 6.1.2, “Protective Coating Systems (Paints) - Organic Materials,” (quantity of unqualified paint); Section 6.5.2, “Containment Spray as a Fission Product Cleanup System,” (fission product controls); Section 3.2.1, “Seismic Classification,” and 3.2.2, “System Quality Group Classification,” (seismic design, quality classification); and Section 16, “Technical Specifications.”

The ITAAC and its regulatory criteria for DCs are covered in Section 14.3.11 of this SE.

The staff reviewed the US-APWR DCD Tier 2 Section 6.2.2 in accordance with the guidance contained in SRP 6.2.2 of NUREG–0800.

This technical evaluation includes the heat removal by the Containment Heat Removal System (CHRS), the RWSP which is the source of water for the cooling systems, and the RWSP strainers, including the effect of accident-generated debris and an assessment of potential loss of long term cooling capabilities.

6.2.2.4.1 Containment Spray System (CSS)

The CSS is designed to remove sufficient heat from the containment atmosphere to limit peak pressures after accidents to values below design limits, and to reduce the peak pressures to less than half the peak values within 24 hours of the initiation of a postulated accident. These analyses are discussed in DCD Tier 2 Chapter 6, "Engineered Safety Features," and Chapter 15, "Transient and Accident Analyses." In particular, Tables 6.2.1-6, "Summary of LOCA Transients Evaluated," and Table 6.2.1-7, "Summary of Sensitivity of ECCS Conditions on the Containment Pressure and Temperature," show pressure and temperature calculations for different LOCAs. Table 6.2.1-8, "Description and Summary Results for Evaluations of Various Pipe Sizes and Break Locations for Postulated Secondary Steam System Piping Failures," (two sheets) shows pressure and temperature calculations for different secondary steam system piping failures. Results are also presented in Figures 6.2.1-18, "Break Mass and Energy Flow for the Blowdown Phase of the DEPSG Break," through 6.2.1-38, "RWSP Water Temperature vs. Time for DEPSG Break with Maximum Accumulator Flowrate," for the LOCA cases, and Figures 6.2.1-39, "Containment Pressure vs. Time for MSLB Case 1," through 6.2.1-65, "RWSP Water Temperature vs. Time for MSLB Case 9," for the secondary steam system piping failures. All the calculated pressures and temperatures are below design values. The calculations took into consideration heat removal by the CSS and heat removed/stored in the structures in the containment. The staff finds that the CSS's capability for removing heat from the containment and maintaining its pressure and temperature at acceptably low levels following a DBA meets the GDC 38 requirements.

The ability of the CSS to remove sufficient heat is accomplished with only two out of the four available trains in operation. DCD Tier 2, Table 6.2.2-3, "Failure Modes and Effects Analysis for CSS," presents the results of failure modes and effects analyses of the various components of the CSS, which show that the CSS is capable of withstanding a single failure with an additional train out of service without loss of safety function. There will be at least two of the four available trains in operation, providing 100 percent capacity. Sufficient instrumentation ensures that failure of an instrument does not impair the readiness of the system. The active components of the CSS are powered from separate busses energized from offsite power supplies. Also, redundant on-site Class 1E emergency electric power sources are available. Each emergency power source can drive all pumps, valves and instruments associated with one train of the CSS. Failure of one onsite or one offsite power system will cause loss of only one train of the four available trains. The staff finds that the CSS meets the redundancy and power source requirements for an engineered safety feature per GDC 38.

The CSS design complies with the following additional GDCs: 2, 4, 5, 17, 39, and 40, as discussed in DCD Tier 2 Section 3.1, "Conformance with NRC General Design Criteria," summarized here as follows:

GDC 2, "Design basis for protection against natural phenomena," is covered in Section 3.1.1.2.

GDC 4, "Environmental and dynamic effects design bases," is covered in Section 3.1.1.4.

GDC 5, "Sharing of structures, systems and components," is covered in Section 3.1.1.5.

GDC 17, "Electrical power systems," is covered in Section 3.1.2.8.

GDC 39, "Inspection of containment heat removal system," is covered in Section 3.1.4.10.

Finally, GDC 40, "Testing of containment heat removal system," is covered in Section 3.1.4.11.

The US-APWR containment has both sprays and fan coolers, but only the CSS is credited as the active system to remove heat from the containment. So, evaluation of the performance capability of the fan cooler is not applicable to the containment heat removal design basis function.

DCD Tier 2, Section 6.2.2.2.3, “Containment Spray Piping,” discusses a delay time of 100 seconds (s), while in DCD Tier 2, Table 6.2.1-5, “Engineered Safety Feature Systems Information,” delay times are shown as 5 sec and 243 sec. The staff issued **RAI 84-796, Question 06.02.02-5,** requesting clarification. In its response to **RAI 84-796, Question 06.02.02-5**, dated November 7, 2008, the applicant explained that there are multiple components factored into the total delay time for CSS initiation. The 243 sec is the maximum spray time delay with all of the components added. The 5 sec and the 100 sec are two different contributors to the total time. The short time (5 sec) is employed for the conservative minimum containment pressure and temperature calculations. Table 6.2.2-1 (see below) is taken from the applicant’s response to **RAI 84-796, Question 06.02.02-5,** and shows the different components of the CSS initiation time delays.

Table 6.2.2-1
Components for Spray Flow Initiation Time Delay

ITEM	Time (s)
Response Time of Sensor and Digital Controller*	3
EPS Starting Time	100
Response Time of Digital Controller and Electrical Circuit**	5
Sequence Time Delay for CS Pump	30
Pump Motor Starting Time to Full Speed	5
Time for the Water to Reach CS Nozzles through CS Line	100
Total Delay Time after SI Signal Release	243

Notes: Maximum values are assumed

* Delay Time that the signal output for the Emergency Power Source (EPS) starting is provided after the containment pressure exceeds the analytical limit

**Delay Time that the breaker for the CS pump motor is closed after the voltage and frequency of the EPS reach the set-points

Long delay times for CSS initiation are employed in containment pressure and temperature calculations. The results of these calculations are conservative (the reduction of high pressure and temperatures will take longer than calculations with more realistic and shorter delay times). Despite these conservatisms in the long delay times, the calculated results are still within acceptable limits that are based on the requirements of GDCs 16, 38, and 50, and described in SRP Section 6.2.1.1.A.

In its response to **RAI 84-796, Question 06.02.02-5**, dated November 7, 2008, the applicant also revised Table 6.2.1-5 and removed the 5 sec time in the column labeled “Full Capacity” (i.e., all four CSS trains in operation, no single failure). The 5 sec time is not used for the containment design evaluation and was removed for practical purposes. The staff confirmed that DCD Revision 3, incorporated the proposed change to Table 6.2.1-5 (i.e., the 5 sec time was removed). Given the discussion above, the staff finds that the applicant’s response to **RAI 84-796, Question 06.02.02-5, acceptably clarified the CSS delay times, and therefore, RAI 84-796, Question 06.02.02-5, is resolved and closed**.

In DCD Tier 2, Section 6.2.2.3, "Design Evaluation," a Sauter mean diameter of 1000 microns (0.039 in) is used for the sprays. The staff issued **RAI 84-797 Question 06.02.02-6**, requesting justification for this value. In its response to **RAI 84-797, Question 06.02.02-6**, dated November 7, 2008, the applicant provided the droplet size distribution for the SPRACO Model 1713A spray nozzle, using a pressure differential across the nozzle of 275.9 kilopascal (kPa) (40 psi). The droplet size distribution confirmed that the selected value, 1000 microns (0.039 in), is conservative since the actual Sauter mean diameter value is 880 microns (0.035 in). The median diameter is 230 microns (0.009 in). Larger droplets have less surface area (for a given total volume of water) and shorter residence times in the containment atmosphere than smaller droplets. As a result, larger droplets lead to less heat and mass transfer with the containment atmosphere than smaller droplets, which is conservative since higher containment pressure and temperature will be calculated with the larger droplets. Since the selection of 1000 (0.039 in) microns droplet is consistent with SRP 6.2.2 recommendations, the staff finds that the applicant's response to **RAI 84-797, Question 06.02.02-6, is acceptable, and therefore, RAI 84-797, Question 06.02.02-6, is resolved and closed**.

The GOTHIC code models explicitly the droplet heat transfer characteristics and fall velocities, with the droplet evaporation and condensation rates depending on the droplet Reynolds number. In **RAI 331-935, Question 06.02.01-12**, the applicant was asked to provide a sensitivity study for the effect of the spray droplet diameter. In its response to **RAI 331-935, Question 06.02.01-12**, dated May 26, 2009, the applicant provided a sensitivity study for the spray droplet diameter, using three different diameters: 1000 microns (0.039 in), which is the nominal diameter; 2000 microns (0.08 in), twice as large; and 500 microns (0.02 in), half the nominal diameter. The results of this sensitivity study yielded very small effects. The containment peak pressure variation between the largest and the smallest droplet diameter was only 4 kPa (0.6 psi). The containment peak temperature variation was only 1°C (2°F).

The staff has performed independent confirmatory calculations with the code MELCOR to check the effectiveness of the CSS using different droplet diameters and water temperatures for LOCAs and MSLBs. Three different spray droplet diameters were investigated: 200, 500 and 1000 microns, and two different spray water temperatures: 300 K (540 R) and 322 K (580 R). Cases without spray actuation were also run for both the LOCA and the MSLB. The results with the MELCOR code agreed with the results reported by the applicant, who used the GOTHIC code. Given the discussion above, the staff finds the applicant's response to **RAI 331-935, Question 06.02.01-12 regarding spray droplet diameter is acceptable, and therefore, RAI 331-935, Question 06.02.01-12, is resolved and closed**.

There are 348 spray nozzles distributed into four separate rings (rings A, B, C and D) at four different elevations – Ring A is at the highest elevation, Ring D is at the lowest elevation. Ring A has 32 nozzles, Ring B has 80 nozzles, Ring C has 116 nozzles, and Ring D has 120 nozzles. Some of the nozzles spray vertically (down), some of the nozzles spray horizontally and some of the nozzles spray at a 45° angle. Some of the horizontal nozzles are used to spray and to vent each ring; there is one such nozzle in each ring. DCD Tier 2, Figure 6.2.2-5, "Containment Spray System Spray Ring Elevations," and Figure 6.2.2-6, "Containment Spray System Spray Nozzle Locations and Predicted Coverage on Operating Floor," show the elevations of the four different rings, the number of nozzles in each ring, and the predicted coverage on the operating floor – there is complete coverage of the operating floor area with significant overlap. The effectiveness of the sprays as a fission product removal system is addressed in Section 6.5.2 of this SER. Based on the discussion above, the staff finds that the spray elevations and spray coverage of the containment are adequate.

The performance data for the CS/RHR pump and HX is shown in DCD Tier 2, Section 5.4.7, “Residual Heat Removal System.” Heat exchanger fouling and its consequences in heat removal capabilities are considered in the design as stated by the applicant. In **RAI 84-798, Question 06.02.02-7** and supplemental **RAI 623-4942, Question 06.02.01-20**, the staff requested an evaluation of the HX fouling in the CSS/RHR system and its effect on heat removal capabilities. In its response to **RAI 84-798, Question 06.02.02-7**, dated November 7, 2008, and **RAI 623-4942 Question 06.02.01-20**, dated September 29, 2010, the applicant stated that the fouling will be accounted for by the vendor using a fouling factor according to standards of the Tubular Exchanger Manufacturers Association (TEMA). In addition, in response to **RAI 840-6096, Question 06.02.02-78**, dated November 22, 2011, the applicant confirmed that the HX vendor will be required to conform to the specified design conditions (i.e., accounting for various fluid constituents for normal, minimum and maximum expected operating conditions (discussed in Section 6.2.2.4.11 of this evaluation)). Additionally, the staff issued **RAI 947-6540, Question 06.02.01-24,** requesting that the applicant include the fouling factors in the DCD. In its response to **RAI 947-6540, Question 06.02.01-24**, dated August 1, 2012, the applicant modified DCD Tier 2, Table 5.4.7-2, “Equipment Design Parameters,” to include the CS/RHR heat exchanger tube-side and shell-side fouling factors of 0.000088 square-meter-Kelvin per watt (m^2-K/W) ((0.0005 hour-square-foot-degree Fahrenheit per British thermal unit) (hr-ft^2-°F/Btu)) used in the CSS/RHR HX safety-related design calculations. Given the discussion above, the staff finds the response to **RAI 84-798, Question 06.02.02-7**, as supplemented by the responses to **RAI 840-6096, Question 06.02.02-78; RAI 623-4942, Question 06.02.01-20; and RAI 947-6540, Question 06.02.01-24 are acceptable**. The staff opened a **Confirmatory Item** to verify that the fouling factor information is incorporated into DCD Table 5.4.7-2 as provided in response to **RAI 947-6540, Question 06.02.01-24**.

Later, as a result of the March 18, 2013, public meeting with the applicant, the staff issued **RAI 1036-7079, Question 06.02.02-94**. The RAI asked the applicant to justify the tube-side and shell-side fouling factors included in the US-APWR DCD Table 5.4.7-2, “Equipment Design Parameters,” as being conservative with respect to the containment peak pressure and temperature, for the prevailing water quality, velocity, and temperature conditions in the CS/RHR HX over the life of the plant. The RAI also asked the applicant to document in the DCD the justification for the fouling factors or an evaluation of the impact of the surface fouling on the heat removal capacity of the safety-related CS/RHR HX that would be relied on for containment spray and RHR under the DBA conditions. **RAI 1036-7079, Question 06.02.02-94 is being tracked as an Open Item.**

Additional information regarding CSS testing and inspection is provided in other sections (Chapter 14, “Verification Programs,” Subsection 14.2.12.1, “Preoperational Tests,” and Chapter 16).

Instrumentation is described in DCD Tier 2 Chapter 5, “Reactor Coolant and Connecting Systems,” Chapter 7, “Instrumentation and Controls,” and Chapter 18, “Human Factors Engineering.” DCD Tier 2 Chapter 5, Section 5.4.7, “Residual Heat Removal System,” discusses the instrumentation to monitor and control the RHR function of this system. DCD Tier 2 Chapter 7, Section 7.3.1.5.3, “Containment Spray Actuation,” describes the instrumentation to actuate the CSS. Finally, Chapter 18, “Human Factors Engineering,” identifies the control panel locations for the CSS and describes the instrumentation and alarm features of the human interface with the CSS information and control. From DCD Tier 2, Subsection 7.3.1.5.3, Figure 7.2-2, “Functional Logic Diagram for Reactor Protection and Control System,” (Sheet 12) and Table 7.3-3, “Engineered Safety Features Actuation Signals,” there are two modes of CSS actuation: automatic or manual. The automatic actuation is set by High-3 containment pressure signal – This signal is the result of two-out-of-four signals from High-3 containment pressure. For manual actuation, two-out-of-two manual controls must be operated concurrently for each train. The High-3 pressure setpoint is 234.2 kPa (34 psig) with a response time of 3 sec.

6.2.2.4.2 Long-Term Recirculation Water Sources

DCD Tier 2, Section 6.2.2 and 6.3 describe the US-APWR systems and components that provide containment heat removal, ECC, and the long-term water source for recirculation cooling. The long-term recirculation flow is achieved through operation of the CSS and the High-Head Injection System (HHSI). The water source for the CSS and the HHSI is the RWSP. The RWSP is located inside containment, at the bottom floor, and contains a large volume of borated water that is monitored for boron concentration, level, and temperature. The inside walls and floor of the RWSP (in contact with boric acid solution) are lined with steel plate, clad with stainless steel. The RWSP ceiling is also clad with stainless steel plate as stated in DCD Tier 2, Section 6.2 "Containment Systems." Water that is sprayed and spilled into containment during an emergency is routed to the RWSP through wall openings, floor openings, and overflow drain pipes. This configuration does not need, as in other containment designs, to switch-over from an external water tank to the containment sump to support long-term recirculation cooling.

The ECC/CS strainers limit debris entering the safety systems that are required to maintain the post-LOCA, long-term cooling performance. Four independent ECC/CS strainers, one for each ECC/CS safety train, are provided inside the RWSP. Each strainer has a minimum of 2,754 square feet (ft^2) (255.9 square meters (m^2)) of strainer surface area. The applicant performed US-APWR design specific testing to demonstrate that the measured debris head loss of the US-APWR strainer is bounded by the design basis head loss assumed in the HHSI and CS pump NPSH evaluations.

The applicant addresses debris concerns associated with long term core cooling in Revision 7 of Technical Report MUAP-08001-P, which is proprietary. Technical Report MUAP-08001-NP is the non-proprietary version. This evaluation relies on MUAP-08001-NP, the non-proprietary version of the technical report, unless noted otherwise.

RG 1.82, Revision 3, as modified and supplemented for PWRs by NEI 04-07 and the associated NRC SE, provide guidance for PWR debris evaluations. The guidance is organized into several subject areas listed below. Each of these areas will be assessed as part of this evaluation in the following subsections:

6.2.2.4.3	Break Selection
6.2.2.4.4	Zone of Influence (ZOI)/Debris Generation
6.2.2.4.5	Debris Characteristics
6.2.2.4.6	Latent Debris
6.2.2.4.7	Coatings Evaluation
6.2.2.4.8	Debris Transport
6.2.2.4.9	Strainer Head Loss Testing
6.2.2.4.10	Net Positive Suction Head (NPSH)
6.2.2.4.11	Upstream Effects (Water Hold-Up)
6.2.2.4.12	Debris Interceptor/Strainer Structural Design
6.2.2.4.13	Downstream Effects on Ex-Vessel Components and Systems

6.2.2.4.14 Downstream Effects on Fuel and Vessel
6.2.2.4.15 Chemical Effects

6.2.2.4.3 Break Selection

The objective of the break selection process is to identify the break size and location that presents the greatest challenge to post-accident sump performance. RG 1.82 position 1.3.2.1 and Sections 3.3 and 4.2.1 of NEI 04-07 and associated NRC SE provide criteria to be considered in the overall break selection process to identify the limiting break. The overall criterion used to define the most challenging break for recirculation operations is the resultant estimated head loss across the sump screen. Therefore, all phases of the accident scenario must be considered for each postulated break location: debris generation, debris transport, debris accumulation, and sump screen head loss. Two attributes of break selection that are emphasized in the approved evaluation methodology that can contribute to head loss are the maximum amount of debris transported to the sump screen and the worst combination of debris mixes that are transported to the sump screen. Additionally, the approved methodology states that breaks should be considered in each high-pressure system that relies on recirculation, including secondary side system piping, if applicable.

In DCD Tier 2, Subsection 6.2.2.3.1, “Break Selection,” as supplemented by GTR2, and MUAP-08001-NP, the applicant considers breaks in the primary coolant system and secondary side system to require RWSP recirculation operation. The break size considered for line breaks are double ended guillotine breaks. The basis for this break selection is to provide the largest volume of debris from insulation and other materials that may be within the region affected by the postulated break. The applicant used break selection criteria guidance from the SE of NEI 04-07 and RG 1.82. MUAP-08001-NP provides additional details of the break selection evaluation.

Section 3.3.5 of the SE to NEI 04-07 describes an approach to the break selection process which includes beginning the evaluation at an initial location along a pipe and stepping along in equal increments (i.e., 5 foot (ft) (1.524 meter (m)) increments) considering breaks at each sequential location. As described in MUAP-08001-NP Section 3.1, “Break Selection,” the applicant chose an alternate approach to the use of equal increments for the US-APWR break selection evaluation, given that the insulation in the ZOI is limited to one type, RMI. Therefore, only RMI debris is considered as potential insulation debris that is dependent on any break location. In order to maximize the RMI contribution, the applicant evaluates all of the RMI installed on the main coolant piping cross-over leg (largest diameter pipe) as debris. This approach, assuming all RMI installed on the main coolant piping cross over leg as debris, when compared to an equal increments approach, produces the largest quantity of RMI. As a result, the standard US-APWR design takes a conservative approach to calculating the amount of insulation debris from a pipe break.

In addition to insulation debris, the US-APWR design evaluates other sources of debris that could transport to the strainer and contribute to head loss: latent debris, coating debris, and post-accident chemical effects debris. For the US-APWR the amount of latent and chemical debris generated and transported to the RWSP strainers is independent of break location. Latent debris and chemical effects are discussed in Subsections 6.2.2.4.6 and 6.2.2.4.15 of this SER, respectively. The coatings debris is also considered to be a constant volume regardless of break location. The coatings debris amount is discussed in Subsection 6.2.2.4.7, “Coatings Evaluation,” of this evaluation. Debris transport is discussed in Subsection 6.2.2.4.8 of this SER.

In DCD Tier 2, Subsection 6.2.2.3.1, the applicant concludes that a break in the main coolant pipe, 31 inch (in.) (78.74 centimeter (cm)) inner diameter, is the limiting break location in terms of debris generation, transport, and head loss for the strainer.

Given the discussion above, the staff finds that the applicant's break selection evaluation is acceptable because the applicant's design and approach simplifies the break selection process and meets the intent of NEI 04-07 and the associated NRC SE and Regulatory Position C.1.3.2.3 of RG 1.82, Revision 3. The staff opened **Confirmatory Item CI-SRP06.02.02-2** to verify that the final design document incorporates the associated supplemental DCD information provided by GTR2.

6.2.2.4.4 Zone of Influence (ZOI)/Debris Generation (Excluding Coatings)

The ZOI is the volume about the break in which the LOCA break jet forces would be sufficient to damage materials. Debris generation is the amount of debris generated by these forces. Sections 3.4 and 4.2.2 of NEI 04-07 and the associated NRC SE provide a methodology to assess ZOI and debris generation. The NEI 04-07 baseline methodology incorporates a spherical ZOI based on material damage pressures. The size of the spherical ZOI is based on experimentally deduced destruction pressures that were determined by applying ANSI/ANS 58.2 1988 standard jet expansion models to various types of insulation. The pressure gradients were used to correlate the damage to insulation blankets or cassettes by air and steam jets during debris generation testing to an equivalent spherical model of destruction. The relationship between the ANSI/ANS 58.2-1988 standard and the staff approved ZOIs was assessed in Appendix I, "ANSI/ANS Jet Model," of the NRC SE on NEI 04-07. Once the ZOI is established for a selected break location, the types and locations of all potential debris sources can be identified using plant specific drawings, specifications, or other such reference materials. The amount of debris generated is then calculated based on the amount of materials within the most limiting ZOI. Section 4.2.2 of NEI 04-07 and the associated NRC SE discuss proposed refinements to the NEI 04-07 baseline methodology that would allow application of debris-specific ZOIs. This refinement allows the use of a specific ZOI for each debris type identified. Using this approach, the amount of debris generated within each material-specific ZOI is calculated, and then these material-specific debris amounts are combined to arrive at a total debris source term. The NRC SE on NEI 04-07 concluded that the definition of multiple, spherical ZOIs at each break location corresponding to damage pressures for potentially affected materials is an appropriate refinement for debris generation. As discussed in Section 4.2.2 of the NRC SE on NEI 04-07, the NRC staff accepted the application of these proposed refinements for PWR sump analyses.

The applicants' ZOI and debris generation evaluations and methods are presented in DCD Tier 2, Subsection 6.2.2.3.3 "Debris Generation," as supplemented by GTR2, and MUAP-08001-NP. The applicant uses the analytical refinements associated with debris-specific ZOIs, consistent with staff guidance. This refinement allows use of a specific ZOI for each debris type identified. The applicant analyzes two insulation debris types following a LOCA, that is, RMI and Nukon. In the US-APWR, RMI is applied to the reactor vessel (RV), the RCPs, the SGs, the pressurizer, main coolant pipes connecting the RV, the SG, and the RCP, and to the main steam and main feed pipe lines. The ZOI radius for RMI debris generation is two times the inside diameter of the pipe at the assumed break location. Fiber insulation debris is excluded from the ZOI, which was evaluated at a ZOI radius of 17 times the inside diameter.

DCD Tier 2, Table 6.2.2-4, "Design Basis Debris," as supplemented by GTR2, summarizes the design basis insulation debris in containment and states the RMI amount is 106 cubic feet (ft^3) (3 cubic meters (m^3)) and fiber insulation is 0.0 ft^3 (0.0 m^3). The applicant also assumes an amount of fiber insulation as part of the debris evaluation. This additional amount of fiber insulation is referred to as operational margin. DCD Tier 2, Table 6.2.2-4 defines the amount of fiber insulation serving as operational margin as 1.875 ft^3 (0.053 m^3) and this fiber insulation is assessed as part of the strainer testing program.

The staff reviewed the applicants' ZOI and debris generation evaluations, as presented in DCD Tier 2, Subsection 6.2.2.3.3 and MUAP-08001-NP, relying on the approved methods documented in Sections 3.4 and 4.2.2 of the NRC SE associated with NEI 04-07. The staff finds that the applicant's approach is acceptable because it is consistent with staff guidance on debris-specific zones of influence. The staff opened **confirmatory item CI-SRP06.02.02-2** to verify that the final design document incorporates the associated supplemental DCD information provided by GTR2.

6.2.2.4.5 Debris Characteristics

The objective of this section is to assess the applicant's analysis related to the characteristics of post-accident debris. Section 3.4.3 of the NEI 04-07 and associated NRC SE provide guidance for debris characteristics. NEI 04-07 describes the debris characteristics in terms of size distribution, size and shape, and density.

The applicant discussion of debris characteristics is contained in DCD Tier 2, Subsection 6.2.2.3.4, "Debris Characteristics," as supplemented by GTR2, and MUAP-08001-NP. The analyzed debris for US-APWR includes RMI, latent, and coatings debris. This evaluation section describes the applicants' assumptions regarding the characteristics of RMI debris. The characteristics of latent debris and coatings are discussed in Sections 6.2.2.4.6 and 6.2.2.4.7 of this SER, respectively. The characteristics of chemical debris are discussed in Subsection 6.2.2.4.15 of this SER.

The applicant's size distribution and size assumed for RMI is 75 percent small pieces and 25 percent large pieces. The staff finds the RMI size and size distribution to be acceptable because it is consistent with guidance contained in the NRC SE associated with NEI 04-07.

The fiber insulation that serves as operational margin is assumed to be 100 percent fiber fines. The staff finds the fiber insulation size distribution and size to be acceptable because it is consistent with guidance contained in the NRC SE associated with NEI 04-07. The applicant uses a 2.4 lbm/ft^3 (38.4 kg/m^3) density for fiber insulation debris, consistent with Nukon fiberglass properties. The bulk density of 2.4 lbm/ft^3 (38.4 kg/m^3) for fiber insulation debris is approved in the staff's SE on NEI 04-07, and is acceptable. In addition, the staff considers the treatment of fiber insulation debris as fiber fines to be a reasonably conservative assumption for operational margin as fiber fines readily transport and are able to accumulate on the strainer.

The staff opened **confirmatory item CI-SRP06.02.02-2** to verify that the final design document incorporates the associated supplemental DCD information provided by GTR2.

6.2.2.4.6 Latent and Miscellaneous Debris

Latent debris is unintended debris present in containment prior to a postulated high-energy line break, which may be composed of various constituent materials including dirt, dust and other particulate or fiber. The objective of the latent debris evaluation is to provide an estimate of the types and amounts of latent debris existing in containment for the purpose of assessing its impact on sump strainer head loss. The applicant performed an evaluation of the potential sources of latent debris within containment using the guidance provided in NEI 04-07 and the associated NRC SE.

In DCD Tier 2, Section 6.2.2.3.4, as supplemented by GTR2, and MUAP-08001-NP, the applicant models latent fiber debris as 100-percent fines that are transportable to the debris screens. The staff considers this size and size distribution to be acceptable because it is consistent with guidance contained in the NRC SE associated with NEI 04-07. The applicant uses a 2.4 lbm/ft^3 (38.4 kg/m^3) density for latent fiber debris and uses Nukon fiberglass as the surrogate debris for latent fiber debris. The bulk density of 2.4 lbm/ft^3 (38.4 kg/m^3) for latent fiber debris was approved in the staff's SE on NEI 04-07, and is acceptable. The use of fines from a low-density fiberglass, such as Nukon, as a surrogate debris for latent fiber in head loss testing is acceptable because this practice was also considered appropriate in the staff's SE on NEI 04-07, as discussed in Section 3.5.2.3 and Appendix VII, "Characterization of Pressurized Water Reactor Latent Debris," of that document. These documents indicate that the hydraulic properties of latent fiber are similar to those of fiberglass (Nukon).

In DCD Tier 2, Subsection 6.2.2.3.4, as supplemented by GTR2, and MUAP-08001-NP, the applicant models latent particulate debris as a mix of dirt and dust (sand) covering a range of sizes consistent with guidance provided in NUREG/CR-6877, "Characterization and Head-Loss Testing of Latent Debris from Pressurized-Water-Reactor Containment Buildings." The staff considers the method used to select size and size distribution to be acceptable because this practice is consistent with this guidance.

In DCD Tier 2, Subsection 6.2.2.3.4, as supplemented by GTR2, and MUAP-08001-NP, the quantity of latent debris specified for evaluation is 200 lbm (90.7 kg). NEI 04-07 guidance indicates this quantity of latent debris likely bounds the maximum mass of latent debris inside operating PWR containments, based upon sampling. The US-APWR design assumes the latent fiber comprises 15 percent (by mass) and latent particulate comprises 85 percent (by mass) of the total latent debris loading. The staff considers the fiber/particulate percentages to be acceptable because this practice is consistent with the staff's SE on NEI 04-07.

The staff concludes that the assumption of 200 lbm (90.7 kg) for the latent debris mass is a practical estimate of the latent debris mass for the US-APWR containment. In accordance with the containment cleanliness program, applicants referencing the US-APWR must include a program to limit the amount of latent debris left inside containment following refueling and maintenance outages to the design basis amounts. The staff finds the 200 lbm (90.7 kg) of total latent debris, consisting of 30 lbm (13.6 kg) of fiber and 170 lbm (77.1 kg) of particulate, inside the US-APWR containment to be acceptable because the quantity and type of latent debris is consistent with NEI 04-07 guidance and the associated staff SE, and there is a COL item to limit latent debris to the design-basis amounts (see COL 6.2-5).

Miscellaneous debris is composed of items such as equipment tags, tape, labels, and placards. Section 3.5.2.2.2, "Evaluate Resident Debris Buildup," of the NEI 04-07 and associated staff SE provide guidance for the considerations to be used in identifying and evaluating potential sources of miscellaneous debris in containment. The objective of the miscellaneous debris evaluation is to provide an estimate of the types and amounts of miscellaneous debris existing in containment for the purpose of assessing its impact on sump strainer head loss. The staff's SE of the NEI guidance indicates that if transportability of miscellaneous debris or the capability of miscellaneous debris to remain intact cannot be determined then it should be assumed that they remain intact and are transported to the sump screen, to preserve conservatism. In the absence of specific data which describes the behavior of miscellaneous debris, the staff's SE of the NEI guidance states that the wetted sump-screen flow area be reduced by an area equivalent to 75 percent of the surface area of the debris.

In DCD Tier 2, Subsection 6.2.2.3.3, as supplemented by GTR2, and MUAP-08001-NP, the types of miscellaneous debris are not specified. Instead, a 200 ft^2 (18.6 m^2) reduction in strainer area is assigned.

The staff concludes that the assumption of 200 ft^2 (18.6 m^2) is a practical estimate of the miscellaneous debris for the US-APWR containment. In accordance with the containment cleanliness program, applicants referencing the US-APWR must include a program to limit the amount of miscellaneous debris in containment. The staff finds the 200 ft^2 (18.6 m^2) assumption of miscellaneous debris to be acceptable because of the COL commitment to limit miscellaneous debris to the design-basis amount (see COL 6.2-5).

In summary, the applicant's method and approach for assessing latent and miscellaneous debris is acceptable because it is consistent with staff guidance on size, size distribution, and amount of debris. Additionally, the US-APWR has a COL commitment to limit latent and miscellaneous debris, consistent with the design basis. The staff opened **confirmatory item CI-SRP06.02.02-2** to verify that the final design document incorporates the associated supplemental DCD information provided by GTR2.

6.2.2.4.7 Coatings Evaluation

To determine if the US-APWR design meets the requirements of GDC 35, GDC 38, and 10 CFR 50.46(b)(5) with respect to protective coatings (paint) in containment, the staff reviewed information in the DCD and supporting documents listed in Section 6.2.2.3, "Regulatory Basis," of this evaluation. The following are key guidance documents for coatings debris, and the first two are exclusive to coatings debris:

- "Revised Guidance Regarding Coatings Zone of Influence for Review of Final Licensee Responses to Generic Letter 2004-02, "Potential Impact of Debris Blockage on Emergency Recirculation during Design Basis Accidents at Pressurized-Water Reactors," dated April 6, 2010 (ADAMS Accession Number ML100960495).

- Enclosure 2, "NRC Staff Review Guidance Regarding Generic Letter 2004-02 Closure in the Area of Coatings Evaluation," to "Revised Guidance for Review of Final Licensee Responses to Generic Letter 2004-02," "Potential Impact of Debris Blockage on Emergency Recirculation during Design Basis Accidents at Pressurized-Water Reactors," dated March 28, 2008 (ADAMS Accession Number ML080230234).

- NEI 04-07, Revision 0 dated December 2004, and the staff's accompanying SE.

As stated in the NRC SE on NEI 04-07 for protective coatings, it is acceptable for applicants to use a coatings ZOI spherical equivalent distance determined by plant-specific analysis or a default value of 10 pipe diameters. Subsequent to the guidance issued in the NRC SE on NEI 04-07, additional coatings testing was completed and demonstrated that some coatings qualified for a ZOI reduction. The staff's March 2008 guidance regarding coatings zone of influence approved a spherical ZOI radius of 4D (four times the pipe diameter at the assumed break location) for epoxy based coatings. The staff's April 2010 coatings guidance confirmed the March 2008 guidance remains valid for epoxy coatings.

Depending on the break location, coated components may or may not exist within the ZOI. When plant-specific information does not exist regarding the amount of coatings within the ZOI, NEI 04-07 assumes that coated components with an area equivalent to the surface area of the sphere will exist within this volume and that the coatings on these components will fail, generating fine particulate debris. The volume of coating debris is a function of the coating thickness and surface area.

DCD Tier 2, Section 6.1.2, "Organic Materials," and Subsection 6.2.2.3.9, "Coatings Evaluation," describe the selection of coatings for the US-APWR. The US-APWR uses a qualified coating system in containment and only epoxy type coatings are used. See Section 6.1.2 of this SE for details on the coatings system evaluation.

DCD Tier 2, Subsections 6.2.2.3.3 and 6.2.2.3.4, as supplemented by GTR2, and MUAP-08001-NP state the coatings ZOI applied to epoxy based coatings is four pipe diameters (4D) and 100 percent of the coating material in the ZOI fails as particulate in the size of 10 microns (0.00039 inches). As stated in the SE for NEI 04-07, the staff found it reasonable to treat coating debris as highly transportable particulates in the size range 10 to 50 microns where there is a possibility of a thin fiber bed. This particle size is based on the basic material constituents for inorganic zinc and epoxy coatings. The staff finds it is appropriate to treat the coatings in the ZOI as particulate for two reasons. First, all the coatings in the containment are qualified and only coatings within the ZOI will fail. Second, coatings within the ZOI are subjected to jet forces that are known to erode the coatings into a particulate form.

The applicant calculates the US-APWR coating debris volume based on the surface area of the spherical ZOI multiplied by a coating thickness. The sphere radius is four times the main coolant pipe break diameter. As a result, DCD Table 6.2.2-4 states the maximum volume of coating debris is $3 ft^3$ ($0.085 m^3$). The applicant assumes the epoxy coating density is $94 lbm/ft^3$ ($1505 kg/m^3$) to determine the mass of coatings. This material density is consistent with NEI 04-07 guidance and is therefore acceptable.

Based on the applicant's analysis approach, the coatings debris amount is the same for all break locations. In **RAI 278-2250, Question 06.02.02-16**, the staff asked the applicant to justify that the surface area of a sphere was a conservative estimate of the coated surface area inside the ZOI and to justify the assumed coating thickness of 650 microns (0.0256 inches). In its response to **RAI 278-2250, Question 06.02.02-16**, dated April 10, 2009, the applicant assessed an existing plant design with a similar layout to the US-APWR and concluded that the assumed coatings thickness and spherical surface area for the US-APWR resulted in a greater volume of coatings in comparison to the operating plant, and was therefore a reasonably conservative approach. The staff concludes that the assumption of $3 ft^3$ ($0.085 m^3$) is a practical estimate of the coating debris for the US-APWR containment. In addition, since the coatings debris amount is a key design parameter, the as-built coatings will be verified to be consistent with the design basis as part of ITAAC 7.b.v found in Table 2.4.4-5 "Emergency Core Cooling System Inspections, Tests, Analyses, and Acceptance Criteria." The applicant also assumes an additional amount of coating debris, with the same characteristics discussed above, as part of the debris evaluation. This additional amount of coatings debris is referred to as operational margin. DCD Tier 2, Table 6.2.2-4 defines the amount of coating serving as operational margin as 200 lbm (90.7 kg) and this additional coatings debris is assessed as part of the strainer testing program.

Given the discussion above, the staff finds the applicant's coatings assessment using the NRC guidance in RG 1.82, NEI 04-07 and associated staff SE, and the additional coating guidance documents, is acceptable with respect to the ZOI, debris generation, and characteristics of coating debris. Therefore, **RAI 278-2250, Question 06.02.02-16 is resolved and closed,** but the staff opened **confirmatory item CI-SRP06.02.02-2** to verify that the final design document incorporates the associated supplemental DCD information provided by GTR2.

6.2.2.4.8 Debris Transport

Debris transport analysis estimates the fraction of debris that would be transported from debris sources within containment to the sump suction strainers. In general, debris transport in the containment can be considered to occur through four primary mechanisms:

- blowdown transport, which is the vertical and horizontal transport of debris throughout containment by the break jet;

- washdown transport, which is the downward transport of debris due to fluid flows from the containment spray and the pipe rupture;

- pool-fill transport, which is the horizontal transport of debris by break flow and containment spray flow to areas of the containment pool that may be active (influenced by recirculation flow through the suction strainers) or inactive (hold-up or settling volumes for fluid not involved in recirculation flow) during recirculation flow; and

- containment pool recirculation transport, which is the horizontal transport of debris from the active portions of the containment pool to the suction strainers through pool flows induced by the operation of the ECCS and CSS in recirculation mode

The applicant discusses debris transport in DCD Tier 2, Subsection 6.2.2.3.5, "Debris Transport," as supplemented by GTR2, and MUAP-08001-NP. The US-APWR has a number of gratings, curbs, and hold-up volumes that can trap debris or cause debris to settle. However, the US-APWR does not credit debris settlement for fiber and coating debris and all this debris is assumed to transport to the RWSP and the operable ECC/CS trains. RMI debris is addressed near the end of this debris transport evaluation section.

The US-APWR has four ECC/CS trains with an independent sump strainer for each train. The design requires a minimum of two trains in operation, assuming one train is out of service for maintenance and the remaining train is out of service due to a single failure. Therefore, transported debris in the RWSP could be distributed to two, three, or four sump strainers. The number of operable ECC/CS trains during a postulated accident determines the debris distribution to each strainer. For the strainer head loss evaluation, the number of available ECC/CS trains in service should maximize the head loss, therefore only two trains are in service. For the strainer debris bypass fraction, the number of available ECC/CS trains should maximize the amount of bypass debris; therefore all four ECC/CS trains could be in service. Debris that is postulated to bypass the strainer is assessed in Subsections 6.2.2.4.13, “Downstream Effects – Ex-Vessel Components and Systems,” and 6.2.2.4.14, “Downstream Effects – Fuel Assemblies,” of this SER.

General arrangement drawings for the US-APWR are provided in DCD Tier 2, Chapter 1 Figure 1.2-14 through Figure 1.2-25 (contain security-related information). Detailed drawings around the RWSP are provided in Appendix-D to MUAP-08001-P (proprietary). The US-APWR has a similar layout to conventional four loop PWR plants, with a SG and RCP located in each of four independent SG compartments. Each SG compartment is enclosed horizontally by primary and secondary shield walls. Access openings, at floor level, are provided between the SG compartments. In addition, a labyrinth access opening penetrates the secondary shield wall, providing access to and from each SG compartment. The labyrinth access floor elevation is 2 in. (5 cm) higher than the nominal floor level.

DCD Tier 1, Figure 2.4.4-1, “Emergency Core Cooling System (Sheet 4 of 4),” as supplemented by GTR2, shows a schematic of the water flow path from the SG compartment to the RWSP. Water flowing through the SG compartment floor openings is directed into three rooms: reactor cavity, header compartment, and containment vessel (C/V) drain pump room. The reactor cavity and header compartment each contain overflow pipes that return water to the RWSP (note, there are no overflow pipes in the C/V drain pump room). There are three sets of four overflow pipes for a total of twelve overflow pipes. One set of four pipes is located in the upper elevation of the reactor cavity and the remaining two sets of four pipes are located in the upper elevation of the header compartment. The reactor cavity and header compartments are connected to equalize water levels between the two compartments.

Although the design basis has a minimum of two ECC/CS trains operating, the applicant assumes all design basis coatings debris is transported to only one train (sump strainer). In a similar manner, the fiber insulation and coatings referred to as operational margin in DCD Table 6.2.2-4 (note 1 and 2) are all transported to only one train (sump strainer). The latent and chemical debris are treated as uniformly distributed in containment and are distributed among the operating strainers. The US-APWR assumes that uniformly distributed debris is equally allocated to the overflow pipes, that is, approximately 33 percent to each set of overflow pipes (three sets of pipes, each set contains 4 pipes). Given the possible strainer combinations, the applicant determined that a reasonably conservative debris allocation for two operable sumps is 85 percent/15 percent, for debris that is considered to be uniformly distributed (see MUAP-08001-NP Figure 3-5, “Schematics of Debris Allocation on Operable Sumps (for head loss)”).

The eighty-five percent debris allocation provides the greatest amount of debris to one strainer and determines the debris amounts used for strainer testing discussed in Subsection 6.2.2.4.9, “Strainer Head Loss Testing,” of this SER. The staff finds the 85 percent/15 percent debris allocation is conservative because it represents the largest debris allocation to one strainer and the applicant does not consider debris to settle behind a curb, collect on grating, or remain in a hold-up volume, such as the reactor cavity, header compartment, or C/V drain pump room.

In the US-APWR design the applicant considers that RMI insulation does not collect on the strainer surfaces because of the strainers low approach velocity, that is, 0.0045 ft/s (0.1372 cm/s). RMI settling velocities and lift over curb velocities are two orders of magnitude greater than the US-APWR strainer approach velocity and therefore any RMI that may reach the RWSP should settle on the floor and not rise up to the strainer surface. The applicant’s evaluation is consistent with staff guidance found in NEI 04-07 and the associated staff SE and is therefore acceptable. In addition, given RMI’s material properties, it is reasonable to expect that a large portion of the RMI debris will be trapped behind curbs, on debris interceptors (e.g., on the SG compartment floor) or inside the reactor cavity, header compartment, and C/V drain pump room. Additional discussion on the treatment of RMI during head loss testing is found in Subsection 6.2.2.4.9 of this SER.

Given the discussion above, the staff finds the applicant's evaluation of debris transport to the RWSP and operable sump strainers is consistent with staff guidance and is acceptable. The staff opened **confirmatory item CI-SRP06.02.02-2** to verify that the final design document incorporates the associated supplemental DCD information provided by GTR2.

6.2.2.4.9 Strainer Head Loss Testing

The objective of this section is to provide the staff's evaluation of the applicant's strainer head loss testing. The applicant conducted strainer head loss testing to demonstrate that the tested strainer head loss is bounded by the assumed strainer head loss used in the NPSH evaluation. The applicant also included additional fiber and coatings material, above the design basis debris amounts (referred to as operational margin), in an effort to develop a conservative strainer head loss.

During a postulated LOCA inside containment, pipe and equipment insulation and coatings can be fragmented by the jet forces emitted from the break location. Chemical precipitate debris may be created from coolant system fluid and the buffering agent solutions interacting with plant materials and the generated debris (chemical debris is discussed in Subsection 6.2.2.4.15, "Chemical Effects," of this evaluation). All this debris potentially transports to the RWSP and sump strainers.

The US-APWR debris strainer is located in the RWSP and upstream of the ECC/CS pumps to minimize debris from entering the pump suction and downstream locations. Debris that transports to the RWSP must not cause a strainer screen head loss that adversely impacts NPSH and satisfactory operation of the ECC/CS pumps during DBA conditions. The overall head loss attributable to the strainer is a combination of the debris deposited on the strainer and the head loss associated with the clean strainer.

As shown in the US-APWR DCD Figures 6.2.2-8, "Plan View of RWSP and ECC/CS Strainers," and 6.2.2-9, "Sectional View of RWSP and ECC/CS Strainers," four independent sets of ECC/CS strainers are provided inside the RWSP. The ECC/CS strainers minimize debris entering the safety systems that are required to maintain the post-LOCA long-term cooling performance. DCD Tier 2, Subsection 6.3.2.2.4, "ECC/CS Strainers," as supplemented by GTR2, discusses that the ECC/CS strainers are installed on the bottom floor of the RWSP, and are designed to be fully submerged during all postulated events requiring the actuation of the ECCS.

Principal design features of the strainers are provided in DCD Tier 2, Table 6.3-5, "Safety Injection System Design Parameters (Sheet 1 of 3)," as supplemented by GTR2. Additional strainer design attributes are described in MUAP-08001-NP. The standard US-APWR design utilizes a passive disk layer type of strainer systems (Sure-Flow Strainer, supplied by Performance Contracting Inc.). The strainer is principally constructed of perforated plate with a square flange at the bottom for attaching to the supporting plate, which covers the sump pit. A manifold core tube connected to the flange penetrates near the center of the layered disks, and guides the clean water filtered by the layered disks into the sump pit. The joint gap between the components of the strainer is controlled to preclude debris from bypassing the perforate plates.

The strainers and supporting plates will be constructed of corrosion-resistant stainless steel. The nominal diameter of holes is designed to be equal or less than 0.066 in. (0.17 cm).

The applicant conducted head loss testing using US-APWR plant-specific debris loads and flow rates to demonstrate the adequacy of the strainer screens. The debris types considered are non-chemical particulates and fibers from latent debris, coatings debris, and chemical precipitates.

The methodology, assumptions, and results of strainer head loss testing is documented in proprietary Appendices provided in MUAP-08001-P. The testing was developed using NRC Staff guidance from Enclosure 1, “NRC Staff Review Guidance Regarding Generic Letter 2004-02 Closure in the Area of Strainer Head Loss and Vortexing,” and Enclosure 3, “NRC Staff Review Guidance Regarding Generic Letter 2004-02 Closure in the Area of Plant-Specific Chemical Effect Evaluations,” to “Revised Guidance for Review of Final Licensee Responses to Generic Letter 2004-02,” “Potential Impact of Debris Blockage on Emergency Recirculation during Design Basis Accidents at Pressurized-Water Reactors,” dated March 28, 2008 (ADAMS Accession Number ML080230234).

The US-APWR strainer head loss testing was performed at Alden Research Laboratory located in Holden, Massachusetts. The scaled down test facility included a large steel tank that contains a strainer, which is representative of a single ECC/CS train in the US-APWR design. The US-APWR strainer head loss test program utilized a mixing tank test protocol and test facility that does not take credit for debris settlement upstream of the strainer. The intent of this protocol and facility is to prevent settling during the tests, and provide a reasonably conservative test result. (See strainer head loss testing audit report, dated September 20, 2011, ADAMS Accession Number ML112371513).

The applicant adopted an area ratio-based geometrical scaling approach. The area ratio is a ratio of the test strainer area and the plant strainer area. Based on this scaling principle, a portion of the plant strainer was placed in the test loop. The test loop flow rate was determined by multiplying the design basis maximum sump flow rate by the ratio of the test strainer surface area to the plant strainer surface area. In this way, the screen surface approach velocity is comparable to the plant. The debris loading in the testing was also scaled based on the area ratio. The area ratio-based scaling approach is consistent with the March 2008 staff guidance and therefore the staff finds this approach acceptable.

The two main tests typically conducted to evaluate the strainer head loss are a full load test and thin bed test. For plants with minimal fiber debris, a test with the upper bound (full load) fiber quantities may also serve to determine whether or not the thin bed configuration can occur. The US-APWR has minimal fiber debris, and the debris introduction procedure was designed to allow gradual debris accumulation on the strainer surface to capture potential thin bed formation. The staff finds this approach consistent with the March 2008 guidance and therefore is acceptable for US-APWR strainer testing. The maximum fiber debris allocated to one strainer results in a theoretical bed thickness of about 1/18th (0.055) of an inch (0.14 cm).

Debris used for testing

DCD Table 6.2.2-4, “Design Basis Debris”, as supplemented by GTR2, lists the types and quantities of debris used to assess the sump strainer performance. The debris loads used for testing were found to be consistent with the debris generation, characterization, and transport analysis. The debris loads used in the tests were scaled down from the plant debris loads based on the ratio of the tested screen area to the plant screen area. As explained in MUAP-08001-P, the screen area scaling factor is [ ]. In addition, the amount of latent and chemical debris used for the strainer test was multiplied by 85 percent to account for the maximum debris allocation to one sump.

The staff compared the characteristics of the surrogate test materials with the corresponding plant material to ensure prototypicality or conservatism. The insulation surrogate material used by the applicant during head loss testing was the same type as the plant material. Latent debris, coating, and chemical surrogate materials were consistent with the March 2008 guidance. For example, [ ], and Aluminum Oxyhydroxide (AlOOH) was used as the chemical surrogate. The staff finds that the applicants' surrogate materials used for head loss testing are acceptable because they are similar in shape, size, and density to plant materials and/or are consistent with staff guidance.

RMI debris head loss assessment

The applicant's transport analysis, found in DCD Tier 2, Section 6.2.2.3.5, "Debris Transport," as supplemented by GTR2, showed that RMI would not collect on the strainer, so strainer head loss testing was not conducted with RMI. This is consistent with the staff's testing guidance presented in the March 28, 2008, document listed above. Additionally, because settled RMI debris could potentially create a surface on which otherwise transportable debris (fiber and particulate) could collect, and therefore limit the amount of debris reaching the strainer, it was considered reasonable to not test with RMI. Also, for the design basis debris loads expected in US-APWR and the applicant's testing approach, that is a stirred or non-settlement test, the staff considers it acceptable to perform testing without introducing RMI into the test flume along with the other debris.

Miscellaneous debris (Tags, Tape, Label) head loss assessment

Based on the US-APWR debris generation analysis, the applicant assumed 200 ft^2 (18.6 m^2) of miscellaneous debris per sump (tags, tape, and labels etc.) is available within containment to potentially obstruct portions of the strainer screen. In the absence of specific data which describes the behavior of miscellaneous debris, staff guidance states that the wetted sump-screen flow area be reduced by an area equivalent to 75 percent of the surface area of the debris. To account for miscellaneous debris, the applicant reduced the screen area by 200 ft^2 (18.6 m^2). The 200 ft^2 (18.6 m^2) area reduction was applied to the plant strainer prior to calculating the scaling area ratio. Reducing the plant strainer area increases the scaling area ratio, resulting in more debris addition to the test rig. In addition, the higher scaling area ratio increases the fluid velocity and therefore increases the head loss across the strainer. Because the applicant's treatment of miscellaneous debris is consistent with staff guidance found in the SE on NEI 04-07, the staff finds the applicant's treatment of miscellaneous debris for head loss testing to be acceptable.

Debris preparation, sequencing and addition

The appropriate types, sizes, and amounts of non-chemical debris were individually placed into buckets with water and mixed to prevent debris agglomeration. The debris was then poured into a tank and pumped into the test flume except for the latent particulate material (dirt and dust), which was added directly to the test flume. The debris was sequenced into the flume in the following order: 1) all non-chemical particulate, 2) fiber. Chemical additions followed completion of all non-chemical particulate and fiber additions.

The applicant's debris preparation, sequencing and additions were consistent with the staff's March 2008, testing guidance and are acceptable. For example, the debris was prepared as fine and readily suspendable material. The debris was sequenced with most transportable debris first and the least transportable last. The debris was introduced slowly into the test tank with the pump running and conservative hydraulic conditions (mixing, non-settlement test) established. Chemical debris in strainer head loss testing is discussed in Section 6.2.2.4.13.5 of this evaluation.

Post-test scaling

US-APWR strainer head loss testing was performed with water at 120-130 °F (48.9-54.4 °C). This is at relatively low temperature when compared to potential sump temperatures following a postulated LOCA (> 212° F (100° C)). The applicant initially performed post-test data scaling using a ratio of the water viscosities to predict strainer head loss at elevated temperatures. The March 2008, head loss testing guidance recommends a flow sweep be conducted at the end of the test to verify that the strainer head loss varies relatively linearly with flow in order to support a temperature scaling approach using a ratio of water viscosities. Based upon review of the flow sweep data, the staff issued **RAI 912-6355, Question 06.02.02-88**, and requested that the applicant provide additional information to justify use of a viscosity correction to scale the US-APWR head loss tests to higher temperatures. In its response to **RAI 912-6355, Question 06.02.02-88**, dated June 13, 2012, the applicant no longer applies a viscosity correction to determine the strainer head loss at elevated temperatures. The staff considers that not using a viscosity correction to scale head loss results collected at a low temperature to determine head loss at a high temperature is conservative and therefore the response to **RAI 912-6355, Question 06.02.02-88, is acceptable**.

Strainer Head Loss Testing Assessment

The applicant's strainer testing is documented in MUAP-08001-P. As discussed above (including chemical effects discussed in Section 6.2.2.4.13 of this evaluation), the applicant's head loss testing approach is consistent with staff guidance and as such, provides confidence that the testing reasonably bounds the peak head loss. DCD Tier 2, Section 6.2.2.3.6, "Debris Head Loss," as supplemented by GTR2, identifies that the strainer head loss tests support the design basis strainer head loss with margin. In other words, the test result for the maximum head loss across the strainer is less than the strainer head loss assumed in the NPSH evaluation. For example, at 120 °F (49 ^{o}C), the tested strainer head loss with non-chemical and chemical debris is about two feet, whereas the design analysis assumes four feet (100 percent margin). The maximum head loss across the strainer screen is also less than the design basis strainer minimum submergence. Therefore, fluid flashing at the strainer debris bed is prevented (see additional discussion in Subsection 6.2.2.4.10, "Net Positive Suction Head (NPSH)," of this SER). In addition, strainer testing was conducted at the minimum submergence level using maximum flowrates and showed no signs of vortex formation, providing evidence to support the applicant's evaluation which showed no potential for formation of a vortex given the low approach velocities, amount of submergence, small opening size of the perforated plate, and overall stacked-disc geometry.

Based on the discussion above, the staff finds that the applicant's strainer testing and analysis provides a reasonable approach, consistent with the staff's March 2008 guidance, to determining the effects of US-APWR plant specific debris on strainer performance and is considered acceptable. The response to **RAI 912-6355, Question 06.02.02-88,** being acceptable, it is **resolved and closed,** but the staff opened **confirmatory item CI-SRP06.02.02-2** to verify that the final design document incorporates the associated supplemental DCD information provided by GTR2.

6.2.2.4.10 Net Positive Suction Head (NPSH)

NPSH review criteria are contained in SRP Section 6.2.2 and RG 1.82. Additional guidance is found in SECY Paper 11-0014, “Use of Containment Accident Pressure in Analyzing Emergency Core Cooling System and Containment Heat Removal System Pump Performance in Postulated Accidents,” dated January 31, 2011. Section 6.3 of this SE, “Emergency Core Cooling Systems,” provides additional discussion on NPSH.

NPSH is the difference between the stagnation pressure at the pump suction and the liquid vapor pressure. It is, therefore, a measure of the energy forcing the liquid into the pump. There are two related quantities, the available NPSH and the required NPSH. The available NPSH (NPSHA) is a function of the system design, the pump flow rate and the temperature of the pumped water. The required NPSH (NPSHR) is the NPSH that produces an acceptable amount of cavitation, measured as a reduction in pump discharge head. NPSHR is a function of the pump design and pump flow rate. NPSHR increases as the pump flow rate increases, whereas NPSHA decreases as the pump flow rate increases.

The US-APWR ESF includes safety injection (SI) pumps and CS/RHR pumps. These pumps are normally aligned to take suction from the RWSP located inside the containment. DCD Tier 2, Figure 6.2.2-1, “Flow Diagram of the Containment Spray” and Figure 6.3-2, “ECCS Piping and Instrumentation Diagram,” show piping and instrumentation diagrams of the CSS and ECCS. The applicant compares the RG 1.82 NPSH positions to the US-APWR design in DCD Table 6.2.2-2, “Comparison of RSWP Recirculation Intake Debris Strainer Design to RG 1.82 Requirements,” as supplemented by GTR2.

DCD Tier 2, Subsection 6.2.2.3.7, “Net Positive Suction Head,” as supplemented by GTR2, and MUAP-08001-NP describe the applicant’s overall NPSH evaluation. Containment pressure, RWSP level, head losses through the sump strainer and pump suction piping, and RWSP water temperature determine the fluid conditions at the suction of the pump. For the NPSHA calculation, MUAP-08001-NP describes the input conditions that would provide a conservative result such as assuming maximum pump flowrates, minimum RWSP water level, maximum strainer head loss (with chemicals), and system resistance from piping and components. Minimum RWSP level is discussed in Section 6.2.2.4.11 of this SER, “Upstream Effects.” Strainer head loss is discussed in Section 6.2.2.4.9 of this SER. For the minimum NPSHA calculation, the applicant assumes that no additional containment pressure is credited above the initial containment pressure for low sump fluid temperatures (i.e., below approximately 212 °F (100 °C)). For higher sump fluid temperatures, the containment pressure is assumed to equal the saturation pressure corresponding to the sump water temperature.

MUAP-08001-NP provides Figure 3-15, “NPSH vs. Temperature;” Figure 3-16, “Assumed Containment Head, Sump Vapor Pressure Head, and Head Losses;” and Figure 3-17, “NPSH Available vs. Time,” that plot NPSH and head loss inputs versus temperature and time consistent with RG 1.82 guidance. These figures demonstrate that the NPSHA exceeds the NPSHR at all times throughout the LOCA transient. Minimum NPSH margin occurs at the RWSP liquid saturation temperature corresponding to the initial containment pressure, assuming maximum pump flowrates, maximum strainer head loss, minimum RWSP level, and design basis NPSH required.

The staff's review of the applicant's NPSHA analysis is discussed in the next several paragraphs starting with containment pressure and addressing the associated draft guidance contained in SECY-11-0014, in particular, operation in the maximum erosion zone and uncertainties in NPSHR.

Containment Pressure

In accordance with SECY-11-0014, when calculating NPSHA, the inclusion of some or all of the pressure developed in the containment during an accident is referred to as containment accident pressure (CAP) credit. In **RAI 626-4750, Question 06.03-87**, the staff asked the applicant about the use of CAP in the NPSH evaluation. In its response to **RAI 626-4750, Question 06.03-87**, dated October 14, 2010, the applicant explains that the NPSHA containment pressure input is assumed to be equal to the initial containment pressure prior to the start of the accident, for sump liquid temperatures below the saturation temperature corresponding to this containment pressure. For RWSP liquid temperatures higher than this initial saturation pressure, the NPSHA containment pressure input is assumed to be equal to the sump fluid vapor pressure. For example, when the temperature of the RWSP water is at elevated temperatures, greater than 212 °F (100 °C), the RWSP water vapor pressure will be greater than the pressure in containment before the postulated accident. Using this approach, that is equating the containment pressure to the RWSP water vapor pressure (at elevated RWSP water temperatures), the applicant relies on containment pressure higher than that present before the postulated accident to provide sufficient NPSH and therefore, the staff finds that the applicant uses CAP credit.

SECY-11-0014 presents a technical discussion on the use of containment accident pressure, provides a regulatory history of this issue, and develops options to permit continued use of containment accident pressure. In staff requirements memorandum (SRM), SRM-SECY-11-0014, "Staff Requirements - SECY-11-0014 - Use of Containment Accident Pressure in Analyzing Emergency Core Cooling System and Containment Heat Removal System Pump Performance In Postulated Accidents," dated March 14, 2011, the Commission approved the staff-recommended option to continue permitting the use of containment accident pressure consistent with past staff practice and draft guidance contained in SECY-11-0014.

The staff finds the applicant's approach toward using CAP credit, that is, setting containment pressure equal to the vapor pressure corresponding to the RWSP temperature, is consistent with operating PWR NPSH analysis and the Commission direction identified in SRM-SECY-11-0014.

The draft guidance contained in SECY-11-0014 places emphasis on boiling water reactor (BWR) applications that credit CAP to support power uprate evaluations. The purpose for this guidance is to ensure that ECCS and containment heat removal pumps will perform their safety functions during postulated DBAs and certain postulated non-DBAs. This is accomplished by focusing on the capability of the pumps to perform their safety functions during postulated accidents. In particular, the guidance seeks to quantify both uncertainty and margin because the NPSH calculation could now credit all of the containment accident pressure. Past industry practice, accepted by the NRC staff, has been to perform conservative analyses for the DBA (e.g., the LOCA), and realistic analyses for the non-DBAs. While the conservative analyses produce bounding results, they do not provide a measure of the uncertainty or of the margin to a more expected result.

The conservative approach to calculating NPSH margin ordinarily assigns bounding values to the parameters used in the calculation of NPSH margin. These bounding values and assumptions are typically based on historically high or low values or on TS limiting conditions for operation. The chosen accident scenario is also limiting. For example, the worst pipe break (giving the most limiting NPSH margin) is assumed for the LOCA, and the worst single failure is assumed. It is also assumed that all these limiting conditions occur simultaneously. As discussed in SECY-11-0014, BWR licensees propose an alternate method to the conservative method for calculating NPSH margin, which uses a Monte Carlo calculation method that permits use of the total containment accident pressure. The US-APWR applicant is not proposing to use an alternate method and does not seek to credit the total containment pressure. Rather, the applicant considers that equating containment pressure to the vapor pressure of the RWSP, as discussed above, is a conservative approach.

The staff agrees with the applicant that equating containment pressure to the vapor pressure of the RWSP is reasonably conservative approach because the RWSP vapor pressure (based on RWSP liquid temperature) is not considered to exceed the total containment pressure. This is due to the dependency between RWSP liquid temperature and containment atmospheric conditions given that mass and energy are released from the break into a large, essentially leak-tight containment volume, no heat source is located within the RWSP, and non-condensable gas (air) is present in the containment atmosphere. The presence of non-condensable gas results in a containment vapor saturation pressure that is lower than the total pressure in containment. The lower vapor pressure provides for lower temperature condensed water in comparison to the total pressure. Treating this approach, that is equating containment pressure to the vapor pressure of the sump liquid, as a conservative approach is consistent with staff refinements being made to the SECY-11-0014 draft guidance and shared with the PWR owners group (see ADAMS Accession Number ML12125A235 and ML12107A076).

As documented in SECY-11-0014, the staff does not consider the issue of CAP credit as a generic safety matter. This is consistent with past staff practice and is supported by the following considerations discussed in the SECY-11-0014:

- No regulation restricts the use of CAP. Existing regulations, staff guidance and plant TS are intended to ensure that the containment is a low-leakage, robust structure, the integrity of which is demonstrated periodically (see Section 5.1 of Enclosure 1 to SECY-11-0014).

- The risk associated with CAP credit is small (see analysis in Chapter 19, “Probabilistic Risk Assessment and Severe Accident Evaluation,” of this evaluation)

- Adequate protection of public health and safety is provided when CAP credit is allowed.

In summary, given the discussion above, the staff finds the applicant’s approach to calculating the NPSHA regarding use of containment accident pressure is acceptable.

Maximum Erosion Zone

SECY-11-0014, Enclosure 1, "The Use of Containment Accident Pressure In Reactor Safety Analysis," states that one of the adverse effects of insufficient NPSH margin is cavitation, which results in erosion (pitting) of the surface of the impeller blades and possibly other parts of the pump from the condensation (implosion) of vapor bubbles near a solid surface. Pump tests indicate that the zone of maximum erosion rate lies between NPSH margin ratios of 1.2 to 1.6, however the time operating in the maximum erosion zone has not been correlated with the degree of damage. For the SECY-11-0014 paper, the staff selected a time limit of 100 hours for the time permitted in the maximum erosion zone. From MUAP-08001-NP, Figure 3-17, "NPSH vs. Time," it can be determined that the US-APWR NPSH margin ratio lies within the specified margin ratio well less than the estimated 100 hours, and therefore the staff conclude that US-APWR satisfies this preliminary guidance.

NPSH Required (NPSHR) Uncertainty

The uncertainties in NPSHR described in the draft guidance (SECY-11-0014) address the possibility that conditions during the pump NPSHR vendor tests could be different from the conditions during operation at the plant, effectively increasing the NPSHR values. The differences could arise due to pump speed, suction piping variation, dissolved gas evolution, etc. Based on an NRC SER that assessed NPSHR uncertainties under similar conditions (large, dry PWR), an average variation in NPSHR from +9 to +21 percent could be expected depending on the differences between the NPSHR test method and actual plant conditions (see ADAMS Accession Number ML112500207). The staff conservatively applied +21 percent to NPSHR as a bounding estimate for this review. When comparing NPSHR to NPSHA, the applicant's NPSH margin is over 21 percent for the CS/RHR pump and the SI pump (see MUAP-08001-NP Section 3.6.2.1.f, "NPSH Required;" Table 3-11, "NPSH available calculation data (CS/RHR Pump);" and Table 3-12, "NPSH available calculation data (SI Pump)"). Because the NPSH margin exceeds 21 percent, the staff finds the applicant's evaluation is consistent with the draft guidance discussed in SECY-11-0014.

Flashing

NPSH can also be impacted if head loss causes flashing of the sump fluid as a result of the fluid undergoing a differential pressure drop at the strainer. When the containment pressure is set equal to the vapor pressure of the sump fluid, the evaluation method to address sump fluid flashing is based on comparing the static head of water above the strainer, associated with the post-LOCA containment minimum water level, to the strainer head loss. If the static water head (strainer submergence) exceeds the calculated head loss across the strainer, then the static water head inhibits flashing. The applicant summarizes their flashing evaluation in MUAP-08001-NP Table 3-9, "Static Pressure at Strainer Outlet vs. Vapor Pressure." The applicant concludes that the strainer is designed with sufficient submergence to preclude the occurrence of two-phase flow (flashing) at the debris bed which may result in an unacceptable increase in strainer head losses. This conclusion is based on the conservative assumption that all the head loss occurs at the top of the strainer. In addition, initial dry air partial pressure will significantly contribute to prevent flashing once the sump water is subcooled. Based on the above discussion, the staff finds the applicant's approach and assessment of the potential for sump fluid flashing at the strainer is acceptable.

Deaeration

Air or gas voids can be generated downstream of the strainer surface as the result of dissolved gas coming out of solution within the sump fluid after undergoing a pressure drop across the debris bed on the strainer. Excessive deaeration resulting from the passage of flow through the debris bed could impact head loss and pump performance. Generation of air or gas voids through deaeration should be avoided by providing sufficient strainer submergence relative to the expected pressure drop. The applicant assessed deaeration in MUAP-08001-NP. For the US-APWR, at temperatures above 150 ºF (66 ^{o}C), strainer submergence exceeds the expected strainer head loss and therefore, strainer submergence is sufficient to prevent deaeration. At temperatures below 150 ºF (66 ^{o}C), the design basis strainer head loss exceeds strainer submergence, and therefore the potential for deaeration exists. The applicant's calculated void fraction at the sump strainer is very small (0.43 percent at 70 ºF (21 ^{o}C)) and air ingestion due to deaeration is not expected to adversely affect strainer or pump performance. RG 1.82 does provide a conservative method to account for the effects of air ingestion by increasing the NPSH required. The applicant applied RG 1.82 guidance, increasing the NPSH required to account for deaeration, and showed sufficient NPSH margin remained available. Based on the discussion above, the staff finds the applicant's deaeration evaluation acceptable.

Risk

SRP Section 6.2.2, "Containment Heat Removal Systems," Revision 5, issued March 2007, stipulates that if containment accident pressure is credited in determining NPSHA, an evaluation of the contribution to plant risk from inadequate containment pressure should be made. Given that the applicant does credit containment accident pressure in determining NPSHA, the applicant assessed the contribution to plant risk. DCD Tier 2, Subsection 6.2.2.3.7, "Net Positive Suction Head," indicates that the contribution to plant risk from using containment accident pressure is discussed in DCD Tier 2, Section 19.1.7, "PRA-Related Input to Other Programs and Processes". The applicant also discusses the contribution to plant risk in MUAP-08001-NP Appendix-F, Section F.3 "Contribution to Plant Risk." The applicant risk assessment, associated with loss of NPSH to the ECCS pumps, estimates the total core damage frequency (CDF) to be approximately two orders of magnitude less than the CDF described in Chapter 19 of the DCD. The staff's evaluation of the applicant's risk assessment can be found in Chapter 19 of this SE.

In summary, given the discussion above, the staff finds the applicant's approach to calculating the NPSHA regarding use of containment accident pressure is acceptable, and **RAI 626-4750, Question 06.03-87, is resolved and closed**. The staff also concludes that the applicant meets the staff's guidance on NPSHR uncertainty and operation in maximum erosion zone. Additional aspects of NPSH are assessed in Section 6.3 of this SER. Based on the overall discussion presented in Section 6.2.2, to include inputs from strainer head loss (see Subsection 6.2.2.4.9 of this SER) and RWSP minimum level (see Subsection 6.2.2.4.11 of this SER) and SER Section 6.3, the staff finds that the applicant has adequate NPSH provided to the CS/RHR and SI pumps. The staff opened **confirmatory item CI-SRP06.02.02-2** to verify that the final design document incorporates the associated supplemental DCD information provided by GTR2.

6.2.2.4.11 Upstream Effects

The purpose of the upstream effects review is to evaluate potential hold up volumes, choke points, and other physical obstructions that could prevent water from draining to the sump. Water contained in hold up volumes would not be available in the RWSP to provide strainer coverage and would result in a reduction of available net positive suction head by impacting the RWSP minimum water level analysis. Section 7.2 of NEI 04-07 and the associated NRC SE provide guidance to be considered in the upstream effects evaluation. Note, the DCD Sections, DCD figures, and DCD tables referred to in this evaluation section are as supplemented by GTR2.

The applicant describes upstream effects in DCD Tier 2, Subsection 6.2.2.3.11, “Upstream Effect.” The applicant describes the RWSP function in DCD Tier 2, Subsection 6.2.2.2.5, “Refueling Water Storage Pit.” DCD Figure 6.2.1-16, “RWSP Upper and Lower Plan View at Elevation 25ft.-3in.,” and Figure 6.2.1-17, “RWSP Panoramic Sectional View,” present the plan and sectional view of the RWSP, respectively. ECC/CS post-accident flow paths and potential ineffective pools (hold-up volumes) within the containment are shown in DCD Figure 6.2.1-9, “Outline of the Post-LOCA Recirculation Pathways.” DCD Figure 6.2.1-10, “Volume of Ineffective Water,” shows the volume of ineffective pools. DCD Table 6.2.1-3, “RWSP Design Features,” summarizes the RWSP volumes used to calculate the minimum liquid volume. Upstream effects are also discussed in MUAP-08001-NP, Section 3.7 “Upstream Effect.”

In DCD Subsection 6.2.1.1.2, “Design Features,” the applicant discusses that containment drainage flows through several large floor openings in the SG compartments to the reactor cavity, header compartment, and C/V drain pump room. Containment drainage also flows from the refueling cavity, through piping, to the header compartment. DCD Figures 6.2.1-12, “Transfer Piping,” and 6.2.1-16, “RWSP Upper and Lower Plan View at Elevation 25 ft. - 3 in.,” show the pipes that transfer water from the reactor cavity and header compartment to the RWSP. These pipes are protected from large debris by debris interceptors, as shown in DCD Figure 6.2.1-14, “Debris Interceptor.”

In DCD Tier 2, Subsection 6.2.2.3.11, the applicant identified the overflow pipes and refueling cavity drains as two possible choke points. Besides the overflow pipes and refueling cavity drains, the applicant assumes no other drains or narrow pathways provide make-up flow to the RWSP. Floor drain piping which collects in the containment normal sump, such as the SG compartment floor and operating floor, is assumed to become blocked. Containment spray water is drained to lower containment levels by way of stairway openings, equipment hatch, or compartment access openings. The applicant does not consider these openings to be narrow pathways vulnerable to blockage.

In the refueling cavity, there is 8 inch (20.32 cm) drain piping that routes water to the header compartment. In **RAI 354-2585, Question 06.02.02-24**, the staff asked the applicant about the potential for debris to clog the refueling cavity drain piping. In its response to **RAI 354-2585, Question 06.02.02-24**, dated July 27, 2012, the applicant proposed to revise the design to include debris interceptors (grating) in the upper core internal laydown pit that is situated directly above the drain piping (e.g., see revised DCD Figure 6.2.1-9 provided in GTR2) to prevent large debris from reaching the refueling cavity drain piping and blocking flow. Given the discussion above, the staff finds that the installation of a debris interceptor is an acceptable approach to prevent large debris from reaching the refueling cavity drain piping, causing blockage, and the applicant’s response to **RAI 354-2585, Question 06.02.02-24** is acceptable.

The reactor cavity and header compartment receive containment drainage from the floor openings in the SG compartments. Debris interceptors are installed at the floor openings or within the compartment to prevent large debris from reaching the overflow piping installed within these compartments. The debris interceptors opening size is 8 inches by 8 inches (20.32 cm x 20.32 cm), which is smaller than the overflow pipe diameter (12 inches (30.48 cm)) in order to prevent blockage of the overflow piping. The staff finds that the installation of debris interceptors is an acceptable approach to prevent large debris from reaching and potentially blocking the overflow piping.

Given the debris source term, debris interceptors, drain pipe sizing, containment layout and multiple drain paths, the staff agrees with the applicant's assessment that blockage of flow paths is unlikely to occur.

The US-APWR categorizes hold-up volumes into two groups, "Return water on the way to the RWSP" and, "Ineffective pools." The calculated values describing these two groups are listed in DCD Table 6.2.1-3, "RWSP Design Features." Additional detail can be found in MUAP-08001-NP Figure 3-11, "Minimum Water Level of the RWSP," and Table 3-10, "Upstream Effect on Hold-up Volumes."

For "Return water on the way to the RWSP," the applicant accounts for containment spray water droplets, saturated steam in the atmosphere, condensate on containment surfaces, and the water stream collected on containment floors, such as the SG compartment floor and the refueling cavity floor. Examples of "ineffective pools" for the US-APWR include the reactor cavity, header compartment, and reactor coolant drain pump room.

According to DCD Table 6.2.1-3, the RWSP normal liquid volume is 79,920 ft^3 (597,800 gallons) (2,263 m^3) with a calculated minimum liquid volume of 18,340 ft^3 (137,200 gallons) (519.3 m^3). The calculated minimum liquid volume in the RWSP is determined by subtracting the return water and hold-up volume (ineffective pool) from the initial water volume in the RWSP. The combined return water and ineffective pools account for approximately 61,580 ft^3 (460,700 gallons) (1,744 m^3), or roughly 75 percent of the initial RWSP volume. The minimum liquid volume remaining in the RWSP is used to assess strainer submergence and the static water pressure (height of water) input to the safety pumps NPSH evaluation.

As a result of an April 7, 2011, audit that reviewed the applicant's water hold-up analysis, the staff issued **RAI 740-5719, Question 06.02.02-64**, requesting the applicant provide information about the treatment of NaTB baskets and associated drain piping in the water hold-up analysis. In addition, the staff asked about dynamic water retention on floors, especially on the 25-ft 3-in. (7.7-m) floor elevation, given that this floor contains curbs around multiple floor openings that route water back to the RWSP. In its response to **RAI 740-5719, Question 06.02.02-64**, dated May 30, 2012, the applicant revised the hold-up volume analysis to add the NaTB basket and piping volumes and provided a justification for dynamic water retention. The dynamic water retention analysis (water depth on floor) is dependent on the curb dimensions and flow rate over the curb, where the water depth increases with a reduction in curb length and higher flow rates. The applicant's analysis reduced the curb length by 20 percent, to account for potential blockage, resulting in an increase to the floors water depth. In addition, the applicant assumed a maximum flow rate (all trains of safety injection and all trains of containment spray in operation) to maximize the floors water depth. The justification for water retention also referred to **RAI 839-6103, Question 06.02.02-73,** which is discussed below. The impact of the response to **RAI 740-5719, Question 06.02.02-64**, is reflected in DCD Table 6.2.1-3 and MUAP-08001-NP Table 3-10, "Upstream Effect Hold-up Volumes." The staff finds the applicant's response to **RAI 740-5719, Question 06.02.02-64,** to be acceptable because the applicant revised the total hold-up volume to account for the NaTB baskets and piping and the applicant provided justification for the approach to dynamic water retention which maximized the floors water depth. Additionally, MUAP-08001-NP, Revision 7, reflects the changes discussed in response to **RAI 740-5719, Question 06.02.02-64**.

In MUAP-08001-NP, Figure 3-11, “NPSH available Calculation Data (CS/RHR Pump),” the applicant lists the margin between the calculated minimum RWSP liquid volume and the design basis RWSP minimum liquid volume. In **RAI 839-6103, Question 06.02.02-70**, based on the relatively small volume margin, the staff requested additional information on the applicant’s approach to calculating the minimum RWSP liquid volume. In its response to **RAI 839-6103, Question 06.02.02-70**, dated May 30, 2012, the applicant provided a summary of the hold-up volumes along with the assumptions made to maximize hold-up and minimize the RWSP liquid volume. Examples of the applicants’ assumptions used to assess water hold-up and minimize the RWSP liquid volume are as follows:

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The staff finds that the applicant’s response to **RAI 839-6103, Question 06.02.02-70, is acceptable** because the applicant provided additional information that demonstrated a reasonable approach to calculating hold-up volumes and determining the minimum liquid RWSP volume. The impact of this response is reflected in a revised DCD Table 6.2.1-3 as shown in GTR2 and MUAP-08001-NP, Table 3-10.

In **RAI 839-6103, Question 06.02.02-71**, the staff asked that the applicant justify the approach used to determine water height above the overflow piping that returns water to the RWSP. In its response to **RAI 839-6103 Question 06.02.02-71**, dated May 30, 2012, the applicant provided a basis for their approach and assessed the sensitivity of the calculated water height. The applicant referenced the United States Department of the Interior, Bureau of Reclamation, “Design of Small Dams,” Third Edition, dated 1987, drop inlet spillway method to calculate the height of water above the overflow pipe entrance. [

] The staff finds the applicant’s response to **RAI 839-6103, Question 06.02.02-71** to be acceptable given the reasonableness of the approach to calculating the height of water above the overflow piping and the limited sensitivity that the height of water has on the hold-up volume margin. The impact of this response is reflected in a revised DCD Table 6.2.1-3, “RWSP Design Features,” as shown in GTR2 and MUAP-08001-NP, Table 3-10, “Upstream Effect Hold-up Volumes.”

In **RAI 839-6103, Question 06.02.02-73**, the staff asked that the applicant justify the approach used to determine the water height on the containment floor at elevation 25 ft 3 in. (7.7 m) outside the SG compartment, that is, located between the secondary shield wall and containment wall. In its response to **RAI 839-6103, Question 06.02.02-73**, dated July 27, 2012, the applicant considered the losses due to water flowing from outside the SG compartment into the SG compartment via the four labyrinth access openings. The resultant water depth outside the SG compartment was greater than the water depth inside the SG compartment. To assess the sensitivity of the selected approach, the applicant [

] as discussed above in response to **RAI 839-6103, Question 06.02.02-71**. The staff finds the applicant's response to **RAI 839-6103, Question 06.02.02-73,** to be acceptable given the reasonableness of the approach to calculating the height of water above the 25 ft 3 in. (7.7 m) floor elevation outside the SG compartment and the sensitivity that the height of water has on the available hold-up margin. The impact of this response is reflected in a revised DCD Table 6.2.1-3 as shown in GTR2 and MUAP-08001-NP, Table 3-10.

The staff finds that the applicant's evaluation of potential hold up volumes, choke points, and other physical obstructions that could prevent water from draining to the sump is consistent with NEI 04-07 and associated SE guidance. Given the discussion above, the staff finds that the applicant's assessment of upstream effects and RWSP minimum water volume analysis is acceptable. In addition, the staff finds that the RWSP level assumed in the NPSH evaluation is acceptable, given the positive margin between the calculated RWSP minimum level and the design basis RWSP level used in the NPSH analysis. The staff opened **Confirmatory Item CI-SRP06.02.02-2,** under which**, RAI Questions 06.02.02-24, 06.02.02-64, 06.02.02-70, 06.02.02-71 and 06.02.02-73 are being tracked** to verify that the final design document incorporates the associated supplemental DCD information provided by GTR2.

6.2.2.4.12 Debris Interceptor/Strainer Structural Analysis

The debris interceptor and strainer module structural evaluation is found in Section 3 of this SER, "Design of Structures, Systems, Components, and Equipment."

6.2.2.4.13 Downstream Effects – Ex-Vessel Components and Systems

The term "downstream ex-vessel effects" refers to effects of post-LOCA debris on the systems and components in the ECCS and CSS flow path located downstream of the recirculation sump strainer. Debris may be carried downstream of the ECCS strainer during post-LOCA operation, thus causing blockage or wear and abrasion in system components. Areas of concern for downstream ex-vessel components include (1) blockage of system flowpaths at narrow flow passages (e.g., containment spray nozzles, pump internal flow passages, and tight-clearance valves), (2) wear and abrasion of surfaces (e.g., pump running surfaces) and heat exchanger tubes and orifices. US-APWR DCD Tier 2, Subsections 6.2.2.1.4, "Reliability Design Basis," and 6.3.2.5, "System Reliability," as supplemented by GTR2, reference Technical Report MUAP-08013-P, "US-APWR Sump Strainer Downstream Effects," for the evaluation of ECCS and CSS systems and their components to operate under post-LOCA conditions. DCD Tier 2, Subsection 6.2.2.3.12, "Downstream Effects – Ex-Vessel," as supplemented by GTR2, summarizes the applicant's downstream effects ex-vessel evaluation. The staff opened **CI-SRP06.02.02-2** to verify that the final design document incorporates the associated supplemental DCD information, discussed above, as provided by GTR2.

The staff reviewed the applicant's evaluation of ex-vessel downstream effects in Technical Report MUAP-08013-P Revision 4, issued August 2012, to provide reasonable assurance that the ECCS and CSS and their associated components will function as designed under post-LOCA conditions for the required mission time. MUAP-08013-P Section 3.0, "Downstream Effects," contains the following subsections:

- MUAP-08013-P, Section 3.1 “System Descriptions (ECCS and CSS).”
- MUAP-08013-P, Section 3.2 “Design Inputs/Evaluation Assumptions.”
- MUAP-08013-P, Section 3.3 “Piping, Valve and Heat Exchanger Evaluations.”
- MUAP-08013-P, Section 3.4 “ECCS and CSS Pump Evaluations.”
- MUAP-08013-P, Section 3.5 “ECCS and CSS Performance Evaluations.”
- MUAP-08013-P, Section 3.6 “Regulatory Summary.”
- MUAP-08013-P, Section 3.7 “Confirmation Items.”
- MUAP-08013-P, Section 3.8 “Summary of Results.”

System Descriptions (ECCS and CSS)

Technical Report MUAP-08013-P, Section 3.1, “System Descriptions,” provides a general description of the ECCS and CSS.

Design Inputs/Evaluation Assumptions

Technical Report MUAP-08013-P, Section 3.2.1, “LOCA Scenarios,” addresses ECCS and CSS operation under small-break LOCA (SBLOCA) and large-break LOCA (LBLOCA) conditions. For the purpose of downstream ex-vessel effects, the SBLOCA is bounded by the LBLOCA. The ECCS flows occurring during a SBLOCA are considerably smaller than flows occurring during a LBLOCA. Also, the debris source term is expected to be much smaller during a SBLOCA. Therefore, the LBLOCA conditions are used to evaluate the effects of post-LOCA debris on the systems and components in the ECCS and CSS flow path located downstream of the recirculation sump strainer.

Technical Report MUAP-08013-P, Section 3.2.2, “Mission Time,” defines mission time as the period for which a system, structure or component (SSC) is required or credited in performing its safety related function. The post-LOCA mission time for the ECCS and CSS is defined as 30 days.

Technical Report MUAP-08013-P, Section 3.2.3, “Component List,” states that Table 3.2-1, “Components in the Flow Path during a LBLOCA,” lists all components and flow paths within the scope of the downstream evaluation(s).

Technical Report MUAP-08013-P, Section 3.2.4, “Post-LOCA Fluid Constituents,” describes the total quantity of debris generated during a LBLOCA in Table 3.2-3, “Debris Source Term,” and the amount of debris that bypasses the containment sump strainer in Table 3.2-4, “Debris Concentration Components.” These tables are derived from the US-APWR design basis debris generation identified in Table 3-4, “Debris Generation,” of Technical Report MUAP-08001-NP, “US-APWR Sump Strainer Performance.” The nominal diameter of the sump strainer holes is equal or less than 0.066 in (0.17 cm). The analysis for ex-vessel downstream effects assume 100-percent latent debris bypass (fiber and particulate), 100-percent epoxy coatings, 50-percent generated fiber bypass (Nukon), and 5-percent RMI bypass through the containment sump strainers. The bypass debris of 50 percent of the generated fiber (Nukon) is based on testing results documented in MUAP-08013-P, Appendix J.

In **RAI 366-2740, Question 06.02.02-45,** the staff requested the applicant to describe further the basis for the RMI bypass listed Table 3.2-3 of Technical Report MUAP-08013-P. In its response to RAI 366-2740, Question 06.02.02-45, dated June 11, 2009, the applicant stated the following with respect to RMI bypass:

> The ex-vessel downstream effects evaluation assumes that five percent of the reflective metal insulation within the ZOI bypass is reduced to fines smaller than 110 percent of the sump screen opening (0.066 in, 0.17 cm) and that 100 percent of this amount stays in solution and passes through the sump screen. No credit is assumed for settling or capture by the sump screen. This is conservative in that some RMI may not transport, some debris may settle due to low velocities in the RWSP and in the reactor vessel, some debris may be caught on components and structures within containment, and some debris will accumulate on the sump screen. RMI typically fails as metal pieces not as fines. The RMI does not readily transport. The conservative assumption of five percent fines bypass was chosen to bound all current studies on RMI and RMI transport.

The staff finds that the applicant's response adequately described the basis for RMI bypass listed Table 3.2-3 of Technical Report MUAP-08013-P. Therefore, **RAI 366-2740, Question 06.02.02-45, is resolved and closed**.

In **RAI 840-6096, Question 06.02.02-74,** the staff requested the applicant to describe further the characteristic size for debris listed Table 3.2-3 of Technical Report MUAP-08013-P and to explain how the sizes were determined. In its response to RAI 840-6096, Question 06.02.02-74, dated November 22, 2010, the applicant stated the following:

> The characteristic debris sizes in Table 3.2-3 of MUAP-08013 (R2) are taken from Table 3-5 of MUAP-08001 (R5), which provides additional discussion. These are characteristic sizes of the debris (e.g., break-generated or latent debris) reaching the RWSP and are derived from NRC testing and NRC guidance. For the plugging evaluation, the maximum size of debris which could pass through the strainer is compared to the downstream orifice sizes. This maximum size conservatively bounds the characteristic debris sizes. Per the NRC SE on TR-WCAP-16406-P (ADAMS Accession Number ML073520295), the maximum length of deformable particulates that may pass through the strainer is two times (2x) the maximum linear dimension of the penetration in the sump screen (0.066-in). For non-deformable particulates, the maximum size is the maximum linear dimension of the penetration in the sump screen.

The staff finds that the applicant's response adequately described the characteristic size for debris listed in Table 3.2-3 of Technical Report MUAP-08013-P. Therefore, **RAI 840-6096, Question 06.02.02-74, is resolved and closed**.

By **RAI 840-6096, Question 06.02.02-75**, the staff requested the applicant to describe further Table 3.2-4 of Technical Report MUAP-08013-P by clarifying that this is the amount of debris that will bypass the strainers and be ingested into the ECCS. The staff also requested the applicant to revise the table to include the debris concentration in PPM and specify the volume of water used in determining the PPM concentration. In its response to RAI 840-6096, Question 06.02.02-75, dated November 22, 2010, the applicant stated the following:

> Table 3.2-4 defines the amount of debris assumed to bypass the strainers and is suspended in the post-LOCA fluid. MHI will revise Table 3.2-4 of MUAP-08013-P to include a PPM column for debris PPM concentration. Table 3.1-1 of MUAP-08006-NP, Revision 1, "US-APWR Sump Debris Chemical Effects Test Plan," was used as the design input for the PPM calculations. The mass of water used

for the PPM calculation was the sum of the minimum post-LOCA recirculation sump water volume (43,930 ft^3 at the long-term sump water temperature of 149°F) and the RCS water volume (699,000 lbm).

The applicant's response states that Table 3.2-4 of Technical Report MUAP-08013-P defines the amount of debris assumed to bypass the strainer and is suspended in the post-LOCA fluid. The applicant also revised Table 3.2-4 to include debris concentrations in PPM and described the water volumes used in determining the PPM concentrations. The applicant's methodology for determining the PPM concentration of post-LOCA debris is consistent with the methodology approved in NRC SE on WCAP-16406-P. The staff finds the applicant's response provides an acceptable description of the debris concentrations in PPM and the methodology used to determine the debris concentration. Table 3.2-4 of MUAP-08013-P was revised as indicated in the RAI response. Therefore, **RAI 840-6096, Question 06.02.02-75, is resolved and closed.**

Technical Report MUAP-08013-P, Section 3.2.5, "ECCS and CSS Flows and Flow Velocities," describes the range of system flow and local velocities expected within the ECCS and CSS piping components in Table 3.2-6, "Affected Equipment / Flow Rates," based on LBLOCA conditions. SBLOCA conditions are bounded by LBLOCA due to the higher flows creating more wear and generating a greater debris load. The flow velocities in Table 3.2-6 are based on the shutoff and runout flow conditions for the Safety Injection and CS/RHR pumps.

Technical Report MUAP-08013-P, Section 3.2.6, "Summary of Analysis Conservatisms," summarizes conservative assumptions for the downstream effects evaluation. The assumptions and conservatisms are listed below:

- Section 3.2.6.1 states SI pump design range of flow is 265 gpm (1003 l/min) at shutoff and 1540 gpm (5830 l/min) at runout. These parameters bound the settling velocities and wear rate evaluations.

- Section 3.2.6.2 states CS/RHR pump design range of flow is 355 gpm (1344 l/min) at shutoff and 3650 gpm (13817 l/min) at runout. These parameters bound the settling velocities and wear rate evaluations.

- Section 3.2.6.3 states wear is calculated from "time zero", i.e., start of the event. Worst-case fluid properties are assumed to be present. This assumption is conservative since it does not credit debris transport or the slow increase of fluid properties due to long term mixing.

- Section 3.2.6.4 states fluid velocity through a single CS/RHR HX tube is assumed to be 15 ft/s (4.6 m/s). A nominal design and operating heat exchanger velocity range is 3 to 10 ft/s (0.91 to 3.05 m/s). Therefore the use of 15 ft/s is conservative from a HX design perspective and bounds the HX design and procurement specifications.

- Section 3.2.6.5 states that this analysis assumes 100-percent latent debris bypass, 50-percent fiber bypass, and 5-percent RMI bypass through the containment sump strainers.

In **RAI 840-6096, Question 06.02.02-76,** the staff requested that the applicant describe how the assumption that fluid velocity of 15 ft/s through a single CS/RHR HX tube is conservative, based on the fact that the CS/RHR HX is not yet designed. In its response to RAI 840-6096, Question 06.02.02-76, dated November 22, 2010, the applicant stated the following:

> The design engineer will specify the operating conditions to which the heat exchanger manufacturer must design, including normal, minimum and maximum expected operating conditions. The vendor will confirm that the heat exchanger is designed in accordance with the purchase specifications, and provide a Heat Exchanger Data Sheet for the final as-designed heat exchanger.
> MHI will verify that the as-designed normal flow velocity through the CS/RHR heat exchanger is less than 15 ft/sec. This is part of the normal procurement acceptance review process for plant equipment and will be part of the Heat Exchanger Data Sheet review for every plant heat exchanger.

The staff finds the applicant's response acceptable because it states that the as-designed normal flow velocity (less than 15 ft/sec (4.6 m/s)) through the CS/RHR HX will be included in the vendor purchase specifications. The applicant also stated that this flow rate will be documented in the HX Data Sheet. Therefore, **RAI 840-6096, Question 06.02.02-76, is resolved and closed**.

Piping, Valve and Heat Exchanger Evaluations (Wear and Blockage)

Technical Report MUAP-08013-P, Section 3.3, "Piping, Valve and Heat Exchanger Evaluations," describes the applicant's evaluation of ECCS and CSS piping, valves and heat HXs with respect to wear and blockage for post-LOCA fluid conditions.

Technical Report MUAP-08013-P, Section 3.3.1, "Wear Rate Evaluation Summary," describes the applicant's evaluation of wear for orifices, spray nozzles and piping in the ECCS and CSS. Using runout flow rates and bypass debris abrasiveness, the piping and component wear was calculated for a 30-day duration. The results of this evaluation are summarized in Table 3.3-1, "ECCS and CSS Components Wear vs. Time," which lists the diametrical wear and associated flow rate increase for the components. As part of the ECCS and CSS performance evaluation in Section 3.5 of MUAP-08013-P, the applicant concluded that orifices, spray nozzles and piping wear will not affect system performance during the course of a LOCA.

In **RAI 840-6096, Question 06.02.02-77**, the staff requested the applicant to describe further the wear rate evaluation for piping and orifice. In its response to RAI 840-6096, Question 06.02.02-77, dated November 22, 2010, the applicant stated the following:

> The wear rate equation and how it relates to debris abrasiveness and debris concentration are contained in the methodology section of the wear rate calculation. Note that it is consistent with industry methods and those used at all current US PWRs. The wear rate increases with debris concentration and debris abrasiveness. The calculated wear rate is based on the debris concentrations discussed in MUAP-08013, Table 3.2-4. The debris abrasiveness is based on using wear rate data for coarse sand. Silica sand generally has a Brinell hardness number (i.e., abrasiveness) ranging from approximately 500 to 700. This is larger than that expected for the majority of post-LOCA debris (i.e., reflective metal insulation), based on the Brinell hardness numbers given for stainless steel in MUAP-08013.

The staff finds the applicant's response acceptable as it further described the methodology used to evaluate the wear rate for piping and orifice. Therefore, **RAI 840-6096, Question 06.02.02-77, is resolved and closed**.

Technical Report MUAP-08013-P, Section 3.3.2, "Heat Exchanger Evaluation," describes the applicant's evaluation of the CS/RHR HX for plugging, performance and wear.

In **RAI 840-6096, Question 06.02.02-78**, the staff requested the applicant to discuss whether the HX equipment specification will identify both the post-LOCA debris conditions and the normal operating fluid conditions, and whether the vendor will be required to evaluate the HX performance and wear for both conditions. In its response to **RAI 840-6096, Question 06.02.02-78**, dated November 22, 2010, the applicant stated the following:

> The specified design conditions for the CS/RHR heat exchanger will include normal, minimum and maximum (i.e., post-LOCA) expected operating conditions. Operating conditions include the fluid constituents for each operating case. The heat exchanger vendor will be required to confirm compliance with the specification.

The staff finds the applicant's response acceptable as it explained that the specification will identify both the post-LOCA debris conditions and the normal operating fluid conditions and require the vendor to evaluate the HX performance and wear for both conditions. Therefore, **RAI 840-6096, Question 06.02.02-78, is resolved and closed**.

Technical Report MUAP-08013-P, Section 3.3.3, "Valve Wear Evaluation," describes the applicant's evaluation for valve wear. Valve and valve trim materials are specified to be wear resistant. Material hardness data for valves and piping is listed in Table 3.2-2, "Material Hardness Data," of MUAP-08013-P. The valve procurement specification will note the constituents of the post-LOCA fluid and require that the valve be able to operate reliably under those conditions for a minimum of 30 days.

In **RAI 840-6096, Question 06.02.02-79**, the staff requested the applicant to describe the valve procurement specifications and the valve qualification methodology. In its response to RAI 840-6096, Question 06.02.02-79, dated November 22, 2010, the applicant stated the following:

> The valve procurement specification will contain its service conditions. It will include details such as the type and quantity of post-LOCA debris and the quality and type of post-LOCA fluid. These are standard (and required) valve specifications and design inputs.
>
> RG 1.100, Revision 3, Seismic Qualification of Electrical and Active Mechanical Equipment and Functional Qualification of Active Mechanical Equipment for Nuclear Power Plants, is applicable to the valve design. Valves will be qualified to QME-1-2007 and they will be designed to operate under normal operating and post-LOCA conditions.
>
> MHI will revise Section 3.3.3 of MUAP-08013-P (R2) to state that valves are to be qualified per QME-1-2007 as endorsed by RG 1.100, Revision 3, for their intended function using the post-LOCA fluid.

The staff finds the applicant's response acceptable as it further described valve procurement specifications will include debris type and quantity, and the valve qualification methodology will be performed in accordance with QME-1-2007. Section 3.3.3 of MUAP-08013-P was revised as indicated in the RAI response. Therefore, **RAI 840-6096, Question 06.02.02-79, is resolved and closed**.

Technical Report MUAP-08013-P, Section 3.3.4, "Piping and Valve Blockage and Debris Settling Evaluation," describes the applicant's evaluation for blockage and debris settling in piping, valves, orifice, and spray nozzles. Piping, valve, orifice, and spray nozzle diameters are significantly larger than the strainer hole size of 0.066 inch (0.17 cm), and flow velocities are above the settling velocities of the post-LOCA fluid. Globe valves used for throttling are expected to be throttled to a minimum of one-in. opening between the valve disc and seat. The one-in. valve opening is considerably larger than any expected particle passing through the sump strainer. Therefore, blockage and settle of debris is not expected to occur in piping, valves, orifices, and spray nozzles. The staff notes the maximum settling velocities identified in Table 3.3-2 of MUAP-08013-P is acceptable based on the staff evaluation of settling velocities in Section 4.2.4 of SE NEI 04-07, "PWR Sump Performance Evaluation Methodology," U.S. Nuclear Regulatory Commission, Washington, DC, December 2004 (ADAMS Accession Number ML050550156).

In **RAI 366-2740, Question 06.02.02-48**, the staff requested the applicant to provide additional information to clarify if ECCS and CSS fluid flow velocities could be less than the velocities required to prevent settling of suspended debris in the downstream flow path, such as during system flow initiation or realignment. In its response to **RAI 366-2740, Question 06.02.02-48,** dated June 11, 2009, the applicant stated that the assumed flow rates are conservatively chosen, and the ECCS and CSS operation does not interrupt flow during any system realignment. The staff finds the applicant's response acceptable because it states that the assumed flow rates are conservatively chosen, and the ECCS and CSS operation does not interrupt flow during any system realignment. Therefore, **RAI 366-2740, Question 06.02.02-48, is resolved and closed**.

The US-APWR credits the CSS for containment pressure suppression and fission product removal. The MUAP-08013-P, Section 3.3.4 report states that the potential for CSS spray nozzle plugging by debris is low. However, the performance of the spray nozzles in accomplishing their necessary safety functions may be affected by changes to the CSS fluid physical or chemical properties, even though the flow rate through the nozzles is not restricted. In **RAI 366-2740, Question 06.02.02-51**, the staff requested the applicant to provide an evaluation regarding the effects of entrained debris, chemicals, and gases on the performance of the CSS spray nozzles, especially regarding the effects on spray droplet size distribution, for containment pressure suppression and removal of fission products from the containment atmosphere. In its response to **RAI 366-2740, Question 06.02.02-51,** dated June 11, 2009, the applicant stated that 0.375 inch (9.5 mm) spray nozzle orifice will wear and open to 0.378 inch (9.6 mm) after 24 hours and have minimal effect on spray function. The applicant stated that entrained gas would remain entrained, and there would be increased flow rate through the nozzles. The applicant also stated that the small amount of entrained debris particles are minute in size and will have a minimal effect on spray function. The applicant further stated that chemicals and precipitates are typically soft, non-abrasive, low-shear, and readily stay in solution due to the fully developed turbulent flow conditions with the piping system. The staff finds this acceptable, since there is significant margin in the spray flow and droplet parameters affecting both the condensation heat transfer and fission product removal functions of the containment spray. Therefore, **RAI 366-2740, Question 06.02.02-51 is resolved and closed**.

Technical Report MUAP-08013-P, Section 3.3.5, “Instrument Clogging Evaluation,” describes the evaluation for potential the clogging of instrument tubing. All connections, by design, are either at the horizontal or above. Flow velocities in all cases are above the settling velocities of the post- LOCA fluid as stated in Technical Report MUAP-08013-P Table 3.3-2. Therefore, the potential for instrument and instrumentation tubing plugging is very low.

Technical Report MUAP-08013-P, Sections 3.3.6 and 3.4.4, “Chemical Effect Evaluation,” state that precipitants and other chemical forms present as a result of the chemical effects testing have no effect on the plugging or wear evaluations. Chemicals and precipitants are typically soft, non-abrasive, low-shear and readily stay in solution due to the fully developed turbulent flow conditions present within the piping system(s). As such, they do not contribute to plugging or change wear characteristics of piping, pump, heat exchangers or valves downstream of the containment sump. [The NRC staff notes that the US-APWR evaluation that precipitants and other chemical forms have no effect on plugging or wear of downstream ex-vessel components is acceptable and consistent with staff positions documented in NRC Memorandum, ”Basis for Excluding Chemical Effects Phenomenon from WCAP-16406-P Ex-vessel Downstream Evaluations,” dated January 21, 2010 (ADAMS Accession Number ML093160100) and Technical Report, “Evaluation of Chemical Effects Phenomena Identification and Ranking Table Results”, dated March 2011 (ADAMS Accession Number ML102280594).]

ECCS and CSS Pump Evaluations

Technical Report MUAP-08013-P, Section 3.4.1.1, “ECCS Pumps,” and Section 3.4.1.2, “CSS Pumps,” describe the design and qualification criteria for the ECCS and CSS pumps for operation with post-LOCA debris fluids for a 30 day mission time. The post-LOCA fluids include the debris that bypasses the containment sump strainer identified in Table 3.2-4 of Technical Report MUAP-08013-P.

Technical Report MUAP-08013-P, Section 3.4.1.1 and 3.4.1.2 states that the ECCS and CSS pumps will be specified to meet the intent of American Petroleum Institute (API) Standard 610 in the context of rotor dynamics. API-610 provides a standard analysis method for severe service pumps and a recognized design standard for the design and operation of centrifugal pumps for petroleum, heavy duty chemical and gas industry services. Details will be provided in the procurement specifications.

In **RAI 840-6096, Question 06.02.02-80,** the staff requested the applicant to describe examples of API-610 details to be provided in the procurement specifications relating to ECCS and CSS pump design. In its response to RAI 840-6096, Question 06.02.02-80, dated November 22, 2011, the applicant stated the following:

> Examples of API-610 details that may be provided in procurement specifications include those contained in:
>
> - Annex H Materials and material specifications for pump parts.
> - Annex I Lateral analysis.
> - Annex J Determination of residual unbalance.
> - Annex L Vendor drawing and data requirements.
> - Annex M Test data summary.
> - Annex N Pump datasheets.

The NRC staff finds the use of API-610 as a guide for the design of ECCS and CSS pumps to be acceptable because API-610 is an industry standard for pump design. However, the staff notes that the ECCS and CSS pumps will meet the design requirements of ASME Code Section III, the qualification requirements of QME-1-2007, and the Inservice Testing requirements of ASME Code for Operation and Maintenance of Nuclear Power Plants (OM Code).

In **RAI 840-6096, Question 06.02.02-80,** the staff also requested the applicant to describe the qualification methodology for the ECCS and CSS pumps in post-LOCA fluids. In its response to RAI 840-6096, Question 06.02.02-80, dated November 22, 2011, the applicant stated the following:

> RG 1.100, Revision 3, Seismic Qualification of Electrical and Active Mechanical Equipment and Functional Qualification of Active Mechanical Equipment for Nuclear Power Plants, is applicable to the pump design. The pumps will be qualified to QME-1-2007 and they will be designed to operate under normal operating and post-LOCA conditions.
>
> MHI will revise Sections 3.4.1.1 and 3.4.1.2 of MUAP-08013-P to state that the pumps, including the mechanical seal, shall be qualified per QME-1-2007 as endorsed by RG 1.100, Revision 3 to operate in post-LOCA fluids for at least 30 days.

QME-1-2007 as endorsed by RG 1.100, Revision 3, is the staff approved methodology for the qualification of ECCS and CSS pumps and is therefore acceptable. Sections 3.4.1.1, “ECCS Pumps,” and 3.4.1.2, “CSS Pumps,” of MUAP-08013-P were revised as indicated in the RAI response.

Technical Report MUAP-08013-P, Sections 3.4.1.1 and 3.4.1.2 state that the fluid characteristics listed in Tables 3.2-3, 4 and 5 conservatively represent the post-LOCA fluid conditions that ECCS and CSS pumps will experience. At the procurement stage, the pump vendor will provide a table listing the material and hardness’s of all wetted pump surfaces (wear rings, pump internals, bearing, casing, etc.).

In **RAI 840-6096, Question 06.02.02-80**, the staff also requested the applicant to describe further the vendor actions to confirm that pump wetted surface materials are acceptable in post-LOCA fluids for at least 30 days. In its response to RAI 840-6096, Question 06.02.02-80, dated November 22, 2011, the applicant stated the following:

> MHI also considers it the vendor's design and qualification action to provide documentation confirming the pump wetted surface material (such as wear rings, pump internals, bearing, and casing) wear rates provide acceptable operation in post-LOCA fluids during the 30 day mission time. This is a standard aspect of pump specification and procurement activities. A list of materials of the wetted pump surfaces and the hardness of each material will be recorded in the pump design and qualification documentation.

The staff finds the applicant’s response acceptable because it states that it is the vendor's design and qualification action to provide documentation confirming the pump wetted surface material (such as wear rings, pump internals, bearing, and casing) wear rates provide acceptable operation in post-LOCA fluids during the 30 day mission time.

Technical Report MUAP-08013-P, Sections 3.4.1.1 and 3.4.1.2, state that at the procurement stage, the pump vendor will provide a table listing the design or specified opening sizes and internal running clearances for the ECCS and CSS pumps.

In **RAI 840-6096, Question 06.02.02-80**, the staff also requested the applicant to describe the pump vendor actions regarding pump opening sizes and internal running clearances. In its response to RAI 840-6096, Question 06.02.02-80, dated November 22, 2011, the applicant stated the following:

> MHI also considers it the vendor's design and qualification action to provide documentation confirming the pump opening sizes and internal running clearances provide acceptable operation in post-LOCA fluids during the 30 day mission time. This is a standard aspect of pump specification and procurement activities. The vendor will record the opening sizes and internal running clearances in the pump design and qualification documentation and are part of standard pump vendor O&M documents.

The staff finds the applicant's response acceptable because it states that it is the vendor's design and qualification action to provide documentation confirming the pump opening sizes and internal running clearances provide acceptable operation in post-LOCA fluids during the 30 day mission time. Therefore, given the discussion above, **RAI 840-6096, Question 06.02.02-80, is resolved and closed**.

In **RAI 840-6096, Question 06.02.02-85**, the staff requested the applicant to specify ITAAC for verification that the ECCS and CSS pumps will operate as designed in post-LOCA debris conditions. In its response to RAI 840-6096, Question 06.02.02-85, dated November 22, 2011, the applicant referenced ITAAC that verify pump NPSH and full flow capabilities. The staff does not consider these ITAAC adequate for verification that pumps will operate as designed in post-LOCA debris conditions. However, during a conference call between the applicant and the staff on July 11, 2012, the staff agreed that this RAI may be closed when associated **RAI 896-6296, Question 3.9.6-69**, which addresses the functional design and qualification for all safety-related pumps and valves, is confirmed acceptable. This is an open item **(RAI 840-6096, Question 06.02.02-85)** pending resolution of **RAI 896-6296, Question 3.9.6-69**. (Note, ITAAC related to containment systems (e.g., strainers) are evaluated in Section 14.3.11 of this SER).

Seal Leakage

Technical Report MUAP-08013-P, Section 3.4.3, "Seal Leakage," describes the leakage criteria for the ECCS safety injection and CS/RHR pumps. Both pumps are specified to maintain a seal leakage rate of less than [ ]. Under complete seal failure conditions, the leak rate is specified to be less than 50 gpm. The pump seal vendor will confirm that their design meets or exceeds these conditions.

In **RAI 840-6096, Question 06.02.02-81,** the staff requested the applicant to identify the seal leakage criteria in Appendices A and B, "Pump Design Parameters" of MUAP-08013-P. In its response to RAI 840-6096, Question 06.02.02-81, dated November 22, 2011, the applicant stated the following:

> ECCS and CS/RHR pumps leak rate information will be added to the tables in Appendices A and B of MUAP-08013-P, as the following note: The leak rate through the mechanical seal shall be less than [ ] (under normal seal conditions). When the mechanical seal has failed, the leak rate shall be less than 50 gpm.

The staff finds the applicant's response to add the seal leak rate information to the tables in Appendices A and B of MUAP-08013-P to be acceptable since these are design parameters that should be identified in Appendices A and B. Appendices A and B of MUAP-08013-P were revised as indicated in the RAI response. Therefore, **RAI 840-6096, Question 06.02.02-81, is resolved and closed**.

In **RAI 840-6096, Question 06.02.02-82,** the staff requested the applicant to identify pumps design parameters such as post-LOCA fluid constituents and seal leakage in Appendices A and B, "Pump Design Parameters" of MUAP-08013-P and in applicable sections of the DCD. In its response to RAI 840-6096, Question 06.02.02-82, dated November 22, 2011, the applicant stated that Tables A-1, "ECC/CS Strainer and Safety Injection Pump Design Parameters," and B-1, "Containment Spray/Residual Heat Removal Pump Design Parameters," of MUAP-08013-P will be revised to include the post-LOCA fluid constituents and seal leakage criteria. However, the applicant stated that these criteria are too detailed to include in the DCD but will revise Sections 6.2 and 6.3 of the DCD to reference MUAP-08013-P as a source of additional requirements (note, the response to **RAI 840-6096, Question 06.02.02-82,** refers to the response to **RAI 840-6096, Question 06.02.02-84**, discussed below, for DCD modifications). The staff finds the applicant's response acceptable to revise Tables A-1 and B-1 of MUAP-08013-P and reference MUAP-08013-P as a source of additional requirements. In August 2012, Tables A-1 and B-1 of MUAP-08013-P were revised as indicated in the RAI response. Therefore, **RAI 840-6096, Question 06.02.02-82 is resolved and closed**.

In **RAI 840-6096, Question 06.02.02-83,** the staff identified that the available and required NPSH values for the ECCS safety injection pump and CS/RHR pump listed in Tables A-1 and B-1 of Technical Report MUAP-08013-P, Revision 2, are not consistent with the NPSH values listed in Tables 3-11, "NPSH available Calculation Data (CS/RHR Pump)," and 3-12, "NPSH available Calculation Data (SI Pump)," of Technical Report MUAP-08001-P, Revision 5. In its response to RAI 840-6096, Question 06.02.02-83, dated November 22, 2011, the applicant stated that the NPSH available and required for the ECCS safety injection pump and CS/RHR pump listed in Tables A-1 and B-1 of MUAP-08013-P will be revised to be consistent with the NPSH values in Tables 3-11 and 3-12 of MUAP-08001-P. The staff finds this response to be acceptable since values have been revised to be consistent and are acceptable per staff evaluation in Section 6.3.4.7 of this report. Tables A-1 and B-1 of MUAP-08013-P were revised as indicated in the RAI response. Therefore, **RAI 840-6096, Question 06.02.02-83 is resolved and closed**.

DCD Section 06.02.02.1.4 for Containment Spray and DCD Section 6.3.2.5 for ECCS do not reference the design and evaluation criteria in MUAP-08013-P for downstream ex-vessel components. In **RAI 840-6096, Question 06.02.02-84**, the staff requested the applicant to revise applicable portions of the DCD to reference the criteria in MUAP-08013-P. In its response to RAI 840-6096, Question 06.02.02-84, dated November 22, 2011, the applicant stated that DCD Sections 6.3.2.5 and 6.2.2.1.4 will be revised to state, "MUAP-08013-P contains requirements for design and evaluation of ECCS and CSS ex-vessel downstream components to ensure the ECCS and CSS systems and their components will operate as designed under post-LOCA conditions." The staff finds the applicant's response to reference to

MUAP-08013-P in applicable sections of the DCD to be acceptable. DCD Sections 6.2.2.1.4 and 6.3.2.5 were modified as indicated in the RAI response and documented in GTR2. The staff opened **CI-SRP06.02.02-2** to verify that the final design document incorporates the associated supplemental DCD information provided by GTR2. Therefore, **RAI 840-6096, Question 06.02.02-84 is considered confirmatory and will be tracked as CI-SRP06.02.02-2**.

ECCS and CSS Performance Evaluations

Technical Report MUAP-08013-P, Section 3.5, "ECCS and CSS Performance Evaluations," states that based on piping wear and pump operation evaluations, the applicant concluded that the system piping and component flow resistances will changes minimally during the course of the LOCA. Therefore flow balances and system performance is not affected in an appreciable manner. The resulting flows and pressures are consistent or conservative with respect to the accident analysis. The minor resistance changes do not affect the system flow calculations and Design Basis analysis.

Regulatory Summary

Technical Report MUAP-08013-P, Section 3.6, "Regulatory Summary," provides a high level summary of the regulatory issues and concerns regarding operation of the ECCS and CSS during post-LOCA debris conditions.

Confirmation Items

Technical Report MUAP-08013-P, Section 3.7, "Confirmation Items," provides a list of confirmation items for ECCS and CSS pumps and valves to ensure system components will meet their design specifications. Confirmation Items 3.7.3 through 3.7.6 are specifically identified in applicable component sections of MUAP-08013-P and are to be confirmed during QME-1-2007 qualification testing of the component. Confirmation Items 3.7.1 and 3.7.2 for the SI pump runout flow less than 2,000 gpm and CS pump runout flow less than 4,000 gpm are pump design parameters and will also be confirmed during pump qualification testing per QME-1-2007.

Summary of Results

Technical Report MUAP-08013-P, Section 3.8, "Summary of Results," states that the intent of this technical report is to assess the US-APWR ECC and CSS to ensure that that these systems and their components will operate as designed under post LOCA Conditions. The report concludes that the US-APWR ECC/CS Systems and components are fully capable of performing their intended functions under post-LOCA operating conditions with regard to ex-vessel downstream effects.

The NRC staff reviewed the US-APWR DC application for compliance with the NRC regulations for the evaluation of ex-vessel downstream effects. Given that there are open and confirmatory items discussed in this ex-vessel section, the staff is unable to find that the applicant has demonstrated that there is reasonable assurance that the ex-vessel effects of debris loading in the recirculating fluid resulting from a postulated LOCA will not adversely affect the US-APWR ECCS and CSS equipment in providing adequate core and containment cooling.

6.2.2.4.14 Downstream Effects – Fuel Assemblies

The downstream effects evaluation on fuel assemblies is found in Section 6.3 of this SER.

6.2.2.4.15 Chemical Effects

6.2.2.4.15.1 Introduction

To determine the compliance of the US-APWR design with the requirements of GDC 35, GDC 38, and 10 CFR 50.46(b)(5) with respect to chemical debris formed in the post-LOCA containment pool, the staff reviewed the information in the DCD and supporting documents using the guidance listed in Section 6.2.2.3 of this SER. Chemical effects are corrosion products, gelatinous material, or other chemical reaction products that form as a result of interaction between the PWR containment environment and containment materials after a LOCA. SRP Section 6.2.2 provides no specific guidance for chemical effects evaluations but references RG 1.82 Revision 3, the NEI 04-07 guidance, and the staff's SE of NEI 04-07 for acceptable guidance for PWR sump debris evaluations.

For PWR plants, RG 1.82 Revision 3 contains the following guidance relative to chemical effects:

- Section 1.1.2.3 states that to minimize potential debris caused by chemical reaction of coolant with metals in the containment, exposure of bare metal surfaces (e.g., scaffolding) to spray impingement or immersion should be minimized either by removal or by using chemical-resistant protection (e.g., coatings or jackets).

- Section 1.3.2.6 states that in addition to debris generated by jet forces from the pipe rupture, debris created by the resulting containment environment (thermal and chemical) should be considered in the analyses. Examples of this type of debris would be disbondment of coatings in the form of chips and particulates or formation of chemical debris (precipitants) caused by chemical reactions in the pool.

The following documents contain additional staff guidance for chemical effects evaluations:

- NEI 04-07, Revision 0 (ADAMS Accession Number ML050550138), and the staff's accompanying SE (ADAMS Accession Number ML043280641).

- Enclosure 3, "NRC Staff Review Guidance Regarding Generic Letter 2004-02 Closure in the Area of Plant-Specific Chemical Effect Evaluations," to "Revised Guidance for Review of Final Licensee Responses to Generic Letter 2004-02," "Potential Impact of Debris Blockage on Emergency Recirculation during Design Basis Accidents at Pressurized-Water Reactors," dated March 28, 2008 (ADAMS Accession Number ML080230234).

- WCAP-16530-NP-A, "Evaluation of Post-Accident Chemical Effects in Containment Sump Fluids to Support GSI-191," dated May 2, 2008 (ADAMS Accession Number ML081150379)

6.2.2.4.15.2 Applicant's Approach to Addressing Chemical Effects

In DCD Tier 2, Subsection 6.2.2.3.2, "Debris Source Term," as supplemented by GTR2, and MUAP-08001-NP, "US-APWR Sump Strainer Performance," the applicant describes the debris source term of the US-APWR that challenges sump performance as consisting of non-chemical debris and chemical debris. Additionally, in DCD Tier 2 Section 6.2.2.3.10, "Chemical Effects Test," the applicant describes the US-APWR chemical effects testing.

MUAP-08001-P, Section 3.8, "Summary of Chemical Effects Tests," states that based on a review of the results presented for Integrated Chemical Effects Test #5 (ICET 5), the US-APWR was expected to have minimal corrosion and reaction products. ICET 5 used similar materials and chemistry to the US-APWR, although with different quantities of the materials (see NUREG/CR-6914 "Integrated Chemical Effects Test Project," issued December, 2006). However, in order to further understand the plant specific interactions between the containment materials and post-LOCA debris with the recirculation sump fluid chemistry, the applicant elected to perform a chemical effects test for the US-APWR. MUAP-08001-P Section 3.8 states that the objective for the chemical effects test was to obtain experimental data under simulated plant conditions on the corrosion products that may form in a post-LOCA environment. These data were used to determine compositions, characterize properties, and calculate masses of chemical reaction products that may develop in the containment under a representative post-LOCA environment. Appendix C, "Evaluation of Chemical Debris (for head loss)," of MUAP-08001-P, contains a summary of the results of the chemical effects test and the prediction of the type and amounts of precipitates expected to form. The staff's evaluation benefited from a visit to the applicant's Research Center in Takasago, Japan, and Engineering Center in Kobe, Japan, on September 17 - 18, 2008, for laboratory tours and discussions of the chemical effects test methodology and results.

6.2.2.4.15.3 Chemical Effects Testing – Chemical Source Term

The applicant's chemical effects testing is described in MUAP-08011-P, "US-APWR Sump Debris Chemical Effects Test Result," Revision 2, issued August 2012 (Proprietary). The testing used a similar methodology to the Integrated Chemical Effects Tests (ICET) documented in NUREG/CR-6914, using materials, buffer, and simulated post-LOCA environment representative of the US-APWR design. The tests were performed to predict the type and amount of chemical reaction products that would form. The applicant's overall chemical effects evaluation involved several steps. The chemical precipitate source term was determined by: (1) autoclave tests to represent the dynamic, higher temperatures early in a LOCA and (2) 30-day steady state recirculation tank tests to represent longer term LOCA conditions. Specifically, the applicant calculated the source term by measuring the concentration of dissolved species in the test solution as a function of time. Since the amounts of containment materials in the test were scaled according to the volume of test solution, the applicant assumed the concentration of dissolved species in the test solution was equivalent to that in the sump pool. After determining the chemical source term, a thermodynamic equilibrium computer code was used to predict the type and amounts of chemical precipitates.

The chemical effects test program used two different types of tests, autoclave and recirculation. The transient temperature autoclave tests were conducted with varying temperature to simulate the first 100 hours of a design basis event, when the temperature is predicted to be changing rapidly. The transient autoclave test temperature varied from 140 °C (284 °F) (representing the peak post-LOCA temperature) to 65 °C (149 ºF) (representing the long-term steady state post-LOCA temperature). The temperature of 140 °C (284 °F) was chosen to be conservatively higher than the actual predicted peak temperature of [ ]. In 2010, the applicant determined that the maximum sump pool temperature would be [ ]. The staff asked if the temperature profile in the test remained bounding in terms of generating more dissolution than the expected transient in the post-LOCA sump pool **(RAI 836-6099, Question 06.02.02-68)**. In its response to **RAI 836-6099, Question 06.02.02-68**, dated November 11, 2011, the applicant stated that the test temperature bounds the re-estimated plant temperature except for a brief period during the first hour when the calculated plant temperature rises faster than the autoclave test temperature. The response included proposed revisions to MUAP-08011-P, including new figures showing the two temperature transients. Based on the response the staff concluded the revised plant temperature profile is bounded by the autoclave test temperature profile and the test provided a conservative amount of dissolution as an input to the chemical debris analysis. The staff also confirmed that MUAP-08011-P, Revision 2, was revised as proposed in the response. Therefore, **RAI 836-6099, Question 06.02.02-68, is resolved and closed**.

The steady-state test temperature of 65 °C (149 °F) was also chosen to be higher than the expected steady-state plant temperature of [ ]. Using the higher temperature is a conservative approach for debris generation because it results in more dissolution, and the applicant assumed the maximum amount of precipitates that could form based on the concentrations of species in solution. The transient temperature autoclave tests were conducted at three different pH levels as a sensitivity study on the effect of pH. One test was conducted at the nominal post-LOCA steady state pH of 7.7-7.8 (“standard condition”), one test was conducted with more acidic conditions (pH 4.3), and one test was conducted at a more alkaline pH (8.4).

The applicant clarified in the response to **RAI 263-2072, Question 06.02.02-14**, dated March 31, 2009, that for each transient temperature autoclave test, a constant temperature autoclave test at the same pH and 65 °C (149 °F) was conducted as a baseline to determine the increased concentrations of chemical species resulting from the higher temperature period of the LOCA. Since the autoclave tests were relatively short (100 hours), the increase in concentration for each species attributed to high temperature (>65 °C) was then added to the concentration measured in the 30-day, low-temperature recirculation test to determine the maximum concentration of each species for each condition (standard, alkaline, acidic). If the concentration of a species reached a peak and then decreased, the applicant treated the peak concentration as a plateau from that point forward. The applicant also describes this procedure in Appendix C of MUAP-08001-P. In the case of aluminum, the maximum concentration calculated according to this method was [ ] beginning at about 50 hours, and the applicant used this value to calculate the amount of aluminum-base precipitate.

Figure C-1, “Impurities concentration trend of the sump water,” of MUAP-08001-P provides the concentration profile resulting from the combination of the recirculation and autoclave test results for the standard pH condition. The pH was achieved based on simulating the post-LOCA sump water chemistry using a representative boric acid solution with the pH increased by mixing with the sodium tetraborate buffer. Section 3.3.2, “Chemistry Parameters,” of MUAP-08011-P indicates that the inputs to the sump chemistry were selected to maximize the pH for the nominal conditions by setting the RWSP and RCS boric acid concentrations at their low limits of 4000 ppm and 0 ppm, respectively. Achieving a more alkaline pH was considered conservative since the dissolution of aluminum, which is the most important metallic element relative to precipitate formation, increases with increasing pH in PWR post-LOCA water.

The recirculation test was conducted at a constant temperature of 65 °C (149 °F) for a duration of 30 days, with a pH of 7.7-7.8. The recirculation test tank had a simulated spray and recirculation of the test solution simulating the operation of the containment spray and ECCS systems. Test materials were either submerged or unsubmerged depending upon their locations in the actual containment.

The staff finds the environmental conditions selected for the chemical effects test program acceptable, since the test conditions selected by the applicant represent plant conditions in a way that produces more dissolved chemical species than the actual expected plant post-LOCA conditions. Therefore, **RAI 263-2072, Question 06.02.02-14, is resolved and closed.**

The materials in the chemical effects test were intended to be representative of the materials that will be present in the US-APWR containment, including both those materials in the ZOI and those that could be transported to the sump through containment washdown. The materials include Nukon fiberglass insulation, concrete particulate, concrete, galvanized steel, aluminum, carbon steel, and copper. The ratio of the amount of each material in the US-APWR containment to the sump water volume was determined. The same ratio was used to determine the amount of material included in the autoclave and recirculation tests, when scaled to the volume of the autoclave and recirculation test tank. As clarified in the responses to **RAI 45-876, Question 06.02.02-1**, dated August 26, 2008, and **RAI 85-1445, Question 06.02.02-10**, dated November 12, 2008, (1) no coatings materials were included in the tests because the US-APWR will use only DBA-qualified epoxy coatings, which the staff has acknowledged as being essentially inert under DBA conditions, (2) no primers containing materials that will contribute to chemical effects will be used, and (3) experience exists with such coatings systems in operating plants. The response to **RAI 263-2072, Question 06.02.02-12**, dated March 31, 2009, provided a proposed ITAAC to verify that all coatings in the containment are DBA qualified. When the applicant issued DCD Revision 3, the proposed coatings ITAAC was modified to verify all coatings are consistent with the ECC/CS suction strainer debris generation, debris transport and downstream effects evaluations. Based on the information discussed above, the staff finds the responses to **RAI 45-876, Question 06.02.02-1, RAI 85-1445, Question 06.02.02-10, and RAI 263-2072, Question 06.02.02-12** are acceptable because the coatings materials to be used in containment are of a type that does not contribute to chemical effects, and this will be verified by an ITAAC. Therefore, **RAI 45-876, Question 06.02.02-1, RAI 85-1445, Question 06.02.02-10, and RAI 263-2072, Question 06.02.02-12, are resolved and closed.**

RG 1.82 Revision 3, Section 1.3.2.5, states that the cleanliness of the containment during plant operation should be considered when estimating the amount and type of debris available to block the ECC sump screens. RG 1.82 also states that the potential for such material (e.g., thermal insulation other than piping insulation, ropes, fire hoses, wire ties, tape, ventilation system filters, permanent tags or stickers on plant equipment, rust flakes from unpainted steel surfaces, corrosion products, dust and dirt, latent individual fibers) to impact head loss across the ECC sump screens should be considered. This type of debris, known as latent or resident debris, is further defined in NUREG/CR-6877. In its response to **RAI 45-876, Question 06.02.02-2**, dated August 26, 2008, the applicant indicated that a portion of the fiber, concrete dust, and carbon steel included in the chemical effects testing was representative of latent debris. The staff found the materials included in the chemical effects tests are acceptable, because the materials included are representative of the actual materials found in containment that have the potential to contribute to chemical effects, and are consistent with the staff review guidance for chemical effects evaluations identified above in Subsection 6.2.2.4.13.1. Therefore, **RAI 45-876, Question 06.02.02-2, is resolved and closed.**

6.2.2.4.15.4 Type and Amount of Chemical Debris

The applicant's chemical effects test results includes the measured concentrations of the various dissolved species from both the autoclave and recirculation tests, such as aluminum, silicon, iron, copper, nickel, magnesium, calcium, zinc, boron, and sodium. Other data included the temperature and pH as a function of time for each test run, weight change of the material coupons, chemical analysis of deposits on the coupons, mass of precipitated solids, etc. The concentrations of the dissolved species, pH, and temperature were inputs to the model predicting the type and amount of precipitates.

The applicant used a commercial thermodynamic simulation software program (OLI Stream Analyzer) to predict the mass of each precipitate formed from the dissolved elements. The applicant found that the only significant precipitates formed were aluminum hydroxide ($Al(OH)_3$) and sodium aluminum silicate ($NaAlSi_3O_8$). Table C-1, "Chemical debris of the US-APWR," of MUAP-08001-P (Revision 2), as supplemented by the response to **RAI 263-2072, Question 06.02.02-15**, dated March 31, 2009, provides the predicted amounts of these precipitates per liter of sump water as [ ] and [ ], and the corresponding precipitate mass (rounded up) for the US-APWR sump of [ ] $Al(OH)_3$ and [ ] $NaAlSi_3O_8$ for a total of [ ]. The applicants' predictions were based on the standard test condition (pH 7.7-7.8) only. Small amounts of Zn_2SiO_4 and CuO were also predicted, but the applicant considered the amounts negligible [ ]. The staff finds it acceptable to neglect the predicted [ ] and [ ] because of the small quantity relative to the quantity of aluminum-base precipitates known to be capable of producing head loss in a debris bed. **RAI 263-2072, Question 06.02.02-15, is resolved and closed.**

As part of its closure plan for GSI-191 issues, (see Note 1 in Table 6.2.2-1 below), the applicant made changes to the plant design that approximately doubled the volume of the recirculating water. The applicant then recalculated the amount of chemical precipitates by assuming the same concentration and increasing the precipitate mass in proportion to the water volume. After performing the calculations and rounding up, the applicant's design-basis quantity for chemical precipitates is 330 pounds sodium aluminum silicate and 300 pounds aluminum hydroxide, for a total of 630 pounds (DCD Tier 2, Table 6.2.2-4). These design-basis and calculated values are listed in Rows 1 and 2, respectively, of Table 6.2.2-1, below. The table also includes staff confirmatory calculations, which are discussed below.

Table 6.2.2-1 Chemical Precipitate Amounts: Applicant Calculations and Staff Confirmatory Calculations

Analysis Performed			**Chemical Precipitates, lbm**		
			$NaAlSi_3O_8$	$Al(OH)_3$	Total
1	Applicant's design basis	Rounded up from standard condition (next row)	330	300	630
2	Applicant's analysis	Standard condition, pH 7.8, with design change[1]	[ ]	[ ]	[ ]
3	Staff confirmatory[2]	Standard condition, pH 7.8, using applicant's measured Si & Al concentrations	[ ]	[ ]	[ ]
4	Staff confirmatory[2]	Standard condition, pH 7.8, original design, using WCAP-16530 release-rate equations	[ ]	[ ]	[ ]

Analysis Performed			Chemical Precipitates, lbm		
			$NaAlSi_3O_8$	$Al(OH)_3$	Total
5	Staff confirmatory	WCAP-16530 equations with pH/temperature profile estimate and design change	[ ]	[ ]	[ ]

Notes

1. A design change in 2011, increased the volume of the recirculating water, thus increasing the ratio of water to aluminum surface area. (Letter from Y. Ogata to the U.S. NRC, "Updated Closure Plan for Issues Associated with GSI-191 for the US-APWR Design Certification," December 21, 2011, Mitsubishi Heavy Industries (MHI) Reference UAP-HF-11449, ADAMS Accession Number ML11362A464.)

2. The staff used these two different methods for the amount of aluminum and silicon released into solution. In both cases, the precipitate quantities were subsequently calculated using equations from WCAP-16530-NP-A.

3. The WCAP-16530-NP-A predicts AlOOH as the aluminum hydroxide form. The staff converted the mass of AlOOH to $Al(OH)_3$ according to the molecular weight ratio. For example, 289 lbm of $Al(OH)_3$ is equivalent to 222 lbm of AlOOH calculated from the WCAP.

The staff previously investigated the use of OLI Stream Analyzer and other thermodynamic predictive codes in resolving GSI-191 chemical effects. This was documented in NUREG/CR-6912 ("GSI-191 PWR Sump Screen Blockage Chemical Effects Tests: Thermodynamic Simulations," December 2006, ML071900449), which states, "The study concluded that the codes as tested were broadly useful in assessing whether precipitation of secondary solid phases was likely under the specified conditions and the quantity of material that was predicted to form." Although the NRC staff thinks thermodynamic equilibrium programs may be broadly useful to inform chemical effects evaluations, the staff questions the ability of these programs to accurately predict the types and quantities of precipitates since thermodynamic equilibrium conditions are not reached in the time frames of interest. Predictions from thermodynamic equilibrium programs, however, may be acceptable to the staff provided the predicted type and quantity of chemical precipitates are conservative.

Therefore, as an independent check on the applicant's predictions using the OLI software, the staff used the equations from WCAP-16530-NP-A, Appendix D, to perform confirmatory calculations of the amount of precipitate that could form based on the concentrations of dissolved elements. The staff approved WCAP-16530-NP-A in a safety evaluation incorporated into the report. These equations are conservative because they assume that all aluminum forms a precipitate in the presence of the sodium tetraborate buffer. If the silicon mass is less than 3.12 times the aluminum mass, the WCAP-16530-NP-A analysis assumes all of the silicon will precipitate as well.

The staff used two different methods to determine inputs for the concentrations of the dissolved elements for the confirmatory calculations. For the first method, the staff used the applicant's measured aluminum and silicon concentrations from the tests with nominal pH conditions, which the applicant illustrated in Figure C-1 of MUAP-08001-P. Equations in WCAP-16530-NP-A use these elemental concentrations to calculate precipitate concentrations. Using the volume of sump water in the plant, the staff calculated the corresponding mass of precipitate. The amount of precipitate is proportional to the water volume because the applicant performed the test with the same material/water ratio as the plant design.

For the second method, the staff used the material release rate equations from WCAP-16530-NP-A to determine the concentrations of the dissolved elements. The release rate equations predict the amount of dissolution or corrosion for the constituents of the materials that can contribute to precipitate formation, such as aluminum, silicon, and calcium, as a function of pH and temperature, and are based on research and testing documented in WCAP-16530-NP-A. The materials and chemistry inputs and the time-temperature profile used by the applicant as the basis of their chemical effects test, were the inputs to this calculation. The staff performed the calculation with two different constant pH levels of 7.8 (standard condition) and 8.4 (alkaline condition). The staff also used this method to calculate the precipitate quantity using the pH transient predicted for the sump pool rather than a constant pH. (The pH transient is described in DCD Tier 2, Section 6.5.2, “Containment Spray Systems,” in the context of confirming that the sodium tetraborate stored in containment will buffer the pH at 7 or higher and maintain it there for 30 days following a LOCA.) This pH transient is conservative with respect to dose assessment, but the constant pH 7.8 conditions are conservative with respect to aluminum corrosion and the corresponding precipitate mass.

Using the first method of determining the concentrations (applicant’s measured concentration profiles) for the standard condition in the autoclave test, the staff calculated [ ] of total precipitate in the sump, consisting of [ ] of sodium aluminum silicate and [ ] of aluminum oxyhydroxide. This result is listed in Row 3 of Table 6.2.2-1. The amounts the staff calculated are approximately the same as the applicants’ predictions. (The staff’s values for Al(OH)3 are actually converted from values for AlOOH, since AlOOH is the hydroxide form of aluminum in all WCAP-16530-NP-A results. The conversion was performed by multiplying the mass of AlOOH by 1.30, which is the ratio of the molecular weight of Al(OH)3 to AlOOH.) Using this method, the staff’s calculated value of [ ] of precipitate is less than the applicant’s design-basis value of 286 kg (630 lbm). The applicant’s prediction of the total precipitate mass is thus conservative compared to the staff’s calculation.

Using the second method – the WCAP-16530-NP-A release rate equations – the staff calculated a total of [ ] total debris, consisting of [ ] of sodium aluminum silicate and [ ] of aluminum hydroxide. This result is listed in Row 4 of Table 6.2.2-1 and is much less than the applicant’s design basis value of 286 kg (630 lbm). At the more alkaline condition (pH 8.4), the staff’s calculated aluminum concentration using this method was 57 percent greater than at pH 7.8, resulting in a greater mass of precipitate for the alkaline condition. The staff’s calculated precipitate mass for both the standard and alkaline conditions, using the WCAP release rate equations, is less than the applicant’s prediction for the standard condition. The applicant’s prediction of total precipitate mass thus bounds the staff’s confirmatory calculations performed using two different methods, for both the standard and alkaline conditions. Row 5 of Table 6.2.2-1 shows the staff’s calculated debris mass using the WCAP-16530-NP-A release rate equations for the standard pH condition and the estimated post-LOCA transient for both the pH and temperature. The calculated debris total in this case is [ ] of sodium aluminum silicate. This method produces the lowest debris total because the pH starts in the slightly acidic range before the buffer takes effect, and the corresponding aluminum corrosion rate is lower.

The reason for the large difference between the applicant’s calculated debris quantity (286 kg (630 lbm)) and the staff’s confirmatory calculation using the WCAP release-rate equations [

] is the result of design changes after Revision 3 of the DCD. The design changes included a large increase in the amount of the post-LOCA recirculating water pool. Since the applicant calculated the quantity of chemical precipitate according to a concentration (mg/L), the total quantity depends on the total recirculating water volume. The applicant determined the concentration values using tests with prototypical ratios of material surface area or mass to the original volume of water. Later, when the volume was increased with the design change, the applicant scaled up the chemical debris quantity to maintain the same concentration of chemicals. However, because the pH, temperature, and material quantities are not significantly affected by the design change, the release rates for these materials are not significantly affected. Therefore, the calculated amount of material actually released (e.g., kg of aluminum) does not change significantly, and the actual concentration of that material (mg/L) would be proportionally lower in the larger water volume. Since the WCAP-16530-NP-A uses temperature and pH to calculate release rate, the amount of chemical precipitate is not directly related to the amount of recirculating water. This is the reason the staff calculated much lower precipitate quantities than the applicant using the increased water volume.

An unexpected aspect of the applicant's data was that the aluminum concentration measured in the autoclave test at nominal pH was higher than the aluminum concentration measured in the autoclave test under alkaline conditions. Since corrosion of aluminum generally increases with increasing pH, the staff requested additional information to explain this result. In its response to **RAI 263-2072, Question 06.02.02-13**, dated March 31, 2009, the applicant speculated that the dissolution of the aluminum might have been less in the alkaline tests due to precipitation or absorption of silicon or zinc on the coupon surfaces. Both silicon and zinc concentrations were higher in the alkaline test than in the standard condition tests. The applicant also verified that the dissolution of aluminum was actually greater in the standard condition than in the alkaline conditions via the coupon weight loss measurements, which showed the aluminum coupon in the standard conditions lost more weight than the aluminum coupon in the alkaline conditions.

A silicon inhibition effect on aluminum corrosion has been observed, as documented in WCAP-16785-NP, "Evaluation of Additional Inputs to the WCAP-16530-NP Chemical Model", but the staff could not confirm the applicant's proposed explanation for the US-APWR test results. Similarly, the unexpectedly low corrosion rate of aluminum in ICET 4, with pH 9.8, was attributed to corrosion inhibition by silicon. Despite the unexpected results, using the material release rates from WCAP-16530-NP-A, the staff calculated less precipitate mass for both the standard and alkaline conditions than the applicant calculated for the standard condition. Therefore, the staff finds that **RAI 263-2072, Question 06.02.02-13 is resolved and closed**, because the applicant's prediction of precipitate mass is conservative compared to the calculated precipitate mass for both the standard and alkaline conditions using a model accepted by the staff.

The staff finds the type and amount of precipitates predicted using the results of the chemical effects test acceptable. This conclusion is based on the staff's review of the chemical effects test plan, which used a similar methodology to the NRC-sponsored ICET test series, the staff's review of the chemical effects test results, and the staff's review of the predicted amount of precipitate. The staff's confirmatory calculations determined that the type of precipitates the applicant predicted is consistent with those calculated using a staff-approved methodology (WCAP-16530-NP-A), and the mass of precipitates predicted to form using the applicant's methodology is greater than the amount of precipitate calculated using a staff-approved methodology (WCAP-16530-NP-A).

6.2.2.4.15.5 Chemical Debris in Strainer Head-Loss Testing

This section describes the staff's review of the chemical debris component in the design-basis strainer testing for the US-APWR. The staff's review of the strainer testing is discussed mainly in Subsection 6.2.2.4.9 of this SER. The applicant performed integrated testing with fiber, particulate, and chemical debris, and described the use of the chemical debris in Appendix B of MUAP-08001-P. The applicant performed the testing with the goal of all chemical debris eventually reaching the strainer (i.e., no settlement credit). The chemical precipitate was added in batches by pumping it to the flume from a mixing tank. All of the chemical precipitate was added in four approximately equal batches after all of the particulate and fiber were in the flume. The procedure included a fifth addition for rinsing the residual precipitate from the mixing tank and piping into the flume. The chemical addition commenced a few hours into the test and was completed within about three hours. After head loss stabilized, the applicant determined visually that the entire strainer was covered with chemical debris. The staff finds this procedure acceptable because adding all of the design basis 30-day chemical precipitate quantity early in the test conforms to the staff's March 2008, strainer testing guidance.

The applicant determined the total amount of chemical debris for the design basis test according to the strainer area and the maximum debris allocation to one strainer. As explained in MUAP-08001-P, the total amount of postulated chemical debris (630 lb) was multiplied by 0.85 for the debris allocation per sump, and then multiplied by the screen area scaling factor [ ] to determine the amount of chemical debris in the testing [ ]. This total represents the sum of the $Al(OH)_3$ and $NaAlSi_3O_8$. The staff finds the quantity of chemical debris in the testing acceptable because including all of the calculated debris mass, scaled according to screen area, conforms to the staff's March 2008 testing guidance.

As stated in DCD Tier 2 Subsection 6.2.2.3.10, the applicant's analysis of NPSH is based, in-part, on the absence of chemical debris when the temperature of the pool is greater than 66 °C (150°F) (based on solubility of aluminum). As described in MUAP-08001, the applicant justified this, in part, because significant aluminum precipitation was not observed in the APWR 30-day chemical effects test (pH 7.8 and 65°C/149°F) or ICET #5 (pH 8.2 – 8.4 and 60 °C/140°F). For additional plant-specific justification, the applicant used published thermodynamic data for aluminum species and the OLI StreamAnalyzer to calculate solubility as a function of temperature for aluminum. For [ ] aluminum, the applicant calculated a precipitation temperature of [ ]. The applicant selected the criterion of [ ] to provide margin above the calculated value.

The staff performed confirmatory calculations for pH 7.8 and [ ] Al using aluminum solubility data from Argonne National Laboratory (ANL) (C.B. Bahn, Ke.E. Kasza, W.J. Shack, and K. Natesan, "Aluminum Solubility in boron Containing Solutions as a Function of pH and Temperature," NRC Contract J-4149, September 19, 2008, ML091610696) and calculated precipitation temperature values of [ ] and [, for the two sets of ANL data. At 150°F (66°C), the solubility calculated from the ANL equations is [ ]. Since solubility for this chemical system increases with temperature, selecting a design basis precipitation temperature higher than the calculated value is conservative. And because the ANL data were generated for boron-containing solutions in the temperature, concentration, and pH ranges of the US-APWR, the staff determined the data are applicable.

In addition, after increasing the amount of post-LOCA water, the applicant assumed the aluminum concentration in the water would remain at [ ], while the staff's calculated value decreased because the corrosion rate and aluminum surface area do not change. The staff's calculations using the WCAP-16530-NP-A methodology for a constant pH of 7.8 indicate the Al concentration in the larger water volume would be about 7.2 ppm, corresponding to a precipitation temperature of about [ ]. For the predicted plant pH and temperature transient, the staff calculated an aluminum concentration of approximately [ ] when the temperature reaches 66 °C (150°F), which is another indication that the applicant's assumption of 66°C (150°F) aluminum precipitation temperature is conservative. The staff finds the applicant's assumption of no detrimental chemical precipitates at temperatures greater than 66°C (150°F) acceptable because it is conservative with respect to a staff-reviewed methodology and it was determined with a conservative input aluminum concentration.

To represent the postulated AlOOH and $NaAlSi_3O_8$ precipitate in the testing, the applicant used AlOOH surrogate generated outside the loop based on the test plan. Sodium aluminum silicate was not used due to safety concerns with handling and using it. This approach conforms to staff guidance based on procedures industry developed and documented in WCAP-16530-NP-A. The staff accepted these procedures – with limitations and conditions - in its SE of the WCAP and included them in the March 2008, guidance. The applicant's procedures include the preparation of the chemicals and the settlement testing with acceptance criteria. In the settling test, prepared samples of each solution are shaken, put aside and allowed to settle for one hour. Settling is visually measured by the clarification of the liquid from the surface of the liquid down to the top level of cloudiness of the solution. For the AlOOH surrogate, the applicant's acceptance criterion was a settled volume of at least 6 mL (0.2 U.S. fluid ounce (fl oz)) in a 10 mL (0.34 fl oz) sample of 2.2 g/L (2200 ppm) concentration. This was based on the criterion approved by the staff in the SER for WCAP-16530-NP-A for head-loss tests in which the objective is to keep the chemical precipitate suspended. Research and testing has demonstrated that AlOOH prepared and tested according to the WCAP causes head loss in strainer tests. The staff's SER on WCAP-16530-NP-A states that surrogate precipitate prepared in accordance with the WCAP provides adequate settlement and filterability characteristics to represent post-LOCA chemical precipitates in strainer head-loss tests. For the two design-basis tests, the applicant reported settled volumes in the range 7.9 ml (0.27 fl oz) to 9.0 ml (0.3 fl oz) (four tests), which are acceptable because they meet the acceptance criterion of 6 ml.

6.2.2.4.15.6 Chemical Debris in Fuel Assembly Testing

This section describes the staff's review of the chemical debris component of the design-basis core-inlet blockage (CIB) head-loss testing performed to establish long-term core cooling for the US-APWR. The tests also included fibrous and particulate debris. The staff's review of the CIB testing is discussed mainly in Section 6.3 of this report. The applicant described the use of chemical debris in the fuel assembly testing in MUAP-08013-P, Revision 4.

MUAP-08013-P describes how the applicant calculated the amount of chemical debris to include in each of the three test conditions (hot-leg break, cold-leg break, and cold-leg break after hot-leg switchover). For the cold-leg break and cold-leg break after hot-leg switchover, the applicant considered all of the postulated chemical debris in the test. The postulated plant quantity [ ] was divided by the number of fuel assemblies (257) to determine the amount of chemical debris in these tests [ ]. For the tests representing a cold-leg break, the applicant considered [ ] percent of the postulated chemical debris based on calculation of the amount of flow reaching the lower plenum. Dividing [ ] percent of the postulated plant debris by 257 fuel assembly yields a total of [ ] in the tests for cold-leg break conditions. The staff performed confirmatory calculations and finds these quantities acceptable because they are based on the chemical debris quantity discussed above in Section 6.2.2.4.13. The staff's review of the [ ] percent flow fraction is discussed in Section 6.3.

The tests used AlOOH surrogate prepared outside the test loop to represent all of the chemical debris in the CIB tests. The staff finds this acceptable because of the demonstrated effect of this surrogate on head loss, as discussed in the staff's March 2008, chemical effects guidance. As discussed above in Section 6.2.2.4.13.5, using AlOOH surrogate also avoids the hazard associated with handling sodium aluminum silicate. According to Appendix B of the CIB test report, MUAP-08013-P, the applicant tested the settling properties of the prepared surrogate using a 2.2 g/L suspension with a required one-hour settled volume of [ ]. The staff finds this acceptable because it meets the staff's guidance discussed above for strainer testing in Section 6.2.2.4.13.5.

6.2.2.5 Combined License Information Items

DCD Tier 2, Section 6.2.8, "Combined License Information," as supplemented by GTR2, establishes COL Information Item 6.2(5) for instituting a containment cleanliness program and COL Information Item 6.2(6) for preparation of administrative procedures to ensure insulation within the ZOI and aluminum in containment will be consistent with the design basis.

The following is a list of COL item numbers and descriptions from Table 1.8-2 of the DCD:

Table 6.2.2-1
US-APWR Combined License Information Items

Item No.	Description	Section
6.2(5)	Preparation of cleanliness, housekeeping and foreign materials exclusion program is the responsibility of the COL Applicant. This program will be established to limit 200 lbs of latent debris, and to limit the allocated 200 ft^2 of miscellaneous debris per sump.	6.2.8
6.2(6)	Preparation of administrative procedures is the responsibility of the COL Applicant. The procedures will ensure that RMI and fiber insulation debris within ZOIs will be consistent with the design basis debris specified in Table 6.2.2-4, and will ensure that the aluminum in containment exposed to water in containment in post-LOCA condition (i.e., spray and blowdown water) is limited to equal or less than 810 ft^2.	6.2.8

COL information items needed, but not listed in Table 1.8-2 of the DCD: None

DCD Tier 2, Subsection 6.2.2.3.2, as supplemented by GTR2, provides information on what is included in the US-APWR containment cleanliness program objectives to include activities related to permanent and temporary modifications, foreign material and maintenance activities. Initially, DCD Tier 2, Section 6.2.8 COL 6.2(5) did not discuss latent (resident) and miscellaneous debris and associated design basis analysis limits. In **RAI 354-2585, Questions 06.02.02-35 and 06.02.02-36**, the staff asked the applicant to clarify the debris limits for latent and miscellaneous debris sources in containment. In its response to **RAI 354-2585, Questions 06.02.02-35 and 06.02.02-36**, dated October 7, 2009, the applicant proposed changes to DCD Tier 2, Section 6.2.2.3 and COL 6.2(5) to include latent and miscellaneous debris limits. Given the discussion above, the staff finds the applicant's response to **RAI 354-2585, Questions 06.02.02-35 and 06.02.02-36, is acceptable, pending modification of the DCD**.

In **RAI 736-5644, Question 06.02.02-63 and RAI 836-6099, Question 06.02.02-66**, the staff asked the applicant to describe the type of aluminum sources in containment and how the licensing basis will ensure that the quantity of aluminum exposed to the water in containment following a LOCA does not exceed the amount used in the debris calculations. In its response to RAI 736-5644, Question 06.02.02-63, dated July 13, 2011, and RAI 836-6099, Question 06.02.02-66, dated July 13, 2011, the applicant responded that the aluminum sources may include items such as instruments, motors, and cranes, and that aluminum would be listed in DCD Section 6.2.2.3 as a potential debris source subject to programmatic controls. In addition, the applicant added COL 6.2 (6) to Revision 3 of the DCD, requiring a COL applicant to identify and control the amount of aluminum exposed to the containment spray water. In its response to RAI 836-6099, Question 06.02.02-66, dated November 11, 2011, the applicant clarified that the aluminum controls would apply to both spray water and blowdown water. This response also proposed a corresponding modification to DCD Table 1.8-2 and COL 6.2(6). These changes limit the amount of aluminum surface area exposed to spray or blowdown water to 810 ft^2 (75 m^2), which is the amount assumed in the chemical effects testing and analysis. Therefore, the staff finds these changes acceptable pending modification of the DCD.

The staff opened confirmatory item **CI-SRP06.02.02-2, which tracks the issues of RAI 354-2585, Questions 06.02.02-35 and 06.02.02-36, RAI 736-5644, Question 06.02.02-63, and RAI 836-6099, Question 06.02.02-66**, to verify that the final design document incorporates the associated supplemental DCD COL 6.2(5) and 6.2(6) information provided in GTR2.

No additional Combined License Information Items were identified in this review.

6.2.2.6 Conclusions

As a result of the open and/or confirmatory items the staff is unable to finalize its conclusion on Section 06.02.02 related to containment heat removal system and the effects of accident-generated and latent debris in accordance with NRC regulations.

6.2.3 Secondary Containment Functional Design

6.2.3.1 Introduction

For the US-APWR design, portions of the primary containment are enclosed by containment penetration areas which function to prevent the direct release of containment atmosphere to the environment through the containment penetrations. The Containment Penetration Areas are served by the Annulus Emergency Exhaust System (AEES) following a DBA. The AEES maintains these areas at a negative pressure during accident conditions.

6.2.3.2 Summary of Application

DCD Tier 1: The Reactor Building (R/B) contains the Containment Penetration Areas and is described in DCD Tier 1, Section 2.2.1, "Building Structures Design Description." The AEES is described in DCD Tier 1, Section 2.7.5.2, "Engineered Safety Features Ventilation Systems."

DCD Tier 2: The applicant provided a Tier 2 description of the containment penetration areas and the AEES in DCD Tier 2, Section 6.2.3, "Secondary Containment Functional Design," summarized here, in part, as follows:

The US-APWR design does not utilize a secondary containment. Rather than a secondary containment, portions of the primary containment are enclosed by containment penetration areas which function to prevent the direct release of containment atmosphere to the environment through the containment penetrations. Containment penetration areas are served by the auxiliary building heating, ventilation, and air conditioning (HVAC) system during normal operation and by the AEES following a DBA. The AEES maintains the containment penetration areas at a negative pressure during accident conditions.

ITAAC: The ITAAC acceptance criteria for the AEES are specified in DCD Tier 1, Table 2.7.5.2-3, "Engineered Safety Features Ventilation System Inspections, Tests Analyses, and Acceptance Criteria." Among other tests, the ITAAC will confirm that an ECCS actuation signal will isolate the Auxiliary Building Ventilation System, (which is the normal operations HVAC system that services the penetration areas, and will start the AEES trains. The ITAAC will also confirm that the AEES provides a negative pressure of at least 6.35 mm (0.25 in.) water gauge, relative to adjacent areas within 240 seconds from the initiation signal.

TS: DCD Tier 2, Chapter 16, Section 3.7.11, "Annulus Emergency Exhaust System," provides limiting conditions for operation and surveillance requirements for the AEES. In addition, it includes required actions in the event that the secondary containment boundary is inoperable.

Topical Reports: There are no topical reports for this area of review.

Technical Reports: MHI Technical Report MUAP-10020, "US-APWR Safety-Related Air Conditioning, Heating Cooling, and Ventilation Systems Calculations," Revision 1, issued March 2011.

US-APWR Interface Issues identified in the DCD: There are no US-APWR interface issues for this area of review.

Site Interface Requirements identified in the DCD: There are no site interface requirements for this area of review.

Cross-cutting Requirements (TMI, USI/GSI, Op Ex): None for this area of review.

RTNSS: There is no RTNSS for this area of review.

10 CFR 20.1406: There are no 10 CFR 20.1406 requirements for this area of review.

CDI: There is no CDI for this area of review.

6.2.3.3 Regulatory Basis

The relevant requirements of the Commission's 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 US-APWR application contain the proposed ITAAC that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the DC is built and will operate in accordance with the DC, the provisions of the Atomic Energy Act of 1954, and NRC regulations.

Acceptance criteria adequate to meet the above requirements include:

1. 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.

Based on review of DCD Tier 2 Section 6.2.3, the staff issued **RAI 990-7011, Question 06.02.03-1, Part 1,** which requested that the applicant state in the DCD or indicate where in the DCD the design basis section of the secondary containment is discussed, including what commission regulations apply to the design, i.e., those regulations discussed in NUREG-0800 SRP Section 6.2.3, Section II acceptance criteria.

In its response to RAI 990-7011, Question 06.02.03-1, Part 1, dated April 4, 2013, the applicant stated that DCD Section 6.2.3 will be revised to include a Design Basis, System Design, Design Evaluation, Tests and Inspection and Instrumentation Requirements section for the Secondary Containment System in order to be consistent with NRC RG 1.206, "Combined License Applications for Nuclear Power Plants"

The staff has reviewed the applicant's response to this RAI and the accompanying DCD markup and finds it acceptable because the added DCD Design Basis Section 6.2.3.1, states the relevant requirements of NRC regulations which the Secondary Containment is designed to meet. The staff found that the stated regulations conform to the areas of review and the associated acceptance criteria specified in NUREG-0800, Section 6.2.3. **RAI 990-7011, Question 06.02.03-1, Part 1 is being tracked as a Confirmatory Item** to ensure the DCD is revised accordingly.

6.2.3.4 Technical Evaluation

6.2.3.4.1 GDC 4 Requirements to protect Structures, Systems and Components Important to Safety Against Dynamic Effects

As described in Section 6.2.4 of this report, "Containment Isolation System." the staff found that since the design of the piping from containment in the annulus and the penetration areas meets the acceptance criteria for design stress and fatigue limits for ASME Code Section III, Class 2 piping as described SRP Section 3.6.2, high energy lines passing through containment are designed conservatively to preclude a breach of piping integrity and therefore guard pipes are not required on these lines. The staff review of system piping against SRP 3.6.2 criteria is in Section 3.6.2 of this report. Therefore, the staff finds that the US-APWR 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.

6.2.3.4.2 Annulus Emergency Exhaust System

As stated in DCD Tier 2, Section 6.5.1, the AEES is one of the ESF filter systems and is designed for fission product removal and retention by filtering the air as it exhausts from the Penetration Areas of the R/B. This is accomplished by maintaining sub-atmospheric conditions in those areas when required. The AEES consists of two 100 percent capacity trains.

One of the 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. As stated in DCD Tier 2, Table 3D-2, the AEES components are located inside the R/B, which is a Seismic Category I Structure.

6.2.3.4.3 Reactor Building

As described in DCD Tier 2, Section 3.8.4.1.1, "R/B," the R/B has five main floors, and surrounds the PCCV. The Reactor Building contains the "annulus". The annulus consists of concrete walled areas around the PCCV that serves a secondary containment function. The annulus is made up of all areas with containment penetrations. These areas are further defined in DCD Tier 2, Figures 6.5-2 through 6.5-9, and DCD Tier 2, Figures 9.4.5-1 and 9.4.5-3. The annulus is maintained at a slightly negative pressure to control the release of any radioactive materials to the environment. This function is achieved by the annulus compartment boundaries in conjunction with the AEES, which maintains the annulus areas at sub-atmospheric pressure under postulated post-accident conditions.

As stated in DCD Tier 2 Section 3.8.4.1.1, the Reactor Building is a Seismic Category 1 Structure. The Reactor Building is a reinforced concrete structure. The Reactor Building surrounds the PCCV.

6.2.3.4.4 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 calls for staff review of the applicant's analysis of the pressure and temperature response of the secondary containment to a LOCA in the primary containment.

As described in DCD Tier 2, Chapter 16 Subsection B 3.7.11, the design basis capacity of the AEES is based on the large break LOCA safety analysis. In conjunction with the LOCA inside containment, the system evaluation assumes a passive failure outside containment, such as valve packing leakage and a passive failure of the ECCS outside containment, such as SI seal pump leakage. The radiological consequence analysis is provided in DCD Tier 2, Chapter 15 Section 15.6.5.5, "Radiological Consequences."

As described in MUAP-10020 Revision 1, "US-APWR Safety-Related Air Conditioning, Heating Cooling, and Ventilation Systems Calculations," the applicant provided calculations of the airflow requirements of the AEES, and the heat removal requirements for the safety related HVAC systems that serve to remove heat from within the annulus compartment boundaries. The analysis was performed by the applicant to demonstrate the ability of the AEES to depressurize the R/B and maintain sub-atmospheric pressure in the R/B annulus compartment boundaries following a design-basis LOCA. This calculation defines the annulus compartment boundary as that which encompasses the safeguard component areas and penetration areas. The LOCA is assumed to occur concurrently with a LOOP and loss of one of the AEES trains.

Based on review of the of MUAP-10020, Revision 1, and DCD Tier 2 Section 9.4.5, "Engineered Safety Feature Ventilation System," the staff found that any heat load due to heat transfer from the primary to the secondary containment would be removed via the Safeguard Component Area Air Handling Units (AHUs) and the Penetration Areas AHUs which are part of the Safeguard Component Area HVAC System and the AEES respectively. Tables 5.3.1-1, "SCAVS Cooling and Heating Loads," and 5.5.1-4, "SRCAVS (Annulus Area) Calculation Sheet," of MUAP-10020, Revision 1, summarizes the heat gains to the Safeguard Component Areas and Penetration Areas respectively due to heat transfer through concrete structures. Table 2.4.5-1, "Heat Gain from Piping and Supports," shows values for heat gain from piping and supports for the Safeguard Component Area and Penetration Area, but the basis of either of these sets of values are not provided. The calculation assumes heat transfer through walls exposed to the outside environment, but it is not apparent if heat transfer via conduction from the primary containment structure to the secondary containment structure is considered.

NUREG-0800 Section 6.2.3 recommends that heat transfer from the primary containment atmosphere to the primary containment structure should be conservative, like those in BTP 6-2. Based on review of the MUAP-10020 Revision 1, supplied to the staff, the staff issued **RAI 990-7011, Question 06.02.03-1, Part 2,** which requested that the applicant state in the DCD or indicate where in the DCD it is stated:

> What calculation was used as the basis for determining the pressure and temperature response of the secondary containment (i.e., add a reference MUAP-10020, or other calculation).

How heat transfer from the primary containment atmosphere to the secondary containment was considered, what heat transfer coefficients were used, and if selection of the coefficients conform to BTP 6-2 guidance.

If conductive heat transfer through the primary containment structure and convective heat transfer to the secondary containment atmosphere was considered when developing the heat loads due to structures shown in MUAP-10020 Revision 1 Tables 5.3.1-1 and 5.5.1-4.

In its response to RAI 990-7011, Question 06.02.03-1, Part 2, dated April 4, 2013, the applicant stated that DCD Section 6.2.3 will be revised to describe how heat transfer from the containment to the containment penetration areas was considered. The applicant added Section 6.2.3.3, "Design Evaluation" of the DCD to state Technical Report MUAP-10020 was the basis of the heat transfer analysis used to establish the required capacity of the penetration area air handling units. In the applicant's RAI response, the applicant stated that the heat gain in the penetration areas from the containment was calculated by assuming a steady-state maximum containment temperature. The applicant used steady state heat transfer coefficients for inside and outside surfaces and a constant conduction heat transfer coefficient through the PCCV wall in lieu of BTP 6-2 guidance, which varies with time. The applicant stated that although BTP 6-2 guidance was not used for the conduction and convection heat transfer coefficients, other conservative assumptions such as use of the steady state maximum containment temperature and a fifteen percent margin on the calculated heat gain provide a conservative result. The staff has reviewed the applicant's response to this RAI and the accompanying DCD markup and finds it acceptable because the added DCD Design Evaluation Section 6.2.3.3, states the basis for the heat removal capability for secondary containment, including a referenced calculation. The staff finds that the applicant used a sufficiently conservative heat transfer calculation method for this application. **RAI 990-7011, Question 06.02.03-1, Part 2 is being tracked as a Confirmatory Item** to ensure the DCD is revised accordingly.

Based on review of MUAP-10020, Revision 1, as clarified by the response to RAI 990-7011, Question 06.02.03-1, Part 2, the staff found that the applicant has considered heat transfer to the primary to the secondary containment when designing the functional capability of the secondary containment. The applicant used sufficiently conservative heat transfer coefficient for conduction and convection for this application. Consequently the staff finds that that Acceptance Criterion 1A is met.

Based on review of the of MUAP-10020 Revision 1, the staff found that because the applicant assumed a constant, zero percent exceedance maximum temperature to accommodate all environmental conditions, and used a conservative, constant surface temperature for exterior roofs and walls, the applicant assumed adiabatic boundary conditions for the surface of the secondary containment structure exposed to the outside environment. Therefore the staff finds that NUREG-0800, Section 6.2.3 AC 1B is met.

Based on the review of Technical Report MUAP-10020, Revision 1, Section 4.1.3, the staff found that the applicant has considered the compressive effect of the primary containment expansion on the secondary containment atmosphere. Therefore the staff finds that the volume decrease and the resulting pressure increase in the penetration areas was considered in accordance with NUREG-0800, Section 6.2.3 AC 1C. Based on the above, the staff finds that this acceptance criterion is met.

NUREG-0800 Section 6.2.3 recommends that secondary containment in-leakage should be considered in the analysis. As stated in DCD Tier 2, Chapter 15 Table 15.6.5-4, a bounding inleakage of 0.15 percent air weight per day from the primary containment was used. The staff found that the value of inleakage to the secondary containment used in the radiological consequence analysis conservative with respect to the maximum value allowed for such leakage in Appendix J tests as described in DCD Tier 2, Chapter 16 GTS Section 5.5.16, “Containment Leakage Rate Testing Program” (0.10 percent air weight per day).

Based on review of the MUAP-10020 Revision 1, supplied to the staff, the staff issued **RAI 990-7011, Question 06.02.03-1, Part 3** which requested that the applicant state in the DCD or indicate where in the DCD it is stated:

> The values and the justification for the assumed inleakage values to the secondary containment spaces.

In its response to RAI 990-7011, Question 06.02.03-1, Part 3, dated April 4, 2013, the applicant stated the assumed inleakage value to the secondary containment. The applicant indicated this value was proprietary information. The applicant stated this value is based on Japanese power plant applications. The applicant stated that initial testing and routine surveillance testing will prove the capability to achieve the design requirement secondary containment negative pressure within the timeframe assumed in the accident analyses. Therefore, the staff finds that the applicant considered secondary containment in-leakage in the analysis used to determine AEES emergency filtration unit fan capacity and that sufficient testing exists to ensure that these areas will function as designed. Accordingly, the staff finds that the applicant adequately addressed this issue and, therefore, considers **RAI 990-7011, Question 06.02.03-1, Part 3** resolved**.**

Therefore the staff finds the values for assumed inleakage values to the secondary containment spaces were considered in the analysis. Consequently, the staff finds that NUREG-0800, Section 6.2.3 AC 1D is met.

Based on review of MUAP-10020, Revision 1, Section 4.1.3, the staff found that the applicant did not take any credit for secondary containment out-leakage. Therefore, the staff finds that NUREG-0800, Section 6.2.3 AC 1E is met.

Based on the review of MUAP-10020, Revision 1, Section 5.6.1, the staff found that the AEES Exhaust fan airflow requirement is determined through a calculation that assumes a draw down the Safeguard Component Areas and Penetration Areas pressure to -6.35 mm (-0.25 in.) water gauge in 180 seconds. The staff found that this time is much less than the is less than the 240 second draw down time assumed in the LOCA radiological consequence analysis, as is therefore conservative with respect to the system performance assumed in the radiological analyses. The calculation also imposed a 15 percent capacity margin on this airflow requirement. Also, MUAP-10020 Revision 1, Section 5.5.2.5 shows that the safety-related AHUs that remove heat from the Safeguard Component Areas and Penetration Areas are sized with at least 15 percent margin. The staff finds that the applicant’s analysis complies with NUREG-0800, Section 6.2.3, Acceptance Criterion 1F, because, by considering the flow rate provided by only one AEES train, the analysis assumes the LOOP and the most severe single active failure in the secondary containment filtration system, with ample margin.

Based on review of MUAP-10020, Revision 1, Section 2, the staff finds that heat loads generated within the secondary containment is considered in the design requirements for the secondary containment functional design. Therefore the staff finds that the design complies with NUREG-0800, Section 6.2.3, Acceptance Criterion 1G.

Based on review of MUAP-10020, Revision 1, Section 2, Section 5.6.1, the staff finds that AEES fan performance characteristics were considered when the applicant determined the period of time for secondary containment depressurization and steady state secondary containment negative pressure. The staff found that the AEES Exhaust fan airflow requirement is determined through a calculation that assumes a draw down the Safeguard Component Areas and Penetration areas pressure to -6.35 mm (-0.25 in.) water gauge in 180 seconds. The staff found that this time is much less than the 240 second draw down time assumed in the LOCA radiological consequence analysis, as is therefore conservative with respect to the system performance assumed in the radiological analyses. The calculation also imposed a 15 percent capacity margin on this airflow requirement. The system including the fan is conservatively sized to achieve a steady state negative pressure of -10.16 mm (-0.4 in.) water gauge.

Based on the above review, the staff judged that several design assumptions regarding AEES design capacity stated in MUAP-10020, Revision 1, should be included in the design basis description of the system in DCD Tier 2. The staff issued **RAI 990-7011, Question 06.02.03-1, Part 3** which requested that the applicant state in the DCD or indicate where in the DCD it is stated:

> That the AEES fan is conservatively designed to achieve a steady state negative pressure of 10.16 mm (-0.4 in.) water gauge in the secondary containment spaces.
>
> The values and the justification for the assumed inleakage values to the secondary containment spaces.

As stated above, in its response to RAI 990-7011, Question 06.02.03-1, Part 3, the applicant provided a response satisfactory to the staff. Accordingly, the staff finds that the applicant adequately addressed this issue and, therefore, considers RAI 990-7011, Question 06.02.03-1, Part 3 resolved.

The staff notes that AEES system performance with respect to minimum draw down time and minimum acceptable negative pressure are verified via system test controlled by preoperational test described in Tier 2, Section 14.2.12.1.70, "Annulus Emergency Exhaust System Preoperational Test," and by an ITAAC test described in Tier 1 ITAAC Table 2.7.5.2-3. The staff notes that satisfactory system performance is periodically monitored and verified via TS Surveillance requirements in DCD Tier 2, Chapter 16, Section 3.7.11.

The staff issued **RAI 990-7011, Question 06.02.03-1, Part 3,** which requested that the applicant state in the DCD or indicate where in the DCD it is stated how or if the as-built fan characteristic curve is or will be used to evaluate secondary containment leak tightness.

As stated above, in its response to RAI 990-7011, Question 06.02.03-1, Part 3, the applicant provided a response satisfactory to the staff. The applicant stated that initial testing and routine surveillance testing will prove the capability to achieve the design requirement secondary containment negative pressure within the timeframe assumed in the accident analyses. Accordingly, the staff finds that the applicant adequately addressed this issue and, therefore, considers RAI 990-7011, Question 06.02.03-1, Part 3 resolved.

The staff finds that these verifications provide assurance that the system will perform when required such that assumptions with regard to secondary containment depressurization time and pressure used in the radiological analyses are met. Therefore the staff finds that the design complies with NUREG-0800, Section 6.2.3, "Acceptance Criterion 1H."

Based on the above discussion, the staff finds that the US-APWR design meets NUREG-0800, Section 6.2.3, Acceptance Criteria 1A through 1H, consequently, the staff finds that GDC 16 requirements for functional capability of the secondary containment by using an analysis of pressure and temperature with meets the position of BTP 6-2 as described above or contains assumptions with suitably conservative margin.

6.2.3.4.5 Annulus Emergency Exhaust 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."

As stated in DCD Section 9.4.5.1.1.1, the Auxiliary Building HVAC system is the full capacity normal operation filtration train designed to maintain the annulus at a negative pressure with respect to adjacent areas and to maintain annulus temperature between 10 °C (50 °F) and 40.5 °C (105 °F). Sub-atmospheric pressure prevents the escape of unfiltered air to the outside environment. The Annulus Emergency Exhaust filtration trains are not required to function during normal operation.

The staff finds that the Auxiliary Building HVAC system conforms to the guidelines of RG 1.52. The staff review off the Auxiliary Building HVAC System with respect to conformance to RG 1.52 is in Section 9.4.3 of this report.

6.2.3.4.6 Annulus Emergency Exhaust System - Accident Operation

As described in DCD Tier 2, Section 9.4.5, the Annulus Emergency Exhaust System is designed for fission product removal by filtering the air it exhausts from the penetration areas and safeguard component areas following an accident.

Each of the two annulus exhaust filtration units contain a motor-controlled damper, pre-filter, upstream high-efficiency particulate air (HEPA) filter, fan, motor-controlled damper, 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 240 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, an ECCS actuation signal causes both AEES trains to energize. The normal filtration train supply and exhaust dampers close. Both accident trains start on receipt of the containment isolation signal.

As stated in DCD Tier 2, Section 9.4.5.1.1.1, "Annulus Emergency Exhaust System," the AEES emergency filtration units (EFUs) are designed in accordance with RG 1.52. Each AEES EFU is designed to reduce and maintain annulus pressure as specified in the TS, and is located in a Seismic Category 1 building. The staff finds that the AEES EFUs meet the guidance of RG 1.52. Details of the staff review of ESF filter systems, including the AEES EFUs and how they meet the guidance of RG 1.52, are in Section 9.4.5 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 AEES EFUs meet 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 DBA even if exfiltration occurs. If the leakage rate exceeds 100 percent of the volume per day, there should be a special exfiltration analysis. Based on staff review of DCD Tier 2, Chapter 6, Figures 6.5-2 through 6.5-9, which consists of plan drawings of the R/B, including the Penetration Areas and the Safeguard Component Areas of the R/B that compose the "annulus", the staff found that there are no penetrations or doors exposed to the environment that are served by the AEES. Therefore, the secondary containment areas 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 DCD Tier 1, Table 2.7.5.2-2, indication of the position of AEES dampers is provided in the main control room (MCR). DCD Tier 2, Section 14.2.12.1.70, "Annulus Emergency Exhaust System Preoperational Test," shows the AEES will be tested to demonstrate the ability to lower pressure. DCD Tier 2, Chapter 16 TS LCO 3.7.11 Required Action B.1 states that the boundary of spaces served by the AEES must be restored to operable status within 24 hours of discovery, with a note stating that the boundary may be opened intermittently under administrative control. NUREG-0800, Section 6.2.3 recommends administrative control and indication of openings in the secondary containment in the MCR.

Based on the review of the above information in the DCD, the staff issued **RAI 990-7011, Question 06.02.03-1, Part 4** which requested that the applicant:

- Describe in the DCD or indicate where in the DCD it is described how preoperational tests (DCD Tier 2, Section 14.2.12.1.70) will confirm the evaluated effect of open doors or hatches on the functional capability of the depressurization and filtration system.

- Describe in the DCD Tier 2 Section 6.2.6 that all secondary containment space openings like personnel doors and equipment hatches are under administrative control, or have position indicators and alarms in the MCR. In addition, the staff requested that the applicant describe how MCR operators would use this or other MCR instrumentation to identify a degraded/inoperable secondary containment barrier (i.e., discovery of and entry into TS 3.7.11 Condition B).

In its response to RAI 990-7011, Question 06.02.03-1, Part 4, dated April 4, 2013, the applicant stated that DCD Subsection 14.2.12.1.70.C.5, "Test Method" will be revised to add the requirement to confirm that a uniform minimum negative pressure is attained throughout the containment penetration areas and safeguard component areas by measuring internal pressure at several locations. The applicant submitted DCD Section 6.2.3.5 "Instrumentation Requirements" to the staff which states that the pressure in the containment penetration areas is monitored and stored by the process computer in the MCR. In the applicant's RAI response, the applicant stated that the doors and hatches are under administrative control. The program is described in DCD Subsection 13.5. The staff found that the program, as described in this section of the DCD in addition to COL item 13.5(1) is sufficient to ensure that personnel doors and equipment hatches are under administrative control. In the applicant's RAI response, the applicant stated that routine surveillance testing of secondary containment pressure would identify degraded/inoperable secondary containment barriers. The staff reviewed the RAI response including the associated DCD mark-ups and finds them acceptable because the DCD description will ensure that secondary containment barriers will be under administrative control and that preoperational and surveillance tests exist to ensure that degraded/inoperable secondary containment barriers can be identified. **RAI 990-7011, Question 06.02.03-1, Part 4 is being tracked as a Confirmatory Item** to ensure the DCD is revised accordingly.

Therefore, based on the DCD 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. The staff finds that the US-APWR secondary containment only partially encloses the primary containment. The staff found that the applicant has provided the magnitude of unprocessed primary containment leakage that is assumed to bypass the secondary containment and is assumed to be released to the environment in the event of a DBA. This amount is conservatively assumed to be the total amount of primary containment Type C leakage. The applicant has specified special testing requirements to ensure that Type C leakage does not exceed a specified value. This special testing requirement is included in the Appendix J testing program described in DCD Tier 2, Section 6.2.6 and in DCD Tier 2, Chapter 16 Section 5.5. As described in Section 6.2.6 of this report the staff finds the special testing requirements for secondary containment bypass leakage acceptable and therefore the staff finds that quantitative credit for holdup by the secondary containment of fission product releases from the primary containment is acceptable. Therefore the staff find that NUREG-0800, Section 6.2.3, Acceptance Criterion 3D, is met.

As stated in DCD Tier 2, Section 3.8.4, the R/B 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 R/B and the staff evaluation and findings as they relate to the external design pressure margin above the maximum expected external pressure of the R/B 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.

DCD Tier 2, Table 14.3-1f, “Radiological Analysis Key Design Features,” describes that the PCCV facility consists of the containment vessel and the annulus that encloses the containment penetration area, and provides an efficient leak tight barrier. The staff found that this feature is verified via testing of the drawdown time and pressure of the AEES system to ensure that it achieves the required negative pressure of 6.35 mm (0.25 in.) H_2O gauge pressure in ≤240 seconds from initiation of signal. These tests include preoperational test described in Tier 2, Section 14.2.12.1.70 and by an ITAAC test described in Tier 1 ITAAC Table 2.7.5.2-3. The staff notes that satisfactory system performance is periodically monitored and verified via TS Surveillance requirements in DCD Tier 2, Chapter 16, Section 3.7.11. The staff finds that these verifications provide assurance that the system will perform when required such that assumptions with regard to secondary containment depressurization time and pressure used in the radiological analyses are met. This ventilation draw-down performance is required to satisfy Radiological Consequences of DBAs described in DCD Tier 2, Section 15.6.5, “Loss of Coolant Accidents Resulting from Spectrum of Postulated Piping Breaks within the Reactor Coolant Pressure Boundary.” In Section 15.6.5.5.1.1, the applicant states that the AEES prevents uncontrolled radioactive release from the containment penetration and safeguard components to the environment.

Therefore, with the exception of the above confirmatory items, the staff finds that the US-APWR meets the NUREG-0800, Section 6.2.3, Acceptance Criterion 3.

6.2.3.4.7 GDC 43 and 10 CFR Part 50, Appendix J, Criteria for Secondary Containment System Testing.

The staff reviewed the US-APWR 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.

6.2.3.4.8 Bypass Leakage

Bypass leakage is leakage that bypasses the Safeguard Component Area and Containment Penetration Areas and escapes to the environment. Potential bypass leakage paths exist through containment isolation valve seat leakage, the double seals of the fuel transfer tube, and containment ventilation system isolation valves.

In DCD Tier 2, Revision 3, Section 6.2.6.5, the applicant stated that because the US-APWR design does not have a secondary containment, there are no special testing requirements for this design feature. However, based on staff review of DCD Tier 2, Revision 3, Sections 6.2.6, 6.2.3, 6.5.3.2, and the Chapter 15 accident analyses, the staff determined that some credit is taken by the applicant for the action of the AEES in concert with the Safeguard Component Areas and Containment Penetration Areas to capture primary containment leakage through an ESF filter system prior to release to the environment. As stated in DCD Tier 2 Table 15.6.5-4, the LOCA radiological consequence analysis assumes a primary containment leak rate of 0.15 percent of the containment air weight per day for the first 24 hours after a postulated accident. The analysis also assumes that 50 percent of this 0.15 percent per day leak rate, or 0.075 percent of the containment air weight per day at Pa, is captured and processed for particulates through a 99 percent efficient ESF filtration system prior to release to the environment. The remaining fifty percent of the primary containment particulate leakage (0.075 percent of the containment air weight per day at P_a), and 100 percent of the primary containment gaseous source term leakage is released directly to the environment. The staff noted that 10 CFR Part 50, Appendix J Part IV B states that multiple barrier containments shall be subject to individual leakage rate tests in accordance with procedures specified in the technical specifications and associated bases. SRP Section 6.2.3 states that primary containment bypass leakage paths are to be identified and associated bypass leakage rates are to be determined.

In **RAI 866-6149, Question 06.02.06-34**, the staff requested that that the applicant clarify US-APWR DCD Revision 3, Tier 2 Section 6.2.6.5, to be consistent with DCD Sections 6.2.3, 6.5.3.2, and the Chapter 15 accident analyses, as they apply to which SSCs outside of the primary containment structures that function to collect and process assumed primary containment leakage. The staff requested the applicant provide testing requirements for such SSCs or justify why special testing requirements to verify performance are not required.

In this RAI, the staff also requested that the applicant address SRP 6.2.3 Acceptance Criterion 4A and 4B as they apply to GDC 43 and 10 CFR Part 50, Appendix J, requirements for secondary containment system testing which state:

A. “The fraction of primary containment leakage bypassing the secondary containment and escaping directly to the environment should be specified. BTP 6-3 provides guidance for detecting leakage paths to the environment which may bypass the secondary containment. The periodic leakage rate testing program for measuring the fraction of primary containment leakage that may directly bypass the secondary containment and other contiguous areas served by ventilation and filtration systems should be described. Individual tests should be according to procedures from technical specifications or their bases.”

With regard to this criterion, the staff requested the applicant to quantify the bypass leakage paths in the secondary containment design in DCD Section 6.2.3. The staff requested the applicant describe the periodic leakage rate testing program for measuring the fraction of primary containment leakage that may directly bypass the containment penetration areas and annulus, served by the AEES in DCD Section 6.2.6.5. The staff also requested the applicant clarify the description of this program, in DCD Chapter 16, TS Section 5.5.16.

B. “There should be provisions in the design of the secondary containment system for inspections and monitoring of the functional capability. Preoperational and periodic test programs determine the depressurization time, the secondary containment in-leakage rate, the uniformity of negative pressure throughout the secondary containment, and other contiguous areas, and the potential for ex-filtration.”

With regard to the above criterion, the staff requested the applicant clarify DCD Tier 2 Section 6.2.3 to describe the provisions for secondary containment functional capability as they relate to the Containment Penetration Areas and the Safeguard Component Areas, served by the AEES.

In its response to **RAI 866-6149, Question 06.02.06-34**, dated January 6, 2012, the applicant confirmed the staff understanding that the penetration areas within the R/B in conjunction with the AEES function to contain and filter for particulates any primary containment leakage to these areas. The applicant stated that all containment penetrations and containment isolation valves with exception of the main steam and feedwater penetrations are enclosed by the penetration areas and do not extend beyond secondary containment barriers provided by these areas. The applicant stated that the AEES in conjunction with the penetration areas act as a partial secondary containment for the US-APWR design. The applicant provided DCD markups of Section 6.2.3 and Table 6.2.4-3 that discuss penetrations that are potential secondary containment bypass leakage paths.

In its RAI response, the applicant stated that all potential bypass leakage paths are tested as part of the Primary Containment Leakage Test Program (CLRT), and this program is the means to quantify and track the amount of potential leakage bypassing the AEES to ensure that the assumptions in the safety analyses are met. The applicant provided DCD markups of Section 6.2.6.5 to include this discussion.

In its RAI response, the applicant stated that Type C leakage as monitored by the CLRT provides a conservative measurement of secondary containment bypass leakage, and Type C test results for individual valves as part of this program will be used to ensure that the total secondary containment bypass leakage is maintained below the bypass leakage amount assumed in the offsite dose analysis. The staff determined that the DCD needed further clarification to reflect the RAI response and issued a follow up RAI.

In a follow up **RAI 918-6361, Question 06.02.06-35**, the staff requested that that the applicant clarify DCD Tier 2, Revision 3, Section 6.2.6.5 to state the additional acceptance criteria that will be used as part of the CLRT to ensure that assumptions used in the safety analyses regarding the secondary containment are met.

In its response to **RAI 918-6361, Question 06.02.06-35**, dated June 7, 2012, the applicant stated that Section 6.2.6.5 of the DCD will be revised to state that leakage paths that may bypass the penetration areas and the AEES are identified in Tier 2 Table 6.2.4-3, "Description of Radiation Shielding for the Control Room in a LOCA," and are Type C tested as part of the containment leakage rate test program with an additional acceptance criterion to ensure that the assumptions of the safety analyses regarding secondary containment performance are met. The total combined leakage from all Type C tests shall be below the amount assumed for bypass leakage to the environment stated in DCD Tables 6.5-5 and 15.6.5-4. The applicant provided a secondary containment bypass leakage test criterion value of < 0.50La for Type C tests. The applicant provided DCD markups of DCD Tier 2, Section 6.2.6.5 to include this discussion and Tier 2, Chapter 16, Section 5.5.16 to include the additional Type C acceptance criterion.

In a follow up **RAI 966-6811, Question 06.02.06-36**, the staff requested that the applicant clarify the proposed revision of DCD Tier 2, Revision 3, Chapter 16 TS Section 5.5.16, provided in the response to **RAI 918-6361, Question 06.02.06-35**, to indicate what leakage rate acceptance criteria apply to the Primary CLRT and which criteria apply to the special testing requirements associated with secondary containment functional performance. In addition, the staff indicated that staff findings on secondary containment functional performance will be reviewed by the staff under Section 6.2.3 of the staff's SER.

In its response to **RAI 966-6811, Question 06.02.06-36**, dated November 6, 2012, the applicant clarified the proposed revision of DCD Tier 2, Revision 3, Chapter 16 TS Section 5.5.16, provided in the response to **RAI 918-6361, Question 06.02.06-35**, to indicate what leakage rate acceptance criteria apply to the Primary CLRT and which criteria apply to the special testing requirements associated with secondary containment functional performance.

The staff has reviewed the applicant's response to this RAI and finds it acceptable because the applicant clearly separated the acceptance criteria of the primary containment leakage tests from that of the secondary containment in Chapter 16 TS Section 5.5.16. Thus, **RAI 918-6361, Question 06.02.06-35, is resolved and closed. RAI 866-6149, Question 06.02.06-34, is resolved and closed, and RAI 966-6811, Question 06.02.06-36 is being tracked as a Confirmatory Item** to ensure that the DCD is revised accordingly.

The staff found that the AEES in conjunction with the penetration areas in the R/B function as a partial secondary containment to capture some primary containment leakage. The staff found that the additional acceptance criteria in the CLRT in conjunction with the Tier 2, Chapter 16, Section 3.7.11 TS surveillance requirements associated with the AEES assure performance of the secondary containment function when required. The staff's review of the AEES is discussed in Sections 6.5.1 and 9.4.5 of this SER. The staff's review of the Containment Leakage Test Program, including the secondary containment bypass leakage test acceptance criterion is discussed in Section 6.2.6 of this SER.

Therefore the staff finds that the partial secondary containment design meets NUREG-0800, Section 6.2.3, Acceptance Criteria 4A as it relates to requirements that secondary containment leakage rate tests conform to procedures specified in the TS so that bypass leakage paths are identified and associated bypass leakage rates are determined. Based on conformance with NUREG-0800, Section 6.2.3, Acceptance Criterion 4A, the staff finds that the US-APWR 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 TS so that bypass leakage paths are identified and associated bypass leakage rates are determined.

6.2.3.4.9 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. The staff found that this feature is verified via testing of the drawdown time and pressure of the AEES system to ensure that it achieves the required negative pressure of 6.35 mm (0.25 in.) H_2O gauge pressure in ≤240 seconds from initiation of signal. These tests include preoperational test described in Tier 2, Section 14.2.12.1.70 and by an ITAAC test described in Tier 1 ITAAC Table 2.7.5.2-3. The staff notes that satisfactory system performance is periodically monitored and verified via TS Surveillance requirements in DCD Tier 2, Chapter 16, Section 3.7.11.

Based on the review of the above information in the DCD, the staff issued **RAI 990-7011, Question 06.02.03-1 Part 4** which requested that the applicant:

- Describe in DCD or indicate where in the DCD it is described how preoperational tests (DCD Tier 2, Section 14.2.12.1.70) will confirm a uniform negative pressure throughout the secondary containment spaces. Please specify in the DCD if the acceptance criteria for this test includes verification of adequate negative pressure and drawdown time in all Safeguard Component Areas and Containment Penetration Areas.

In its response to RAI 990-7011, Question 06.02.03-1, Part 4, dated April 4, 2013, the applicant stated that DCD Subsection 14.2.12.1.70.C.5, "Test Method," will be revised to add the requirement to confirm that a uniform minimum negative pressure is attained throughout the containment penetration areas and safeguard component areas by measuring internal pressure at several locations. The applicant submitted DCD Section 6.2.3.5, "Instrumentation Requirements," to the staff which states that the pressure in the containment penetration areas is monitored and stored by the process computer in the MCR. The staff reviewed the RAI response including the associated DCD mark-ups and finds them acceptable because the DCD description will ensure that surveillance testing will confirm a uniform negative pressure throughout the secondary containment spaces. **RAI 990-7011, Question 06.02.03-1 Part 4 is being tracked as a Confirmatory Item** to ensure the DCD is revised accordingly.

The staff finds that these verifications provide assurance that the system will perform when required such that assumptions with regard to secondary containment depressurization time and pressure used in the radiological analyses are met. Therefore, the staff finds that the US-APWR 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 4A and 4B, the staff finds that the requirements of GDC 43, and 10 CFR Part 50, Appendix J, 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.

6.2.3.4.10 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 AEES ITAAC meet the guidance of SRP Section 14.3 AC 1 through 5. Therefore the staff finds that ITAAC acceptance criteria given in DCD Tier 1, Table 2.7.5.2-3, "Engineered Safety Features Ventilation Systems Inspections, Tests, Analyses, and Acceptance Criteria," will adequately demonstrate the acceptability of the AEES design, which assures the safety function of the secondary containment. The requirements of 10 CFR 52.47(b)(1) are met in that the DCD contains proposed ITAAC that are necessary and sufficient to provide reasonable assurance that a plant that incorporates the DC is built and will operate in accordance with the DC.

6.2.3.4.11 TS

TS relating to the AEES are presented in DCD Tier 2, Chapter 16, Section 3.7.11. The staff found that the TS to verify the functional capability of the AEES ESF filtration trains along with their associated room boundaries. 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 TS requirements adequate to verify that the AEES and associated boundaries 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 DCD. No additional COL information items need to be included in DCD 2, Table 1.8-2 for secondary containment functional design consideration.

6.2.3.6 Conclusions

The staff reviewed information provided in DCD Section 6.2.3 against the requirements of GDC 4, GDC 16, and GDC 43 and 10 CFR 50, Appendix J. The staff reviewed information presented in DCD Tier 1, Sections 2.7.5 and in DCD Tier 2, Chapter 16.

As discussed above the staff finds the US-APWR 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, through conformance with acceptance criteria for design stress and fatigue limits for ASME Code Section III, Class 2 piping as described Section 3.6.2 of this report. The staff finds the US-APWR design meets GDC 16 requirements as they relate to containment and associated systems establishing an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment, by meeting SRP 6.2.3 acceptance criteria 1A through 1H and 3A through 3E as described above. The staff finds the US-APWR design meets GDC 43 requirements as they relate to the reactor containment and associated systems designed to permit appropriate periodic pressure and functional testing to assure structural integrity and operability and the staff finds the US-APWR design meets 10 CFR 50, Appendix J as it relates to secondary containment leakage rate testing in accordance with the procedures specified in the technical specifications and associated bases, so that bypass leakage paths are identified and associated bypass leakage rates are determined, by meeting SRP 6.2.3 acceptance criteria 4A and 4B as described above.

6.2.4 Containment Isolation System

6.2.4.1 Introduction

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

6.2.4.2 Summary of Application

DCD Tier 1: The Tier 1 information pertaining to the CIS is given in DCD Tier 1, Section 2.11, “Containment System.”

DCD Tier 2: The applicant provided a Tier 2 description of the CIS in DCD Tier 2, Section 6.2.4, “Containment Isolation System,” summarized here, in part, as follows:

The containment prevents or limits the release of fission products to the environment. The containment isolation system allows the free flow of normal or emergency-related fluids through the containment boundary in support of reactor operations, but establishes and preserves the containment boundary integrity. The containment isolation system includes the system and components (piping, valves, and actuation logic) that establish and preserve the containment boundary integrity. This section of the DCD describes the design and functional capabilities of the US-APWR containment isolation system in compliance with these GDC.

ITAAC: The ITAAC for the CIS are specified in DCD Tier 1, Table 2.11.2-2, “Containment Isolation System Inspections Tests, Analyses and Acceptance Criteria.”

TS: DCD Tier 2, Chapter 16, Section 3.6, “Containment Systems,” provides limiting conditions for operation and surveillance requirements for the CIS.

Topical Reports: There are no topical reports for this area of review.

Technical Reports: There are no technical reports for this area of review.

US-APWR Interface Issues identified in the DCD: There are no US-APWR interface issues for this area of review.

Site Interface Requirements identified in the DCD: There are no site interface requirements for this area of review.

Cross-cutting Requirements (TMI, USI/GSI, Op Ex):

TMI Item II.B.8, “Three- Foot Diameter Containment Penetration.”
TMI Item II.E.4.2, “Actuation and Control Features for Isolation Valves.”

TMI Item II.E.4.4, “Vent/Purge Valve Positions.”

RTNSS: There is no RTNSS for this area of review.

10 CFR 20.1406: There are no 10 CFR 20.1406 requirements for this area of review.

CDI: There is no CDI for this area of review.

6.2.4.3 Regulatory Basis

The relevant requirements of the Commission’s regulations for this area of review and the associated acceptance criteria are specified in Section 6.2.4, “Containment Isolation System,” of NUREG-0800, the SRP, and are summarized below. Review interfaces with other SRP sections can be found in SRP Section 6.2.4.

1. GDC 1, “Quality standards and records,” requires that SSC’s important to safety shall be designed, fabricated, erected and tested to quality standards commensurate with the importance of the safety function to be performed.

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

3. GDC 4, “Environmental and dynamic effects design bases,” requires that SSC’s important to safety shall be designed to accommodate the effects and to be compatible with the environmental conditions associated with normal operation, maintenance, testing and postulated accidents.

4. GDC 16, Containment design,” requires that the reactor containment and associated systems shall be provided to establish an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment and to assure that the containment design conditions important to safety are not exceeded for as long as postulated accident conditions require.

5. GDC 54, “Systems penetrating containment,” requires that piping systems penetrating the containment be provided with leak detection, isolation, and containment capabilities having redundancy, reliability, and performance capabilities which reflect the importance to safety of isolating these piping 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.

6. GDC 55, “Reactor coolant pressure boundary penetrating containment,” and GDC 56, “Primary containment isolation,” require that 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), be provided with containment isolation valves as follows, unless it can be demonstrated that the containment isolation provisions are acceptable on some other defined basis:

   a. One locked-closed isolation valve inside and one outside containment; or
   b. One automatic isolation valve inside and one locked-closed isolation valve outside containment; or
   c. One locked-closed isolation valve inside and one automatic isolation valve outside containment; or
   d. One automatic isolation valve inside and one outside containment.

7. GDC 57, "Closed systems isolation valves," requires 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.

8. 10 CFR 50.34(f)(2)(xiv)(A) requires that CIS' are provided that ensure all non-essential systems are isolated automatically by the CIS.

9. 10 CFR 50.34(f)(2)(xiv)(B) requires that CIS'are provided that for each non-essential penetration (except instrument lines) have two isolation barriers in series.

10. 10 CFR 50.34(f)(2)(xiv)(C) requires that CIS' are provided that do not result in the reopening of the containment isolation valves on resetting of the isolation signal.

11. 10 CFR 50.34(f)(2)(xiv)(D) requires that CIS'are provided that utilize a containment setpoint pressure for initiating containment isolation as low as is compatible with normal operation.

12. 10 CFR 50.34(f)(2)(xiv)(E) requires that CIS'are provided that include automatic closing on a high radiation signal for all systems that provide a path to the environs.

13. 10 CFR 50.34(f)(2)(xv) requires that CIS'are provided that provide a capability for containment purging/venting designed to minimize the purging time consistent with ALARA principles for occupational exposure, and that provide and demonstrate high assurance that the purge system will reliably isolate under accident conditions.

14. 10 CFR 50.34(f)(3)(iv) requires one or more dedicated containment penetrations, equivalent in size to a single 3 ft (0.9 m) diameter opening in order not to preclude future installation of systems to prevent containment failure such as a filtered, vented containment system.

15. 10 CFR 50.63(a)(2) requires that appropriate containment integrity is maintained in the event of a station blackout (SBO) for a specified duration.

16. 10 CFR 52.47(a)(8) requires that an application for a standard DC contain the information necessary to demonstrate compliance with any technical requirements set forth in 10 CFR 50.34(f).

17. 10 CFR 52.47(b)(1) requires that an application for a standard DC contain the proposed ITAAC that are necessary and sufficient to provide reasonable

assurance that if the inspections, tests, analyses and acceptance criteria are performed, a facility that incorporates the design certification has been constructed and will be operated in conformity with the design certification, the provisions of the Atomic Energy Act of 1954, and the Commission's rules and regulations.

18. 10 CFR 20.1406, "Minimization of Contamination," requires that applicants for a standard DC shall describe how facility design and procedure for operation will minimize, to the extent practicable, contamination of the facility and the environment, facilitate eventual decommissioning, and minimize, to the extent practicable, the generation of radioactive waste.

Acceptance criteria adequate to meet the above requirements are described in NUREG-0800, SRP 6.2.4, "Containment Isolation System", and RG 1.141, "Containment Isolation Provisions for Fluid Systems".

The applicant's CIS is acceptable if it meets the relevant regulatory guidance. This will ensure that the relevant requirements of 10 CFR 50.34, "Contents of Construction Permit and Operating License Applications; Technical Information," 10 CFR 50.63, "Loss of All Alternating Current Power," 10 CFR 52.47, "Contents of Applications; Technical Information," 10 CFR Part 50, Appendix A, GDC 1, 2, 4, 16, 54, 55, 56 and 57, and 10 CFR 20.1406 are met.

6.2.4.4 Technical Evaluation

The staff reviewed the US-APWR DCD, Revision 3, Section 6.2.4, "Containment Isolation System", in accordance with NUREG-0800, SRP6.2.4, "Containment Isolation System", Revision 3, March 2007, and RG 1.141, "Containment Isolation Provisions for Fluid Systems". In addition, US-APWR DCD Section 6.2.8, "Combined License Information," is included in this review.

The staff's review encompassed the following areas specified by Section 6.2.4 of the SRP, 10 CFR 50.34(f)(2)(xiv) (A) through (E), 10 CFR 50.34(f)(2)(xv) and 10 CFR 50.63(a)(2):

- CIS design, including:
    - the number and location of isolation valves (e.g., the isolation valve, arrangements, location of isolation valves with respect to the containment wall, purge and vent valve conformance to BTP 6-4, and instrument line conformance to RG 1.11),
    - the actuation and control features for isolation valves,
    - the normal positions of valves, and the positions valves take in the event of failures,
    - the initiating variables for isolation signals, and the diversity and redundancy of isolation signals,
    - the basis for selecting closure time limits for isolation valves,
    - the redundancy of isolation barriers,
    - the use of closed systems inside containment as isolation barriers,
- the protection provided for the CIS against loss of function caused by missiles, pipe whip, and natural phenomena,

- environmental conditions in the vicinity of the CIS and equipment and its potential effect,

- the mechanical engineering design criteria applied to isolation barriers and equipment,

- the provisions for alerting operators of the need to isolate manually controlled isolation barriers in the event of leakage or an accident,

- the provisions for, and TS pertaining to, operability and leak rate testing of isolation barriers,

- the calculation of containment atmosphere released prior to isolation valve closure for lines that provide a direct path to the environs,

- containment purging/venting requirements of 10 CFR 50.34(f)(2)(xv),

- containment penetration requirements of 10 CFR 50.34(f)(3)(iv)

The following sections provide a discussion of the staff's findings and conclusions for each of the above review areas.

6.2.4.4.1 Number, Location and Arrangement of Isolation Valves

The regulatory requirements relating to number, location, and arrangement of isolation valves serving containment piping penetrations are specified in GDC 55, “Reactor coolant pressure boundary penetrating containment," GDC 56, “Primary containment isolation,” and GDC 57, “Closed system isolation valves.” GDC 55 and 56 require two isolation valves, one inside and one outside containment, per penetration, and the valves must be locked closed or automatic, with the restriction that a simple check valve may not be used as an automatic valve outside containment. GDC 57, which applies to penetrations for which there is a closed system inside containment, requires one locked closed, automatic (but not simple check) or remote manual isolation valve outside containment. The staff reviewed the applicant’s proposed use of containment isolation valves, as described in DCD Table 6.2.4-3, “List of Containment Penetrations and System Isolation Positions,” for conformance with these GDC. The staff reviewed the valve arrangement information for each penetration and confirmed that the number, location, and arrangement conform to the acceptance criteria. DCD Table 6.2.4-3, identifies the penetrations. The staff used acceptance criteria described in NUREG-0800 Section 6.2.4, and guidance described in RG 1.141 which endorses ANSI N271-1976, “Containment Isolation Provisions for Fluid Systems”.

SRP 6.2.4, Subsection II, acceptance criteria 4 and 5 provide review guidance for review of containment isolation provisions on another design basis than GDC 55 and 56, for lines consisting of two valves located outside of containment or a single valve outside of containment respectively. This “other defined basis” for GDC 55 and 56 is described in detail in ANSI N271-1976. The staff noted that the US-APWR design does not include other design basis for GDC 55 and 56, for lines consisting of two valves located outside of containment. Therefore the staff determined that SRP 6.2.4, Subsection II, acceptance criteria 4 does not apply to the US-AWPR design.

In Sections 6.2.4.3.1, “Evaluation of Conformance to General Design Criterion 55 of 10 CFR 50, Appendix A,” and 6.2.4.3.2, “Evaluation of Conformance to General Design Criterion 56 of 10 CFR 50, Appendix A,” the applicant has stated that Table 6.2.4-2 lists GDC 55 and 56 systems with single valve isolation and provides justification for containment isolation on another defined basis than that specified in GDC 55 and 56, pursuant to these GDC which allow an applicant to justify another defined basis, in the footnotes in Table 6.2.4-3. The staff has reviewed the list of these configurations and the corresponding justification statements against guidance for containment isolation provisions on another defined basis, described in paragraph 3.6 of ANSI N271-1976.

Each of the four RHRS hot leg CS/RHR pump suction lines (Figure 6.2.4-1, “Containment Isolation Configurations,” sheet 12 of 52, “Containment Isolation Configurations”) consists of a single locked closed motor-operated valve (MOV) isolation valve located inside containment. Based on review of the design as summarized in Table 6.2.4-2, the staff found that RHRS is credited as a closed system outside containment. The RHRS is protected from missiles, is classified as Seismic Category 1, and Quality Group B standards. The RHRS has a design temperature and pressure rating at least equal to that for the containment.

The staff reviewed the basis for this containment isolation configuration as described in Table 6.2.4-3 Note 6 against the guidance for “other defined” basis described in Section 3.6 and Appendix B of ANSI N271-1976. The staff finds that the justification conforms to the guidance and therefore SRP 6.2.4, Subsection II, Acceptance Criterion 5 for this penetration. Therefore the staff finds that the containment isolation configuration for these lines have an “other defined basis” as allowed by GDC 55, that is acceptable to the staff.

Each of the four SI pump suction lines (Figure 6.2.4-1 Sheet 11 of 52) consists of a single remote manual motor operated valve located outside containment. Based on review of the design as summarized in Table 6.2.4-2, “Associated Containment Isolation Configurations,” the staff found that the SIS is credited as a closed system outside containment. The system is protected from missiles, is classified as Seismic Category 1, and Quality Group B standards. The SIS has a design temperature and pressure rating at least equal to that for the containment. In accordance with SRP 6.2.4 Acceptance Criterion 5, since the design of the piping from containment to the single isolation valve meets the acceptance criteria for design stress and fatigue limits for ASME Code Section III, Class 2 piping in containment penetration areas as described SRP Section 3.6.2, the staff finds that the piping is designed conservatively to preclude a breach of piping integrity and therefore a guard pipe is not required on these lines. The staff review of system piping against SRP 3.6.2 criteria is in Section 3.6.2 of this report.

Each of the four RWSP CS/RHR pump suction lines (Figure 6.2.4-1 Sheet 18 of 52) consists of a single remote manual motor operated valve located outside containment. Based on review of the design as summarized in Table 6.2.4-2, the staff found that the CS/RHR system is credited as a closed system outside containment. The system is protected from missiles, is classified as Seismic Category 1, and Quality Group B standards. The SIS has a design temperature and pressure rating at least equal to that for the containment. In accordance with SRP 6.2.4 Acceptance Criterion 5, since the design of the piping from containment to the single isolation valve meets the guidance for design stress and fatigue limits for ASME Code Section III, Class 2 piping in containment penetration areas as described SRP Section 3.6.2, the staff finds that the piping is designed conservatively to preclude a breach of piping integrity and therefore a guard pipe is not required on these lines. The staff review of system piping against SRP 3.6.2 criteria is in Section 3.6.2 of this report.

The staff reviewed the containment isolation design of the four RWSP CS/RHR pump suction lines and the four SI pump Suction lines against SRP 6.2.4 acceptance criterion 5 as it relates to provisions for a leak-tight or controlled leakage housing surrounding lines which consist of a single CIV located outside of containment. In RAI **279-1899, Question 06.02.04-38,** and follow-up **RAI 729-5667, Question 06.02.04-55, and RAI 790-5916, Question 06.02.04-56,** the staff requested that the applicant provide more justification for compliance with this guidance.

In its response to **RAI 790-5916, Question 06.02.04-56**, dated September 1, 2011, the applicant clarified that the piping and valve are designed conservatively to preclude a breach of piping integrity. The containment isolation valve is designed and fabricated to RG 1.26, "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive- Waste-Containing Components of Nuclear Power Plants" Quality Group B standards. In addition, provision for detection and termination of valve leakage is provided by the safeguard component area in which the valves are located. This area contains a leak detection system which is described in DCD Section 9.3.3, "Equipment and Floor Drainage Systems." The staff's review of this system is provided in Section 9.3, "Process Auxiliaries," of this report. In the RAI response, the applicant stated that upon detection of leakage from the valve shaft or bonnet, leakage is terminated by closing the affected valves. Closing the valve isolates the valve bonnet area. Each safeguard component area has sealed pipe penetrations and doorways, and is capable of holding a volume of water up to 8 ft (2.4 m) in depth as described in DCD Subsection 3.4.1.5.2, "Reactor Building Flooding Events."

The staff has reviewed the applicant's response to this RAI and finds it acceptable because the design of the piping and the valves meet the SRP Section 3.6.2 acceptance criteria for design stress and fatigue limits for ASME Code Section III, Class 2 piping in containment penetration areas as described Section 3.6.2, "Determination of Rupture Locations and Dynamic Effects Associated with the Postulated Rupture of Piping," of this report. Therefore, the staff finds that the piping and valves are designed conservatively to preclude a breach of piping integrity and therefore a leak-tight or controlled leakage housing is not required to surround these valves.

Therefore, **RAI 279-1899 Question 06.02.04-38,** and follow-up **RAI 729-5667, Question 06.02.04-55 and RAI 790-5916, Question 06.02.04-56, are resolved and closed.**

Therefore, based on review of the information provided in the DCD as described above, the staff has determined that the US-APWR design satisfies SRP 6.2.4 Acceptance Criteria items 4 and 5, and therefore the staff finds that the design meets the requirements of GDC 55 and 56 as it relates to provisions for containment isolation on another defined basis as allowed by those GDCs.

As stated in SRP 6.2.4, Subsection II, acceptance criterion 1, RG 1.11 describes acceptable containment isolation provisions for instrumentation lines. In Section 6.2.4.1, the applicant has stated that instrument lines are designed in accordance with RG 1.11 and RG 1.141, and that instrument lines that are closed both inside and outside containment are designed to withstand pressure and temperature conditions following a LOCA and dynamic effects.

As stated in US-APWR DCD Table 6.2.4-2, each of the five containment pressure instrumentation lines (shown on US-APWR DCD Figure 6.2.4-1 Sheet 17 of 52) consists of a sealed bellows which is enclosed in a protective casing. The instrument lines are credited as closed systems outside containment. As stated in this table, the lines are protected from missiles, are classified as Seismic Category 1, and Quality Group B standards. The instrument lines have a design temperature and pressure rating at least equal to that for the containment.

Based on review of the above design information, the staff has determined that the applicant has met the guidance in RG 1.11 and 1.141 for instrument lines closed both inside and outside containment without isolation valves and therefore the staff finds that SRP 6.2.4, Subsection II, acceptance criterion 1 is met. Therefore the staff finds that the requirements of GDC 55 and 56 as they apply to provisions for containment isolation of instrumentation lines have been met.

Per SRP 6.2.4, Subsection II, acceptance criterion 9, the staff has reviewed the CIS design description as it relates to design requirements for location of the outside containment isolation valve as close to containment as practical. In Section 6.2.4.1, the applicant has stated that the CIS is designed in accordance with GDC 55, 56 and 57.

In Section 6.2.4.3, "Design Evaluation," the applicant stated that valves outside containment in systems designed in conformance with GDC 55, 56 and 57 are located as close to containment as practical. The staff has reviewed Table 6.2.4-3, "List of Containment Penetrations and System Isolation Positions," which contains a value, for each penetration, of the maximum length of pipe from containment to the outermost isolation valve which will not be exceeded in further design. The staff reviewed several licensed plant designs, and has found these values similar to those in currently operating plants. Based on this review and engineering judgment, the staff finds these lengths acceptable. Consequently, the staff finds that SRP 6.2.4, Subsection II, acceptance criterion 9 is met.

Therefore based on a review of the information provided in the DCD the staff finds that the applicant has met the requirements of GDC 55, 56 and 57 as it relates to locating containment isolation valves outside containment as close to containment as practical.

6.2.4.4.2 Actuation and Control Features for Isolation Valves

As stated in SRP 6.2.4, Subsection II, acceptance criterion 8, in accordance with 10 CFR Section 50.34(f)(2)(xiv)(A), (TMI Action plan item II.E.4.2) all nonessential systems should be automatically isolated upon initiation of an appropriate containment isolation signal. Nonessential systems are generally those which are neither ESF systems nor systems which accomplish a function similar to an ESF system. However, non-ESF and non-safety-grade systems should be classified as essential, if their continued operation under post-accident conditions will improve the reliability of a safety function.

In Section 6.2.4.1, the applicant has stated that the CIS is designed in accordance with the TMI-related requirements of 10 CFR 50.34(f)(2)(xiv)(A) through (E).

Per NUREG-0718, "Licensing Requirements for Pending Applications for Construction Permits and Manufacturing Licenses," Appendix B (SRP 6.2.4 Reference 25), all plants shall give careful consideration to the definition of essential and non-essential systems, identify each system determined to be essential, identify each system determined to be non-essential, and describe the basis for selection of each essential system.

The staff reviewed the ESF classification and the actuation and control features (e.g., automatic, manual, or remote manual) for each isolation device as listed in US-APWR DCD Table 6.2.4-3 and has determined that the applicant has classified each isolation device as either essential or non-essential, and provides for automatic isolation of nonessential systems. As stated in paragraph 6.2.4.3 of the DCD, piping systems penetrating containment are provided with leak detection, isolation and containment capabilities. Therefore the staff finds that the design has provisions for leakage detection for remote manual controlled systems, and automatic isolation of non essential systems

Therefore the staff finds that SRP 6.2.4, Subsection II, acceptance criterion 8 is met because the applicant has classified systems penetrating the containment as essential or nonessential, has provided for automatic isolation of nonessential systems, and has provided for detection of leakage for lines outside of containment. Therefore, the staff has found that the applicant met the requirements of 10 CFR 50.34(f)(2)(xiv) (A) through (E) as it relates to the classification of systems penetrating the containment as either essential or nonessential, provisions for automatic isolation of nonessential systems, and provisions for detecting leakage from lines outside containment.

In Section 6.2.4.2, the applicant stated that US-APWR DCD Table 6.2.4-1 provides the design information regarding provisions for isolating containment penetrations. The staff has reviewed Table 6.2.4-1 to identify those containment penetrations that rely on relief valves as containment isolation valves. Per SRP 6.2.4, Subsection II, acceptance criterion 7, relief valves may be used as isolation valves provided the relief setpoint is greater than 1.5 times the containment design pressure. The four CS/RHR pump suction lines (US-APWR DCD Figure 6.2.4-1 Sheet 12 of 52) and the four Main Steam Supply lines (US-APWR DCD Figure 6.2.4-1 Sheet 15 of 52) rely on relief valves for such purpose. Since the four Main Steam Supply line relief valves remain closed at main steam pressure, their relief setpoint is much greater than 1.5 times containment design pressure, and they are therefore acceptable. Since the four CS/RHR pump suction relief valves have a relief setpoint of 3.2 MPa (470 psig), as stated in US-APWR DCD Table 5.2.2-2, “CS/RHR Pump Suction Relief Valve Design Data,” the staff finds that their relief setpoint is much greater than 1.5 times containment design pressure, and they are therefore acceptable.

Per ANSI N271-1976, Section 4.74, when relief valves that discharge into containment are also used for containment isolation barriers, the discharge side of the valve shall be designed to withstand and be tested at the containment design pressure. As stated in US-APWR DCD Section 6.2.4.1, the discharge side of the relief valves in the CS/RHR pump suction lines are designed to withstand and be tested at containment design pressure.

Therefore based on review of the above information in the DCD, the staff has determined that SRP 6.2.4, Subsection II, acceptance criterion 7 is met. Consequently, the staff finds that the applicant has met the requirements of GDC 57 and 55 as it relates to usage of relief valves as containment isolation barriers

The staff has reviewed provisions in the US-APWR design to address the requirements of GDC 54 as they relate to provisions in the design of the CIS to reduce the possibility of unintended isolation valve reopening following isolation.

The staff evaluated these provisions described in US-APWR DCD Section 6.2.4.1 against the acceptance criteria for those provisions contained in Section 6.2.4 Subsection II of the SRP. 10 CFR 50.34(f)(2)(xiv) requires control systems for automatic containment isolation valves be designed for resetting the isolation signal without automatically reopening the valves.

In Section 6.2.4.1 the applicant stated that the CIS is designed in accordance with the TMI-related requirements of 10 CFR 50.34(f)(2)(xiv)(A) through (E). In Section 7.3.1.5, “ESF Initiating Signals, Logic, Actuation Devices and Manual Controls,” the applicant stated that all actuation signals are latched at the train level and require manual reset.

Based on review of this design information in the DCD, the staff concludes that SRP 6.2.4, Subsection II, acceptance criterion 19 is met, and consequently, the staff finds that the US-APWR design addresses the requirements of GDC 54 as they relate to provisions in the design in the CIS to reduce the possibility of unintended isolation valve reopening after valve reset.

The staff has reviewed provisions in the US-APWR design to address RG 1.155, “Station Blackout,” Regulatory Position C.3.2.7 which states:

“The ability to maintain appropriate containment integrity" should be addressed. "Appropriate containment integrity" for SBO means that adequate containment integrity is ensured by providing the capability, independent of the preferred and blacked out unit's onsite emergency ac power supplies, for valve position indication and closure for containment isolation valves that may be in the open position at the onset of a SBO. The following valves are excluded from consideration:

- Valves normally locked closed during operation,
- Valves that fail closed on a loss of power,
- Check valves,
- Valves in nonradioactive closed-loop systems not expected to be breached in a station blackout (this does not include lines that communicate directly with containment atmosphere), and
- Valves of less than three-inch nominal diameter.”

The staff reviewed the design requirements of the CIS as described in US-APWR DCD Table 6.2.4-3 and Figure 6.2.4-1 against the acceptance criteria for those provisions contained in Section 6.2.4 Subsection II of the SRP and has confirmed that, as described in DCD Sections 8.3, “Onsite Power Systems,” and 8.4, “Station Blackout,” there is alternate AC power supply available within 100 seconds which will allow closure of containment isolation valves which may be open at the onset of a SBO. As stated in DCD Section 6.2.4.2 all power operated isolation valves have position indication in the MCR.

Therefore based on review of the above information in the DCD, the staff has found that the US-APWR addresses the guidance contained in RG 1.155 Regulatory Position C.3.2.7, therefore SRP 6.2.4, Subsection II, Acceptance Criterion 21 is met. Consequently, the staff finds that the US-APWR design complies with the SBO rule, 10 CFR 50.63(a)(2), as it applies to containment isolation valves and valve position indication.

6.2.4.4.3 Normal and Fail Positions of Isolation Valves

Per SRP 6.2.4, Subsection II, acceptance criterion 10, the staff has reviewed the CIS design as it relates to meeting the following criteria:

- Upon loss of actuating power, automatic isolation valves should take the position of greatest safety, usually the post-accident position, or the usage of redundant isolation barriers ensure that the isolation function for the line is satisfied.

- If a fluid system has not post-accident function, the isolation valves in the lines should be closed automatically.

- For ESF or ESF-related systems, isolation valves in the lines may remain open or can be opened,

- All power operated isolation valves should have position indication in the MCR.

The staff reviewed US-APWR DCD Table 6.2.4-3 and US-APWR DCD Figure 6.2.4-1 to determine the position of automatic isolation valves upon loss of power in order to ensure the valve takes the position of greatest safety.

The staff determined that the position of automatic isolation valves upon loss of power takes the position of greatest safety. The staff's review confirmed that non motor-operated automatic isolation devices fail in the closed position upon loss of power source (air or electrical power). For lines equipped with motor-operated valves, a loss of actuating power leaves the affected valve in the “as-is” position, which may be the open position; however redundant isolation barriers ensure that the isolation function for the line is satisfied. MOVs are powered by Class 1E DC power. A single power system failure will not prevent closure of both isolation valves in a containment penetration. These features ensure single-failure-proof isolation capability for all penetrations that might be opened during operation.

The staff reviewed US-APWR DCD Table 6.2.4-3 and US-APWR DCD Figure 6.2.4-1 to determine the post-accident automatic isolation of fluid systems that do not have a post-accident function. Using this table, the staff compared the ESF/ESF support designation for each penetration to the post-accident position of the valves. The staff has determined that the US-APWR design ensures that fluid systems that do not have a post-accident function are automatically isolated.

The staff reviewed US-APWR DCD Table 6.2.4-3 and US-APWR DCD Figure 6.2.4-1 to determine the post-accident position of ESF or ESF-related systems isolation valves upon loss of power. Using this table, the staff has determined that engineered safety feature or engineered safety feature related system isolation valves remain open or can be opened upon loss of actuating power.

Based on review of US-APWR DCD Section 6.2.4.2, the staff finds that electrical redundancy via two independent power sources is provided on those lines that utilize two power operated isolation valves in series on the same penetration, and that all power-operated isolation valves have position indications in the MCR.

Therefore based on review of information in the DCD as discussed above, the staff has determined that SRP 6.2.4, Subsection II, acceptance criterion 10 is met. Consequently, the staff finds that the applicant has met the requirements of GDC 55 and 56 as they relate to the above criteria.

6.2.4.4.4 Initiating Variables for Isolation, Diversity and Redundancy of Isolation Signals

Per SRP 6.2.4, Subsection II, acceptance criterion 12, there should be diversity in the parameters sensed for the initiation of containment isolation to satisfy the GDC 54 requirement for reliable isolation capability. The staff reviewed US-APWR DCD Section 7.3, “Engineered Safety Feature Systems,” in order to evaluate the requirements of GDC 54 as they relate to diversity in the parameters sensed for the initiation of containment isolation. The staff also reviewed US-APWR DCD Table 6.2.4-3 in order to verify that there was diversity in the means of isolation of lines that penetrate containment.

The staff found that the following signals initiate closure of containment isolation valves, as indicated in US-APWR DCD Table 7.3-3, “Engineered Safety Features Actuation Signals”:

- A containment isolation Phase A signal (DCD Section 7.3.1.5.4) is generated from any of the following monitored variables:
  - o ECCS actuation signal.
  - o Manual actuation.
- A containment isolation Phase B signal (DCD Section 7.3.1.5.5) is generated from any of the following monitored variables:
  - o High-3 containment pressure signal (2-out of 4 signals).
  - o Manual CS system actuation.
- An ESF actuation signal for ECCS function (DCD Section 7.3.1.5.1) is initiated by any one of the following monitored variables:
  - o Low pressurizer pressure.
  - o Low main steamline pressure.
  - o High containment pressure.
  - o Manual initiation.

RG 1.141 states that CIS designs shall have diversity in the parameters sensed for the initiation of containment isolation, in accordance with SRP Section 6.2.4, “Containment Isolation System.” Based on the above design information, the staff determined that SRP 6.2.4, Subsection II, acceptance criterion 12 is met. Consequently, the staff finds that the design meets the GDC 54 requirements for reliable isolation capability as they relate to diversity in the parameters sensed for the initiation of containment isolation.

Per SRP 6.2.4, Subsection II, acceptance criterion 11, the staff reviewed DCD Section 7.3 and Tables 7.3-3 and 7.3-4, “Engineered Safety Features Actuation Variables, Ranges, Accuracies, Response Times, and Setpoints (Nominal),” in order to evaluate them against the requirements of 10 CFR 50.34(f)(2)(xiv) as they relate to the reduction of the containment set point pressure that initiates containment isolation for nonessential penetrations to the minimum value compatible with normal operating conditions. Containment isolation of nonessential penetrations is initiated via the Containment Isolation phase A function upon receipt of a valid ECCS actuation signal, which can be generated by receipt of a valid high containment pressure signal.

The staff reviewed the actuating logic, set point, the range, accuracy response time as listed in US-APWR DCD Figure 7.2-2, “Functional Logic Diagram for Reactor Protection and Control System,” (Sheets 11 and 12 of 21) and Tables 7.3-3, “Engineered Safety Features Actuation Signals,” and 7.3-4, “Engineered Safety Features Actuation Variables, Ranges, Accuracies, Response Times, and Setpoints (Nominal),” and compared this with the guidance for TMI Action Item II.E 4.2, contained in NUREG-0737, “Clarification of TMI Action Plan Requirements.” TMI Item II.E.4.2 states that the containment setpoint pressure that initiates containment isolation for nonessential penetrations must be reduced to the minimum compatible with normal operating conditions. It further states that the pressure setpoint selected should be far enough above the maximum expected pressure inside the containment during normal operation so that inadvertent containment isolation does not occur during normal operation from instrument drift or fluctuations due to the accuracy of the pressure sensor. A margin of 6.9 kPa (1 psi) above the maximum expected containment pressure should be adequate to account for instrument error. Any proposed values greater than 6.9 kPa (1 psi) will require detailed justification. Applicants for an operating license should use pressure history data from similar plants that have operated more than one year, if possible, to arrive at a minimum containment setpoint-pressure.

In US-APWR DCD Table 7.3-4, the applicant indicated that an ESF actuation signal for ECCS function, and consequently, a containment isolation phase A isolation signal is generated at a high containment setpoint pressure of 46.9 kPa (6.8 psig).

In its response to **RAI 57-852, Question 06.02.04-15, and RAI 279-1899, Question 06.02.04-46**, dated April 8, 2009, the applicant provided the basis for the setpoint in Section 6.2.4.2 of the DCD. This setpoint was selected as 10 percent of the containment design pressure of 570.19 kPa (68 psig). The applicant indicated that the maximum expected pressure inside containment during normal operation is 115.4 kPa (2.0 psig), and the accuracy of the pressure instrument channel of 17.24 kPa (2.5 psi) was estimated by combining instrumentation factors that affect the accuracy of each component in the channel (as explained in DCD Section 7.2.2.7.1, “Methodology for Instrument Channel Statistical Accuracy Calculation”). In order to prevent an inadvertent actuation of containment isolation, the applicant applied an additional margin of 15.85 kPa (2.3 psi). The applicant performed the accident dose evaluation with this setpoint.

Based on review of information provided in US-APWR DCD Section 6.2.4.2 and the response to Questions 06.02.04-15 and 06.02.04-46, the staff determined that SRP 6.2.4, Subsection II, acceptance criterion 12 is met because the applicant’s dose consequence analyses are based on this setpoint and have been found acceptable to the staff. The staff’s review of the applicant’s accident analyses is in Chapter 15 of this report. Consequently, **RAI 279-1899, Question 06.02.04-46, and RAI 57-852, Question 06.02.04-15,** are resolved. The staff finds that the design meets the requirements of 10 CFR 50.34(f)(2)(xiv) as they relate to the reduction of the containment setpoint pressure that initiates containment isolation for nonessential penetrations to the minimum value compatible with normal operating conditions.

Per SRP 6.2.4, Subsection II, acceptance criterion 13, GDC 56 system lines which provide open paths from the containment to the environs (e.g. purge and vent lines) should be equipped with radiation monitors capable of isolating these lines upon a high-radiation signal, which should not be considered a diverse containment isolation parameter. The staff reviewed US-APWR DCD Sections 9.4.6, “Containment Ventilation System,” 7.3, Table 6.2.4-3, and Figure 6.2.4-1 in order to evaluate if the US-APWR design meets this criterion.

The containment isolation valves for the containment purge system will close within five seconds upon initiation of a valid containment purge isolation signal. The containment purge isolation signal is generated on receipt of two-out-of-four High Containment High Range area radiation signals. The signals have a response time of 15 seconds. The containment purge isolation signal is not considered and is not listed as a diverse containment isolation parameter. Based on this design information, the staff has determined that the design features of the purging/venting system minimize purging time and that there is a high degree of assurance that the purge system will isolate reliably under accident conditions. The system is equipped with radiation monitors capable of isolating the lines on a high radiation signal. Therefore, SRP 6.2.4, Subsection II, acceptance criterion 13 is met. Consequently, the staff finds the design meets the requirements of GDC 56 as they relate to the reliability of containment isolation of system lines which provide open paths from the containment to the environs have been met. **RAI 279-1899, Question 06.02.04-46, is resolved and closed.**

6.2.4.4.5 Basis for Selection of Closure Time Limits

Per SRP 6.2.4, Subsection II, Acceptance Criterion 14, the NRC staff has reviewed the US-APWR CIS design as it relates meeting the following criteria:

- Containment isolation closure times should be selected for rapid isolation of containment following postulated accidents.

- Isolation valve closure times should be five seconds or less for lines providing open paths from the containment to the environs. Radiological consequences and the effect on the containment back-pressure on the release of containment atmosphere should justify the selected valve closure time.

- BTP 6-4 provides additional guidance on the design and use of containment purge systems which may be used during the normal plant operating modes (i.e., startup, power operation, hot standby, and hot shutdown).

- Containment purge valves that do not satisfy the operability criteria of BTP 6-4 must be sealed closed as defined in Subsection II.6 of this SRP section during operational conditions 1, 2, 3, and 4. Furthermore, closure of these valves must be verified at least every 31 days. These requirements should be incorporated into the TS for plant operation.

The staff reviewed the basis of the US-APWR containment isolation valve closure times stated in US-APWR DCD Section 6.2.4.2 and the values of the closure time for each valve listed in Table 6.2.4-3, against guidance contained in Section 6.2.4, Subsection III of the SRP and the guidance in paragraph 4.4.4, “Valve Closure Time”, of ANSI N271-1976, endorsed by RG 1.141.

Based on review of information provided in the DCD the NRC staff has determined that the applicant’s stated containment isolation valve closure time design basis and valve closure times are in accordance with paragraph 4.4.4, of ANSI N271-1976 and thus RG 1.141 for containment isolation valves other than purge, vent or other valves which may be open during operation and provide an open path from the containment atmosphere to the environs.

The largest piping penetration that provides a direct path to the atmosphere consists of the two 8 in (20 cm) low volume purge system valves. The isolation valves in these lines are specified as having a five-second closure time. The staff noted that this closure time is consistent with the assumptions and criteria for radiological dose analyses used in US-APWR DCD Chapter 15 analyses.

Based on review of information provided in the DCD and the response to **RAI 57-852, Question 06.02.04-17**, which provides detail on how the US-APWR design complies with BTP-6-4 acceptance criteria 5A through 5D, the staff determined that the design and use of the US-APWR containment purge system is in compliance with BTP 6-4. The staff found that there is a proposed TS in Chapter 16 of the DCD which requires that the 36 in (91 cm) high volume purge valves are sealed closed in plant modes 1, 2, 3 and 4 and verified sealed closed every 31 days. **Thus, RAI 57-852, Question 06.02.04-17, is resolved and closed.**

Based on compliance with RG 1.141, as described above, the staff has found that SRP 6.2.4, Subsection II, Acceptance Criterion 14 is met. Consequently, the staff finds that the US-APWR containment isolation closure times are in accordance with GDC 54 requirements as they relate to the rapid isolation of containment following postulated accidents.

6.2.4.4.6 Sealed Closed Barriers

Per SRP 6.2.4, Subsection II, Acceptance Criterion 6, the staff reviewed provisions for systems which utilize blank flanges as containment isolation barriers. In US-APWR DCD Section 6.2.4.3.2, the applicant stated that Table 6.2.4-2 lists GDC 55 and 56 systems which utilize blank flanges as containment isolation barriers and provides justification for their use. The staff has reviewed this listing. Based on review of this information, the staff finds that US-APWR containment design has the following flanged or sealed penetrations:

- The Containment Leak Rate Testing System (LTS) has a penetration (US-APWR DCD Figure 6.2.4-1 Sheet 47) that consists of two blank flanges in series. The LDS penetration is classified as Seismic Category 1 and Quality Group B standards. The sealed lines contain flanged closures supplied with testable seals. The LDS penetration has a design temperature and pressure rating at least equal to that for the containment.

- Two oil supply and drain lines for the RCP motor (US-APWR DCD Figure 6.2.4-1 Sheet 48) each of which consists of two blank flanges in series. These lines are classified as Seismic Category 1 and Quality Group B standards. The lines contain flanged closures supplied with testable seals. The lines have a design temperature and pressure rating at least equal to that for the containment.

- The personnel airlock (US-APWR DCD Figure 6.2.4-1 Sheet 49) consists of two flanged doors in series. The doors are interlocked to ensure that both doors are not opened simultaneously. The airlock is designed to Seismic Category 1 and Quality Group B standards. Each door is provided with a testable seal. The airlock has a design temperature and pressure rating at least equal to that for the containment

- The equipment hatch (US-APWR DCD Figure 6.2.4-1 Sheet 50) consists of a single flanged door. The door is not opened during normal or accident conditions. The hatch is designed to Seismic Category 1 and Quality Group B standards. The hatch is provided with a testable seal. The hatch has a design temperature and pressure rating at least equal to that for the containment.

- The fuel transfer tube (US-APWR DCD Figure 6.2.4-1 Sheet 39) consists of a single flange. The penetration is not opened during normal or accident conditions. The tube is designed to Seismic Category 1 and Quality Group B standards. The tube is a flanged closure supplied with a testable seal. The tube has a design temperature and pressure rating at least equal to that for the containment.

- There are fourteen spare penetrations (US-APWR DCD Figure 6.2.4-1 Sheet 52). The penetrations are designed to Seismic Category 1 and Quality Group B standards. The spare penetrations are welded closed. The penetrations have a design temperature and pressure rating at least equal to that for the containment.

- There are sixty one electrical penetrations (US-APWR DCD Figure 6.2.4-1 Sheet 51). The penetrations are designed to Seismic Category 1 and Quality Group B standards. Each penetration is provided with a testable seal. The penetrations have a design temperature and pressure rating at least equal to that for the containment.

The applicant stated in Section 6.2.4 of the DCD that flanged closures, the personnel airlock and the equipment hatch are under administrative control. The staff determined that the single-barrier penetration closures are passive and are not subject to single-active failures during plant operation. The staff finds that this conforms to ANSI N271-1976, Paragraph 4.10, endorsed by RG 1.141.

Based on review of information provided in the DCD the NRC staff has found that SRP 6.2.4, Subsection II, Acceptance Criterion 6, is met. Therefore the staff had found that the US-APWR design satisfies the requirements of GDC 55 and 56 for systems which utilize blank flanges as containment isolation barriers and provides justification for their use.

6.2.4.4.7 Use of Closed Systems as Isolation Barriers

Per SRP 6.2.4, Subsection II, Acceptance Criterion 15, the staff has reviewed the use of closed systems inside containment as one of the isolation barriers as they relate to the following criteria:

a. The system does not connect with either the RCS or the containment atmosphere.
b. The system is protected against missiles and pipe whip.
c. The system is designated seismic Category I.
d. The system is classified Quality Group B.
e. The system is designed to withstand temperatures equal to at least that of the containment design.
f. The system is designed to withstand the external pressure from the containment structure acceptance test.
g. The system is designed to withstand the LOCA transient and environment.

US-APWR DCD Section 6.2.04.3.3 lists those systems (or portions of systems) that meet the criteria of GDC 57. These are as follows:

- MSFWS feedwater line.

- SGBDS SG blowdown line.

- MSFWS main steam line.

- CCWS inlet and outlet for letdown HX.

- CCWS inlet and outlet for excess letdown HX.

The staff reviewed the design of those systems, as described in Chapters 3, "Design of Structures, Systems, Components and Equipment," 9, "Auxiliary Systems," and 10, "Steam and Power Conversion System," of the US-APWR DCD, with acceptance criteria for those systems contained in Section 6.2.4 Subsection II of the SRP.

DCD Section 3.2.2, "System Quality Group Classification," lists the classification of mechanical fluid systems, components and equipment. The staff reviewed this list and confirmed that the specified portions of the above systems are designated Seismic Category 1 and classified Quality Group B. The staff has determined that the systems do not connect with either the RCS or the containment atmosphere. The staff noted that Section 3.1.1.4.1 of the DCD stated that the safety-related SSCs are designed to accommodate the effects of, and to be compatible with, the environmental conditions associated with the normal operation, maintenance, testing, and postulated accidents, including LOCAs. DCD Section 3.5, "Missile Protection," states that Safety-related SSCs are identified in DCD Section 3.2, "Classification of Structures, Systems, and Components," and Section 3.11, "Environmental Qualification of Mechanical and Electrical Equipment," and are protected from missiles. The staff noted that DCD Section 10.3.1.1, "Safety Design Bases," specified the scope of the safety related portion of the Main Steam System (MSS), and compliance with GDC 4. The staff has concluded that the safety related portion of the MSS includes that portion of the system that penetrates containment. The staff noted that DCD Section 10.4.7.1.2, "Safety Design Basis," specifies the scope of the safety related portion of the Feedwater System, and compliance with GDC 4. The staff has concluded that the safety related portion of the Feedwater System includes that portion of the system that penetrates containment. The staff noted that DCD Section 10.4.8.1.1, "Safety Design Bases," specifies the scope of the safety related portion of the SGBDS, and compliance with GDC 4. The staff has concluded that the safety related portion of the SGBDS includes that portion of the system that penetrates containment. The staff noted that DCD Section 9.2.2.1.1, "Safety Design Basis," specifies the scope of the safety related portion of the CCWS, and compliance with GDC 4. The staff has concluded that the safety related portion of the CCWS includes that portion of the system that penetrates containment. The staff noted that DCD Section 6.2.4.1 specifies that the containment isolation barriers are required to be protected from missiles.

Based on this review, the staff has concluded that isolation barriers are located behind missile barriers, pipe whip was considered in the design of pipe restraints. The staff also concluded that the sections of piping penetrating the containment to the isolation barriers, including the piping between isolation valves, are designated Seismic Category I and are therefore adequately protected from missiles and pipe whip.

Based on review of the above design information contained in the DCD the staff has concluded that SRP 6.2.4, Subsection II, Acceptance Criterion 15, is met. Therefore the use of closed systems inside containment as one of the isolation barriers in the US-APWR CIS design, meets the requirements of GDC 1, 2, 4, and 16 as these GDC apply to this purpose. Additional staff findings regarding compliance of these systems to these GDCs are in Chapters 3, 9 and 10 of this report.

6.2.4.4.8 Protection of CIS' against Loss of Function As a Result of Missiles, Pipe Whip, and Natural Phenomena

As mentioned in section 6.2.4.4.7 of this report, "Use of Closed Systems as Isolation Barriers," the staff confirmed that the CIS design bases include protection from missiles, pipe breaks, earthquakes, fire, internal and external flooding, ice, wind, and tornados. Other sections of this report discuss specific features and design criteria for the protection of systems, structures, and equipment from these phenomena.

6.2.4.4.9 Environmental Conditions in the Vicinity of Containment Isolation Components

Containment isolation equipment may be subject to potentially harsh conditions resulting from pressure, temperature, flooding, jet impingement, radiation; missile impact, and seismic response. The staff review confirmed that the US-APWR CIS has been properly classified to ensure that protection from these environmental hazards is encompassed by the mechanical and electrical design bases and quality standards of the isolation system. Section 3.11 of this report discusses the staff's review of the environmental qualification of the US-APWR SSCs, including containment isolation equipment.

The US-APWR design identifies the containment penetrations as components that are relied upon to be functional after core damage in a severe accident. The severe accident analysis, Described in US-APWR DCD Section 19.2.3.3.7, “Equipment Survivability,” identifies that the electrical penetrations as those that were scoped in to the applicant’s equipment reliability study and must survive a period of time of very high temperature and pressure. In **RAI 553-4357, Question 06.02.04-53**, dated March 16, 2010, the staff asked the applicant the process by which procured components will be verified that they are designed to survive the environmental conditions described in the Chapter 19 analysis.

In its response to **RAI 553-4357, Question 06.02.04-53,** dated April 19, 2010, the applicant provided further justification for the severe accident performance requirements for containment penetrations, and provided DCD revisions to DCD Section 19.2.3.3.7 that provide more detail of these requirements. The staff reviewed the response and determined that additional information was needed in the DCD to clarify how these severe accident performance requirements will be verified on as-procured equipment. The staff evaluated this issue using criteria for equipment survivability described in RG 1.7, “Control of Combustible Gas Concentrations in Containment,” the staff review is further described in Section 6.2.5, “Combustible Gas Control in Containment,” of this report.

In follow-up **RAI 803-5891, Question 06.02.05-45**, dated August 11, 2011, the staff asked the applicant to clarify the DCD to ensure that type tests will be performed on procured equipment that is required to function in a severe accident.

In its response to **RAI 803-5891, Question 06.02.05-45**, dated September 9, 2011, the applicant stated that the DCD will be revised to include COL Information Item 19.3(7). This COL item requires an applicant that references the US-APWR design to perform an equipment survivability assessment of the as-built equipment required to mitigate severe accidents (electrical penetrations, hydrogen, igniters, and the containment pressure instrument (wide range) to provide reasonable assurance that they will operate in the environmental conditions resulting from hydrogen burns associated with severe accidents for which they are intended and over the time span for which they are needed.

The staff has reviewed the RAI responses and finds them acceptable because the applicant has clarified the severe accident performance requirements for the containment electrical penetrations and has identified an information item for a COL applicant that will provide reasonable assurance that the procured containment electrical penetrations will operate in the environmental conditions resulting from hydrogen burns associated with severe accidents for which they are intended and over the time span for which they are needed. Consequently, RAI

553-4357, Question 06.02.04-53 is resolved. **RAI 803-5891, Question 06.02.05-45, is being tracked as a Confirmatory Item to ensure the DCD is changed** in accordance with the RAI response.

Based on review of the information in the DCD described above, and pending the closure of the above confirmatory item, the staff finds that the US-APWR containment isolation components are designed to be compatible with the environmental conditions in which they are expected to perform their functions.

6.2.4.4.10 Mechanical Engineering Design Criteria Applied to the Containment Isolation System, Structure, and Components

Per SRP 6.2.4, Subsection II, Acceptance Criterion 16, the staff has reviewed the reliability and performance considerations used in the design of the US-APWR isolation barriers as they relate to the requirements of GDCs 1, 2, 4, and 54. The design criteria for components performing a containment isolation function, including the isolation barriers and the piping between them or the piping between the containment and the outermost isolation barrier, was found acceptable if:

A. Group B quality standards, as defined in RG 1.26, apply to the components, unless the service function dictates that Group A quality standards apply.

B. The components are designated Seismic Category I in accordance with RG 1.29, “Seismic Design Classification”

The staff reviewed the design of those systems, and portions of those systems which serve as containment isolation barriers as described in US-APWR DCD Section 3.2 and Table 3.2-2, “Classification of Mechanical and Fluid Systems, Components, and Equipment.”

The staff has determined that:

A. Group B quality standards, as defined in RG 1.26, apply to the components, unless the service function dictates that Group A quality standards apply.

B. The components are designated Seismic Category I in accordance with RG 1.29.

Therefore the staff determined that SRP 6.2.4, Subsection II, Acceptance Criterion 16, is met. Consequently, the staff found that the requirements of GDCs 1, 2, 4, and 54, have been met as they relate to the inclusion of appropriate reliability and performance considerations in the US-APWR design that reflect the safety importance of containment capability under accident conditions.

6.2.4.4.11 Provisions for Alerting Operators of the Need to Actuate Manual Isolation Devices in the Event of leakage or an Accident

Per SRP 6.2.4, Subsection II, Acceptance Criteria 2 and 3, the staff reviewed provisions to detect leakage outside containment for essential system lines that penetrate containment. In Table 6.2.4-3, “List of Containment Penetrations and System Isolation Positions,” in the DCD application, the applicant has provided a listing of containment penetrations and system isolation positions. This listing includes provisions for containment isolation by means of remote-manual valves in engineered safety feature related systems including some valves in the SIS, RHRS, MSS, CSS, PSS and CCWS. This listing also includes provisions for containment isolation by means of remote-manual valves in lines in systems needed for safe shutdown of the plant (e.g., liquid poison system, reactor core isolation cooling system, and isolation condenser system) including some valves in the CVCS.

As discussed in US-APWR DCD Section 6.2.4.2, "System Design," the US-APWR design provides means of detection of possible leakage from these lines. Based on review of information presented in this DCD paragraph, which describes design provisions to detect leakage, the staff has determined that the US-APWR design is capable of detecting leakage in these lines and providing appropriate indication in the MRC. Therefore the staff found that SRP 6.2.4, Subsection II, Acceptance Criteria 2 and 3 are met. Consequently, the staff finds that the applicant has met the requirements of GDC 54 as it relates to provisions for leak detection for those ESF system and ESF related system containment penetrations that utilize remote-manual valves for containment isolation.

Per SRP 6.2.4, Subsection II, Acceptance Criterion 17, the staff has reviewed provisions in the US-APWR design to address the requirements of GDC 54 as they relate to reliable isolation capability. For remote manual isolation valves, the design of the CIS was found acceptable if there are provisions to allow the operator in the MCR to know when to isolate fluid systems equipped with remote manual isolation valves.

The staff evaluated these provisions as described in US-APWR DCD Section 6.2.4.2 against the acceptance criteria for those provisions contained in Chapter 6.2.4 Subsection II of the SRP. In this section of the DCD, the applicant stated that detection of possible leakage is provided for those systems where remote-manual isolation valves are employed. The applicant described the instrumentation and methods by which leakage outside containment can be detected.

In **RAI 362-2278, Question 09.02.02-34**, dated April 13, 2011, the staff questioned the containment isolation configuration of the CCWS lines to the RCP pumps and motors, specifically how they conform to SRP 9.2.2 Section III, Item 4.F guidance which states that there should be design provisions for isolation of component cooling water supply and return lines to the RCP by remote manual means only.

In its response to RAI **362-2278, Question 09.02.02-34**, dated July 4, 2011, the applicant proposed revisions to Section 6.2.4.2 of the DCD and Table 6.2.4-3 to indicate that these lines are isolated via remote manual means. The staff reviewed the response and the proposed revisions to the DCD and finds them acceptable because the applicant included justification for this configuration and included a description of the means by which operators in the MCR can detect leakage and isolate these lines. **RAI 362-2278, Question 09.02.02-34 is being tracked as a Confirmatory Item** to ensure the DCD is revised accordingly.

Based on the review of information presented in the DCD, and pending the closure of the above confirmatory item, the staff has found that SRP 6.2.4, Subsection II, Acceptance Criteria 2 and 3 are met. Consequently, the staff has found that the requirements of GDC 54 as they relate to reliable isolation capability for remote manual isolation valves have been met.

6.2.4.4.12 Provisions for, and Technical Specifications Pertaining to Operability and Leakage Rate Testing of Isolation Barriers

Per SRP 6.2.4, Subsection II, Acceptance Criterion 18, the staff has reviewed provisions in the US-APWR design to address the requirements of GDC 54 as they relate to provisions for operability testing of the containment isolation valves and leakage rate testing of the isolation barriers. In order to permit periodic Type A, Type B, and Type C testing of the containment and its piping penetrations, test connections must be provided on the containment and on penetrations in order to permit application and measurement of test air pressure and venting of leakage air.

The staff evaluated these provisions as described in US-APWR DCD Section 6.2.4.4 and in Figure 6.2.4-1. In Section 6.2.4.4 of the DCD, the applicant stated that test connections are provided for 10 CFR 50 Appendix J Type C leakage rate testing. US-APWR DCD Chapter 16, "Technical Specifications," Section 3.6 specifies periodic Type B and C leakage rate testing.

Based on review of information provided in the DCD sections listed above, the staff confirmed that test, vent, and drain connections are provided at suitable locations. The staff determined that the US-APWR design provides provisions for operability testing of the containment isolation valves and leakage rate testing of the containment isolation barriers. The staff found that the content of the containment isolation barrier leakage and operability surveillance requirements described in DCD Chapter 16 are consistent with NRC requirements for isolation barriers. Therefore the staff determined that SRP 6.2.4, Subsection II, Acceptance Criterion 18 is met. Consequently, the staff finds that the design meets the requirements of GDC 54 as they relate to provisions for operability testing of the containment isolation valves and leakage rate testing of the isolation barriers. Section 6.2.6 of this report provides the staff's evaluation of the US-APWR containment leakage testing program.

6.2.4.4.13 Calculation of Containment Atmosphere Released before Isolation Valve Closure for Lines that Provide a Direct Path to the Environs

Per SRP 6.2.4, Subsection II, Acceptance Criterion 22, the staff reviewed the US-APWR design to address the extent to which 10 CFR Part 50 Appendix K is used for the determination of the extent of fuel failure (source term) in the radiological calculations.

The staff reviewed US-APWR DCD Section 11.1.3 which states that the realistic reactor coolant source term represents the expected average concentrations of radionuclides contained in the reactor coolant and the secondary coolant. These concentrations are calculated according to the modeling procedures in ANSI/ANS-18.1 "Radioactive Source Term for Normal Operation for Light Water Reactors," and NUREG-0017 "Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Pressurized-Water Reactors (PWR-GALE Code)." The reference plant values provided in ANSI/ANS-18.1 were adjusted to be consistent with the US-APWR plant values listed in Table 11.1-8 by using adjustment factors. The staff found this calculation method conforms to the guidance in RG 1.112 "Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Light-Water-Cooled Power Reactors."

The NRC staff reviewed DCD Chapter 15. Per Table 15.6.5-4, the assumed radionuclide inventory in the reactor coolant is based on an assumed one percent fuel defect and is listed in Table 11.1-2, "Design Basis Reactor Coolant Activity," of the DCD. A 15 second timeframe is assumed as the duration of purge from the accident initiation, and 100 percent of reactor coolant inventory is released to the containment at the initiation of the LOCA. The staff also reviewed DCD Appendix 15A. The US-APWR uses the RADionuclide TRansport, Removal, and Dose (RADTRAD) computer code for estimating doses at offsite locations such as the exclusion area boundary (EAB) and the low-population zone (LPZ), as well as onsite locations (e.g., MCR) due to postulated radioactivity releases from DBA conditions. RADTRAD calculates dose consequences for different specified time intervals based on user-input information on the amount, form, and species of the radioactive material released in the reactor plant. For US-APWR, RADTRAD is configured to use the assumptions and source terms discussed in NUREG-1465 “Accident Source Terms for Light-Water Nuclear Power Plants,” as specified in RG 1.183 “Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors.”

Based on this review, the staff finds that the US-APWR DCD utilizes 10 CFR Part 50, Appendix K, for the determination of the extent of fuel failure (source term) in the radiological calculations and therefore finds that SRP 6.2.4, Subsection II, Acceptance Criterion 22 is met.

6.2.4.4.14 TMI Item II.E.4.4. Vent/Purge Valve Positions

Per SRP 6.2.4, Subsection II, Acceptance Criterion 20, the staff has reviewed provisions in the US-APWR design to address the requirements of 10 CFR 50.34(f)(2)(xv) purging requirements as they relate to the regulatory guidance of BTP 6-4 “Containment Purging During Normal Plant Operations.”

In US-APWR DCD Section 1.9, the applicant has stated that the US-APWR design complies with SRP Section 6.2.4 Acceptance Criterion 14 with no exceptions. The NRC staff reviewed the design requirements of the containment purge system as described in Chapter 9 of the DCD and has confirmed that:

- The performance and operability assurance program for the containment purge system is consistent with SRP Section 3.10,
- The number of supply and exhaust lines is limited to one supply and one exhaust line,
- The size of the lines do not exceed eight inches in diameter,
- The containment isolation provisions for the purge system meet the standards appropriate for engineered safety features,
- Instrumentation and control systems isolating the purge system lines are independent and actuated by diverse parameters,
- Purge system isolation valve closure times do not exceed five seconds,
- Isolation valve closure is not prevented by debris which could become entrained in the escaping air and steam,
- The purge system is not designed for temperature and humidity control within the containment,
- The need for purging of the containment is minimized by the presence of other containment atmospheric cleanup systems within containment,
- The containment purge valves isolation function and leakage rate are periodically tested during reactor operation.

The bases for TS 3.6.3 indicate that the eight inch Low Volume Purge System valves will be opened as needed in Modes 1, 2, 3, and 4. BTP 6-4 guidance states that the opening of large valves that provide a direct path from the containment atmosphere to the environs should be minimized during power operation. The staff also notes that the plant design has very few safety-related items in containment that would require containment entry while at power. Therefore, venting or purging should occur infrequently. As a result, the containment vent/purge system should only be used for containment pressure control, as low as is reasonably achievable, or air quality considerations for personnel entry, or TS surveillances. TS SR 3.6.3.2 includes this restriction.

In RAI 57-852, Question 6.2.4-17, the staff requested that the applicant clarify the calculation of the low volume CIV closure time conforms to the guidance of BTP 6-4, “Containment Purging During Normal Plant Operations.” In its response to RAI 57-852, Question 6.2.4-17, dated September 22, 2008, the applicant provided details of how the US-APWR design complies with BTP-6-4 acceptance criterion 5A through 5D. The staff reviewed the applicant’s response to this RAI and found it acceptable because it clarified the staff’s understanding of the basis of the closure times of these valves. **Therefore the staff considers RAI 57-852, Question 6.2.4-17 resolved and closed.**

As discussed above, the staff found that the design satisfies SRP 6.2.4, Subsection II, Acceptance Criterion 14. Based on review of information provided in the DCD, and the response to **RAI 57-852, Question 6.2.4-17**, the staff found that the US-APWR utilizes the regulatory guidance of BTP 6-4 and therefore SRP 6.2.4, Subsection II, Acceptance Criterion 20 is met. Therefore the staff found that the US-APWR design complies with 10 CFR 50.34(f)(2)(xv) purging requirements.

6.2.4.4.15 TMI Item II.B.8 Three-Foot Diameter Containment Penetration

The staff has reviewed provisions in the US-APWR design to address the requirements of 10 CFR 50.34(f)(3)(IV) for provisions for a dedicated three foot diameter containment penetration so as not to preclude later installation of a ventilated containment system. The staff evaluated these provisions as described in US-APWR DCD Section 19.2.3.3.9, “Other Severe Accident Mitigation Features,” with the requirements as stated in the regulation and Reference 1. In US-APWR DCD Section 1.9, Table 1.9.3-2, the applicant has stated that the requirements of TMI Item II.B.8 are addressed in Section 19.2.3.3.9. In DCD Section 19.2.3.3.9, the applicant stated that the containment high volume purge opening satisfies the requirement of the regulation. The staff reviewed the design description of the high volume purge opening and has concluded that the containment high volume purge opening satisfies the requirement of the regulation, therefore the requirements of 10 CFR 50.349(f)(3)(IV) have been met.

6.2.4.4.16 Minimization of Contamination

In consideration of 10 CFR 20.1406, “Minimization of Contamination,” the staff reviewed the US-APWR design in order to determine how the design will minimize to the extent practicable, contamination of the facility and the environment, facilitate eventual decommissioning and minimize to the extent practicable the generation of radioactive waste. US-APWR DCD Table 12.3-8, “Regulatory Guide 4.21 Design Objective and Applicable DCD Subsection Information for minimizing Contamination and Generation of Radioactive Waste,” describes the provisions related to the CIS for providing for adequate leak detection capability to provide prompt detection of leakage for any structure, system, or component which has the potential for leakage. The DCD states that the piping systems penetrating the containment are provided with leak detection, isolation, and containment capabilities. These piping systems are designed with the capability to test, periodically, the operability of the isolation valves and associated apparatus and determine if valve leakage is within acceptable limits.

The staff finds that these design provisions meet the requirement of 10 CFR 20.1406 and are consistent with guidelines of RG 4.21 “Minimization of Contamination and Radioactive Waste Generation: Life-Cycle Planning” since the existence of leak detection systems ensure that operating personnel will be alerted in a timely manner. Prompt action by plant personnel to address detected leakage will minimize radioactive contamination of spaces containing CIS components. Section 12.3 of this report further addresses the US-APWR design in accordance with 10 CFR 20.1406.

6.2.4.5 Combined License Information

There are no COL item numbers in DCD Table 1.8-2 related to this review area. The staff finds that there are no COL information items that are needed for this review area.

6.2.4.6 Conclusions

The staff has reviewed the CIS in accordance with SRP 6.2.4 and BTP 6-4. On the basis of its review, the staff concludes that the CIS complies with the requirements of GDC 1, 2, 4, 16, 54, 55, 56 and 57 10 CFR 50.63, and 10 CFR 20.1406 as related to containment isolation. The CIS complies with the requirements of the NUREG-0737 TMI action plan items incorporated in 10 CFR 50.34.

6.2.5 Combustible Gas Control in Containment

6.2.5.1 Introduction

The control of combustible gases in the containment following a beyond DBA involving 100 percent fuel clad coolant reaction or postulated accident is necessary to ensure conformance with the requirements of GDCs 5, 41, 42, and 43, and 10 CFR 50.44. Following an accident, hydrogen and oxygen may accumulate inside the containment.

If a sufficient amount of combustible gas is generated, it may react with the oxygen present in the containment at a rate rapid enough to breach the containment or cause a leakage rate in excess of TS limits. Additionally, the associated pressure and temperature increase could damage systems and components essential to continued control of the post accident conditions.

6.2.5.2 Summary of Application

DCD Tier 1: The Tier 1 information associated with this section is found in DCD Tier 1, Section 2.11.4.1.

DCD Tier 2: The applicant has provided a DCD Tier 2 description in Section 6.2.5, summarized here in part, as follows:

> The CHS consists of: (1) the hydrogen monitoring system, and (2) the hydrogen ignition system. The hydrogen monitoring system is nonsafety-related and consists of a single hydrogen detector located outside of containment that samples the hydrogen concentration of containment air samples extracted from the containment via the post-accident containment atmospheric sampling line. The hydrogen ignition system consists of twenty hydrogen igniters that are distributed throughout the containment and its subcompartments where hydrogen may be produced, transmit or collect. The hydrogen ignition system is automatically initiated by the ECCS actuation signal but can also be initiated manually. The hydrogen igniters are meant to limit uniformly distributed post-accident combustible gas concentrations in the containment to less than 10 percent (by volume).

The applicant indicated in US-APWR DCD Tier 2, Revision 3, Section 6.2.5 that the CHS is designed in accordance with 10 CFR50.44, and GDC 41. In addition, the applicant states that the systems also address the recommendations of RG 1.7 and NUREG 0373 and NUREG 0660. In Table 1.9.2-6 of the DCD the applicant also indicates conformance with no exceptions with GDCs 42 and 43 as they relate to designing containment ESF atmosphere cleanup systems to permit inspection, and to permit pressure and functional testing, respectively. The applicant indicated that the Hydrogen Ignition System is designed to limit the combustible gas concentration in the containment following an accident, uniformly distributed to less than 10 percent by volume. Uniform mixing of the containment atmosphere is achieved via a combination active-passive atmospheric mixing system. Atmospheric mixing is achieved by convective heat transfer and hydrogen diffusivity in conjunction with containment spray discharges.

In DCD Tier 2, Revision 3, Section 19.2, the applicant discusses the results of calculations that demonstrate that the requirements of 10 CFR50.44(c)(1), 10 CFR50.44(c)(2), and 10 CFR50.44(c)(5) are met with respect to maintaining containment integrity against hydrogen combustion events. In this same section, the applicant discusses the results of the equipment survivability analysis, which purpose is to identify equipment required to function in a severe accident, to show the expected severe accident conditions and required time such equipment must withstand, such that equipment survivability under conditions created by hydrogen burning meets 10 CFR50.44(c)(3).

ITAAC: The ITAAC associated with DCD Tier 2, Section 6.2.5 are given in DCD Tier 1, Section 2.11.4.

TS: There are no TS for this area of review.

Topical Reports: There are no topical reports for this area of review.

Technical Reports:

- MHI Technical Report MUAP-10004-P, “Additional Sensitivity Analyses for the DDT Potential and the Mixing in the Containment,” Revision 0, issued March 2010

- MHI Technical Report MUAP-07030, “US-APWR Probabilistic Risk Assessment,” Revision 3, issued June 2011.

US-APWR Interface Issues identified in the DCD: There are no US-APWR interface issues for this area of review.

Site Interface Requirements identified in the DCD: There are no site interface requirements for this area of review.

Cross-cutting Requirements (TMI, USI/GSI, Op Ex): None for this area of review.

RTNSS: There is no RTNSS for this area of review.

10 CFR 20.1406: There are no 10 CFR 20.1406 requirements for this area of review.

CDI: There is no CDI for this area of review.

6.2.5.3 Regulatory Basis

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

1. 10 CFR 50.44(c)(1) requires that the containment must have the capability for ensuring a mixed atmosphere during design-basis and significant beyond DBAs.

2. 10 CFR 50.44(c)(2) requires that the containment must limit the hydrogen concentrations in containment during and following an accident that releases an equivalent amount of hydrogen as 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.

3. 10 CFR 50.44(c)(3) requires that the containment must be able to maintain structural integrity with systems and components capable of performing their function during and after exposure to the environmental conditions created by the burning of hydrogen. Environmental conditions caused by the local detonation of hydrogen must also be included, unless such detonations can be shown as unlikely to occur.

4. 10 CFR 50.44(c)(4)(ii) requires that equipment must be provided for monitoring hydrogen in the containment atmosphere following a significant beyond DBA.

5. 10 CFR 50.44(c)(5) requires that an analysis must be performed that demonstrates containment structural integrity. The analysis must address an accident that releases hydrogen generated from 100 percent fuel clad-coolant reaction accompanied by hydrogen burning. Systems necessary to ensure containment integrity must also be demonstrated to perform their functions under these conditions.

6. GDC 5 requires that SSCs important to safety shall not be shared among nuclear power units unless it can be shown that such sharing will not significantly impair their ability to perform their safety functions.

7. GDC 41 requires that systems are provided as necessary to control the concentration of hydrogen that may be released into the reactor containment following postulated accidents in order to assure that containment integrity is maintained.

8. GDC 42 requires that containment atmospheric cleanup systems be designed to permit periodic inspection of important components in order to assure the integrity and capability of the systems.

9. GDC 43 requires that containment atmospheric cleanup systems be designed to permit periodic functional testing to assure the operability and function of the active components and the operability of the system as a whole, including the

performance of the full sequence which brings the systems into operation, including operation of applicable portions of the protection system, the transfer between normal and emergency power sources and the operation of associated systems.

10. 10 CFR 52.47(a)(8) requires that an application for a standard DC contain the information necessary to demonstrate compliance with any technical requirements set forth in 10 CFR 50.34(f), with the exception of paragraphs (f)(1)(xii), (f)(2)(ix) and (f)(3)(v).

11. 10 CFR 52.47(b)(1) requires that an application for a standard DC contain the proposed ITAAC that are necessary and sufficient to provide reasonable assurance that if the inspections, tests, analyses and acceptance criteria are performed, a facility that incorporates the DC has been constructed and will be operated in conformity with the design certification, the provisions of the Atomic Energy Act of 1954, and the Commission’s rules and regulations.

SRP 6.2.5 Acceptance criteria adequate to meet the above requirements include:

1. In meeting the requirements of 10 CFR 50.44 and GDC 41, to provide systems to control the concentration of hydrogen in the containment atmosphere, materials within the containment that would yield hydrogen gas due to corrosion from the emergency cooling or containment spray solutions should be identified, and their use should be limited as much as practicable.

2. In meeting the requirements of 10 CFR 50.44 and GDC 41, to provide systems to control the concentration of hydrogen or oxygen in the containment atmosphere, the applicant should demonstrate by analysis, for non-inerted containments, that the design can safely accommodate hydrogen generated by an equivalent of a 100 percent fuel clad-coolant reaction, while limiting containment hydrogen concentration, with the hydrogen uniformly distributed, to less than 10 percent (by volume), and while maintaining containment structural integrity.

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 its safety function 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.

4. In meeting the requirements of 10 CFR 50.44, to provide the capability for ensuring a mixed atmosphere in the containment during design bases and significant beyond-DBAs, and of GDC 41, to provide systems as necessary to ensure that containment integrity is maintained, this capability may be provided by an active, passive, or combination system. Active systems may consist of a fan, a fan cooler, or containment spray. For passive or combination systems that use convective mixing to mix the combustible gases, the containment internal structures should have design features, which promote the free circulation of the atmosphere. For all containment types, an analysis of the effectiveness of the method used for providing a mixed atmosphere should be provided. This

analysis is acceptable if it shows that combustible gases will not accumulate within a compartment or cubicle to form a combustible or detonable mixture that could cause loss of containment integrity.

Atmosphere mixing systems prevent local accumulation of combustible or detonable gases, which could threaten containment integrity or equipment operating in a local compartment. Active systems installed to mitigate this threat should be reliable, redundant, single-failure proof, able to be tested and inspected, and remain operable with a loss of onsite or offsite power.

5. In meeting the requirements of 10 CFR 50.44 and GDC 41, regarding the functional capability of the combustible gas control systems to ensure that containment integrity is maintained, the design should meet the provisions of RG 1.7, Revision 3, Section C.1.

6. To satisfy the design requirements of GDC 41:

   A. Performance tests should be performed on system components, such as hydrogen igniters and combustible gas monitors. The tests should support the analyses of the functional capability of the equipment.

   B. Combustible gas control system designs should include instrumentation needed to monitor system or component performance under normal and accident conditions. The instrumentation should be capable of determining that a system is performing its intended function, or that a system train or component is malfunctioning and should be isolated. The instrumentation should have readout and alarm capability in the control room. The containment hydrogen and oxygen monitors should meet the provisions of RG 1.7, Revision 3, Section C.2.

7. To satisfy the inspection and test requirements of GDCs 41, 42, and 43, combustible gas control systems should be designed with provisions for periodic inservice inspection, operability testing, and leak rate testing of the systems or components.

8. In meeting the requirements of 10 CFR 50.44(c)(5), regarding containment structural integrity, an analysis must demonstrate containment structural integrity, using an analytical technique that is accepted by the NRC staff and including sufficient supporting justification to show that the technique describes the containment response to the structural loads involved. The analysis must address an accident that releases hydrogen generated from 100 percent fuel clad-coolant reaction accompanied by combustible gas burning. Systems necessary to ensure containment integrity must also demonstrate the capability to perform their functions under these conditions. One acceptable analytical technique is a demonstration that specific criteria of the ASME Boiler and Pressure Vessel Code, described in RG 1.7, Revision 3, Section C.5, are met.

6.2.5.4 Technical Evaluation

The staff reviewed the US-APWR DCD Tier 2, Revision 3, Section 6.2.5, “Combustible Gas Control in Containment” in accordance with NUREG-0800 SRP Section 6.2.5, “Combustible

Gas Control in Containment" and RG 1.7 Revision 3, "Control of Combustible Gas Concentrations in Containment." The applicant's combustible gas control encompasses the hydrogen monitoring system and the hydrogen ignition system, the combination of which is termed the Containment Hydrogen Monitoring and Control System (CHS) in the DCD.

The staff's review encompassed the following areas specified by Section 6.2.5 of the SRP, 10 CFR 50.34(f)(2)(xvii), 10 CFR 50.44(c), Appendix A to Part 50 GDC 5, 41, 42 and 43, and RG 1.7 Revision 3, and RG 1.155 Appendices A and B:

- Sharing of SSCs 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, as per the requirements of GDC 5.

- Provision for systems with suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities, etc., to control the concentration of hydrogen in the containment atmosphere following postulated accidents to assure that containment integrity is maintained, as per the requirements of GDC 41, 10 CFR 50.44(c)(4)(ii), and 10 CFR 50,34(f)(2)(xvii). The staff used the guidance provided in RG 1.7 Revision 3 and the RG 1.155 appendices, as well as SRP Section 6.2.5 Acceptance Criterion AC 5 and 6B.

- Provision that these systems are designed to permit appropriate periodic inspection and functional testing of important components to assure the integrity and capability and operability of the systems, as per the requirements of GDCs 42 and 43. The staff used the guidance provided in RG 1.7 and the RG 1.155 appendices, as well as SRP Section 6.2.5 Acceptance Criterion 6A and Acceptance Criterion 7.

- Ensuring a mixed containment atmosphere during design-basis and significant beyond design-basis accidents, and that concentrations of hydrogen will not collect in areas where unintended combustion or detonation could cause loss of containment integrity or loss of appropriate mitigating features, as per the requirements of 10 CFR 50.44(c)(1). The staff used the guidance provided in SRP 6.2.5 Acceptance Criterion 4.

- Combustible gas control to limit hydrogen concentrations in containment during and following an accident that releases an equivalent amount of hydrogen as 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, as per the requirements of 10 CFR 50.44(c)(2). The staff used the guidance provided in SRP 6.2.5 Acceptance Criterion 1 and 2.

- Equipment survivability so that containment structural integrity is maintained with systems and components capable of performing their functions during and after exposure to the environmental conditions created by the burning of hydrogen from a fuel clad-coolant reaction involving 100 percent of the fuel cladding surrounding the active fuel region, as per the requirements of

10 CFR 50.44(c)(3). The staff used the guidance provided in SRP 6.2.5 Acceptance Criterion 3.

- Containment structural integrity, demonstrated using an analytical technique that is accepted by the NRC, and which addresses an accident that releases hydrogen generated from 100 percent fuel clad-coolant reaction accompanied by hydrogen burning. Systems necessary to ensure containment integrity must be demonstrated to perform their function under these conditions, as per the requirements of 10 CFR 50.44(c)(5). The staff used the guidance provided in SRP 6.2.5 Acceptance Criterion 8.

6.2.5.4.1 Sharing of Structures, Systems and Components

GDC 5 governs the sharing of SSCs 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. Since the US-APWR standard design describes installation as a single unit, there are no design provisions for sharing combustible gas control system equipment between multiple units. Therefore the staff finds that the requirements of GDC 5 are not applicable to the US-APWR combustible gas control system.

6.2.5.4.2 Provision for Systems to Control the Concentration of Hydrogen in the Containment Atmosphere

The staff reviewed the design to ensure that the relevant requirements of GDC 41 are met as they apply to the functional capability of the combustible gas control systems to ensure that containment integrity is maintained. The staff performed a review of the design features of the combustible gas control system using the guidance in RG 1.7 Revision 3 Section C.1.

The staff reviewed the general description of the design features of the combustible gas control system equipment provided in DCD Tier 2 Section 6.2.5.1 and Tier 2 Section 19.2 of the DCD application. The staff noted that In DCD Tier 2 Chapter 1 Table 1.9.1-1 the applicant stated the application complies with guidance in RG 1.7 Revision 3 with no exceptions.

The staff found that the combustible gas control system in the US-APWR has a design requirement to mitigate the risk associated with combustible gas generation attributable to beyond- DBAs. The system must be designed to ensure that it will operate in the severe accident environment for which it is intended, and over the time span for which it is needed. As described in RG 1.7 Revision 3, Section C.1, equipment survivability expectations under severe accident conditions should consider the circumstances of applicable initiating events, and the environment in which the equipment is relied on to function. The staff noted that in DCD Tier 2 Section 6.2.5.2, the containment hydrogen monitoring and control equipment is designed to have a high confidence of low probability of failure for seismic events of 0.5 g. DCD Tier 2 Section 8.3.2.1 states that the hydrogen igniter batteries are located in the Auxiliary Building, ensuring that they will be available during severe accidents initiated by flooding events. To enhance their reliability to function following seismic events, the batteries will be the same type components as those used in the Class 1E batteries, but procured through the nonsafety-related procurement process. Since seismic events are generally the worst case initiating events both in terms of timing and the amount of equipment that would fail , the staff concludes that the design of the CHS adequately addresses the circumstances of applicable initiating events of a severe accident.

As described in RG 1.7 revision 3, Section C.1, since the design requirements of the combustible gas control system address beyond design-basis combustible gas control, the requirements for QA and redundancy/diversity were evaluated by the staff using appendix A and B of RG 1.155. The applicable review criteria are the following:

- Redundancy, diversity, seismic, single failure, installation of non safety related equipment such that it does not adversely affect safety related equipment (for common cause failure (CCF) and for separation from class 1E power sources)

- Measures should be established to assure that all design related guidelines used to comply with 10 CFR 50.44 and 10 CFR 50.34(f)(2)(ix), GDC 5, 41, 42 and 43 are carried forward to procurement documents and that deviations there from are controlled.

- Measures should be established to ensure that purchased material, equipment, and services conform to the procurement documents.

The staff noted that in DCD Tier 2 Section 6.2.5.2, the hydrogen igniters are installed to ensure that failure does not adversely affect safety related equipment or cause CCF of such equipment. Based on its review of Table 6.2.5-1, “Containment Hydrogen Monitoring and Control DesignParameters,” the staff finds that the combustible gas control equipment located inside containment includes parameters for temperature that is based on the applicant’s severe accident environmental analysis.

In its response to **RAI 803-5891, Question 06.02.05-45**, dated September 9, 2011, the applicant stated that the DCD will be revised to include COL Information Item 19.3(7). This COL item requires an applicant that references the US-APWR design to perform an equipment survivability assessment of the as-built equipment required to mitigate severe accidents (electrical penetrations, hydrogen, igniters, and the containment pressure instrument (wide range)) to provide assurance that they will operate in the environmental conditions resulting from hydrogen burns associated with severe accidents for which they are intended and over the time span for which they are needed.

The staff has reviewed the applicant’s response to this RAI and finds it acceptable because the staff determined that the COL information item ensures that the procured equipment that is relied upon to mitigate severe accidents will consider equipment reliability and operability in the beyond-design-basis accident environmental conditions for the specific facility, consistent with RG 1.7 C 1.

RAI 803-5891, Question 06.02.05-45, is being tracked as a Confirmatory Item.

The staff determined that the severe accident performance parameters in the DCD, along with the requirement for a COL applicant that references the US-APWR standard design, to validate the performance of the procured equipment to these parameters, are sufficient measures to ensure design requirements of the combustible gas control system will be carried forward to procurement documents and deviations from the requirements are controlled, and that purchased materials will conform to the procurement documents. Therefore, the staff concludes the design of the CHS meet the guidance for redundancy, diversity, seismic, and single failure contained in RG 1.155.

As described in RG 1.7 Revision 3, Section C.2, equipment provided for monitoring hydrogen in the containment must be functional, reliable, and capable of continuously measuring the concentration of hydrogen in the containment atmosphere following a beyond-design-basis accident for accident management, including emergency planning. Non-safety related commercial-grade hydrogen monitors can be used to meet these criteria, if such instrumentation meet criteria specified in RG 1.7 Section C.2.

The staff noted that DCD Tier 2 Section 6.2.5.2 states that the containment hydrogen monitor is maintained by the plant surveillance test program. The monitor is located outside containment, and is designed to continuously indicate hydrogen concentration in the MCR after being placed in operation. The hydrogen monitor is supplied by redundant non-class 1E power supplies. Based this DCD information, the staff finds that the system meets the design criteria for non-safety-related commercial grade hydrogen monitors described in RG 1.7 Section C.2.

The staff has reviewed the provisions for the US-APWR combustible gas control system instrumentation that is required to monitor system performance under normal and accident conditions against the requirements of GDC 41. Such instrumentation should be capable of determining that a system is performing its intended function, or that a system train or component is malfunctioning and should be isolated. The staff has reviewed the provisions for the US-APWR combustible gas control system hydrogen monitoring instrumentation against the provisions of 10 CFR 50.34(f)(2)(xvii) and RG 1.7, Revision 3, Section C.2 and SRP Section 6.2.5 Acceptance Criterion 6B that the instrumentation should have readout and alarm capability in the control room.

In Tier 2 Section 6.2.5.5 of the DCD, the applicant stated that the hydrogen monitoring system has readout capability in the MCR. DCD Tier 1 Section 2.11.4 includes ITAAC to verify the existence of these design features. The MCR hydrogen alarm and display is described in DCD Tier 2 Section 6.2.5.2. The staff noted that power is supplied to the instrumentation from two non-Class 1E busses and one non-Class 1E alternate AC gas turbine generator (GTG). Since the instrumentation is located outside containment, the staff found that the materials used to construct the instrumentation would not be subjected to a severe accident environment. Since the monitoring and recording functions are designed to the guidance in RG 1.7 Section C.2.1, the staff concludes the instrumentation is designed to be functional.

After review of the design features of the US-APWR hydrogen monitoring system as described above, the staff concludes the system meets provisions of 10 CFR 50.34(f)(2)(xvii) and RG 1.7, Revision 3, Section C.2, and SRP Section 6.2.5 Acceptance Criterion 6B that the instrumentation should have readout and alarm capability in the control room.

Based on the above, and pending the resolution of the above confirmatory item, the staff has determined that the functional capability of the US-APWR combustible gas control system meets the provisions of RG 1.7, Revision 3, Sections C.1 and C.2. Therefore the staff determined that SRP Acceptance Criterion 5 and 6B are met. Consequently, the staff found that the US-APWR design meets the requirements of 10 CFR 50.44, and GDC 41 as they relate to the functional capability of the combustible gas control systems to ensure that containment integrity is maintained.

6.2.5.4.3 Provision That These Systems Are Designed To Permit Appropriate Periodic Inspection and Functional Testing

The staff reviewed the design to ensure that the relevant requirements of GDCs 42 and 43, regarding periodic inspection and system performance testing which supports analyses are met. Performance tests should be performed on system components, such as hydrogen igniters and combustible gas monitors. The tests should support the analyses of the functional capability of the equipment.

As described in RG 1.7 Revision 3, Section C.1, since the design requirements of the combustible gas control system address beyond design-basis combustible gas control, requirements for QA and redundancy/diversity were evaluated by the staff by usage of Appendices A and B of RG 1.155. The staff used the applicable review criteria therein:

- Inspections, tests, administrative controls, and training necessary for compliance with § 50.44 should be prescribed by documented instructions, procedures, and drawings and should be accomplished in accordance with these documents.

- A program for independent inspection of activities required to comply with § 50.44 should be established and executed by (or for) the organization performing the activity to verify conformance with documented installation drawings and test procedures for accomplishing the activities.

- A test program should be established and implemented to ensure that testing is performed and verified by inspection and audit to demonstrate conformance with design and system readiness requirements. The tests should be performed in accordance with written test procedures; test results should be properly evaluated and acted on.

- Measures should be established to identify items that have satisfactorily passed required tests and inspections.

- Measures should be established to control items that do not conform to specified requirements to prevent inadvertent use or installation.

- Measures should be established to ensure that failures, malfunctions, deficiencies, deviations, defective components, and nonconformances are promptly identified, reported, and corrected.

- Records should be prepared and maintained to furnish evidence that the criteria enumerated above are being met for activities required to comply with § 50.63.

- Audits should be conducted and documented to verify compliance with design and procurement documents, instructions, procedures, drawings, and inspection and test activities developed to comply with § 50.63.

In Tier 2 Section 6.2.5.4 of the DCD application, the applicant stated that preservice and inservice testing and inspections are performed on the hydrogen monitoring system and the hydrogen ignition system. For the hydrogen monitor, the DCD Tier 2 Section 6.2.5.4.1 description of the preoperational test includes the requirement of a test report that includes the test results and an evaluation to verify the performance capability of the hydrogen monitor. This DCD section also describes the preoperational test of the hydrogen ignition system that includes a verification that the igniter surface temperature meets or exceeds the hydrogen ignition temperature specified in Table 6.2.5-1. The staff noted that Tier 1 ITAAC items number 1 and 6 exist in Tier 1 Table 2.11.4-1, to verify the quantity and location of the hydrogen igniters. Based on this description of the applicant's preservice tests of the hydrogen monitoring and ignition system, and the requirement for establishment of a QAP to document and correct deficiencies, the staff determined that such tests will support the analyses of the functional capability of the equipment. Therefore the staff finds the above RG 1.155 review criteria are met. Consequently the staff found that, the US-APWR design adequately addresses GDC 42 and 43 requirements that periodic inspections and performance tests will be performed on system components, such as hydrogen igniters and combustible gas monitors.

After review of information provided in the DCD as discussed above, the staff found that the US-APWR design adequately addresses the GDC 42 and 43 requirements that periodic inspections and performance tests will be performed on system components, such as hydrogen igniters and combustible gas monitors. The staff has concluded that such tests will support the analyses of the functional capability of the equipment.

6.2.5.4.4 Ensuring a Mixed Containment Atmosphere

The staff reviewed the design to ensure that the relevant requirements of GDC 41 are met as it applies to controlling the concentration of hydrogen following postulated accidents to ensure that containment integrity is maintained, and the requirements of 10 CFR 50.44(c)(1) are met regarding provisions for the capability for ensuring a mixed atmosphere in the containment during design basis and significant beyond-design-basis accidents.

As described in RG 1.7, Section C.3, the capability for ensuring a mixed atmosphere can be provided by an active, passive or combination system. The containment should have an analysis of the effectiveness of the method used, and the analysis should demonstrate that combustible gases will not accumulate within a compartment or cubicle to form a combustible or detonable mixture that could cause loss of containment integrity. As described in DCD Tier 2 Section 6.2.5, the US-APWR design relies on a combination system of convective heat transfer in conjunction with containment spray discharges to ensure uniform mixing of the hydrogen and contact with the hydrogen igniters.

The active, safety-related portion of the hydrogen mixing system (containment spray) is described in Tier 2 Section 6.2.2 of the DCD application. The acceptability of the design of the Containment Spray System is discussed in Section 6.2.2 of this report. The staff finds that the Containment Spray System is a safety-related system and has been designed to be reliable, redundant, single-failure proof, able to be tested and inspected, and remain operable with a loss of onsite or offsite power.

The staff reviewed the undimensioned plan and elevation drawings of the containment provided in DCD Chapter 3. The staff found that the physical arrangement of the US-APWR containment is similar to existing large dry ambient containments, with regard to the existence of large open areas that promote a well-mixed atmosphere. However the staff found that a review of the undimensioned drawings alone is not sufficient for a staff finding under 10 CFR 50.44(c)(1), and review of a quantitative hydrogen mixing and analysis was needed.

In DCD Tier 2 Section 19.2.3.3.2, the applicant provided the results of their analysis that the containment design can safely accommodate hydrogen generated by an equivalent of a 100 percent fuel clad-coolant reaction, while limiting containment hydrogen concentration with the hydrogen uniformly distributed to less than 10 percent by volume. Accident progression analysis for hydrogen generation and control was performed by the licensee using the MAAP and GOTHIC computer codes.

In **RAI 480-3711, Question 19-430**, the staff requested that the applicant provide a quantitative assessment of the detonation to deflagration transition (DDT) potential in containment during severe accidents and to also provide quantitative arguments that gases in containment compartments remain well mixed for design basis events to include a spectrum of piping break sizes, containment compartment heights and orientations, and all post blowdown phases.

In its response to **RAI 480-3711, Question 19-430**, dated November 26, 2009, the applicant stated that a quantitative report on DDT potential in containment and containment mixing will be provided in a future report. In March 2010, this report, MUAP-10004-P Revision (0), "Additional Sensitivity Analyses for the DDT potential and the Mixing in the Containment" was provided to the staff for review. The applicant analyzed the containment using a more detailed version of the GOTHIC containment model described above. The report concluded in the report that there is no DDT potential during design basis and severe accident sequences in the US-APWR containment and that the containment atmosphere would be well mixed.

The staff performed confirmatory calculations using MELCOR (Reference 9). The staff's results agreed with the applicants for all DBA events. For beyond DBAs the MELCOR confirmatory analysis identified potential detonable conditions in the RWSP in some severe accident sequences where both containment spray and the hydrogen igniters are unavailable.

On August 31, 2010, the staff issued a follow-up **RAI 627-4926, Question 19-449,** which requested the applicant provide details of the GOTHIC containment model to include input parameters, nodalization, and calculation results to aid in resolving this issue. The staff also requested the applicant to investigate and report on the conditions under which detonable mixtures can be formed in the RWSP. The RAI also stated that scoping calculations performed by the staff indicate that loads from a detonation in the RWSP could be large. The staff requested that the applicant provide an assessment of detonation loads as well as structural capacity of containment for impulse loads. The RAI requested that the applicant assess the risk impact of hydrogen detonation in the RWSP.

In its response to **RAI 627-4926, Question 19-449**, dated November 1, 2010, the applicant provided the results of further analyses that are consistent with the conclusions of the staff regarding the build-up of potentially detonable mixtures in the RWSP under certain severe accident conditions. The applicant included these additional accident scenarios in Revision 3 of the US-APWR Probabilistic Risk Assessment (PRA). (Reference 12)

The staff performed confirmatory calculations and sensitivity studies using MELCOR on these scenarios (Reference 11). The results of the confirmatory analyses agreed with the applicant, and the staff determined that there is a significant increase in the conditional containment failure probability (CCFP) if a combustion event in the RWSP is assumed to cause containment failure. The staff noted that the hydrogen igniters rely upon non-safety-related ac power sources, which are not credited to be available in a SBO scenario. Therefore, the staff issued a followup question, **RAI-752-5614, Question 19-522**, which requested clarification on how non-safety-related power sources were modeled in these accident scenarios in the PRA, and, based on the resulting changes in the CCFP if a combustion event in the RWSP is assumed to cause containment failure, how the underlying containment performance goals are still met.

In February 24, 2011, the staff issued **RAI 696-5543, Question 06.02.05-41**. In May 2, 2011, the staff issued **RAI 751-5709, Question 06.02.05-43**, and In August 11, 2011, the staff issued **RAI 803-5891, Question 06.02.05-44**. These questions requested that the applicant provide a discussion on design features to address potential hydrogen accumulation within the RWSP and more generally clarify how the US-APWR design complies with requirements for ensuring a mixed atmosphere required by 10 CFR 50.44(c)(1) and requirements to limit the concentration of hydrogen below 10 percent by volume in 10 CFR 50.44(c)(2) without such design features.

In a letter dated September 9, 2011, the applicant responded to **RAI 803-5891, Question 06.02.05-44**. In this response, the applicant clarified earlier RAI responses to **RAI 696-5543 Question 06.02.05-41, and RAI 751-5709 Question 06.02.05-43**. The applicant clarified Tier 2 of the DCD to state the hydrogen mixing design basis function of the safety-related containment spray system. The staff has reviewed the applicant's response to this RAI and finds it to be acceptable because the applicant has clearly declared a DBA function of the containment spray system to ensure that the containment is fully mixed in a DBA in DCD Tier 2 Section 6.2.2.2.

RAI 803-5891, Question 06.02.05-44 is being tracked as a Confirmatory Item to ensure that the DCD is revised accordingly.

In the same RAI responses, the applicant stated that severe accident scenarios that result in hydrogen accumulating in the RWSP to greater than 10 percent by volume are possible only when both safety-related ac and alternate ac, nonsafety-related power are unavailable to the hydrogen mixing and control systems. Based on review of the above RAI responses and the design description of the containment atmosphere mixing system in Revision 3 of the US-APWR Tier 2, and Revision 3 of the US-APWR PRA, the staff determined that severe accident scenarios that consider the non-availability of both safety-related and nonsafety-related ac power sources to the active mixing systems are significant contributors to the CDF for this design. The staff noted that the system performance criteria for reliable power for the igniters, in order to ensure that the containment is fully mixed, is not dependent solely on conformance with guidance for design criteria in RG 1.7 Section C 1, but should also be based on design goals that ensure that the containment performance guidelines set forth in SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs," are met. That is, that the CCFP is less than approximately 0.1, and the deterministic goal that containment can maintain its role as a reliable leak-tight barrier approximately 24 hours following the onset of core damage under the more likely severe accident challenges.

Therefore, the staff issued **RAI 873-6168, Question 06.02.05-46**, which requested the applicant to clarify how the structures, systems, and components installed to mitigate the hazard from the generation of combustible gas in containment are designed to ensure that they will operate in those beyond DBA scenarios for which they are intended and over the time span for which they are needed. The staff requested that the response to this question to be consistent with the response to **RAI-752-5614, Question 19-522** discussed above.

In its response to **RAI 873-6168, Question 06.02.05-46**, dated June 27, 2012, the applicant stated that in response to **RAI 871-6121, Question 19-560**, the design of the combustible gas control system has been modified to address the question as described below.

In **RAI 871-6121, Question 19-560**, the staff requested the applicant to clarify how the probabilistic and deterministic containment performance guidelines of SECY-93-087 are met for all severe accident scenarios, including those scenarios in plant damage state (PDS) 5E, which were added to the PRA in response to **RAI 627-4926, Question 19-449,** discussed above.

In its response to **RAI 871-6121, Question 19-560,** dated April 25, 2013, the applicant confirmed that the PDS 5E is considered a more likely severe accident scenario for this design because it is included in those PDS that contribute to 90 percent of the CDF. To mitigate the risk associated with this scenario, the applicant proposed design changes to the combustible gas control system. The applicant changed the hydrogen igniter design from the ac powered configuration to an ac with battery back-up configuration by providing dedicated nonsafety-related batteries to 11 out of 20 igniters. The RAI response included a level 2 PRA evaluation of the modified igniter design which showed that the design change limits the hydrogen concentration in the RWSP to less than 10 percent by volume and shows a decrease in large release frequency. The staff has reviewed the applicant's response to this RAI and the associated DCD Tier 1 and Tier 2 revisions and finds them acceptable because the applicant has shown, for all significant beyond DBAs, that hydrogen concentrations in all containment subcompartments are maintained below 10 percent by volume or have an inerted atmosphere due to high steam concentration, thus the revised design is capable of maintaining hydrogen concentrations in any part of containment to below a level that supports a combustion or a detonation that could cause a loss of containment integrity or loss of appropriate mitigating features. Consequently, the staff finds that the design ensures a mixed atmosphere as defined in 10 CFR 50.44(a)(2).

RAI 873-6168, Question 06.02.05-46, is being tracked as a Confirmatory Item to ensure that the DCD is revised accordingly.

As stated in Tier 2 Section 6.2.5.2 and Section 8.3.2.1, "System description," of the DCD, the containment hydrogen monitoring and control system, which includes the nonsafety-related hydrogen igniters are powered by two non-Class 1E busses capable of cross-connection and backed up by non-Class 1E alternate ac GTG power supplies. In addition, 11 out of 20 of these igniters are backed up by nonsafety-related dedicated batteries with the capacity to provide power for at least 24 hours. The locations of those igniters which are backed up by batteries are identified in Tier 2 Section 6.2.5.2 and conform to the same locations used in the applicant's GOTHIC containment model that demonstrated a mixed containment atmosphere.

The staff noted however, that details of the locations of those igniters that are backed up by nonsafety-related dedicated batteries should be provided in Tier 1 Section 2.11.4.1. In October 2012, the staff informed the applicant that the response to RAI 871-6121, Question 19-560, should be revised to include this information.

In its response to **RAI 871-6121, Question 19-560**, dated April 25, 2013, the applicant committed to modify the DCD to indicate the location of those igniters that are backed up by nonsafety-related dedicated batteries in Tier 1 Section 2.11.4.1. The staff has reviewed the applicant's response to this RAI and finds it to be acceptable because the applicant has provided sufficient Tier 1 information on important details and assumptions associated with the design of the CHS that are used in the severe accident analyses.

RAI 871-6121, Question 19-560 is being tracked as a Confirmatory Item to ensure that the DCD is revised accordingly.

Based on this DCD information, the staff finds that the hydrogen igniter power supply design features meet the guidance for reliability and redundancy for such equipment described in RG 1.7 Revision 3, Section C.1, and therefore the staff concludes that the hydrogen igniters would likely have power in a severe accident.

The staff considers the likelihood of a scenario in which the hydrogen igniter function is unavailable in conjunction with no or late spray actuation is very low because the igniters are capable of being powered from the alternate ac power sources, 11 of which are also backed up by nonsafety related dc batteries. The containment spray system is powered by safety-related power sources and alternate ac power sources.

Based on the above discussion the staff found that the US-APWR combustible gas control system has the capability of ensuring a mixed atmosphere for all design basis and significant beyond DBAs, via reliable systems that are expected to remain available during these postulated accidents, and conforms to RG 1.7 Section C.1. Therefore the staff finds that SRP 6.2.5 Acceptance Criterion 4 is met. Consequently, pending the resolution of the confirmatory items described above, the staff finds that the design meets the requirements of 10 CFR 50.44(c)(1).

6.2.5.4.5 Combustible Gas Control to Limit Hydrogen Concentrations in Containment

The staff reviewed the design to ensure that the relevant requirements of GDC 41, and 10 CFR 50.44(c)(2) are met. In meeting these requirements to provide systems to control the concentration of hydrogen or oxygen in the containment atmosphere, the applicant should demonstrate by analysis, that uniformly distributed hydrogen concentrations in the containment do not exceed 10 percent (by volume) during and following an accident that releases an equivalent amount of hydrogen as would be generated from a 100 percent fuel clad metal-water reaction, or that the post-accident atmosphere will not support hydrogen combustion.

In Tier 2 Section 19.2.3.3.2 of the DCD application, the applicant provided the results of an analysis that demonstrates that the containment design can safely accommodate hydrogen generated by an equivalent of a 100 percent fuel clad-coolant reaction, while limiting containment hydrogen concentration with the hydrogen uniformly distributed to less than 10 percent by volume, and while maintaining containment structural integrity.

Based on the staff review of the applicants analysis described above, the staff finds that the analysis demonstrates that the US-APWR containment design can safely accommodate hydrogen generated by an equivalent of a 100 percent fuel clad-coolant reaction, while limiting containment hydrogen concentration, with hydrogen uniformly distributed, to less than 10 percent (by volume), and while maintaining containment structural integrity. Therefore the staff finds that the guidance of RG 1.7 Section C 3 is met and that SRP 6.2.5 Acceptance Criterion 2 is met. Consequently, the staff finds that the US-APWR containment design meets the requirements of 10 CFR 50.44 (c)(2), and GDC 41 to provide systems to control the concentration of hydrogen or oxygen in the containment atmosphere.

The staff reviewed the design to ensure that the guidance of RG 1.7 Section C 4 on hydrogen gas production is met as it applies to provisions to identify, materials within the containment that would yield hydrogen gas due to corrosion from the emergency cooling or containment spray solutions, and that the use of such materials in the design is limited as much as practicable.

The staff noted that DCD Tier 2 Chapter 1, “Introduction and General Description of the Plant,” Table 1.9.2.-6, “US-APWR Conformance with Standard Review Plan Chapter 6 Engineered Safety Features,” states that there are no exceptions for SRP 6.2.5 Acceptance Criteria 1 and references DCD Tier 2 Section 6.2.5 as the area in the application which contains the supporting justification information. DCD Tier 2 Chapter 1 Table 1.9.1-1 states there are no exceptions for RG 1.7 and references DCD Tier 2 Section 6.2.5.1 and Tier 2 Section 19.2, “Severe Accident Evaluation.”

DCD Tier 2 Subsection 6.1.1.2.1 states that aluminum and zinc have been identified as materials that would yield hydrogen gas by corrosion from the emergency cooling or containment spray solutions in the containment and the use of such materials is limited as much as practicable.

Based on the review of information supplied in the DCD as described above, and considering previous NRC sponsored studies of DBA hydrogen production and control issues (such as NUREG 1150) as described in SECY 03-127, SECY-02-080 and SECY-00-0198, the staff determined that combustible gas generated from DBAs of the US-APWR is adequately addressed. This is because the staff determined that the US-APWR containment is similar to the large, dry containment designs examined in previous NRC studies. These studies showed that the risks associated with hydrogen produced in DBAs, primarily from corrosion from the emergency cooling or containment spray solutions is not significant for those containment designs. Therefore staff determined that the US-APWR adequately addresses hydrogen gas production due to corrosion from the emergency cooling or containment spray solutions, and therefore conforms to the guidance of RG 1.7 Section C 4 and the staff finds that SRP 6.2.5 Acceptance Criterion 1 is met.

Based on the above discussion, the staff found that the US-APWR design meets the relevant requirements of GDC 41, and 10 CFR 50.44(c)(2) as they apply to provisions to control the concentration of hydrogen in the containment atmosphere.

6.2.5.4.6   Equipment Survivability

The staff reviewed the design to ensure that the relevant requirements of 10 CFR 50.44(c)(3) are met regarding equipment survivability. Equipment necessary for achieving and maintaining safe shutdown of the plant and maintaining containment structural integrity should perform its safety function 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.

10 CFR 50.44(c)(3) requires that “Containments that do not rely upon an inerted atmosphere to control combustible gases must be able to establish and maintain safe shutdown and containment structural integrity with systems and components capable of performing their functions during and after exposure to the environmental conditions created by the burning of hydrogen. Environmental conditions caused by local detonations of hydrogen must also be included, unless such detonations can be shown unlikely to occur. The amount of hydrogen to be considered must be equivalent to that generated from a fuel clad-coolant reaction involving 100 percent of the fuel cladding surrounding the active fuel region.” SRP Section 6.2.5, “Combustible Gas Control in Containment,” has as one of the specific areas of review the analyses of the capability of systems or system components to withstand dynamic effects, such as transient differential pressures that would occur early in the blowdown phase of an accident. Based on the applicant’s containment mixing and combustible gas control analysis as discussed above that show either inerted conditions or hydrogen levels less than 10 percent by volume in all subcompartments, the staff finds that detonations of hydrogen are unlikely to occur. Therefore the evaluation of environmental conditions caused by local detonations of hydrogen need not be included in the applicants’ equipment survivability analysis.

In DCD Tier 2 Subsection 19.2.3.3.7, “Equipment Survivability,” the applicant provided the results of their analysis of the survivability of equipment deemed necessary for achieving and maintaining safe shutdown of the plant and maintaining containment structural integrity. Accident progression analysis for post severe accident environmental conditions was performed by the licensee using the MAAP and GOTHIC computer codes. The effectiveness of systems and components was compared against the calculated environmental conditions. Components were screened out based on the location, design and redundancy. The applicant identified Containment Penetrations, Hydrogen Igniters, the Depressurization Valve and the Containment pressure wide range instrument that are exposed to a severe accident environment, and equipment necessary to achieve and maintain safe shutdown of the plant and maintain containment structural integrity.

The staff noted that the applicant included design criteria insights from the equipment survivability study in the DCD for this equipment. In Tier 2 Section 6.2.5.2 of the DCD, the applicant stated that the hydrogen ignition system must maintain its function longer than 10 minutes in a 205 °C (400 °F) atmosphere, and must survive momentary peak temperature of 649 °C (1200 °F) In Tier 2 Section 19.2.3.3.7 of the DCD, the applicant stated that the containment electrical penetrations must maintain their function to supply power to the hydrogen igniters and maintain leak tightness at containment design pressure for 24 hours at design temperature of 149 °C (300 °F) and must survive momentary peak temperature of 205 °C (400 °F). In Tier 2 Section 19.2.3.3.7 of the DCD, the applicant stated that for the containment pressure wide range instrument, it must maintain its function for longer than 2 minutes at 205 °C (400 °F) and must survive momentary peak temperature of 427 °C (800 °F). DCD Tier 2 Section 19.2.3.3.7 states that these design criteria be carried forward to procurement documents and that type tests or analyses will be performed to confirm that the equipment will meet these criteria.

In its response to **RAI 803-5891, Question 06.02.05-45**, dated September 9, 2011, the applicant stated that the DCD will be revised to include COL Information Item 19.3(7). This COL item requires an applicant that references the US-APWR standard design to perform an equipment survivability assessment of the as-built equipment required to mitigate severe accidents (electrical penetrations, hydrogen, igniters, and the containment pressure instrument (wide range)) to provide assurance that the procured equipment will operate in the environmental conditions resulting from hydrogen burns associated with severe accidents for which they are intended and over the time span for which they are needed.

The staff has reviewed the applicant's response to this RAI and finds it acceptable because the staff determined that the COL information item ensures that the procured equipment that is relied upon to mitigate severe accidents will consider equipment reliability and operability in the beyond-DBA environmental conditions for the specific facility, consistent with RG 1.7 C 1.

RAI 803-5891, Question 06.02.05-45, is being tracked as a Confirmatory Item to ensure that the DCD is revised accordingly.

In addition, as stated in DCD Tier 2 Section 6.2.5.4.1, the staff notes that procurement procedures and specifications will be available for inspection at the plant site.

Therefore, based on review of information provided in the DCD as discussed above the staff determined that that the US-APWR design meets SRP Section 6.2.5 Acceptance Criterion 3. Consequently, pending resolution of the confirmatory item mentioned above, the staff finds that the design meets the requirements of 10 CFR 50.44(c)(3) as it relates to equipment survivability, for that equipment necessary for achieving and maintaining safe shutdown of the plant and maintaining containment structural integrity.

6.2.5.4.7 Containment Structural Integrity

The staff reviewed the design to ensure that the relevant requirements of 10 CFR 50.44(c)(5), regarding containment structural integrity are met.

As discussed above, on August 31, 2010, the staff issued **RAI 627-4926, Question 19-449,** which requested the applicant to provide an assessment of detonation loads as well as structural capacity of containment for impulse loads due to hydrogen combustion. The RAI specifically requested that the applicant assess the risk impact of hydrogen detonation in the RWSP compartment. Based on the applicant's response to this RAI, dated November 1, 2010, the staff issued **RAI 764-5805 Question 19-531**. This RAI requested the applicant to perform a structural calculation, consistent with the methods described in RG 1.216, to demonstrate that containment structural integrity requirements of 10 CFR 50.44(c)(5) are satisfied.

In its response to **RAI 764-5805 Question 19-531**, dated June 27, 2012, the applicant stated that the challenge to the containment structural integrity caused by the hydrogen detonation scenario within the RWSP is mitigated by the design hydrogen igniter design modification, and accordingly it is not necessary to provide a detailed evaluation of the effects of hydrogen detonation in the RWSP.

The staff reviewed the applicant's response to this RAI and finds it acceptable because the staff found that the applicant's hydrogen analysis discussed above, which shows containment mixing and hydrogen control, obviates the need for a detailed detonation load analysis on this subcompartment. Therefore, **RAI 764-5805, Question 19-531, is resolved and closed**.

The staff reviewed the applicants' analysis of containment structural integrity to verify that it demonstrates containment structural integrity, using an analytical technique that is accepted by the staff, and to determine if the design meets SRP Section 6.2.5 Acceptance Criterion 8.

As described in RG 1.7, Revision 3, Section C.5, a demonstration that shows that the containment design satisfies Subarticle CC-3720 of Section III of the ASME code is an analytical technique acceptable to the staff to demonstrate that that containment structural integrity is met. Additional guidance on this technique is described in RG 1.136, “Containment Structural Integrity Evaluation for Internal Pressure Loadings Above Design Basis Pressure” Revision 3, Section C.5.

The staff reviewed DCD Tier 2, Chapter 3, Subsection 3.8.1.3.2.2 and Tier 2, Chapter 19, Subsection 19.2.3.3.2 which discusses an analysis that demonstrates that containment structural integrity is maintained in the event of an accident that releases hydrogen generated from 100 percent fuel clad-coolant reaction accompanied by combustible gas burning.

DCD Tier 2 Subsection 19.2.3.3.2 states that the hydrogen generation and control severe accident analysis indicated a maximum pressure in the containment under the adiabatic isochoric complete combustion of hydrogen condition of 875.6 kPa, absolute (127 psi absolute). This pressure is lower than the containment ultimate pressure of 1489.3 kPa, absolute (216 psi absolute).

DCD Tier 2 Subsection 3.8.1.3.2.2 states that the US-APWR containment integrity is maintained by satisfying Subarticle CC-3720 of the ASME code, Section III which considers the pressure and dead load combination independently during an accident loading that releases hydrogen generated from 100 percent metal-water reaction of the fuel cladding and accompanied by hydrogen burning. As described in DCD Tier 2 Subsection 19.2.4.1, the staff found that the applicant has shown that under these conditions, the loadings do not produce strains in the PCCV liner in excess of the limits established in Subarticle CC-3720 of the ASME Code Section III.

Therefore, after reviewing information provided in the DCD the staff has found that the applicant provided an analysis of containment structural integrity that was in accordance with RG 1.7 Section C 5 and RG 1.136 Section C 5 guidance. Therefore the staff found that SRP 6.2.5 Acceptance Criterion 8 is met. Consequently, the staff finds that the US-APWR meets the requirements of 50.34(f)(3)(v)(A)(1) and 10 CFR 50.44(c)(5) in regards to containment structural integrity. Additional staff findings on containment structural integrity in response to hydrogen loads associated with a severe accident are in Section 19.2.4 of this report.

6.2.5.5 Combined License Information

There are no COL item numbers in DCD Table 1.8-2 related to this review area. The staff finds that there are no COL information items that are needed for this review area.

6.2.5.6 Conclusions

The staff has reviewed combustible gas control in containment for the US-APWR design in accordance with SRP 6.2.5. On the basis of its review, pending the resolution of the confirmatory items mentioned above, the staff concludes that the US-APWR design meets the requirements of GDC 41, 42, 43, 10 CFR 50.44(c)(1) through (5) and 10 CFR 52.47(a)(8) by means of a system for hydrogen control and accompanying analyses that demonstrate that, for postulated accidents, containment atmosphere is well mixed, combustible gas is controlled to suitably limit hydrogen concentrations in containment, necessary equipment will survive the post accident environment, and that containment structural integrity is maintained. Because the US-APWR design is a single unit, GDC 5 is not applicable.

6.2.5.7 References

1. U.S. Nuclear Regulatory Commission. NUREG-1150, "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants," NRC: Washington, DC. Issued February 1987. (ADAMS Accession Number ML063540601).

2. U.S. Nuclear Regulatory Commission. NUREG/CR-5662, "Hydrogen Combustion, Control, and Value-Impact Analysis for PWR Dry Containments." NRC: Washington, DC. Issued May 1991. (ADAMS Accession Number ML063460463).

3. U.S. Nuclear Regulatory Commission. NUREG/CR-2726, "Light Water Reactor Hydrogen Manual." NRC: Washington, DC. Issued August 1983. (ADAMS Accession Number ML071620344).

4. SECY-00-0198 from NRC Executive Director for Operations (EDO) to The Commissioners, "Status report on Study of Risk-Informed Changes to the technical Requirements of 10 CFR Part 50 (Option 3) and Recommendations on Risk-Informed changes to 10 CFR 50.44 (Combustible Gas Control), issued September 14, 2000. (ADAMS Accession Number ML003747725).

5. SECY-02-080 from NRC Executive Director for Operations (EDO) to The Commissioners, "Proposed Rulemaking-Risk –Informed 10 CFR 50.44, "Combustible Gas Control in Containment" (WITS 20020003)," issued May 13, 2002. (ADAMS Accession Number ML021080664).

6. SECY-03-127 from NRC Executive Director for Operations (EDO) to The Commissioners, "Final Rulemaking-Risk –Informed 10 CFR 50.44, "Combustible Gas Control in Containment." Issued July 24, 2003. (ADAMS Accession Number ML031670912).

7. U.S. Nuclear Regulatory Commission. NUREG/CR-6433, "Containment Performance of Prototypical Reactor Containments Subjected To Severe Accident Conditions." NRC: Washington, DC. Issued December 1996. (ADAMS Accession Number ML063400168).

8. E. C. Neidel, J. G. Castle, Jr. and J. E. Gover, "A Review of H2 Detection in Light Water Reactor Containments," NUREG/CR-2080, SAND8I-0326,R3, (Dec.), 1981, Sandia National Laboratories, Albuquerque, NM.

9. A.Krall, S.Choi, M.Zavisca and M. Khatib-Rahbar, "Analysis of Combustion-Induced Containment Failure Probability for U.S. Advance Pressurized Water Reactor", ERI/NRC 10-205, August 2010, Energy Research Inc.

10. SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs," issued April 2, 1993. (ADAMS Accession Number ML003708021).

11. A.Krall and M. Khatib-Rahbar, "Hydrogen Distribution Inside US-APWR Containment (Scenario AP102S-NG)", August 2011, Energy Research Inc.

12. Mitsubishi Heavy Industries, “US-APWR Probabilistic Risk Assessment,” MUAP-07030(R3), 2011.

6.2.6 Containment Leakage Testing

6.2.6.1 Introduction

Section 6.2.6, “Containment Leakage Testing” of the US-APWR DCD, Revision 3, addresses the leakage rate testing program for the reactor containment. Testing requirements provide assurance that the containment leak tight integrity can be verified throughout the service lifetime. Additionally, periodic Type A, B, and C testing must be performed to assure that leakage through the containment systems and components that penetrate the primary containment do not exceed allowable leakage rate values of the TS.

6.2.6.2 Summary of Application

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

DCD Tier 2: The applicant provided a Tier 2 description of containment leak testing in DCD Tier 2, Section 6.2.6, “Containment Leak Testing,” summarized here, in part, as follows:

> As specified in DCD Tier 2, Revision 3, Section 6.2.6 and by DCD Tier 2, Revision 3, Section 16, TS 5.5.16, the applicant indicated the following:
>
> Section 6.2.6 of the US-APWR DCD states that the US-APWR leakage rate testing program implements RG 1.163 and includes the following elements:
>
> - Maximum allowable containment leakage rate.
> - Design to permit leak rate testing.
> - Pretest requirements.
> - Venting of fluid systems in containment atmosphere.
> - Stabilization of containment condition (temperature, pressure, and humidity).
> - Testing methodology.
> - Acceptance criteria.
>
> Section 6.2.6 of the US-APWR DCD states that the containment leakage testing program and limits implement the performance-based leakage testing requirements of 10 CFR Part 50 Appendix J, (Option B) using the specific methods and guidance provided in NEI 94-01, “Industry Guideline for implementing Performance-Based Option of 10 CFR Part 50, Appendix J”, and ANSI/ANS 56.8-1994, “Containment System Leakage Testing Requirements” as modified and endorsed by the NRC in RG 1.163 and is designed to comply with the requirements of GDC 52, 53, 54, of Appendix A of 10 CFR 50.
>
> The program elements and limits are identified in Chapter 16 (TS) in DCD Tier 2, Revision 3, which is based on NUREG-1431, Revision. 3.1 “Standard Technical Specifications Westinghouse Plants”, Section 5.5, Appendix J Option B.
>
> DCD Tier 2, Revision 3, Sections 6.2.6.1 through 6.2.6.5 discuss the main aspects of the containment leakage rate testing program including integrated leak rate testing,

penetration testing, valve testing, scheduling and reporting of periodic tests, and special testing requirements. DCD Tier 2, Revision 3, Figure 6.2.4-1, "Containment Isolation Configurations" illustrate the provisions for containment penetration testing. DCD Tier 2, Revision 3, Table 6.2.4-3, "List of Containment Penetrations and System Isolation Positions" include the leak rate test types to be performed for each penetration/valve.

The applicant indicated that the proposed schedule and test report content requirements associated with performing pre-operational and periodic leakage rate testing is in accordance with the guidance provided in, NEI 94-01,and ANSI/ANS 56.8-1994 as modified and endorsed by the NRC in RG 1.163.

ITAAC: The ITAAC associated with DCD Tier 2, Section 6.2.6 are given in DCD Tier 1, Section 2.2.

TS: The TS associated with DCD Tier 2, Section 6.2.6 are given in DCD Tier 2, Chapter 16, Sections 3.6.1, B3.6.1, 3.6.2, B3.6.2, and 5.5.16. TS 5.5.16, "Containment Leakage Rate Testing Program." Key TS parameters and the bases on which the staff accepts these parameters are summarized in Table 6.2.6-1 in Section 6.2.6.4.5 of this SER.

Topical Reports: There are no topical reports for this area of review.

Technical Reports: There are no technical reports for this area of review.

US-APWR Interface Issues identified in the DCD: There are no US-APWR interface issues for this area of review.

Site Interface Requirements identified in the DCD: In US-APWR DCD Revision 3, Section 6.2.8 the applicant provides the following items associated with DCD Section 6.2.6 that will be provided by the COL applicant:

COL6.2 (8) The COL applicant is responsible for identifying the implementation milestone for the containment leakage rate testing program described under 10 CFR 50 Appendix J.

Cross-cutting Requirements (TMI, USI/GSI, Op Ex): None for this area of review.

RTNSS: There is no RTNSS for this area of review.

10 CFR 20.1406: There are no 10 CFR 20.1406 requirements for this area of review.

CDI: There is no CDI for this area of review.

6.2.6.3 Regulatory Basis

The relevant requirements of the Commission's regulations and the associated acceptance criteria are given in Section 6.2.6, "Containment Leakage Testing," of NUREG-0800, the SRP, and are summarized below. Review interfaces with other SRP sections can be found in Section 6.2.6 of NUREG-0800.

1. 10 CFR 50.54(o), "Conditions of licenses," requires the primary reactor containment to 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. GDC 52 requires that the reactor containment and other equipment that may be subjected to containment test conditions shall be designed so that periodic integrated leakage rate testing can be conducted at containment design pressure.

3. GDC 53 requires that the reactor containment be designed to permit (1) periodic inspection of all important areas such as penetrations, (2) an appropriate surveillance program and (3) periodic testing at containment design pressure of the leak tightness of penetrations which have resilient seals and expansion bellows.

4. GDC 54 requires that piping systems penetrating primary containment shall be provided with leak detection, isolation, and containment capabilities having redundancy, reliability, and performance capabilities which reflect the importance to safety of isolating these piping systems. Such piping systems shall be designed with a capability to test periodically the operability of the isolation valves and associated apparatus and to determine if valve leakage is within acceptable limits.

5. 10 CFR 100.11 requires that as an aid in evaluating a proposed nuclear power plant site, an applicant should assume the expected demonstrable leakage rate from the containment. As stated in SRP Section 6.2.6 Section II, nuclear power plant leakage rate testing experience shows that a design leakage rate of 0.1 percent per day provides adequate margin above typically measured containment leakage rates and is compatible with current leakage rate test methods and test acceptance criteria. Therefore, the minimum acceptable design containment leakage rate should not be less than 0.1 percent per day.

6. 10 CFR 100.10 addresses factors to be considered when evaluating nuclear power plant sites and includes the safety features that are engineered into the facility. The secondary containment of dual type containments, which provide for a controlled, filtered release to the environs of leakage from the primary reactor containment, is such an engineered safety feature, whose effectiveness should be periodically tested as stated in Appendix J, Option A, in Section IV.B.; Option B plants should also be tested in the same way. In so doing, the leakage limit of the secondary containment is acceptable if it is based on the limit used in the analysis of the secondary containment depressurization time. The test should be conducted at each refueling outage or at a comparable frequency. The test limit should be consistent with the limit used for direct leakage in the analysis of the radiological consequences by the organization responsible for analysis of radiological consequences. Potential bypass leak paths (identified in accordance with BTP 6 3, “Determination of Bypass Leakage Paths in Dual Containment Plants”) should be locally leakage rate tested in accordance with the requirements of Appendix J.

7. 10 CFR Part 50, Appendix J, states that the primary reactor containment must be designed such that the maximum allowable leakage rate, L_a, as specified in the TS or associated bases when tested at the calculated peak internal containment

pressure, P_a, related to the DBA 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.

8. 10 CFR 52.47(b)(1) requires that an application for a standard DC contain the proposed ITAAC that are necessary and sufficient to provide reasonable assurance that if the inspections, tests, analyses and acceptance criteria are performed, a facility that incorporates the DC has been constructed and will be operated in conformity with the DC, the provisions of the Atomic Energy Act of 1954 and the Commission's rules and regulations.

Acceptance criteria adequate to meet the above requirements include:

1. Appendix J, Option B, RG 1.163 endorses NEI 94-01, Revision 0 (with certain exceptions), which provides similar guidance in Sections 8.0 and 9.0. Instrumentation lines that penetrate containment, however, are sometimes isolated for the containment integrated leak rate test (ILRT). To ensure that they are included in the test, the following should be done. Leakage rate testing of instrumentation lines that penetrate containment may be done in conjunction with either the local leak rate tests or the ILRT. Instrumentation lines that are not locally leakage rate tested should not be isolated from the containment atmosphere during the performance of the ILRT. The measured leakage rates from instrumentation lines that are locally leakage rate tested, and also isolated during ILRTs, should be added to the ILRT result. Provisions should be made to ensure that instrumentation lines isolated during the ILRT are restored to their operable status following the test.

2. The reactor containment leakage rate testing program, as described in the DCD is acceptable if, under Option B, it meets the requirements stated in Option B of Appendix J to 10 CFR Part 50 and, under V.B.2 and V.B.3 of Option B, either complies with methods approved by the Commission and endorsed in RG 1.163 and includes a requirement to do so in the TS, or complies with the provisions of some other implementation document which has been adequately justified by the staff, with supporting analyses, and is cited as a requirement in the TS.

3. All leakage rate tests, performed by either pneumatic or hydrostatic means, should have the capability to quantify the leakage rates either explicitly or by a conservative bounding method to satisfy test acceptance criteria in Appendix J and the TS.

4. Appendix J, Option B, RG 1.163 endorses NEI 94-01, Revision 0 (with certain exceptions), which provides similar guidance in Sections 8.0 and 10.0. At the CP or standard DC stage, the applicant should identify all containment isolation valves that will be locally (Type C) leakage rate tested with the test pressure applied in a direction opposite to that which would occur under accident conditions and should commit to justify, at the operating license or COL stage, that such testing will result in equivalent or more conservative results.

5. With regard to the application of Appendix J, Option B, Section III.B., for leakage rate testing of main steam isolation valves (MSIVs) in BWR plants, if a test

pressure of less than P_a (calculated peak containment accident pressure) is necessary, the test pressure and the test acceptance criteria should be justified and included in the plant TS. Further, this will require an exemption from the applicable Appendix J requirement and the applicant or licensee must request one, with appropriate justification. In addition, it is typical for BWR applicants or licensees to request that MSIV local leakage rates be excluded from the Type A test leakage rate and the sum of Type B and Type C leakage rates, which would require exemption from Option A, Sections III.A, III.B.3, and III.C.3, or Option B, Sections III.A and III.B. Such exemptions must also be requested and justified, and may readily be combined with an exemption request regarding test pressure and acceptance criteria, mentioned above.

6. NEI 94-01, Revision 0 (Section 6.0), and ANSI/ANS 56.8 1994 (Section 3.3.1) state that Type B or Type C tests are not required for the following cases:

   a. Containment boundaries that do not constitute potential containment atmospheric leakage pathways during and following a design basis (DB) (LOCA);

   b. Containment boundaries sealed with a qualified seal system;

   c. Test connections, vents, and drains between containment isolation valves, which:

      i. are one inch or less in size,
      ii. administratively secured closed, and
      iii. consist of a double barrier (e.g., two valves in series, one valve with a nipple and cap, one valve and a blind flange).

This guidance may be applied to either Option A or Option B of Appendix J. Examples of Case No. 1 are lines that terminate below the minimum post-accident water level of the suppression pool in a BWR or the recirculation sump in a PWR.

For Case No. 2, a qualified seal system is defined in ANSI/ANS 56.8 1994 as a system that is capable of sealing the leakage with a liquid at a pressure no less than 1.1 Pa, for at least 30 days following the DB LOCA. The staff's position is that the analysis of the sealing capability includes the assumption of the most limiting single failure of any active component. Also, unless there is a virtually unlimited supply of sealing liquid (such as from a suppression pool or recirculation sump), limits for liquid leakage rate should be assigned to these valves based on analysis and included in the plant technical specifications. Periodic leakage rate testing, using the sealing liquid as the test medium, is then needed to ensure that the technical specification limits are maintained.

For Case No. 3, to ensure that containment integrity is restored following testing, the test, vent, and drain connections that are used to facilitate local leakage rate testing and the performance of the ILRT should be under administrative control and should be subject to periodic surveillance, to ensure their integrity and to verify the effectiveness of administrative controls.

6.2.6.4 Technical Evaluation

The staff's review of this section is focused to assure that the US-APWR design provisions will permit containment leakage rate testing to be done in accordance with the requirements of Appendix J, 10 CFR Part 50.

The staff reviewed the US-APWR DCD Tier 2, Revision 3, Section 6.2.6 "Containment Leakage Testing" and the proposed TS of DCD Tier 2, Revision 3, Chapter 16 Section 5.5.16 "Containment Leakage Rate Testing Program" in accordance with NUREG-0800 SRP Section 6.2.6, "Containment Leakage Testing" and RG 1.163 "Performance-Based Containment Leak-Test Program". In addition, review of applicable portions of US-APWR DCD Tier 2, Revision 3, Section 6.2.8 "Combined License Information", is included in this review. The applicant's containment leakage testing program is acceptable if it meets the regulatory guidance commensurate with the safety function to be performed. This will ensure that the relevant requirements of 10 CFR 50.54(o), Appendix J "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors", 10 CFR 100.10 "Factors To Be Considered When Evaluating Sites", 10 CFR 100.11 "Determination of Exclusion Area, Low Population Zone, and Population Center Distance", 10 CFR 52.47 "Contents of Applications; Technical Information", 10 CFR Part 50, Appendix A GDC 52, 53, and 54 are met. These requirements are discussed below.

The staff review of the US-APWR containment leakage rate testing program encompassed the following review areas, as identified in SRP Section 6.2.6 and SRP Chapter 16, TS 5.5.16:

- ILRT (Type A tests as defined by Appendix J), including pretest requirements, general test methods, acceptance criteria for preoperational and periodic leakage rate tests, provisions for additional testing in the event of failure to meet acceptance criteria and scheduling of tests,

- Containment penetration leakage rate tests (Type B tests as defined by Appendix J), including identification of containment penetrations general test methods, test pressures and acceptance criteria,

- Containment isolation valve local (Type C tests as defined by Appendix J), including identification of isolation valves, design provisions for such testing, general test methods, test pressures, and acceptance criteria,

- Scheduling and reporting of periodic tests,

- Special testing requirements,

- Proposed technical specification requirements pertaining to containment leakage rate testing,

- COL Action Items.

The staff's findings for each of the above areas are discussed below. The staff's evaluation of the ITAAC is in Chapter 14 of this report.

6.2.6.4.1 Containment Integrated Leakage Rate (Type A) Testing

ILRTs serve to provide assurance that the containment leakage rate, in the event of an accident, will not exceed the values assumed in the analyses of the radiological consequences of DBAs.

6.2.6.4.1.1 Pretest Requirements and General Test Method for Type A Tests

The staff found that Sheets 46 and 47 of DCD Tier 2, Figure 6.2.4-1, show the penetrations to be used in the ILRT. Based on review of this figure, the staff found that permanently installed, penetrations are provided to facilitate controlled pressurization and depressurization of the containment. As stated in DCD Tier 2, Section 6.2.6.1, compressed air equipment suitable to perform the test will be connected to the penetrations to perform the testing.

In Section 6.2.6.1 of the DCD the applicant provided information on the test method for Type A Tests. The staff found that the applicant stated that during the period between the initiation of the containment inspection and the performance of the Type A test, no repairs or adjustments shall be made so that the containment can be tested as close to the “as-is” condition as practical.

Based on review of DCD Tier 2, Revision 3, Section 6.2.6.1 the staff found that Type A Tests will be performed at a test pressure equal to P_a (the calculated accident peak containment pressure) using either a pressure-decay method or a Flowmeter method. The staff found that conforms to the guidance in RG 1.163 and NEI 94-01 Section 5.4, and is therefore acceptable.

The staff finds that the type A test description conforms to the SRP 6.2.6 acceptance criteria because the applicant has specified that no repairs or adjustments be made to the containment prior to the performance of the ILRT. In addition the staff found that the applicant has described LIRT test prerequisites that conform to the guidance in RG 1.163 and NEI 94-01, to the level of detail described in RG 1.206 part III, Section C.I.6.2.6.1. Therefore, the staff found that the Type A test has the capability to quantify the leakage rate either explicitly or by a conservative bounding method to satisfy test acceptance criteria in Appendix J and the TS. Consequently, the staff found that the requirements of GDC 52, GDC 53, Appendix J to 10 CFR Part 50, and 10 CFR 100.11 have been met as these regulations apply to the test method for type A tests.

6.2.6.4.1.2 Acceptance Criteria for Type A Tests; Provisions for Additional Testing

The staff has reviewed the Type A leakage rate testing acceptance criteria provided in DCD Section 6.2.6 and Chapter 16. In US-APWR DCD Tier 2 Revision 3 Chapter 16, TS B 3.6.4-1 the applicant stated that the peak containment internal pressure for the design basis LOCA or DBA, $P_{a,}$ is equal to 59.5 psig (410.28 kPa). In US-APWR DCD Tier 2 Revision 3, Chapter 16, Section 5.5.16, the applicant stated that the maximum allowable leakage rate (L_a) is 0.10 percent of the containment air weight per day at P_a. During the first startup following testing, the leakage rate acceptance criterion will be less than or equal to 0.75 L_a, the staff found that this is in accordance with the provisions of Appendix J to 10 CFR Part 50, SRP Section 6.2.6, and RG 1.163. The staff found that the allowable leakage rate of 0.10 percent per day is conservative with respect to the value used in analyses of the radiological consequences of a LOCA, as cited in DCD Tier 2, Revision 3, Table 15.6.5-4, (i.e. 0.15 percent of the containment air weight per day at P_a.) and is consistent with the provisions of Section 6.2.6 of the SRP. It is, therefore, an acceptable leakage rate acceptance criterion for type A tests.

The staff found that the provisions for additional testing, in the event of failure to meet acceptance criteria conform to the guidance in RG 1.163 and NEI 94-01 Section 9.2.6, to the level of detail described in RG 1.206 part III, Section C.I.6.2.6.1. Consequently, the staff found that the requirements of GDC 52, GDC 53, Appendix J to 10 CFR Part 50 and 10 CFR 100.11 have been met as these regulations apply to the acceptance criteria and provisions for additional testing for type A tests.

6.2.6.4.1.3 Scheduling and Reporting of Type A Tests

As stated in Section 6.2.6.4 of the DCD, the proposed schedule and test report content requirements associated with performing pre-operational and periodic leakage rate testing are in accordance with NEI 94-01, as modified and endorsed by the NRC in RG 1.163. NEI 94-01 in turn references ANSI/ANS-56.8-1994 for the technical contents of the reports. Per DCD COL 6.2(8), the implementation milestones for pre-operational and periodic Type A leak rate tests will be developed by each COL applicant. The staff finds this information acceptable because the DCD description of scheduling and reporting of periodic tests conform to the guidance in RG 1.163 and NEI 94-01 Section 12, to the level of detail described in RG 1.206 Part III, Section C.I.6.2.6.1. Consequently, with exception of the information to be supplied by COL Information Item 6.2(8), the staff found that the requirements of GDC 52, GDC 53, Appendix J to 10 CFR Part 50, and 10 CFR 100 have been met as these regulations apply to the scheduling and reporting of type A tests.

6.2.6.4.2 Containment penetration (Type B) Leakage Rate Testing

Type B tests are intended to detect or measure the leakage rate across pressure-retaining or leakage-limiting boundaries other than containment isolation valves.

6.2.6.4.2.1 Identification of Containment Penetrations

The staff found that DCD Tier 2, Revision 3, Table 6.2.4-3, includes the complete list of penetrations that will receive preoperational and periodic Type B tests. In addition, this table identifies the containment penetrations that are exempt from leakage rate testing and states the reason for their exemption. The staff found that the US-APWR design does not rely on containment boundaries sealed with a qualified seal system. The penetrations subject to Type B testing consist of: one equipment hatch, two air locks, one fuel transfer tube, one penetration for the containment leak rate testing pressure detection line, two penetrations for the containment leak rate testing air supply and return lines, two penetrations for the oil supply and drain line to the RCP motor, and 61 electrical penetrations. The staff finds the listing of containment penetrations acceptable because the DCD description conforms to NEI 94-01, Section 6.0, and ANSI/ANS-56.8-1994 Section 3.3.1 guidance as it applies to penetration and valves for which Type B or Type C tests are not required. Therefore the staff finds that the design conforms to SRP 6.2.6 Section II acceptance criteria as it applies to the identification of penetrations that are subject to type B tests and the identification and justification of those penetrations for which B testing is not required. The staff found that the design conforms to the guidance in RG 1.206 part III, Section C.I.6.2.6.2. Consequently, the staff found that the requirements of GDC 52, GDC 53, Appendix J to 10 CFR Part 50 and 10 CFR 100 have been met as these regulations apply to the identification of containment penetrations.

6.2.6.4.2.2 Design Provisions for Type B Testing

The staff reviewed DCD Tier 2, Revision 3, Section 6.2.4.4 and DCD Figure 6.2.4-1 and found, as stated in DCD Section 6.2.4.4, that there are test connections for the Type B penetrations, including airlocks, hatch, fuel transfer tube, and electrical penetrations. The staff has reviewed the DCD descriptions of these connections as illustrated in DCD Tier 2 Figure 6.2.4-1 and has found them acceptable because the staff found that the design conforms to the guidance in RG 1.163 and NEI 94-01 Section 6 as it applies to design provisions for type B testing that enable application of test pressure in the same direction that would occur during a DB LOCA. Therefore the staff finds that the design meets SRP 6.2.6 Section II acceptance criteria as it applies to design provisions for type B testing. Consequently, the staff found that the requirements of GDC 52, GDC 53, Appendix J to 10 CFR Part 50 and 10 CFR Part 100 have been met as these regulations apply to the design provisions for type B testing.

6.2.6.4.2.3 Provide Analysis of Cooling Requirements for Concrete Adjacent to Hot Penetrations

In **RAI 50-329, Question 06-02.06-11, and RAI 552-4358, Question 06.02.06-29**, the staff requested that the applicant clarify details associated with design features that will provide cooling to the “hot” penetrations of the main steam, blow down, feedwater, RHR, CVCS or any other system piping where the internal temperature exceed 65.5 ºC (150 ºF).

DCD Chapter 3, Figure 3.8.1-8, “Containment Penetrations,” Sheets 12, 13, and 14 depict the containment penetrations for main steam, feed water, and blowdown piping, respectively. These drawings show insulation around the pipes passing through the respective penetrations. However, the shell of each penetration is welded to the wall of the penetrating pipe and the penetration itself has gussets imbedded in the containment concrete. The staff asked the applicant to demonstrate, by providing a heat transfer calculation, how the high temperature of these pipes is dissipated such that the containment concrete does not exceed the 93.3 ºC (200 ºF) limit locally around the penetration as stated in DCD Section 3.8.1.5.3, “Acceptance Criteria with respect to concrete temperatures.”

In its response to **RAI 50-329, Question 06-02.06-11**, and **RAI 552-4358, Question 06.02.06-29,** dated April 16, 2010, the applicant stated that with the results of a heat transfer calculation that demonstrates that the concrete temperatures do not exceed the applicant’s acceptance criteria. Based on review of the RAI response the NRC staff determined that the temperatures in these lines is dissipated such that the surrounding concrete would be less than the 93.3 ºC (200 ºF) limit locally around the penetration as stated in DCD Section 3.8.1.5.3, and **the RAIs were therefore resolved and closed**.

6.2.6.4.2.4 Provisions for Inspection of Containment Penetrations

In **RAI 267-2016, Question 06-02.06-16, RAI 472-3794, Question 06.02.06-27, and RAI 648-4872, Question 06-02.06-31** the staff requested the applicant to confirm and clarify accommodation for inspection of containment penetrations in accordance with GDC 53.

In its response to **RAI 267-2016, Question 06-02.06-16**, dated September 17, 2008, the applicant stated that DCD Section 6.2.4.4 would be revised to include a paragraph that states that inspections will be performed in accordance with requirements of GDC 53, the US-APWR leakage rate testing program, which implements RG 1.163 and the endorsed NEI 94-01 guidance.

In its response to **RAI 472-3794, Question 06.02.06-27,** dated November 27, 2009, the applicant stated that DCD Section 6.2.1.6 would be revised to refer to Section 3.8.1.7 for the description of the initial test program for the containment, which includes construction, preoperational and startup testing. The staff reviewed the response and requested the applicant to clearly state the commitment to design the containment in accordance with GDC 53 in Tier 2 of the DCD. By specifying what areas are considered important are to be inspected.

In its response to **RAI 648-4872, Question 06-02.06-31**, dated November 11, 2010, the applicant stated that the DCD Section 6.2.1.6 would be revised to include a statement that the containment will be designed to permit appropriate periodic inspection of all important areas such as penetrations, the liner intersection with the base concrete inside containment, locations where the floors or platforms are adjacent to the liner and the vicinity of the crane brackets. The staff has reviewed the response and finds it acceptable because the response and the DCD revisions clarify and specify the important areas of the US-APWR containment that should be inspected, and declare that those areas are designed to facilitate inspection. The staff confirmed that DCD Revision 3, dated March 31, 2011, was revised as committed in the RAI response. Accordingly, the staff finds that the applicant has adequately addressed the issue and therefore, **RAI 267-2016, Question 06-02.06-16, RAI 472-3794, Question 06.02.06-27, and RAI 648-4872, Question 06-02.06-31, are resolved and closed**.

6.2.6.4.2.5 Test Method for Type B Tests

Based on review of DCD Tier 2, Revision 3, Section 6.2.6.2 the staff found that Type B Tests will be performed at a test pressure equal to or greater than P_a (the calculated accident peak containment pressure) using either a pressure-decay method or a flowmeter method. The staff found that conforms to the guidance in RG 1.163 and NEI 94-01 Section 6.4, and is therefore acceptable.

The staff has reviewed the Type B test method as described in the DCD Section 6.2.6.2 and found that because the description conforms to the guidance in RG 1.163 and NEI 94-01, to the level of detail described in RG 1.206 part III, Section C. I.6.2.6.2, the staff finds that the type B test description conforms to the SRP 6.2.6 acceptance criteria. Therefore, the staff found that the Type B test has the capability to quantify the leakage rate either explicitly or by a conservative bounding method to satisfy test acceptance criteria in Appendix J and the TS. Consequently, the staff found that the requirements of GDC 52, GDC 53, Appendix J to 10 CFR Part 50, and 10 CFR 100 have been met as these regulations apply to the test method for type B tests.

6.2.6.4.2.6 Acceptance Criteria for Type B Tests

The staff reviewed the Type B leakage rate test acceptance criteria contained in the US-APWR DCD Tier 2 Revision 3 Chapter 6 Section 6.2.6 and Chapter 16, TS Section 5.5.16. The staff found that the Type B leakage rate test results will be combined with the Type C results, in accordance with Appendix J to 10 CFR Part 50. During the first startup following testing, the leakage rate acceptance criterion will be less than 0.6 L_a. The staff found that this acceptance criterion for combined Type B and C leakage is acceptable because it conforms to the guidance of NEI 94-01 Paragraph 10.2, which is endorsed by RG 1.163. In addition the staff found that this criterion is consistent with the Westinghouse Plants, Revision 3.1 Standard TS (NUREG-1431), Section 5.5, Appendix J Option B. Consequently, the staff found that the criterion is in accordance SRP Section 6.2.6, and thus is in accordance with the provisions of Appendix J to 10 CFR Part 50.

In addition, air lock chambers and individual doors must meet the specific leakage rate acceptance criteria identified US-APWR DCD Tier 2 Revision 3 Chapter 16, TS Section 5.5.16. These are:

a) Overall air lock leakage rate is less than or equal to 0.05 L_a when tested at greater than or equal to P_a.
b) For each door, leakage rate is less than or equal to 0.01 L_a when tested at greater than or equal to 10 psig.

The staff found the above acceptance criteria acceptable because the staff found them to be consistent with air lock leakage acceptance criteria in the Westinghouse Plants, Revision 3.1 Standard TS (NUREG-1431), Section 5.5., Appendix J Option B. Because the staff found that US-APWR plant design is no different from current designs in which NUREG 1431 is applicable, in regard to containment airlock design, the staff found that the acceptance criteria are acceptable. Consequently, the staff found that airlock leakage test acceptance criteria are in accordance SRP Section 6.2.6, and thus is in accordance with the provisions of Appendix J to 10 CFR Part 50.

6.2.6.4.2.7 Scheduling and Reporting of Type B Tests

In DCD Tier 2, Revision 3, Section 6.2.6.2, the applicant states that 10 CFR 50 Appendix J Option B Type B testing is initially performed during preoperational testing following completion of the R/B construction, and performed periodically thereafter, as specified in TS, Section 5.5.16, “Containment Leakage Rate Testing Program.”

As stated in Section 6.2.6.4 of the DCD, the proposed schedule and test report content requirements associated with performing pre-operational and periodic leakage rate testing are in accordance with NEI 94-01, as modified and endorsed by the NRC in RG 1.163. NEI 94-01 in turn references ANSI/ANS-56.8-1994 for the technical contents of the reports. Per DCD COL 6.2(8), the implementation milestones for pre-operational and periodic Type B leak rate tests will be developed by each COL applicant. The staff finds this information acceptable because the DCD description of scheduling and reporting of periodic tests conform to the guidance in RG 1.163 and NEI 94-01 Section 12, to the level of detail described in RG 1.206 Part III, Section C.I.6.2.6.1. Consequently, with exception of the information to be supplied by COL Information Item 6.2(8), the staff found that the requirements of GDC 52, GDC 53, Appendix J to 10 CFR Part 50 and 10 CFR 100 have been met as these regulations apply to the scheduling and reporting of type B tests.

6.2.6.4.3 Containment Isolation Valve Local (Type C) Leakage Rate Testing

Type C tests measure containment isolation valve leakage rates.

6.2.6.4.3.1 Identification of Isolation Valves Subject to Type C Testing

The staff found that DCD Tier 2, Revision 3, Table 6.2.4-3, includes the complete list of isolation valves that will receive preoperational and periodic Type C tests. In addition, this table identifies the isolation valves that are exempt from leakage rate testing and states the reason for their exemption. The staff found that the US-APWR design does not rely on containment boundaries sealed with a qualified seal system. The staff found that DCD Tier 2, Revision 3, Figure 6.2.4-1 illustrates the containment valves that will receive preoperational and periodic Type C tests. The staff finds that all containment valves in lines that constitute potential containment atmospheric leakage pathways during and following a DB LOCA are Type C tested. The staff found the listing of valves acceptable because, the listing follows the guidance of NEI 94-01, Section 6.0, and ANSI/ANS-56.8-1994, Section 3.3.1, which states that Type C testing is not required on test connections, vents and drains between containment isolation valves which are one inch size or less, and are administratively secured closed and consist of a double barrier. Therefore the staff finds that the design conforms to SRP 6.2.6 Section II acceptance criteria as it applies to the identification of penetrations that are subject to Type C tests and the identification and justification of those penetrations for which C testing is not required. The staff finds the design conforms to the guidance in RG 1.206 part III, Section C.I.6.2.6.2. Consequently, the staff found that the requirements of GDC 52, GDC 53, Appendix J to 10 CFR Part 50 and 10 CFR 100 have been met as these regulations apply to the identification of containment isolation valves subject to Type C testing.

6.2.6.4.3.2 Design Provisions for Type C Testing

The staff reviewed DCD Tier 2, Revision 3, Section 6.2.4.4 and DCD Figure 6.2.4-1 and found, as stated in DCD Section 6.2.4.4, that there are test connections for containment isolation valves subject to Type C testing.

The staff found that the test connections, vents and drains between containment isolation barriers enable Type C testing in the direction that would be experienced by the valve in a DB LOCA. Such test connections, vents and drains are under administrative control and are subject to periodic surveillance to ensure their integrity and to verify the effectiveness of administrative controls.

The staff has reviewed the DCD descriptions of these connections as illustrated in DCD Tier 2 Figure 6.2.4-1 and has found them acceptable because the design conforms to the guidance in RG 1.163 and NEI 94-01 Section 6 as it applies to design provisions for Type C testing that enable application of test pressure in the same direction that would occur during a DBLOCA. Therefore the staff finds that the design meets SRP 6.2.6 Section II acceptance criteria as it applies to design provisions for Type C testing. Consequently, the staff found that the requirements of GDC 52, GDC 53, Appendix J to 10 CFR Part 50 and 10 CFR 100 have been met as these regulations apply to the design provisions for Type C testing.

6.2.6.4.3.3 Test Method for Type C Tests

In DCD Tier 2, Revision 3, Section 6.2.6.3, the applicant stated that Type C test methods and techniques are consistent with ANSI/ANS 56.8-1994. In DCD Tier 2 Section 6.2.6.3 the applicant stated that no containment isolation valves are to be locally (Type C) tested with the test pressure opposite to that which would occur during accident conditions. Based on review of DCD Tier 2, Revision 3, Section 6.2.6.3, Tier 2 Table 6.2.4-3, Tier 2 P&IDs of systems that are listed on Tier 2 Table 6.2.4-3, and Tier 2 Figure 6.2.4-1, the staff found that no containment isolation valve is configured to be locally (Type C) tested with the test pressure opposite to that which would occur during accident conditions. Based on review of DCD Tier 2, Revision 3, Section 6.2.6.3, the staff found that since Type C Test methods and techniques are consistent with ANSI/ANS 56.8-1994, Type C tests will be performed at a test pressure equal to or greater than P_a (the calculated accident peak containment pressure) using either a pressure-decay method or a flowmeter method.

The staff has reviewed the Type C test method as described in the DCD and found that because the description conforms to the guidance in RG 1.163 and NEI 94-01, to the level of detail described in RG 1.206 part III, Section C. I.6.2.6.3, the Type C test description conforms to the SRP 6.2.6 acceptance criteria. Therefore, the staff found that the Type C test has the capability to quantify the leakage rate either explicitly or by a conservative bounding method to satisfy test acceptance criteria in Appendix J and the TS. Consequently, the staff found that the requirements of GDC 52, GDC 53, Appendix J to 10 CFR Part 50 and 10 CFR 100 have been met as these regulations apply to the test method for Type C tests.

6.2.6.4.3.4 Acceptance Criteria for Type C Tests

The staff has reviewed the Type C leakage rate testing acceptance criteria contained in US-APWR DCD Tier 2, Revision 3, Chapter 6, Section 6.2.6 and Chapter 16, TS Section 5.5.16. The staff found that the Type C leakage rate test results will be combined with the Type B results, in accordance with Appendix J to 10 CFR Part 50. During the first startup following testing, the leakage rate acceptance criterion will be less than 0.6 L_a. The staff found that this acceptance criterion for combined Type B and C leakage is acceptable because it conforms to the guidance of NEI 94-01 paragraph 10.2, which is endorsed by RG 1.163. In addition the staff found that this criterion is consistent with the Westinghouse Plants, Revision 3.1 Standard TS (NUREG-1431), Section 5.5., Appendix J Option B. Consequently, the staff found that this value is in accordance SRP Section 6.2.6, and thus is in accordance with the provisions of Appendix J to 10 CFR Part 50.

6.2.6.4.3.5 Scheduling and Reporting of Type C Tests

In DCD Tier 2, Revision 3, Section 6.2.6.3, the applicant stated that 10 CFR 50 Appendix J Option B Type C testing is initially performed during preoperational testing following completion of the R/B construction. The first periodic Type C tests are performed at a frequency of at least once per 30 months until acceptable performance is established in accordance with NEI 94-01 with subsequent testing frequencies determined in accordance with NEI 94-01 as specified in the containment leakage rate testing program, not to exceed 60 months consistent with RG 1.163.

As stated in Section 6.2.6.4 of the DCD, the proposed schedule and test report content requirements associated with performing pre-operational and periodic leakage rate testing are in accordance with NEI 94-01, as modified and endorsed by the NRC in RG 1.163. NEI 94-01 in turn references ANSI/ANS-56.8-1994 for the technical contents of the reports. Per DCD COL 6.2(8), the implementation milestones for pre-operational and periodic Type C leak rate tests will be developed by each COL applicant. The staff finds this information acceptable because the DCD description of scheduling and reporting of periodic tests conform to the guidance in RG 1.163 and NEI 94-01 Section 12, to the level of detail described in RG 1.206 Part III, Section C.I.6.2.6.1. Consequently, with exception of the information to be supplied by COL item 6.2(8), the staff found that the requirements of GDC 52, GDC 53, Appendix J to 10 CFR Part 50, and 10 CFR 100 have been met as these regulations apply to the scheduling and reporting of Type C tests.

6.2.6.4.4 Special Testing Requirements

Based on review of DCD Tier 2, Revision 3, Section 6.2.6, the staff finds that the US-APWR design does not rely on containment boundaries sealed with a qualified seal system.

Consequently the staff found that there are no special testing requirements that are associated with a seal system. In DCD Tier 2, Revision 3, Section 6.2.6.5, the applicant stated that because the US-APWR design does not have a secondary containment, or a sub-atmospheric primary containment, there are no special testing requirements for these design features. However, based on staff review of DCD Tier 2, Sections 6.2.6, 6.2.3, 6.5.3.2, and the Chapter 15 accident analyses, the staff determined that some credit is taken by the applicant for the action of the AEES in concert with the annulus and containment penetration areas to capture primary containment leakage through an ESF filter system prior to release to the environment. As stated in DCD Tier 2 Table 15.6.5-4, the LOCA radiological consequence analysis assumes a primary containment leak rate of 0.15 percent of the containment air weight per day for the first 24 hours after a postulated accident. The analysis also assumes that 50 percent of this 0.15 percent per day leak rate, or 0.075 percent of the containment air weight per day at P_a, is captured and processed through a 99 percent efficient ESF filtration system prior to release to the environment. The remaining fifty percent of the primary containment leakage (0.075 percent of the containment air weight per day at P_a) is released directly to the environment. The staff noted that 10 CFR Part 50, Appendix J Part IV B states that multiple barrier containments shall be subject to individual leakage rate tests in accordance with procedures specified in the TS and associated bases. SRP Section 6.2.3 states that primary containment bypass leakage paths are to be identified and associated bypass leakage rates are to be determined.

In **RAI 866-6149, Question 06.02.06-34**, the staff requested that that the applicant clarify US-APWR DCD Revision 3, Tier 2 Section 6.2.6.5, to be consistent with DCD Sections 6.2.3, 6.5.3.2, and the Chapter 15 accident analyses, as they apply to the SSCs outside of the primary containment structures that function to collect and process assumed primary containment leakage. The staff requested the applicant provide testing requirements for such SSCs or justify why special testing requirements to verify performance are not required.

In this RAI, the staff also requested that the applicant address SRP 6.2.3 Acceptance Criterion 4A and 4B as they apply to GDC 43 and 10 CFR Part 50, Appendix J, requirements for secondary containment system testing which state:

A. The fraction of primary containment leakage bypassing the secondary containment and escaping directly to the environment should be specified. BTP 6-3 provides guidance for detecting leakage paths to the environment which may bypass the secondary containment. The periodic leakage rate testing program for measuring the fraction of primary containment leakage that may directly bypass the secondary containment and other contiguous areas served by ventilation and filtration systems should be described. Individual tests should be according to procedures from TS or their bases.

   With regard to this criterion, the staff requested the applicant to quantify the bypass leakage paths in the secondary containment design in DCD Section 6.2.3. The staff requested the applicant describe the periodic leakage rate testing program for measuring the fraction of primary containment leakage that may directly bypass the containment penetration areas and annulus, served by the AEES in DCD Section 6.2.6.5. The staff also requested the applicant clarify the description of this program, in DCD Chapter 16, TS Section 5.5.16.

B. There should be provisions in the design of the secondary containment system for inspections and monitoring of the functional capability. Preoperational and periodic test programs determine the depressurization time, the secondary containment in-leakage rate, the uniformity of negative pressure throughout the secondary containment, and other contiguous areas, and the potential for ex-filtration.

With regard to the above criterion, the staff requested the applicant clarify DCD Tier 2 Section 6.2.3 to describe the provisions for secondary containment functional capability as they relate to the containment penetration areas and annulus, served by the AEES.

In its response to **RAI 866-6149, Question 06.02.06-34**, dated January 6, 2012, the applicant confirmed the staff understanding that the penetration areas within the R/B in conjunction with the AEES function to contain and filter any primary containment leakage to these areas. The applicant stated that all containment penetrations and containment isolation valves with exception of the main steam and feedwater penetrations are enclosed by the penetration areas and do not extend beyond secondary containment barriers provided by these areas. The applicant stated that the AEES in conjunction with the penetration areas act as a partial secondary containment for the US-APWR design. The applicant provided DCD markups of Section 6.2.3 and Table 6.2.4-3 that discuss penetrations that are potential secondary containment bypass leakage paths.

In the RAI response, the applicant stated that all potential bypass leakage paths are tested as part of the Primary CLRT, and this program is the means to quantify and track the amount of potential leakage bypassing the AEES to ensure that the assumptions in the safety analyses are met. The applicant provided DCD markups of Section 6.2.6.5 to include this discussion.

In the RAI response, the applicant stated that Type C leakage as monitored by the CLRT provides a conservative measurement of secondary containment bypass leakage, and Type C test results for individual valves as part of this program will be used to ensure that the total secondary containment bypass leakage is maintained below the bypass leakage amount assumed in the offsite dose analysis. The staff reviewed the RAI response and determined a follow up RAI was necessary.

In a follow up **RAI 918-6361, Question 06.02.06-35**, the staff requested that that the applicant clarify DCD Tier 2, Revision 3, Section 6.2.6.5 to state the additional acceptance criteria that will be used as part of the CLRT to ensure that assumptions used in the safety analyses regarding the secondary containment are met.

In its response to **RAI 918-6361, Question 06.02.06-35**, dated June 7, 2012, the applicant stated that Section 6.2.6.5 of the DCD will be revised to state that leakage paths that may bypass the penetration areas and the AEES are identified in Tier 2 Table 6.2.4-3, and are Type C tested as part of the containment leakage rate test program with an additional acceptance criterion to ensure that the assumptions of the safety analyses regarding secondary containment performance are met. The total combined leakage from all Type C tests shall be below the amount assumed for bypass leakage to the environment stated in DCD Tables 6.5-5 and 15.6.5-4. The applicant provided a secondary containment bypass leakage test criterion value of < $0.50L_a$ for Type C tests. The applicant provided DCD markups of DCD Tier 2, Section 6.2.6.5 to include this discussion and Tier 2, Chapter 16 Section 5.5.16 to include the additional Type C acceptance criterion. The staff reviewed the response and determined a follow up RAI was needed.

In a follow up **RAI 966-6811, Question 06.02.06-36**, the staff requested that the applicant clarify the proposed revision of DCD Tier 2, Revision 3, Chapter 16 TS Section 5.5.16, provided in the response to **RAI 918-6361, Question 06.02.06-35**, to indicate what leakage rate acceptance criteria apply to the Primary CLRT program and which criteria apply to the special testing requirements associated with secondary containment functional performance. In addition, the staff indicated that staff findings on secondary containment functional performance will be reviewed by the staff under Section 6.2.3 of the staff's SER.

In its response to **RAI 966-6811, Question 06.02.06-36**, dated November 6, 2012, the applicant stated that the Containment Leakage Rate test program described in Tier 2, Chapter 16 GTS section 5.5.16 of the DCD will be revised to separate the acceptance criteria associated with primary containment functional performance verification versus the acceptance criterion associated with secondary containment functional performance verification. The applicant provided DCD markups with the response reflecting the change.

The staff has reviewed the applicant's responses to **RAI 866-6149, Question 06.02.06-34; RAI 918-6361, Question 06.02.06-35; and RAI 966-6811, Question 06.02.06-36** and finds them to be acceptable because the staff found that the applicant has identified in the DCD potential secondary containment bypass leakage paths in accordance with BTP 6-3, "Determination of Bypass Leakage Paths in Dual Containment Plants", and has provided an acceptable separate special testing requirement and acceptance criterion to quantify and monitor this leakage to monitor secondary containment functional performance in accordance with SRP 6.2.6 Acceptance Criterion 5.

The staff found applicant's bypass leakage acceptance criterion acceptable because the criterion contains margin.

The criterion contains acceptable margin because the value of the combined as-left primary containment leakage rate for all penetrations subject to Type B and Type C preoperational and periodic operational testing is less than 0.60 L_a. As discussed above, this Type B and C leakage criterion value conforms to the guidance of NEI 94-01 paragraph 10.2, which is endorsed by RG 1.163, and has a margin of 40 percent below L_a.

In addition the value of the integrated primary containment leakage rate test (Type A test) is 0.75 L_a. This criterion value conforms to the guidance of NEI 94-01 paragraph 9.2.5, which is also endorsed by RG 1.163. This represents a margin of 25 percent below L_a. Therefore, the staff has previously concluded that, for plants without any secondary containment, it would be unlikely that a plant, that suffers a LOCA with operating combined Type B and Type C leakage below 0.60 L_a, or integrated leakage below 0.75 L_a would have a post accident primary containment leakage rate greater than 1.0 L_a released to the environment.

As discussed above, the applicant's accident analysis assumes that $0.75L_a$ of particulates is released directly to the environment with no filtration by an ESF system (i.e. 0.075 percent of the containment air weight per day at P_a). Since Type C leakage, which contains all the potential secondary containment bypass leakage pathways, is maintained below the operational leakage 0.5 L_a criterion proposed by the applicant, worst case Type C operational leakage alone would have to increase by an additional 25 percent post accident in order to challenge the $0.75L_a$ unfiltered release assumption in the analysis. In addition, all of this Type C leakage would have to consist of containment isolation valve disk leakage, since bonnet leakage would be released in compartments serviced by the AEES and therefore processed by the secondary containment. The staff considers that a 25 percent increase in type C leakage post-accident unlikely and therefore finds that there is acceptable margin for this secondary containment bypass leakage test criterion.

The staff review of the secondary containment is in Section 6.2.3 of this report. **RAI 866-6149, Question 06.02.06-34; RAI 918-6361, Question 06.02.06-35; and RAI 966-6811, Question 06.02.06-36 are being tracked as Confirmatory Items** to ensure that the DCD is revised accordingly.

Based on the review of the functional performance of the secondary containment as described in Section 6.2.3 of this report, the staff found that the AEES in conjunction with the penetration areas in the R/B function as a partial secondary containment to capture some primary containment leakage. The staff found that the additional acceptance criteria in the CLRT in conjunction with the Tier 2 Chapter 16 Section 3.7.11 TS surveillance requirements associated with the AEES assure performance of the secondary containment function when required. The staff's review of the AEES is discussed in Section 6.5.1 of this SER.

Therefore the staff finds that the partial secondary containment design meets SRP 6.2.6 Section II acceptance criteria as it applies to review of periodic testing of safety features that provide for a controlled, filtered release to the environs of leakage from the primary reactor containment. Consequently, the staff found that the requirements of 10 CFR 100.10 as they relate to the evaluation of engineered safety features of a secondary containment, or a qualified seal system, have been met.

6.2.6.4.5 Technical Specifications

TS 5.5.16, "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 6.2.6-1 below:

Table 6.2.6-1 - Key TS Parameters and Bases

Technical Specification	TS Value	Acceptable	Acceptance Basis
Test Pressure (P_a)	410.2 kPa (59.5 psig)	Yes	P_a is the peak calculated accident pressure of 410.2 kPa (59.5 psig). (DCD Tier 2, Table 6.2.1-1).
Test Pressure (P_a)	410.2 kPa (59.5 psig)	Yes	P_a is less than containment design pressure of 468.8 kPa (68 psig) (DCD Tier 2, Table 6.2.1-1).
Maximum allowable leakage rate (L_a) at P_a	0.1% per day by weight	Yes	DCD Tier 2 Table 15.6.5-4 shows containment leakage at 0.15 percent per day starting at time zero to 24 hours into DBA-LOCA. Since this primary containment leakage rate assumption is greater than the test acceptance criterion, The TS acceptance criterion is conservative with respect to the radiological consequence analysis assumption and therefore is

Technical Specification	TS Value	Acceptable	Acceptance Basis
			acceptable.
Leakage rate acceptance criteria [as-found condition]	≤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]	≤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	≤ 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.

As described in Chapter 16 of this report, the proposed TS for the US-APWR conform to NUREG-0800, Section 16 and NUREG-1431, “Standard Technical Specifications (STS) for Westinghouse Plants.”

Based on the evaluation described above, the staff finds that the CLRT Program described in DCD Tier 2, Section 6.2.6, and the requirements stated in TS Section 5.5.16, satisfy the provisions of RG 1.163. Therefore, the staff finds that the US-APWR TS meets SRP 6.2.6 Section II acceptance criteria as it applies to review of TS related to the CLRT. Consequently, the staff finds that the containment leakage testing program satisfies the requirements of 10 CFR Part 50, Appendix J, Option B and, 10 CFR 50.54(o), as these regulations apply to the TS requirements for the CLRT Program.

6.2.6.5 Combined License Information Items

The following are the COL information items from DCD Table 1.8-2.

Table 6.2.6-2 US-APWR Combined License Information Items

Item No.	Description	Section
6.2(8)	The COL applicant is responsible for identifying the implementation milestone for the containment leakage rate testing program described under 10 CFR 50 Appendix J.	6.2.8

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 DCD Tier 2, Table 1.8-2, for containment leakage testing consideration.

6.2.6.6 Conclusions

With the exception of the above confirmatory items, the staff found that the US-APWR CLRT program as described in the US-APWR Design Certification Document along with the Combined Operating License Action item identified by the applicant, conform 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 5.5.16 includes a requirement for the CLRT program to satisfy the provisions of RG 1.163. As discussed above the staff found that the US-APWR CLRT meets the acceptance criteria in SRP Section 6.2.6 in the areas of review discussed above.

Conformance with the criteria in Section 6.2.6 of the SRP, as described in this section, constitutes an acceptable basis for satisfying 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.

Such compliance assures that leaktight integrity of the containment can be verified periodically throughout its service life to ensure that leakage rates are maintained within the limits of the TS. Maintaining containment leakage rates within such limits assures that, the containment design reflects those assumptions used for the design and performance of engineered safety features in site selection as required by 10 CFR 100.10 and 10 CFR 100.11, and that, in the event of any radioactivity releases within the containment, the radiological release through containment leak paths will not be in excess of the acceptable limits for the site.

6.2.7 Fracture Prevention of Containment Pressure Vessel

6.2.7.1 Introduction

The US-APWR reactor containment system consists of a PCCV with a ¼ in (6.4 mm) steel plate anchored to the concrete inner wall. As defined in the SRP, 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 the operating, maintenance, testing and postulated accident conditions. Reactor containment pressure boundary materials in the US-APWR reactor containment system include the ferritic portions of the containment vessel and all penetration assemblies or appurtenances attached to the containment vessel; all piping, pumps and valves attached to the containment vessel, or to penetration assemblies out to and including the pressure boundary materials of any valve required to isolate the system and provide a pressure boundary for the containment function.

6.2.7.2 Summary of Application

DCD Tier 1: The Tier 1 information associated with this section is found in Tier 1 Section 2.11, “Containment Systems,” which states that the PCCV is designed to retain its structural integrity under design pressures of 68 psig (570 kPa).

DCD Tier 2: The applicant has provided a Tier 2 description of the design features that prevent fracture of the containment pressure vessel in Section 6.2.7, “Fracture Prevention of Containment Pressure Vessel,” which states the following:

> Ferritic containment pressure boundary materials meet the fracture toughness criteria and requirements for testing identified in Article NE-2000 of Section III, Division 1 or Article CC-2000 of Section III, Division 2 of the ASME Code.

ITAAC: The ITAAC associated with Tier 2 Section 6.2.7 are given in Tier 1 Section 2.11 in Table 2.11.1-2, “Containment Vessel Inspections, Tests, Analyses, and Acceptance Criteria,” which requires that a structural integrity test be performed in accordance with Section III of ASME Code requirements.

TS: The TSs associated with Tier 2, Section 6.2.7, are given in Tier 2 Chapter 16, Section 3.6.1 and B3.6.1. These TS relate to the operability and leak rate testing of the containment, containment air locks, and containment isolation valves.

Topical Reports: There are no topical reports for this area of review.

Technical Reports: There are no technical reports for this area of review.

US-APWR Interface Issues identified in the DCD: There are no US-APWR interface issues for this area of review.

Site Interface Requirements identified in the DCD: There are no site interface requirements for this area of review.

Cross-cutting Requirements TMI, USI/GSI, Op Ex): None for this area of review.

RTNSS: There is no RTNSS for this area of review.

10 CFR 20.1406: There are no 10 CFR 20.1406 requirements for this area of review.

CDI: There is no CDI for this area of review.

6.2.7.3 Regulatory Basis

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

1. GDC 1, “Quality standards and records,” found in Appendix A to Part 50, 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, “Fracture prevention of containment pressure boundary,” 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 non-brittle manner and (2) the probability of rapidly propagating fracture is minimized.

4. 10 CFR 52.47(b)(1), which requires that a DC application include the proposed ITAAC that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the DC is built and will operate in accordance with

the DC, the provisions of the Atomic Energy Act of 1954,, and the NRC's regulations.

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.

3. To meet the requirements of GDC 1, 16 and 51, ferritic containment pressure boundary materials should meet the fracture toughness criteria and requirements for testing identified in Article NE 2300 of Section III, Division 1 or Article CC 2520 of Section III, Division 2 of the ASME Code or, for materials that were not fracture toughness tested as discussed below, the fracture toughness criteria for Class 2 components identified in the Summer 1977 Addenda to Section III, Division 1, Subsection NC of the ASME Code.

4. Mandatory fracture toughness testing of ASME Code Section III, Class 2 materials was first identified in the Summer 1977 Addenda Code Class 2 rules. As a result, cases exist where Class 2 ferritic materials of the reactor containment pressure boundary were not fracture toughness tested, because the ASME Code Edition and Addenda in effect at the time the components were ordered, did not require that they be tested. The staff's assessment of the fracture toughness of materials that were not fracture toughness tested is based on the metallurgical characterization of these materials and fracture toughness data presented in NUREG 0577, "Potential for Low Fracture Toughness and Lamellar Tearing on PWR Steam Generator and Reactor Coolant Pump Supports," and ASME Code Section III, Summer 1977 Addenda, Subsection NC. The metallurgical characterization of these materials, with respect to their fracture toughness, is developed from a review of how these materials were fabricated and what thermal history they experienced during fabrication. The metallurgical characterization of these materials, when correlated with the data presented in NUREG 0577 and the Summer 1977 Addenda of ASME Code Section III, provides the technical basis for the staff's evaluation of the compliance with Code Class 2 requirements of the materials, which were not fracture toughness tested.

6.2.7.4 Technical Evaluation

The staff reviewed the US-APWR measures involving fracture prevention of ferritic materials used in the containment pressure boundary in accordance with the guidelines of SRP 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 US-APWR containment pressure boundary utilizes A516, Grade 60 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 paragraph CC-2520 of the ASME BPV Code, Section III, Division 2. 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 finds the fracture-prevention measures used in the US-APWR containment design to be acceptable.

Although the US-APWR DCD Section 6.2.7 does not contain any COL information items, Section 3.8.1, "Concrete Containment," under "Liner Plate" stated, "It is the responsibility of the COL applicant to produce a site-specific specification to define the material and welding requirements, testing and quality requirements." There was also a corresponding COL information item in Section 3.8.6 "Combined License Information". In **RAI 347-2650, Question 06.02.07-1**, the staff requested that the applicant address why these liner plate specifications should be site-specific and discuss why this information should not be standardized for all COLs referencing the US-APWR design. In its response to **RAI 347-2650, Question 06.02.07-1**, dated November 7, 2008, the applicant stated that Section 3.8.1 and Section 3.8.6 would be revised to delete the requirement that the COL applicant produce a site-specific specification. The staff finds this response acceptable. The staff confirmed that Revision 2 of the US-APWR DCD, dated October 2009, contains the changes committed to in the RAI response. On this basis, the staff finds the applicant response to **RAI 347-2650, Question 06.02.07-1, acceptable, and thus, RAI 347-2650, Question 06.02.07-1, is resolved and closed.**

6.2.7.5 Combined License Information Items

There are no COL information items for this area of review identified in DCD Section 6.2.7 or Table 1.8-2. No additional COL information items need to be included for this area of review.

6.2.7.6 Conclusions

Based on the review of the information included in the US-APWR DCD, the staff finds 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.

The staff, therefore, 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, 16, and 51 will be met.

6.3 Emergency Core Cooling Systems

6.3.1 Introduction

The primary function of the ECCS is to remove stored or residual heat and fission product decay heat from the reactor core following an accident. It consists of the SIS, which includes the advanced accumulator system, the high head injection system, and the emergency letdown system. The SIS, in conjunction with the insertion of the control rods is designed to shutdown and cool the reactor during the following events: LOCAs, ejection of a control rod cluster assembly, secondary steam system piping failure, inadvertent opening of main steam relief or safety valve, and SG tube rupture. The emergency letdown system provides redundancy to the normal CVCS in achieving cold shutdown boration conditions. The ECCS also provides the capability of maintaining the desired post-accident pH conditions in the RWSP located inside containment.

6.3.2 Summary of Application

DCD Tier 1: The DCD Tier 1 information associated with this section is found in DCD Tier 1 Section 2.4.4, "Emergency Core Cooling System." US-APWR DCD GSI-191 Tracking Report (July 2011 Version), dated July 29, 2011 (hereafter referred to as GTR1) and US-APWR DCD GSI-191 Tracking Report (August 2012 Version), dated August 30, 2012 (hereafter referred to as GTR2).

The Tracking Reports communicate essential information addressing GSI-191 issues that the applicant committed to provide as part of the GSI-191 closure activities. The Tracking Reports contain information that is reflected in DCD Revision 3, which the applicant plans to incorporate into a future DCD revision.

In this SE section, when the DCD information is listed and is supplemented by a Tracking Report, the Tracking Report is identified and the Tracking Report's associated supplemental information is treated as a confirmatory action item.

DCD Tier 2: The applicant has provided a system description in DCD Tier 2 Section 6.3, "Emergency Core Cooling System," summarized here in part, as follows:

> The advanced accumulator system consists of four accumulators and the associated valves and piping, one for each loop. The system injects borated water into the cold legs of the reactor coolant piping. The accumulators incorporate internal passive flow dampers which function to inject a large flow to refill the reactor vessel in the first stage of the injection, and then reduce the flow as the water level in the accumulator drops.
>
> When the water level is above the top of the standpipe, water enters the flow damper through both inlets at the top of the standpipe and at the side of the flow damper, and injects water with a large flow rate. When the water level drops below the top of the standpipe, the water enters the flow damper only through the side inlet, and injects water with a relatively low flow rate.
>
> The high-head injection system consists of four independent trains, each containing a SI pump and the associated valves and piping. The SI pumps start automatically upon receipt of the SI signal. The SI pumps are aligned to take suction from the RWSP and

deliver borated water through the DVI nozzles into the reactor vessel. The operator switches over from the DVI injection line to the hot leg injection line (the simultaneous direct vessel and hot leg injection mode) to prevent boric acid precipitation in the reactor core during the long-term cooling phase of a LBLOCA.

The emergency letdown system consists of two emergency letdown lines from the RCS hot legs to the RWSP. In the event that the normal CVCS letdown and boration capability is not available, the feed-and-bleed emergency letdown and boration operations can be utilized to achieve a cold-shutdown boration level in the reactor coolant prior to the safe shutdown operation. Details of the safe shutdown design bases are discussed in DCD Tier 2 Section 5.4.7, "Residual Heat Removal System."

Containment pH control is provided by twenty three baskets containing NaTB as a buffer agent, located inside three containers at an elevation that is below the lowest containment spray ring. The solution containing NaTB is then discharged through a solution transfer pipe to the RWSP.

DCD Tier 2, Table 1.6-1, "Material Referenced," as supplemented by GTR1, incorporates-by-reference Technical Report MUAP-08001-P, "US-APWR Sump Strainer Performance;" Technical Report MUAP-08011-P, "US-APWR Sump Debris Chemical Effects Test Result;" and Technical Report MUAP-08013-P, "US-APWR Sump Strainer Downstream Effects." The staff opened **confirmatory item CI-SRP06.03-2** to verify that the final design document incorporates the associated supplemental DCD information provided by GTR1.

ITAAC: The ITAAC associated with DCD Tier 1 Section 2.4.4 are given in DCD Tier 1 Section 2.4.4.2, "Inspections, Tests, Analyses, and Acceptance Criteria."

TS: The TS associated with DCD Tier 2 Section 6.3 are given in DCD Tier 2, Chapter 16, "Technical Specifications," Section 3.5, "Emergency Core Cooling Systems (ECCS)."

Topical Reports: MHI Topical Report MUAP-07001-P, "The Advanced Accumulator"

Technical Reports:

- MHI Technical Report MUAP-08001-P, "US-APWR Sump Strainer Performance."
- MHI Technical Report MUAP-08006-P, "US-APWR Sump Debris Chemical Effects Test Plan," Revision 0.
- MHI Technical Report MUAP-08011-P, "US-APWR Sump Debris Chemical Effects Test Result."
- MHI Technical Report MUAP-08013-P, "US-APWR Sump Strainer Downstream Effects."
- MHI Technical Report MUAP-10021-P, "US-APWR Core Inlet Blockage Test," Revision 0.
- MHI Technical Report MUAP-11022-P, "US-APWR Additional Core Inlet Blockage Test," Revision 0.
- MHI Technical Report MUAP-12004-P, "US-APWR Core Inlet Blockage Test for Test Conditions with Design Changes in Recirculation Water Flow Path to Refueling Water Storage Pit," Revision 0.

US-APWR Interface Issues identified in the DCD: There are no US-APWR interface issues for this area of review.

Site Interface Requirements identified in the DCD: There are no site interface requirements for this area of review.

Cross-cutting Requirements (TMI, USI/GSI, Op Ex): GSI 191, "Assessment of Debris Accumulation on PWR Sump Performance," closure activities (ML120820125).

Compliance of the ECCS design with relevant items of the TMI Action Plan, specified in 10 CFR 50.34(f), is described in DCD Tier 2 Section 6.3, Table 6.3-1, "Response of US-APWR to TMI Action Plan."

Compliance with the resolution of Unresolved Safety Issues and relevant medium- and high-priority GSIs that are specified in NUREG-0933 is described in DCD Tier 2 Section 6.3, Table 6.3-2, "Response of US-APWR to Unresolved Safety Issues," and Table 6.3-3, "Response of US-APWR to Generic Safety Issues."

Operating experience insights from Generic Letters (GLs) and Bulletins incorporated in the design of the ECCS are presented in DCD Tier 2 Section 6.3, Table 6.3-4, "Response of US-APWR to Generic Letters and Bulletins."

The cross-cutting requirements for this area of review are discussed in detail in Section 6.3.4.6 of this SER below.

RTNSS: There is no RTNSS for this area of review.

10 CFR 20.1406: There are no 10 CFR 20.1406 requirements for this area of review.

CDI: There is no CDI for this area of review.

6.3.3 Regulatory Basis

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

1. GDC 2, "Design bases for protection against natural phenomena," as it relates to the seismic design of SSCs whose failure could cause an unacceptable reduction in the capability of the ECCS to perform its safety function.

2. GDC 4, "Environmental and dynamic effects design bases," as it relates to dynamic effects associated with flow instabilities and loads (e.g., water hammer).

3. GDC 5, "Sharing of structures, systems and components," as it relates to the prohibition against SSCs important to safety being shared among nuclear power units unless it can be demonstrated that sharing will not impair their ability to perform their safety function.

4. GDC 17, "Electrical power systems," as it relates to the design of the ECCS having sufficient capacity and capability to assure that specified acceptable fuel design limits

and the design conditions of the reactor coolant pressure boundary are not exceeded during AOOs and that the core is cooled during accident conditions.

5. GDC 27, “Combined reactivity control systems capability,” as it relates to the system design having the capability to assure that under postulated accident conditions and with appropriate margin for stuck rods, the capability to cool the core is maintained.

6. GDCs 35, “Emergency core cooling,” 36, “Inspection of Emergency Core Cooling System,” and 37, “Testing of Emergency Core Cooling System,” as they relate 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, to permit appropriate periodic inspection of important components, and to permit appropriate periodic pressure and functional testing.

7. 10 CFR 50.46, in regard to the ECCS being designed so that its cooling performance is in accordance with acceptable evaluation models.

Acceptance criteria adequate to meet the above requirements include:

1. With regard to the ECCS acceptance criteria of 10 CFR 50.46, the five major performance criteria deal with:

    a. Peak cladding temperature.
    b. Maximum calculated cladding oxidation.
    c. Maximum hydrogen generation.
    d. Coolable core geometry.
    e. Long-term cooling.

2. Guidance, procedures and methods which are acceptable for meeting the requirements for a realistic or best-estimate evaluation model for ECCS performance can be found in RG 1.157. This method must identify and account for uncertainties in the analysis method and inputs such that there is a high level of probability that the acceptance criteria are not exceeded.

3. The ECCS must meet the requirements of GDC 35. The system must have alternate sources of electric power, as required by GDC 17, and must be able to withstand a single failure.

4. The combined reactivity control system capability associated with ECCS must meet the requirements of GDC 27 and should conform to the recommendation of RG 1.47, “Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems.”

5. The design of the ECCS should conform to the recommendations of RG 1.1, “Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal System Pumps (Safety Guide 1).”

6. American Society for Testing and Materials (ASTM) Standard D 3911, “Test Method for Evaluating Coatings Used in Light-Water Nuclear Power Plants at Simulated Design Basis Accident (DBA) Conditions,” identifies test parameters,

conditions, and procedures related to high temperatures and highly acidic and caustic environments.

6.3.4 Technical Evaluation

6.3.4.1 Advanced Accumulators and Support Systems

The ITAAC requirements for the accumulators are described in DCD Tier 1, 2.4.4, “Emergency Core Cooling System,” Table 2.4.4-5, “Emergency Core Cooling System Inspections, Tests, Analyses, and Acceptance Criteria.” The following design commitment acceptance criteria are identified:

(a) Acceptance Criterion 7.b.i.a: (1) The total water volume injected from each as-built accumulator into the reactor vessel is ≥ 2,126 ft^3 (60 m^3), (2) The water volume injected from each accumulator into reactor vessel at large flow rate (prior to flow switching to small flow rate) is ≥ 1,326.8 ft^3 (37.5 m^3).

(b) Acceptance Criterion 7.b.i.b: The calculated resistance coefficients of the as-built accumulator system (based on a cross-section area of 0.6827 ft^2, 0.06 m^2) meet the requirements shown in Table 2.4.4-6, “Requirement for Accumulator System Resistance Coefficient.”

(c) Acceptance Criterion 7.b.iii a: The volume of each as built accumulator is at least 3,180 ft^3 (90 m^3).

(d) Acceptance Criterion 10.c: The as-built accumulator discharge valves identified in Table 2.4.4-2, “Emergency Core Cooling System Equipment Characteristics,” automatically opens upon either the receipt of simulated ECCS actuation or above low pressurizer pressure signal.

The TS requirements for the accumulators are provided in TS 3.5, “Emergency Core Cooling Systems (ECCS),” 3.5.1, “Accumulators.”

6.3.4.1.1 System Description

There are four independent and dedicated SIS trains. The four advanced accumulators are vertically mounted cylindrical tanks located outside each SG/RCP cubicle, inside containment. They are non-insulated carbon steel vessels with stainless steel cladding, designed to Equipment Class 2 and Seismic Category I standards. The safety related piping and components are designed to Equipment Class 2, Seismic Category I standards. The accumulators are filled with boric acid (borated) water and pressurized with nitrogen gas. The accumulators are passive devices and inject borated water into each associated cold leg. The accumulators incorporate internal passive flow dampers, which function to inject a large flow rate to refill the reactor vessel in the first stage of injection, and then reduce the flow rate as the accumulator water level drops. DCD Tier 2 Figure 6.3-6, “Overview of the Accumulator,” presents a simplified view of the dual flow rate accumulator design. DCD Tier 2 Figure 6.3-2, “ECCS Piping and Instrumentation Diagram (Sheet 3 of 4),” provides an overview of the accumulator piping and valve layout.

The accumulator design pressure is 4.93 MPa (700 psig), and the design temperature is 148.89 ºC (300 ºF). The nominal operating conditions are expected to be 4.51 MPa (640 psig) with a temperature range of 21.11 ºC to 48.89 ºC (70 ºF to 120 ºF). The two check valves in each line which isolate the accumulator from the RCS pressure boundary are designed to Equipment Class 1 standards. The total volume is 90.05 m^3 (3,180 ft^3) each, with a borated water volume of 60.20 m^3 (2,126 ft^3) each (excluding the dead volume below the vortex chamber). The boron concentration is about 4,000 mg/Kg (4,000 ppm). Each accumulator has redundant pressure and level instrumentation with alarms. No temperature monitoring is provided.

In **RAI 391-2974, Question 06.03-43**, the staff requested that the applicant summarize the equipment class for the ECCS components and piping inside containment to ensure that they meet, at a minimum, Equipment Class 2 and Seismic Category I. The staff also requested that the applicant identify the seismic design category associated with Equipment Class 3 and Class 4 components and piping. In its response to **RAI 391-2974, Question 06.03-43 (RAI 06.03.02.03-1)**, dated July 28, 2009, the applicant provided the reference to the seismic qualifications for the Equipment Class 3 and 4 components. The seismic qualifications are provided in DCD Tier 2 Table 3.2-2, "Classification of Mechanical and Fluid Systems, Components, and Equipment." In **RAI 391-2974, Question 06.03-44**, the staff requested that the applicant describe the material specification characteristics for ECCS valves, both for the seating surfaces and stems, to prevent failures and to reduce wear. In its response to **RAI 391-2974, Question 06.03-44 (RAI 06.03.02.04-1)**, dated July 28, 2009, the applicant described special considerations for the selections of valves. The surfaces of the SIS valve seating are hard-faced to prevent failure and to reduce wear. In addition, the applicant states that the valve stem materials are selected taking into consideration corrosion resistance, high-tensile properties and resistance to surface scoring by packing. The staff confirmed that Revision 3 to DCD Tier 2, Section 6.3.2.4, "Material Specifications and Compatibility," includes the proposed changes. As noted in the DCD, the complete material specifications are presented in DCD Tier 2 Section 6.1, "Engineered Safety Feature Materials." System and component purchasing and procurement activities are performed within the guidelines provided by DCD Tier 2 Chapter 17, "Quality Assurance." The responses to **RAI 391-2974, Questions 06.03-43 and 06.03-44,** provided the information requested and the applicant provided a DCD revision clarifying the material specifications of the ECCS valves.

Since the applicant provided the DCD Tier 2 reference to the seismic and equipment class design for the safety related SSCs and the applicant revised DCD Tier 2 to provide the reference to the material specifications for the ECCS valves, the staff finds the applicant's responses to **RAI 391-2974, Questions 06.03-43 and 06.03-44, acceptable**. Therefore, **RAI 391-2974, Questions 06.03-43 and 06.03-44, are resolved and closed.**

The operational restrictions and ECCS performance criteria are controlled by DCD Tier 2 TS 3.5, "Emergency Core Cooling Systems," and 3.5.1, "Accumulators." In **RAI 391-2974, Question 06.03-54**, the staff requested that the applicant address discrepancies between the information provided in Table 15.6.5-1,"US-APWR Major Plant Parameter Inputs Used in the Best-Estimate Large break LOCA Analysis," and TS 3.5.1, "Accumulators." In its response to **RAI 391-2974 Question 06.03-54 (RAI 06.03.03.01-1)**, dated July 28, 2009, the applicant addressed the limiting conditions for operation (LCOs) for the accumulator. Consistent with NUREG-1431, "Standard Technical Specifications — Westinghouse Plant," Revision 3.1, there is no LCO for the accumulator temperature allowable range. The upper value is consistent with the upper allowable value for an operable containment. The accumulator pressure range used in the LBLOCA analysis (600 psia ≤ P_{ACC} ≤ 710 psia), (4.1 MPa ≤ P_{ACC} ≤ 4.9 MPa), is the range whose values were converted into absolute pressure, and then rounded for the TS SR 3.5.1.3 range (586 psig ≤ P_{ACC} ≤ 695 psig) (4 MPa ≤ P_{ACC} ≤ 4.8 MPa). The accumulator pressure range used in the LBLOCA analysis corresponds to the TS SR 3.5.1.3 range. The accumulator water volume was addressed in UAP-HF-09031, "MHI's Responses to US-APWR DCD RAI No. 135-1818 Revision 0", dated February 4, 2009 (ADAMS Accession Number ML090370444) in response to **RAI 135-1818, Question 16-49**. The safety analysis assumes values of 19,338 gallons and 19,734 gallons (2,127 ft^3 to 2,179 ft^3), therefore the TS values were revised to reflect analysis assumptions.

In its response to **RAI 391-2974, Question 06.03-54**, dated July 28, 2009, the applicant provided the LCOs for the accumulators in TS 3.5.1. In follow-up **RAI 695-4934, Question 06.03-89**, the staff requested that the applicant describe how the accumulator volume TS value (specified in gallons) is confirmed since only the accumulator pressure and level are known, but not the temperature, and also to describe how the RWSP volume TS value (specified in gallons) is confirmed since only the RWSP pressure and level are known, but not the temperature. In its response to **RAI 695-4934, Question 06.03-89**, dated March 17, 2011, the applicant described the relationship between the level and the volume. The applicant stated the accumulator volume TS value will be confirmed using the accumulator water level instruments. The relationship between the actual volume in the accumulator and the indication of the accumulator water level instruments will be prepared during the detailed design phase before fuel load. Therefore, the operators can confirm the actual water volume in the tank by using the accumulator water level instruments. The operational temperature range for the accumulator is the same as for the containment because the accumulator is not insulated and contains no internal heaters. Therefore the change in the fluid density from 45 °F (7.2 °C) to 120 °F (49 °C) is about one percent. The accumulator safety analysis values are covered by this change. The applicant also stated that the RWSP volume TS value will be confirmed using the RWSP water level instruments. The relationship between the actual volume in the RWSP and the indication of the RWSP water level instruments will be prepared during the detailed design phase before fuel load. Therefore, the operators can confirm the actual water volume in the RWSP by using the RWSP water level instruments. The operational temperature range for the RWSP is the same as for the containment because the RWSP is open to the containment and contains no internal heaters. Therefore the change in the fluid density from 45 °F (7.2 °C) to 120 °F (49 °C) is about one percent.

The staff reviewed the information provided in the March 17, 2011, response to **RAI 695-4934, Question 06.03-89,** and found it acceptable as the applicant has committed to develop a relationship between accumulator and RWSP level and water volume before fuel load to account for the very small changes in water density between 7.2 °C (45 °F) and 49 °C (120 °F). Therefore, **RAI 695-4934, Question 06.03-89, is resolved and closed.**

In **RAI 391-2974, Question 06.03-57**, the staff requested that the applicant identify, by number, those tests in DCD Section 14.2, "Initial Plant Test Program," that specifically address ECCS performance. In its response to **RAI 391-2974, Question 06.03-57 (RAI 06.03.04.01-1)**, dated July 28, 2009, the applicant provided the list of the ECCS test procedures in the Initial Plant Test Program. DCD Tier 2, Section 6.4.2.1, was revised to list the ECCS test procedures:

- Safety Injection System (SIS) Preoperational Test.
- ECCS Actuation and Containment Isolation Logic Preoperational Test.
- Safety Injection Check Valve Preoperational Test.
- Safety Injection Accumulator Test.

The staff finds that the response to **RAI 391-2974, Question 06.03-57,** is acceptable because it properly identified the tests that specifically address ECCS performance and the applicant committed to include that information in the DCD. The staff reviewed Revision 3 of DCD Tier 2, Section 6.3.4.1, “ECCS Performance Tests,” and confirmed that the proposed changes in the July 28, 2009, response to **RAI 391-2974, Question 06.03-57,** were made to the DCD. Therefore, **RAI 391-2974, Question 06.03-57, is resolved and closed**.

In **RAI 407-3082, Question 06.03-76**, the staff requested that the applicant explain the statement in test 14.2.12.1.57, "Safety Injection Accumulator Test," D. Acceptance Criteria Item 1, that "The pressure is controlled properly as designed." The staff also requested that the applicant explain how this statement is verified by the test, and clarify if this statement refers to the nitrogen charging and overpressure control. In its response to **RAI 407-3082, Question 06.03-76 (RAI 06.03.04.01-6)**, dated August 5, 2009, the applicant clarified the control pressure. The test description, Acceptance Criteria item D.1, was revised to “The nitrogen pressure of each accumulator is controlled by the accumulator nitrogen supply pressure control valve within design limits.” The staff finds that the response is acceptable because it described how the pressure is controlled within the design limits and the applicant included that clarification in Subsection 14.2.12.1.57 of the DCD. The staff confirmed that Revision 3 of DCD Tier 2, Section 14.2.12.1.57 contains the proposed changes in response to **RAI 407-3082, Revision 1, Question 06.03-76**. Therefore, **RAI 407-3082, Question 06.03-76, is resolved and closed**.

In **RAI 407-3082, Question 06.03-77**, the staff requested that the applicant clarify the reference to DCD Tier 2, Sections 6.3.1.1, “Safety Injection,” and 6.3.2.2, “Equipment and Component Descriptions,” in Test 14.2.12.1.57, D. Acceptance Criteria Item 2. In its response to **RAI 407-3082, Question 06.03-77 (RAI 06.03.04.01-7)**, dated August 5, 2009, the applicant revised Acceptance Criteria Item D.2 of the test description to point to DCD Tier 2, Section 6.3.2.2.2, for the accumulator discharge performance. The staff finds that the response is acceptable because it corrects the DCD to reference the correct DCD subsections. The staff confirmed that Revision 3 of DCD Tier 2, Section 14.2.12.1.57, contains the proposed changes in response to **RAI 407-3082, Revision 1, Question 06.03-77**. Therefore**, RAI 407-3082, Question 06.03-77, is resolved and closed**.

In **RAI 695-4934, Question 06.03-97**, the staff requested that the applicant explain for test 14.2.12.1.57, how a “partially pressurized” accumulator satisfies the test Acceptance Criteria to demonstrate the discharge performance in the design specifications since the RCS is open and at atmospheric pressure. In its response to **RAI 695-4934, Question 06.03-97**, dated March 17, 2011, the applicant described how the test will demonstrate the discharge performance.

The purpose of the accumulator injection test is to confirm the accumulator flow characteristics are appropriate for the safety analysis. The safety analysis uses the accumulator flow damper characteristic equation (the relationship between cavitation factor and flow coefficient). During the injection test, the cavitation factor and flow coefficient are calculated using various measured parameters, and are verified to be within the expected limits.

The flow coefficients were obtained from the full-height, 1/2 scale test, which was performed under the conditions simulating pre-operational testing. These conditions were verified to cover the flow characteristics during an accident in Topical Report MUAP-07001-P, “The Advance Accumulator.” Table 4.2.4-1, “Test Conditions of Full Height 1/2 Scale Test,” contains the flow characteristics for the assumed pre-operational test condition (Case 7). It was also confirmed that the flow characteristics of the flow damper during both large and small flow were consistent with those in the other tests cases. Therefore acceptable test data is available to verify the discharge performance with a partially pressurized accumulator.

The staff reviewed the information provided in the March 17, 2011, response to **RAI 695-4934, Question 06.03-97,** and found it acceptable because the ITAAC test conditions are consistent with the pre-operational test conditions as documented by Case 7 in the advance accumulator topical report. Therefore, **RAI 695-4934, Question 06.03-97, is resolved and closed.**

The staff noted that Test 14.2.12.1.57 contained no tests of the instrumentation used to monitor the accumulator temperature, level (volume), pressure, or alarms. In **RAI 407-3082, Question 06.03-79**, the staff requested that the applicant explain how this instrumentation would be tested. In its response to RAI **407-3082, Question 06.03-79 (RAI 06.03.04.01-9)**, dated August 5, 2009, the applicant revised the test procedure to address the accumulator instrumentation. Test Method Item C.4 was added: "Accumulator instrumentation for level (volume) and pressure, and associated alarms, are verified," and Acceptance Criterion D.4 was added: "Accumulator instrumentation for level (volume) and pressure, and associated alarms perform as described in Section 6.3.5.2." The response appropriately updated the testing method and acceptance criteria for ITAAC Test 14.2.12.1.57. Therefore, the staff finds the applicant's response to **RAI 407-3082, Question 06.03-79 acceptable**. The staff confirmed that Revision 3 of DCD Tier 2, Section 14.2.12.1.57, contains the proposed changes in response to **RAI 407-3082, Question 06.03-79**. Therefore, **RAI 407-3082, Question 06.03-79 is resolved and closed**.

The accumulators are located in the Containment. This structure is seismic category I and provides tornado/hurricane missile barriers to protect the ECCS. There are four accumulators, one accumulator in RCS each loop. The accumulator sizing is based on three accumulators to account for loss of coolant from the accumulator located in the broken loop during a postulated LOCA. DCD Tier 2 Section 6.2.1 discusses the containment environmental conditions during accidents, and DCD Tier 2 Chapter 3, Section 3.11, "Environmental Qualification of Mechanical and Electrical Equipment," discusses the suitability of equipment for design environmental conditions. All valves required to be actuated during ECCS operation are located to prevent vulnerability to flooding. Protection of the ECCS from missiles is discussed in DCD Tier 2 Chapter 3, Section 3.5. Protection of the ECCS against dynamic effects associated with the rupture of piping is described in Section 3.6, "Protection Against Dynamic Effects Associated with Postulated Rupture of Piping." Protection from flooding is discussed in Section 3.4, "Water Level (Flood) Design."

The treatment of single failures is discussed in DCD Tier 2 Section 6.3.3.3, "Single Failure Considerations."

The staff noted that the DCD states: "The ECCS is designed with redundancy so that the specified safety functions are performed assuming a single failure of an active component for a short term following an accident, and assuming either a single failure of an active component or a single failure of a passive component for a long-term following an accident." However in DCD Tier 2 Section 6.3.2.5, "System Reliability," it is stated "During long term cooling, the most limiting active failure, or a single passive failure, equal to the leakage that would occur from a valve or pump seal failure, may occur." In **RAI 391-2974, Question 06.03-53**, the staff requested that the applicant clarify the intent of the description in Section 6.3.2.5. In its response to **RAI 391-2974, Question 06.03-53 (RAI 06.03.03.03-1)**, dated August 5, 2009, the applicant clarified its treatment of single failures. The applicant stated the description in DCD Tier 2 Section 6.3.2.5 means the two failure considerations are either (1) a limiting active failure with up to the loss of one train of ECCS function to inject water to the core, or (2) a limiting passive failure with ECCS leakage resulting in the loss of one train of the ECCS function to inject water to the core (The total loss of ECCS fluid is prevented by isolating the leaked train from RWSP. The isolated ECCS train loses the function to inject water to the core). The response to **RAI 391-2974, Question 06.03-53,** provided the descriptions of a limiting active failure and a limiting passive failure and the staff agrees with these descriptions and is acceptable. Therefore, **RAI 391-2974, Question 06.03-53, is resolved and closed.**

In **RAI 407-3082, Question 06.03-82**, the staff requested that the applicant identify whether check valves are considered to be passive components for the failure modes and effects evaluation. The staff's concern was that if the check valves were considered to be passive components, then a passive failure could result in total loss of flow. In its response to **RAI 407-3082 Question 06.03-82 (RAI 06.03.03.03-2)**, dated August 5, 2009, the applicant clarified its treatment of check valve failures. Check valves are considered to be passive components. If ECCS water leaks in the reactor building due to the passive failure, the leaking train is isolated by the operator closing the safety injection pump suction isolation valve from the RWSP. The isolated train loses the function to deliver water to the core; however, the delivery of water to the core is maintained by the remaining ECCS trains. The response to RAI 407-3082, Question 06.03-82 stated that check valves are considered to be a passive component. The staff agrees that check valves are passive components and a failure of a check valve would necessitate closure of the safety injection isolation valve resulting in loss of flow from that ECCS train but the remaining trains would provide the needed ECCS flow. The response to **RAI 407-3082, Question 06.03-82, is acceptable**. Therefore, **RAI 407-3082, Question 06.03-82, is resolved and closed.**

The size and injection capacity requirements for the accumulators are described in DCD Tier 2 Section 6.3.2.2.2, "Accumulators." The staff determined that the basis for the capacity requirements needed further discussion to identify the basis for small flow rate capacity and the LOCA analysis used to size the accumulator, and if the assumed volume losses were used in the LOCA analysis. In its response to **RAI 391-2974, Question 06.03-19 (RAI 06.03.02.02-3)**, dated July 27, 2009, the applicant provided a proposed revision to DCD Tier 2, Section 6.3.2.2.2, to point to a new DCD reference (DCD Reference 6.3-3, "The Advanced Accumulator, MUAP-07001-P, Revision 2, September 2008) to support and provide the accumulator small flow rate capacity. The staff verified the change was incorporated into DCD Tier 2, Revision 3. The response to **RAI 391-2974, Question 06.03-19, is therefore acceptable**. Thus, **RAI 391-2974, Question 06.03-19, is resolved and closed.**

In its response to **RAI 391-2974, Question 06.03-33 (RAI 06.03.02.02-7)**, dated July 27, 2009, the applicant clarified the losses described in DCD Tier 2, Section 6.3.2.2.2. The assumption of only three available accumulators, with a one third of the water volume lost to the break, was only used to size the accumulator and is not necessarily the values used in the LOCA analysis. The response to **RAI 391-2974, Question 06.03-33** is acceptable because the applicant clarified the losses which are only used to conservatively size the accumulators. Therefore, **RAI 391-2974, Question 06.03-33, is resolved and closed.**

DCD Tier 2, Section 6.3.2.2.2, states that the required capacity of each accumulator at the large injection flow rate is approximately 1,307 ft^3, which is increased to approximately 1,342 ft^3, to provide margin. However, DCD Table 6.3-5, "Safety Injection System Design Parameters," states that the required capacity is greater than or equal to 1,326.8 ft^3 (37.5 m^3).

This value is also stated in DCD Tier 1, Table 2.4.4-5, “Emergency Core Cooling System Inspections, Tests, Analyses, and Acceptance Criteria,” Design Commitment 7b, Acceptance Criterion 7.b.i. In **RAI 407-3082, Question 06.03-60**, the staff requested that the applicant modify the text in DCD Tier 2, Section 6.3.2.2.2, to be consistent with acceptance criteria and include the evaluation used to develop the required capacity for the large injection flow rate or provide a reference. The staff also requested that the applicant revise the discussion concerning the downcomer and lower plenum volume values to be consistent with the acceptance criterion value as appropriate. In its response to **RAI 407-3082, Question 06.03-60 (RAI 06.03.02.02-8)**, dated August 5, 2009, the applicant clarified the value for the required accumulator capacity to be greater than or equal to 1,326.8 ft^3 (37.5 m^3), with 1,307 ft^3 (37 m^3) of water required to completely fill the reactor downcomer and lower plenum region during a LBLOCA. The large flow injection design water volume is 1,342 ft^3 (38 m^3), which equals the reactor downcomer volume of 1,307 ft^3 (37 m^3) plus a design margin. In the safety analysis, the uncertainty in the flow switching water level is considered and the duration of the large flow rate phase is shortened. The staff’s review of the switchover uncertainty is documented in the SE associated with MUAP-07001. This extends the time to fill the lower plenum during the refill period and to raise the water level in the downcomer, which leads to a longer period of core heatup and a conservative peak cladding temperature (PCT) prediction. The minimum value of 1,326.8 ft^3 (37.5 m^3) is used in the safety analysis to provide this conservatism. Because the applicant uses the minimum accumulator inventory in the safety analysis to conservatively predict the PCT, the staff finds the August 5, 2009 response to **RAI 407-3082, Question 06.03-60, acceptable**. Therefore, **RAI 407-3082, Question 06.03-60, is resolved and closed.**

The staff understands that the safety analysis value uses 1,326.8 ft^3 (37.5 m^3) while the design value is 1,342 ft^3 (38 m^3), but does not agree the DCD Tier 1, Table 2.4.4-5, “Emergency Core Cooling System Inspections, Tests, Analyses, and Acceptance Criteria,” Design Commitment 7b, Acceptance Criterion 7.b.i.a. provides sufficient detail to ensure the large flow injection volume is satisfied. The adequacy of the DCD Tier 1, Table 2.4.4-5, Items 7b and 7 b.i.a. is being tracked by RAI941- 6465, Question 14.3.4-48 and will be evaluated in the staff’s Chapter 14 SE. Therefore, **RAI 941-6465, Question 14.3.4-48 is being tracked as a Confirmatory Item.**

In **RAI 391-2974, Question 06.03-35**, the staff noted that no ITAAC existed to verify the small injection flow rate required capacity, and requested that the applicant explain why this value does not need to be verified. In its response to **RAI 391-2974, Question 06.03-35 (RAI 06.03.02.02-11)**, dated July 27, 2009, the applicant explained why the small injection flow rate does not need to be included in the ITAAC. ITAAC 2.4.4, Acceptance Criterion 7.b.i verifies the accumulator water volume and the large flow rate injection volume. In addition, this ITAAC also verifies the resistance coefficient of the as-built accumulator system, as used in the safety analyses. The staff does not agree that small flow water volume is adequately defined by ITAAC 2.4.4, Acceptance Criterion 7.b.i and hence **RAI 391-2974, Question 06.03-35 remains an Open Item**. This open item will be addressed by the staff’s Section 14.3.4, “Chapter 2 of Tier 1, Development of Specific ITAAC,” SER.

In **RAI 407-3082, Question 06.03-81**, the staff requested that the applicant explain discrepancies between the values listed in Table 15.6.5-2, "US-APWR Major Plant Parameter Inputs Used in the Appendix-K based Small Break LOCA Analysis," TS 3.5.1, "Accumulators," SR 3.5.1.2, and Table 15.6.5-1, "USAPWR Major Plant Parameter Inputs Used in the Best-Estimate Large break LOCA Analysis." In its response to **RAI 407-3082, Question 06.03-81 (RAI 06.03.03.02-1)**, dated August 5, 2009, the applicant addressed the differences in the DCD

Tier 2 Chapter 15 accumulator volumes used in the LBLOCA and SBLOCA analyses. The difference was a result of how the values were rounded off.

The applicant proposed a revision to DCD Tier 2 Table 15.6.5-2, "US-APWR Major Plant Parameter Inputs Used in the Appendix-K based Small Break LOCA Analysis," to be consistent with the value used in the LBLOCA analyses. The staff reviewed the proposed changes to the DCD and finds them acceptable. The staff confirmed that Revision 3 of DCD Tier 2, Table 15.6.5-2 contained the proposed changes. Therefore, **RAI 407-3082, Question 06.03-81, is resolved and closed**.

6.3.4.1.2 Nitrogen Gas Supply Support System

The accumulators are pressurized with nitrogen gas. An air-operated globe valve controls the pressure in the common nitrogen supply line, outside containment.

This valve fails closed. A check valve inside containment and an air-operated control valve outside containment, which closes on a containment phase “A” isolation signal, provide containment isolation for the common nitrogen supply line. The check valve has leak testing capabilities. Inside containment each accumulator line contains a normally closed motor-operated accumulator nitrogen supply line isolation globe valve.

In **RAI 391-2974, Question 06.03-40**, the staff requested that the applicant explain the need for overpressurization protection on the nitrogen supply line segment outside containment as stated in Figure 6.3-2, "ECCS Piping and Instrumentation Diagram (Sheet 3 of 4)," and clarify if that statement means there is a necessary COL action item to provide this feature in the design. In its response to **RAI 391-2974, Question 06.03-40 (RAI 06.03.02.02-16),** dated July 27, 2009, the applicant clarified Note 4 on DCD Tier 2, Figure 6.3-2, “ECCS Piping and Instrumentation Diagram.” Note 4 applies to pressure gauge PI-916. The measuring range of this pressure gauge is 0 to 700 psig, however, the gauge is installed on the nitrogen supply line outside containment, which has a design pressure of 2485 psig and hence the gage is designed to withstand 2485 psig. No COL item is required as this part of the plant design is covered by the DCD. The staff also noted that Figures 6.3-2 and 6.2.4-1, “Containment Isolation Configurations (Sheet 3 of 50),” referred to a system abbreviated as “CVDT,” but this system was not defined in the Acronyms and Abbreviations Table. In **RAI 391-2974, Question 06.03-41**, the staff requested that the applicant define CVDT and add the definition to the table. In its response to **RAI 391-2974, Question 06.03-41 (RAI 06.03.02.02-17),** dated July 27, 2009, the applicant identified the “CVDT” system shown in the figure. The applicant stated that the CVDT is a tank in the Liquid Waste Management System and “CVDT” is the abbreviation for “containment vessel reactor coolant drain tank.” The applicant explained that the CVDT is described in DCD Tier 2, Chapter 11, Section 11.2, “Liquid Waste Management System.” The applicant proposed a revision to the DCD that added CVDT to the “Acronyms and Abbreviations” section in DCD Tier 2, Chapter 6. The staff noted that the Accumulator Nitrogen Supply Containment Isolation Valve (SIS-AOV-114) did not appear to have position indication in the MCR and remote shutdown console (RSC). As a containment isolation valve, the open or closed valve position should be indicated in the MCR and RSC. In **RAI 391-2974, Question 06.03-42**, the staff asked the applicant if the valve position is indicated in the MCR and RSC, and if not, for an explanation on why this position indication is not required. In its response to **RAI 391-2974, Question 06.03-42 (RAI 06.03.02.02-18),** dated July 17, 2009, the applicant confirmed that the containment isolation valve outside containment does have position indication. DCD Tier 2, Section 6.3.2.2.6.20, was revised to include the indications and to state that “the open or closed valve position is indicated in the MCR and RSC.” The DCD revisions made as part foregoing responses have improved the clarity of the DCD; the responses are therefore acceptable. The staff confirmed that Revision 3 of DCD Tier 2, Chapter 6 contained the proposed changes. Therefore, **RAI 391-2974, Questions 06.03-40, 06.03-41, and 06.03-42, are resolved and closed.**

The nitrogen supply system header has a safety valve inside containment to protect the piping and accumulators from overpressure resulting from a failure of the nitrogen supply control valve. An air-operated vent valve in the nitrogen supply header, inside containment, may be opened by the operator to discharge nitrogen from the accumulators to containment. Two normally-closed, MOVs in the nitrogen supply header inside containment can be used by the operator to discharge nitrogen gas from the accumulators to the containment to prevent the discharge of nitrogen into the RCS during safe shutdown if the accumulator discharge valve fails to open. This operator action was not described in DCD Tier 2 Section 6.3.2.8, “Manual Actions.” The staff issued a RAI to request that the applicant include this action in DCD Tier 2 Section 6.3.2.8, “Manual Actions.” In its, response to **RAI 391-2974, Question 06.03-52 (RAI 06.03.02.08-2)**, dated July 27, 2009, the applicant revised DCD Tier 2, Section 6.3.2.8, to include the following:

> During safe shutdown, operator closes remotely the accumulator discharge valves by the operator's manual action before the RCS pressure decreases to the accumulator operating pressure in order to prevent the discharge of nitrogen from accumulators to the RCS. If the accumulator discharge valve could not be closed due to a single failure, operator opens remotely the accumulator nitrogen supply line isolation valve and the accumulator nitrogen discharge valve by the operator's manual action, and discharges the nitrogen in the accumulator to containment atmosphere and depressurizes the accumulator.

The staff reviewed the proposed change to DCD Tier 2 Section 6.3.2.8 and finds the changes acceptable because the applicant provided the operator action to prevent inadvertent discharge the accumulator nitrogen gas into the RCS minimizing possible non-condensable gas accumulation during a safe shutdown. The staff confirmed that Revision 3 of DCD Tier 2, Section 6.3.2.8, contained the proposed changes. Therefore, **RAI 391-2974, Question 06.03-52, is resolved and closed**.

6.3.4.1.3 Inventory Supply Support System

The accumulator inventory is maintained with borated water using two of the four SI pumps. An air-operated modulating globe valve in the accumulator makeup line is controlled by the operator as needed to provide borated water makeup to an accumulator from the RWSP. The valve fails closed. There is also a normally closed air-operated valve on each of the four accumulator makeup lines, also controlled by the operator, when makeup is needed to maintain the accumulator inventory. These valves fail closed. One normally closed air-operated globe valve, which has its control power locked out, is located in each of the two accumulator makeup lines which branch downstream of the containment isolation check valves in two of the four SI pump discharge lines. The valves are opened by operator action when required to provide makeup borated water to the accumulators. These valves fail closed. The staff finds the inventory support system design acceptable. Operator action is required to provide makeup. Valve failure closes the makeup lines to prevent over-pressurization of the accumulators.

6.3.4.1.4 Major Valves

A safety valve on each accumulator provides overpressure protection should the nitrogen flow control valve fail or should overcharging by an SI pump occur. In **RAI 391-2974, Question 06.03-38**, the staff asked the applicant to provide the evaluation for this safety valve sizing and evaluate the impact of potential water and nitrogen discharge. In its response to **RAI 391-2974, Question 06.03-38 (RAI 06.03.02.02-14)**, dated July 27, 2009, the applicant provided the evaluation. Each of the accumulator safety valves is required to release the maximum accumulator makeup flow rate of 100 gpm (13.4 ft^3/min) from the SI pump. The release set pressure for the safety valve is 700 psig (715 psia). Converting the release capacity to standard conditions (15 psia), shows the release capacity needs to be at least 639 ft^3/min (4,780 gpm) for the SI pump flow. The design release capacity of the safety valve is set to 1500 ft^3/min (11,220 gpm). The response shows that valve has adequate flow margin to prevent accumulator over pressure even if the accumulator completely fills with water. A flow-restraining orifice is provided in the accumulator makeup line to prevent the accumulator makeup water flow rate from exceeding 100 gpm (13.4 ft^3/min) even if makeup flow control valve fails fully open. The excessive makeup from this failure would be identified by the accumulator high-water-level alarm. The operator can terminate the excessive makeup by stopping the SI pump within a reasonable time period since it would take about 45 minutes for an accumulator to be filled completely with water from the time the high-water-level alarm is reached. Therefore, the applicant concluded it is unlikely that water would be released from the safety valve due to the failure. The staff agrees with this conclusion because the accumulator high-water-level alarm would alert the operator and there is sufficient time available to stop the SI flow. Therefore, **RAI 391-2974, Question 06.03-38, is resolved and closed**.

During normal operation the motor-operated accumulator discharge gate valve is open with its control power locked out. Two Equipment Class 1 check valves isolate the accumulator injection line from the RCS pressure boundary. These check valves have leak testing capabilities.

6.3.4.1.5 Failure Modes and Effects Analysis

The failures modes and effects analysis (FMEA) for the accumulators is presented in DCD Tier 2 Table 6.3-6, “Failure Modes and Effects Analysis - Safety Injection System.” The staff identified a number of concerns with the FMEA and could not determine if all failures have been considered, or if the instrumentation was also available at the RSC. Therefore, the staff requested additional information as described below.

In **RAI 391-2974, Question 06.03-45**, the staff requested that the applicant describe the level of detail used to develop Table 6.3-6, or provide a reference to the detailed FMEA evaluation. In its response to **RAI 391-2974, Question 06.03-45 (RAI 06.03.02.05-1),** dated July 27, 2009, the applicant stated the FMEA was developed based on the failures described in the first column of the table. Item 1 is the mechanical failure of SI pump. The ac electrical failure is described in the Item 9, and the I&C failure is described in Item 8. The failure to deliver flow due to a valve failure is addressed in the Items 2 and 3. In additional, failure of the accumulator discharge valve is described in item 4 and failure of the nitrogen discharge value is described in item 5. Failure of the nitrogen supply valve is described in Item 6. Item 7 describes the failure of the emergency letdown isolation valves. The dc electric failure is described in Item 10.

In **RAI 391-2974, Question 06.03-46**, the staff requested that the applicant clarify if the "Failure Detection Method" mentioned in Table 6.3-6 is based on information provided in the MCR and if the same information is available from the RSC. In its response to **RAI 391-2974, Question 06.03-46 (RAI 06.03.02.05-2),** dated July 27, 2009, the applicant stated the information provided in the MCR described in Table 6.3-6, “Failure Detection Method," is also applicable to the RSC.

In **RAI 391-2974, Question 06.03-47**, the staff requested that the applicant clarify Item 4. Accumulator discharge valve, in Table 6.3-6, Column "Effect on System Operation" should be clarified as "No effect on plant safety because the accumulator nitrogen gas volume can be vented to the containment atmosphere by opening the accumulator nitrogen discharge valve SIS-MOV-121A or B." In its response to **RAI 391-2974, Question 06.03-47 (RAI 06.03.02.05-3),** dated July 27, 2009, the applicant revised column titled “Effect on System Operation,” for item 4 in DCD Tier 2, Table 6.3-6 to indicate that nitrogen venting is to the containment atmosphere. The staff reviewed the proposed changes to Table 6.3-6 and finds them acceptable. The staff confirmed that the proposed changes were made to DCD Tier 2, Table 6.3-6 and therefore,**RAI 391-2974, Question 06.03-47, is resolved and closed**.

The staff notes that the long term cooling limiting failure is based on leakage from a valve or pump seal. Leakage is detected and alarmed in the MCR. **In RAI 391-2974, Question 06.03-48**, the staff requested that the applicant clarify if the same information available from the RSC. In its response to **RAI 391-2974, Question 06.03-48 (RAI 06.03.02.05-4),** dated July 27, 2009, the applicant stated that valve or pump seal leakage is detected and alarmed at the RSC as well as in the MCR. As discussed above, the responses to **RAI 391-2974, Questions 06.03-45 through 06.03-48** provided the clarifications sought by the staff and increased the staff’s understanding of the failure modes and effects analysis. They are therefore acceptable. Accordingly, **RAI 391-2974, Questions 06.03-45 through 06.03-48, are resolved and closed**.

6.3.4.2 Safety Injection Pumps and Support Systems

The ITAAC requirements for the SI pumps are described in DCD Tier 1, Section 2.4.4, “Emergency Core Cooling System,” Table 2.4.4-5, “Emergency Core Cooling System Inspections, Tests, Analyses, and Acceptance Criteria.” The following design commitment acceptance criteria are identified:

(a) 7.b.ii Acceptance Criteria: A report exists and concludes that each as-built safety injection pump has a pump differential head of no less than 3937 ft (1200 m) and no more 4527 ft (1380 m) at the minimum flow, and injects no less than 1259 gpm (4.8 m^3/min) and no more than 1462 gpm (5.5 m^3/min) of RWSP water into the reactor vessel at atmospheric pressure.

(b) 7.d Acceptance Criteria: A report exists and concludes that the as-built NPSH available to each safety injection pump is greater than the NPSH required.

(c) 10.a Acceptance Criteria: Controls in the as-built MCR start and stop the as-built SI pumps identified in Table 2.4.4-4, .“Emergency Core Cooling System Equipment Characteristics.”

The TS requirements for the SI pumps are provided in TS 3.5, “Emergency Core Cooling Systems (ECCS),” 3.5.2, “Safety Injection System (SIS) – Operating,” and 3.5.3, “Safety Injection System (SIS) – Shutdown.”

6.3.4.2.1 System Description

There are four independent and dedicated SIS trains. The safety related piping and components are designed to Equipment Class 2, Seismic Category I standards. The SI pump trains are automatically initiated by an "S" signal, and supply borated water from the RWSP to the reactor vessel through the DVI nozzles.

Each 50 percent capacity train includes a safety injection pump suction isolation valve, a dedicated 50 percent capacity SI pump, a SI pump discharge containment isolation valve, a DVI line isolation valve, and a hot leg injection isolation valve. The DVI nozzles are located at approximately the same vessel elevation as the reactor coolant hot and cold leg nozzles, but slightly below their centerline. DCD Tier 2 Figure 6.3-2, "ECCS Piping and Instrumentation Diagram (Sheet 2 of 4) and (Sheet 3 of 4)," provides an overview of the SI pump piping and valve layout.

The buildup of boric acid concentration in the reactor core during the long-term cooling phase of a LOCA may occur due to boiling and reach the precipitation concentration. Boric acid precipitation in the core could affect the core cooling. The operator switches over from the operating DVI line to the hot leg injection line (the simultaneous direct vessel and hot leg injection mode) to prevent boric acid precipitation. The evaluation of boric acid precipitation, including the time available for the operator to switch to hot leg injection, is provided in DCD Tier 2 Section 15.6.5.3.1.3, "Post-LOCA Long term Cooling Evaluation Model."

The SI pumps are stainless steel horizontal multi-stage centrifugal pumps designed to Equipment Class 2, Seismic Category I standards.

The design pressure is 14.82 MPa (2,135 psig) and the design temperature is 148.89 ºC (300 ºF). The design flow is 5829.53 liter/min (1,540 gpm) and the design head is 499.87 meters (1,640 ft). The SI pump flow requirements are shown in DCD Tier 2 Figure 6.3-4, "Safety Injection Pump Performance Flow Requirement." The three check valves in each train, two in the DVI line and one in the hot leg injection line, which isolate the pumps from the RCS pressure boundary, are designed to Equipment Class 1 standards.

The staff requested the applicant to provide additional details on the design characteristics of the pumps. In **RAI 391-2974, Question 06.03-18**, the staff requested that the applicant provide test data which demonstrates that Figure 6.3-15, "High Head Safety Injection Flow Characteristic Curve (Minimum Safeguards)," is conservative relative to actual pump performance, provide details on how the test data was generated, and describe the testing conditions and their relevance to the actual system conditions during normal operation and postulated accidents. In its response **to RAI 391-2974, Question 06.03-18 (RAI 06.03.02.01-3),** dated July 28, 2009, the applicant stated that DCD Tier 2 Figure 6.3-4 was developed based on experience and not on actual pump testing data. The actual pump has not been manufactured. The applicant stated that Table 6.3-4 will be a requirement to the pump manufacturer, and that pump testing will be performed during normal operation, or plant shutdown, to confirm the design requirement. The applicant stated that because the pump discharge pressure is affected by the fluid temperature, the fluid density at the expected accident temperature was used in developing DCD Tier 2 Figure 6.3-15, "High Head Safety Injection Flow Characteristic Curve (Minimum Safeguards)." The response is acceptable because it assures that the HHSI flow curve will be a manufacturing requirement, thus guaranteeing that the minimum safeguards flow given in Figure 6.3-15 and pump head in Figure 6.3-4 are conservative. Therefore, **RAI 391-2974, Question 06.03-18, is resolved and closed**.

In Section 6.3.2.2.1, “Safety Injection Pumps,” the second sentence states that “This SI pump flow rate is based on two SI pumps operating (active failure of one SI pump and one SI pump out of service), with each SI pump delivering 1,057 gpm against near atmospheric pressure.” The design flow given on Table 6.3-5 is 1,540 gpm (206 ft^3/min). In **RAI 391-2974, Question 06.03-20**, the staff requested that the applicant explain why the design flow of 1,540 gpm (206 ft^3/min) is greater than the run out flow of 1,057 gpm (141 ft^3/min). In its response to **RAI 391-2974, Question 06.03-20 (RAI 06.03.02.02.01-1),** dated July 27, 2009, the applicant explained why the design flow of 1,540 (206 ft^3/min) gpm is greater than the run out flow of 1,057 gpm (141 ft^3/min).

The SI pump design flow of 1,540 gpm (206 ft^3/min) is greater than the safety injection flow rate of 1,057 gpm (141 ft^3/min) because it includes a safety margin, including the minimum flow rate. The response is acceptable because it explains the difference between design and run out flows. Therefore, **RAI 391-2974, Question 06.03-20, is resolved and closed**.

In **RAI 391-2974, Question 06.03-31**, the staff requested that the applicant summarize or reference the NPSH analyses performed to ensure there is an adequate head available for all accident conditions. In its response to **RAI 391-2974, Question 06.03-31 (RAI 06.03.02.02-5)**, dated July 27, 2009, the applicant provided the requested DCD Tier 2 location of the NPSH analysis The applicant stated that the NPSH analysis is summarized in DCD Tier 2 Section 6.2, “Containment Systems,” Section 6.2.2, “Containment Heat Removal Systems,” Section 6.2.2.3, “Design Evaluation.” The staff finds the response to **RAI 391-2974, Question 06.03-31** acceptable because it provides references to the NPSH analysis. Accordingly, **RAI 391-2974, Question 06.03-31, is resolved and closed**.

The operational restrictions and ECCS performance criteria are controlled by DCD Tier 2 TS 3.5, “Emergency Core Cooling Systems,” 3.5.2 “Safety Injection System (SIS) – Operating,” and 3.5.3, “Safety Injection System (SIS) - Shutdown.”

Preoperational testing for the SIS is described in DCD Tier 2 Section 14.2.12.1.54, “Safety Injection System (SIS) Preoperational Test,” and DCD Tier 2 Section 14.2.12.1.56, “Safety Injection Check Valve Preoperational Test.” The staff asked the applicant to capture all of the design features of the SI pumps identified in the proposed test descriptions in DCD Tier 2 Sections 14.2.12.1.54 and 14.2.12.1.56.

The staff noted that for Test 14.2.12.1.54, "Safety Injection System (SIS) Preoperational Test," Objective 2 is "To verify that the head/flow characteristics of each safety injection pump is approximately the same." In **RAI 407-3082, Question 06.03-72**, the staff requested that the applicant clarify this statement and explain how far apart the characteristics can be and still be acceptable. In its response to **RAI 407-3082, Question 06.03-72 (RAI 06.03.04.01-2)**, dated August 5, 2009, the applicant modified the test objective for test 14.2.12.1.54 to be consistent with the acceptance criteria. DCD Tier 2 Section 14.2.12.1.54, item A.2 was revised to state: “To verify that the performance characteristics of each safety injection pump is *[sic]* within design specifications.” The staff reviewed the response and proposed DCD changes and found them to be acceptable because the response correctly defines the objectives of test 14.2.12.1.54. The staff confirmed that the proposed changes were made to DCD Tier 2, Section 14.2.12.1.54, Item A.2 and **therefore, RAI 407-3082, Question 06.03-72 is resolved and closed**.

The staff noted that for Test 14.2.12.1.54, the applicant did not provide verification of alarms which would indicate pump degradation by unacceptable temperatures within the pump motors, seals, or environment. In **RAI 407-3082, Question 06.03-73**, the staff requested that the applicant clarify if these alarms were checked elsewhere, and explain how the design objectives could be assured without these checks. In its response to **RAI 407-3082 Question 06.03-73 (RAI 06.03.04.01-3)**, dated August 5, 2009, the applicant added criteria for the testing and verification of the alarms which indicate SI pump operation to preoperational test abstract 14.2.12.1.54. Test Method Item C.3: “Alarms which indicate SI pump operation are verified.” Acceptance Criteria D.6: “Alarms which indicate SI pump operation perform as described in Section 6.3.5.3.” The response resulted in the DCD correctly defining the test method and acceptance criteria for test 14.2.12.1.54. Accordingly, the staff finds the response and proposed DCD changes acceptable.

The staff confirmed that the proposed changes were made to DCD Tier 2, Section 14.2.12.1.54, items C.3 and D.6, and **therefore, RAI 407-3082, Question 06.03-73, is resolved and closed**.

The staff noted that Test 14.2.12.1.54 stated, in part: "The RWSP contains an adequate supply of demineralized water for test performance." In **RAI 407-3082, Question 06.03-74**, the staff requested that the applicant explain the term “adequate supply,” in terms of level and volume, since this should be considered in the duration of the SI flow test time and the determination of adequate NPSH. In its response to RAI **407-3082, Question 06.03-74 (RAI 06.03.04.01-4)**, dated August 5, 2009, the applicant adequately clarified the meaning of “The RWSP contains an adequate supply of demineralized water for test performance." The applicant stated that the RWSP lowest water level for the test is the level where the ECCS/CS strainers are completely submerged to prevent air entrainment from the RWSP water surface to the SI pump suction pipe. The applicant stated that this water level is sufficient to ensure the NPSH of SI pumps in the test. During the test, the RWSP water level is maintained constant because the RWSP water used in the test is returned to RWSP using the CS/RHR pumps and the CS/RHR full-flow test lines. The staff reviewed the information contained in the applicant’s August 5, 2009, response to **RAI 407-3082 Question 06.03-74**, and finds the information acceptable because the applicant clarified the level to be the lowest level to prevent air entrainment into the SI pump and the SI pumped water is returned to the RWSP through the CS/RHR system.

Use of the lowest level ensures the adequacy of the pump NPSH. The RWSP water is recirculated during the test so no additional water is needed. Therefore, **RAI 407-3082, Question 06.03-74, is resolved and closed**.

The staff noted that for Test 14.2.12.1.56, "Safety Injection Check Valve Preoperational Test," the Acceptance Criteria included item 1. "The accumulator discharge and injection line check valves operate as demonstrated by verification of flow through the check valves as described Subsections 6.3.2.2.1 and 6.3.2.2.2." The staff issued **RAI 407-3082 Question 06.03-75** to request that the applicant modify this text to be consistent with the test objectives. In its response to **RAI 407-3082, Question 06.03-75 (RAI 06.03.04.01-5)**, dated August 5, 2009, the applicant modified Acceptance Criteria item 1 for Test 14.2.12.1.56 in the DCD to make it clear that the test refers to the SI line isolation valves. The staff reviewed the applicant’s response to **RAI 407-3082, Question 06.03-75** and finds that the response clarifies the acceptance criteria and ensures consistency between the test objectives and test criteria. Accordingly, **RAI407-3082, Question 06.03-75, is resolved and closed**.

DCD Tier 2 Figure 6.3-3, “SIS Elevation Diagram,” presents an elevation drawing of the SIS. The system piping would normally be filled and vented from the RWSP to the reactor vessel injection nozzles and the injection piping is completely filled with water to prevent water hammer. The staff requested the applicant explain the discrepancy between the elevations presentation in DCD Tier 2 Section 6.3.2.1.1 “High Head Injection System” which presented elevations in fractional feet and that in Figure 6.3-3 which presented elevations in the feet-inches. In its response to **RAI 391-2974, Question 06.03-16 (RAI 06.03.02.01-1)**, dated July 27, 2009, the applicant modified the DCD Tier 2 to write-up to use feet-inches for the elevations. The staff finds the response acceptable because it revised the DCD to have the consistency as requested by the staff. The staff confirmed that Revision 3 of DCD Tier 2, Section 6.3.2.1.1 contained the proposed DCD changes. Therefore, **RAI 391-2974, Question 06.03-16, is resolved and closed**.

In **RAI 391-2974, Question 06.03-27**, the staff requested that the applicant describe the void formation phenomena in the SI pump injection lines. In its response to **RAI 391-2974, Question 06.03-27 (RAI 06.03.02.02-1),** dated July 27, 2009, the applicant provided the analysis used to assess void formation due to water column separation in the SI piping. The applicant stated that a static head of 30 ft (9 m) would be required for column separation. The available head, 24 ft – 2 in, in the SI piping is less than 30 ft (9 m) and water column separation will not occur. The staff reviewed the information in the response to **RAI 391-2974, Question 06.03-27** and concurs with the analysis provided by the applicant as a conservative reactor building temperature was chosen to determine both the liquid vapor pressure and the available head at atmospheric conditions. The available head minus the liquid vapor pressure converted in static head was used to determine the 30 ft (9m) needed for column separation. Accordingly, the staff finds that the RAI response is acceptable and **RAI 391-2974, Question 06.03-27, is resolved and closed**.

The staff noted that during the blowdown phase, quick depressurization would occur through the entire primary system. Flashing could occur in the un-isolated sections of SI piping (between the reactor pressure vessel (RPV) DVI nozzle and the closest check valve. The staff determined that the applicant should demonstrate that, if flashing occurs, water hammer would not occur in these sections during a LOCA, or would not result in damage to the piping sufficient to degrade the SI injection. In **RAI 391-2974, Question 06.03-28**, the staff requested that the applicant address the above concerns. In its response to **RAI 391-2974, Question 06.03-28 (RAI 06.03.02.02-2),** dated July 27, 2009, the applicant provided its response to the staff’s concern about how void formation in the SI piping can be precluded during a postulated LOCA. The applicant presented a qualitative description of potential void formation and conditions which would lead to a water hammer event. The two conditions must be met for water hammer generation: (1) a two-phase environment where vapor and liquid exist together and (2) the steam void is surrounded by cold water. The applicant concluded for the SI piping in the US-APWR, even if condition (1) occurs, condition (2) cannot occur since the inner diameter of SI piping is as small as 3.44 inch (8.7 cm) and the flow velocity will be too high. The applicant concluded the steam void would not be entrained into the liquid phase, and would be pushed out into the RPV without a water hammer occurrence. However the response did not include sufficient detail to determine the adequacy of the evaluation presented. **RAI 391-2974, Question 06.03-28, is closed, but unresolved.**

In **RAI 695-4750, Question 06.03-90**, the staff requested that the applicant augment the discussion presented in the response to RAI 391-2974, Question 06.03-28, and provide a reference, that includes details based on flow regime mapping (expected flow, pressure temperature ranges) to support this conclusion. In its response to **RAI 695-4750, Question 06.03-90**, dated March 17, 2011, the applicant provided the requested information.

Water hammer during SI injection would occur due to steam being covered by water. However, it does not occur when the piping is completely water-solid (i.e., zero void fraction). The laminar flow condition limit which corresponds to a zero void fraction in the piping cross-section (i.e., vapor-liquid separation does not occur) was evaluated using the Froude Number (Fr) as shown in “An Evaluation of PWR Steam Generator Water Hammer,” NRC Report NUREG-0291 (1977). Using representative conditions for the SI piping during a LBLOCA, the Fr at these conditions is about 14. This value exceeds the limit shown NUREG-0291, Fr2 ≥ 0.25. This means the SI flow in the piping is in a water-solid condition and a steam void would not develop.

In addition, based on flow regime mapping for horizontal piping shown in Figure 5 in “A Model for Predicting Flow Regime Transition in Horizontal and Near Horizontal Gas-Liquid Flow,” AIChE Journal, Vol.22, Issue 1 Pages.47-55 (January 1976), even if steam and water flow simultaneously in the piping under LBLOCA conditions, water hammer will not occur because the steam-water mixture is in a dispersed flow condition. The response to **RAI 695-4750, Question 06.03-90,** is acceptable because the applicant provided the requested information to support the conclusion that a steam void would not be entrained into the SI liquid phase (condensing the steam) and would be pushed out into the RPV without water hammer occurring. Therefore, **RAI 695-4750, Question 06.03-90, is resolved and closed.**

6.3.4.2.2 SI Pump Support Systems

The SI pumps are horizontal, multi-stage centrifugal type pumps. Typically these pumps require cooling (from non-safety related and safety-related cooling systems) to protect the motors and seals to ensure they can provide the required flow for the duration of the accident. The staff requested the applicant include a description of the cooling system(s) and the associated failure modes and effects analysis in the DCD.

In **RAI 391-2974, Question 06.03-29**, the staff requested that the applicant describe the cooling system(s) and include the cooling system(s) in the failure modes and effects analysis. In its response to **RAI 391-2974, Question 06.03-29 (RAI 06.03.02.02-3),** dated July 27, 2009, provided a description of the SI pump cooling system and included the cooling system in the failure modes and effects analysis. The applicant stated that the SI pump is supplied with cooling water from the CCWS. During an accident, the CCWS is divided into four independent trains, and the failure of one CCWS train does not result to the simultaneous loss of function of more than two SI pumps (on failed and one out of service for maintenance).

The CCWS is described in DCD Tier 2 Chapter 9, Section 9.2.2, "Component Cooling Water System." DCD Tier 2 Section 6.3, Table 6.3-6 Failure Mode and Effect Analysis – Safety Injection System was revised to include the "Component Cooling Water" as item 11.

The staff reviewed the information contained in the July 27, 2009, response to **RAI 391-2974 Question 06.03-29** and determined that more information was necessary. The staff issued follow-up RAI **695-4934, Question 06.03-91,** to request that the applicant modify DCD Tier 2, Chapter 6 to support a conclusion that the US-APWR design meets GDC 35. In its response to **RAI 695-4934, Question 06.03-91**, dated March 17, 2011, the applicant committed to add the description mentioned in the July 27, 2009, response to **RAI 391-2974, Question 06.03-29** to DCD Subsection 6.3.2.2.1. The staff reviewed the proposed changes to DCD Tier 2, Section 6.3.2.2.1 and finds them acceptable because they describe the SI pump cooling system, using the CCWS, and describe the safety design of the CCWS. **RAI 695-4934, Question 06.03-91, is being tracked as a Confirmatory Item to ensure the DCD is modified consistent with the RAI response.**

The staff noted that operator actions may be required to protect the SI pumps, and that the MCR and RSC should have alarms that indicate unacceptable parameters such as high bearing oil, motor winding, or motor air temperatures. In **RAI 391-2974, Question 06.03-58**, the staff requested that the applicant clarify if these alarms were available. In its response to **RAI 391-2974, Question 06.03-58**, dated July 27, 2009, the applicant revised DCD Tier 2 Chapter 6, Section 6.3.5.3, "Safety Injection Pumps," to describe the following MCR and RSC alarms which indicate unacceptable parameters for an SI pump and motor:

- Pump bearing temperature- High.
- Pump bearing oil pressure- Low.
- Motor stator temperature- High.
- Motor cooling air temperature- High.

The July 27, 2009, response to **RAI 391-2974, Question 06.03-58 (RAI 06.03.05.03-1),** is acceptable because it provided a DCD revision that the staff requested to improve the clarity of the DCD. The staff confirmed that Revision 3 of DCD Tier 2, Section 6.3.5.3, contained the proposed DCD changes. Therefore, **RAI 391-2974, Question 06.03-58, is resolved and closed.**

The building area which houses the SIS pumps and instrumentation should be provided with HVAC to protect the pumps and instrumentation from excessive temperatures. The staff requested the applicant include a description of the HVAC system and the associated failure modes and effects analysis in the DCD. In its response to RAI 391-2974, Question 06.03-30 (RAI 06.03.02.02-4), dated July 27, 2009, the applicant provided a description of the HVAC and included the HVAC in the failure modes and effects analysis. The environmental condition in the Safeguard Component Area in the R/B, where the SIS components are located, is maintained by the Safeguard Component Area HVAC system. The Safeguard Component Area HVAC System consists of four completely independent trains; therefore, the failure of one train will not result in the simultaneous loss of the SIS function of more than two trains (one from a failure and one assumed out for maintenance). The Safeguard Component Area HVAC system is described in DCD Tier 2 Chapter 9, Section 9.4.5, "Engineered Safety Feature Ventilation System." DCD Tier 2 Section 6.3, Table 6.3-6 Failure Mode and Effect Analysis - Safety Injection System was revised to include the "Safeguard Component Area HVAC" as Item 12.

The staff reviewed the information in the applicant's July 27, 2009, response to **RAI 391-2974, Question 06.03-30**, and determined that additional information was necessary. The staff noted that the response did not indicate that the DCD Tier 2, Section 6.3, would be modified to provide the description of the HVAC for SI pump cooling during an accident. In followup **RAI 695-4934, Question 06.03-92**, the staff requested that the applicant add sufficient information to DCD Tier 2, Section 6.3, to support a conclusion that the US-APWR design meets GDC 35. In its response to **RAI 695-4934, Question 06.03-92**, dated March 17, 2011, the applicant committed to revise DCD Tier 2, Section 6.3.2.2.1, to include the description mentioned in the July 27, 2009, response to **RAI 391-2974 Question 06.03-30**. **RAI 695-4934, Question 06.03-92, is being tracked as a Confirmatory Item.**

In **RAI 391-2974 Question 06.03-50**, the staff requested that the applicant clarify if the SIS design considers protection from fires and identify the DCD section(s) which address SIS fire protection. In its response to **RAI 391-2974 Question 06.03-50 (RAI 06.03.02.06-1),** dated July 27, 2009, the applicant provided a reference to fire assessment for the SIS enclosures. The applicant stated that the SIS design considers protection from fires in accordance with the fire protection requirements. The applicant also stated that the fire protection is described in DCD Tier 2, Chapter 9, Section 9.5, "Other Auxiliary Systems," Section 9.5.1, "Fire Protection Program" and Appendix 9A, "Fire Hazard Analysis." The staff's evaluation of fire protection is contained in Section 9.5 of this SER. **RAI 391-2974 Question 06.03-50, is resolved and closed.**

6.3.4.2.3 Emergency Letdown System

The emergency letdown system provides redundancy to the normal CVCS in achieving cold shutdown boration conditions. Two emergency letdown lines (one each from reactor hot legs B and D) direct reactor coolant to spargers in the RWSP. The SI pumps return borated RWSP water to the reactor vessel through each pump's DVI nozzle, or through each associated reactor hot leg injection line. The staff requested the applicant address the adequacy of the sparger design. In its response to **RAI 391-2974, Question 06.03-17 (RAI 06.03.02.01-2)**, dated July 27, 2009, the applicant stated the hydrodynamic loads evaluation for the emergency letdown system spargers in the RWSP has not yet been completed. However, the applicant stated that it does not expect the hydrodynamic loads to be a problem because the reactor coolant discharged from the sparger during emergency letdown is small, about 265 gpm (35 ft^3/min). The staff issued **RAI 982-6036, Question 06.03-111,** to request that the applicant demonstrate that the hydrodynamic load evaluation of the sparger design shows adequate performance. **RAI 391-2974, Question 06.03-17, is closed, but unresolved and follow up RAI 982-6036, Question 06.03-111, was written to address the sparger design. RAI 982-6036, Question 06.03-111 is being tracked as an Open Item.**

One normally closed motor-operated gate valve and one normally closed motor-operated globe valve in series are aligned in each of the two emergency letdown lines. As noted in DCD Tier 2 Figure 6.3-2, "ECCS Piping and Instrumentation Diagram (Sheet 2 of 4) and (Sheet 3 of 4)," the globe valve has throttling capabilities and the open or closed position is monitored. These valves are remotely opened by operator action during a safe shutdown for a feed and bleed emergency letdown/boration with the SI pump operation. The valves are designed to Equipment Class 1, Seismic Category I standards.

In its response to **RAI 391-2974, Question 06.03-39 (RAI 06.03.02.02-15),** dated July 27, 2009, the applicant provided a discussion on the globe valve's throttling capability, including its purpose. The function of the Emergency Letdown System is to divert the letdown to the RWSP as a substitute for the CVCS during safe shutdown. The valves are provided with throttling capability to enable the control of letdown flow rate. The valves are not used for flow control except in the safe shutdown.

During an accident or a transient, these valves are used from a fully closed position; therefore, the correct position of these valves during normal operation must be fully closed. The text "in the correct position" in SR 3.5.2.2 of DCD Tier 2 TS 3.5.2 means "in fully closed position" for these valves. The display of a throttled position for these valves does not need to be provided in the MCR or RSC for verifying the correct position, and it is ensured that these valves do not throttle open during an event by verifying their fully closed position. During a safe shutdown, the emergency letdown flow rate is controlled by the operator by monitoring the pressurizer water level and safety injection flow rate. DCD Tier 2 Section 6.3.2.2.6.17 was revised to include the reason for the throttling capability: "2nd. Emergency Letdown Line Isolation Valves (SIS-MOV-032B and D) have the throttling capability to enable the control of letdown flow rate." The response to Question 06.03-39 provided the explanation sought by the staff and is therefore acceptable. Accordingly, **RAI 391-2974, Question 06.03-39, is resolved and closed.**

6.3.4.2.4 Major Valves

There is a normally open motor-operated gate valve in each of the four SI pump suction lines from the RWSP. These valves remain open during normal and emergency operations. The valves may be closed by operator action if an SIS line has to be isolated from the RWSP to terminate a leak, or if maintenance is required.

There is a normally open motor-operated gate valve in each pump discharge line that serves as the outboard containment isolation valve. These valves can be closed remotely by operator action if containment isolation is required.

There is a normally open motor-operated globe valve, with throttling capability to control the flow downstream of each of the four DVI lines, inside the containment. The open or closed position is monitored; however these valves may be throttled resulting in partial opening. The valves are remotely closed for switchover to the hot leg injection by operator action in the event of a LOCA, to preclude boric acid precipitation in the reactor core. These valves also provide the capability to control the SI pump flow to maintain the RCS inventory during safe shutdown.

In its response to **RAI 391-2974, Question 06.03-36 (RAI 06.03.02.02-12),** dated July 28, 2009, the applicant provided a staff-requested discussion on the globe valve's throttling capability, including its purpose. The response provided the information requested and is therefore acceptable. The valves are not used for flow control except during safe shutdown. During an accident or a transient, these valves are used from a fully open position; therefore, the correct position of these valves during normal operation must be fully open. The phrase, "in the correct position," in SR 3.5.2.2 of DCD Tier 2 TS 3.5.2 means in fully open position for these valves. The display of a throttled position for these valves does not need to be provided in the MCR or RSC for verifying the correct position, and it is ensured that these valves do not throttled open during an event by verifying their fully open position. During a safe shutdown, the emergency letdown flow rate is controlled by the operator by monitoring the pressurizer water level and safety injection flow rate. The applicant having provided the requested information, which was acceptable, **RAI 391-2974, Question 06.03-36, is resolved and closed.**

There is a normally closed motor-operated globe valve in each of the four hot leg injection lines. These valves are remotely opened by operator action to initiate hot leg injection. The four hot-leg injection isolation valves are Equipment Class 1, Seismic Category I.

- One swing check valve is aligned in each safety injection pump discharge line as a containment isolation valve.

- Two swing check valves in series are aligned in each direct vessel injection line. These valves are designed to Equipment Class 1, Seismic Category I standards. One swing check valve is aligned in each hot leg injection line. These valves are designed to Equipment Class 1, Seismic Category I standards.

- One swing check valve is aligned in each safety injection pump discharge line to prevent discharge line drain-down.

- One normally closed, motor-operated globe valve, with a throttling capability, is installed in each of four SI pump test lines. These valves have their control power locked out during normal plant operation. The test lines are located inside the containment and are routed from the pump test discharge lines to the RWSP.

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. In **RAI 695-4934, Question 06.03-93,** the staff requested that the applicant address four issues related to gas accumulation in the US-APWR SI SIS and CS/RHRS.

In Item (1) of this RAI question, the staff requested that the applicant identify and discuss potential pathways for gas intrusion into the SIS and CS/RHRS, and to identify features present in the US-APWR design to prevent gas accumulation to ensure SIS and CS/RHRS operability, for example the number and location of high point vents, and piping slopes.

The applicant provided the requested information in response to **RAI 695-4934, Question 06.03-93**. The potential pathways for gas intrusion in the US-APWR SIS and CS/RHRS are:

- Open connections to the RWSP via SIS and CS/RHRS. (Such as pump suction lines and pump discharge lines).

- Open connections to the containment atmosphere via RHRS suction lines during RCS mid-loop operation (i.e., during refueling).

The following design features prevent or control gas accumulation to acceptable levels:

- Pump suction piping is designed to be a continuous downward slope and does not include inverse slope which may trap or prevent venting of accumulated gases.

- Pump discharge piping is designed to be a continuous upward slope and does not include inverse slope which may trap or prevent venting of accumulated gases. Some portions of the piping such as downstream of the check valve in the pump discharge piping are difficult to passively vent, but the air in such portions will be removed by active venting.

- Full-flow test lines are located at the high point of the SIS and CS/RHRS and discharge into the RWSP. Periodic active venting will be applied to the SIS and CS/RHRS. Inservice testing required by DCD Tier 2 Section 3.9.6.2 includes periodic testing through the full flow test lines, which discharge to the RWSP (DCD Tier 2 Chapter 16 TS SR 3.6.6.2). These tests periodically discharge voids, minimize unacceptable dynamic effects such as water hammer, and ensure operability of the suction and discharge lines.

- Sump strainers are installed in the RWSP to prevent vortex formation when the SI pumps and CS/RHR pumps are taking suction from the RWSP.

The applicant provided the requested information for the SIS and CS/RHRS concern potential gas accumulation and design features and testing to minimize gas accumulation.

A similar gas accumulation question was asked as part of the DCD Chapter 5.4.7 review. Some the information provided in the response to RAI 464-3520, Question 05.04.07-11 is also applicable to the DCD Section 6.3 review. In **RAI 464-3520, Question 05.04.07-11,** the staff asked the applicant to address the Final Interim Staff Guidance (ISG) DC/COL-ISG-019 (ML111110572) which provides staff guidance with regard to (1) potential gas accumulation locations and intrusion mechanisms, (2) P&ID/isometric drawing confirmation with the as-built configuration and (3) surveillance and venting procedures. In the applicant's amended response to **RAI 464-3520, Question 05.04.07-11**, dated August 6, 2012, the staff found the response adequate and consistent with the guidance of ISG DC/COL-ISG-019. In the amended response to **RAI 464-3520, Question 05.04.07-11,** the applicant detailed design features which would minimize gas accumulation, a commitment to install vents at high point locations in the as-built design and to develop venting and surveillance procedures to properly fill and vent ECCS and RHR piping. The applicant's response to **RAI 464-3520, Question 05.04.07-11,** described the following design features for the CS/RHR and ECCS vent valves and piping design features:

1. Vents will be located at piping high points.
2. Vents will be installed on upper pipe surfaces, not sides or bottom.
3. Horizontal piping runs will be sloped upward in the direction of flow on the discharge side of pumps and downward in the direction of flow on the suction side of pumps.
4. Vents will be installed at high points of inverted-U piping sections.
5. Pump suction piping will be connected on lower sides of inlet headers.
6. Vent accessibility will be assured by provision of adequate space, ladders, platforms, etc.
7. Vent areas will be thermally shielded to protect operators during venting operations.
8. Vent areas will be radiation shielded to protect operators during venting operations in accordance with ALARA principles.
9. Vent valves will be clearly identified by tagging.
10. Vent areas will be adequately lighted.

The applicant has agreed to modify Tier 1, ITAAC Table 2.4.4-5, "Emergency Core Cooling System Inspections, Test, Analyses and Acceptance Criteria," Items 2.b.i and 2.b.ii, to include a verification that design documents contain the above features to limit gas accumulation and that the as-built design correctly incorporates those design features. **The modification to Tier 1, ITAAC Table 2.4.4-5 items 2.b.1 and 2.b.ii as proposed in the response to RAI 464-3520, Question 05.04.07-11, is being tracked as a Confirmatory Item**.

In addition to the inclusion of design features to limit gas accumulation, the applicant's amended response to **RAI 464-3520, Question 05.04.07-11,** also addressed venting and SRs. The applicant identified the SIS discharge line, upstream of the SIS0VLV-010 check valve as an area susceptible to gas accumulation. TS SR 3.5.2.7 was added to verify that ECCS locations susceptible to gas accumulation are sufficiently filled with water. In addition, this area susceptible to gas accumulation is dynamically swept quarterly to the RWSP by the Inservice testing program as described in DCD Section 3.9.6.2 and TS 5.5.

In Item (2) of **RAI 695-4934, Question 06.03-93**, the staff requested the applicant explain why the test conditions for the SIS, DCD Tier 1 Table 2.4.4-5 Emergency Core Cooling System Inspections, Tests, Analyses, and Acceptance Criteria, Item 7.d (Tests to measure the as-built safety injection pump suction pressure will be performed. Inspections and analysis to determine NPSH available to each safety injection pump will be performed.), are conservative especially with regard to gas entrainment and its effect on NPSH.

The applicant provided the requested information in response to **RAI 695-4934, Question 06.03-93,** dated March 18, 2011. The ITAAC test conditions for the SIS test will be done at the minimum RWSP water level in the event of an accident. The lowest water level of the RWSP is the most conservative with regard to gas entrainment (i.e., vortex) and NPSH. The applicant provided the requested information and the staff agrees the minimum RWSP level is conservative with respect to gas entrainment and NPSH.

In Item (3) of **RAI 695-4934 Question 06.03-93**, the staff requested the applicant explain why the test conditions for the CS/RHRS, (DCD Tier 1 Table 2.4.5-5 Residual Heat Removal System Inspections, Tests, Analyses, and Acceptance Criteria, and Acceptance Criteria, Item 8.f (Tests to measure the as built CS/RHR pump suction pressure will be performed. Inspections and analysis to determine NPSH available to each CS/RHR pump will be performed.), are conservative especially with regard to gas entrainment and its effect on NPSH.

The applicant provided the requested information in response to **RAI 695-4934, Question 06.03-93,** dated March 18, 2011. The ITAAC test conditions for a CS/RHR pump test will be done at the minimum RWSP water level in the event of an accident. The lowest water level of the RWSP is the most conservative with regard to gas entrainment (i.e., vortex) and NPSH. With regard to vortex, mid-loop operation is the most severe condition for a CS/RHR pump since the distance between CS/RHR pump suction nozzle on the main coolant pipe and water surface is the smallest. DCD Tier 1 Table 2.4.5-5, Item 8a.ii provides the ITAAC for this operation mode. Therefore, current ITAAC covers the conservative conditions with regard to gas entrainment and its effect on NPSH. The applicant provided the requested information and the staff agrees the minimum RWSP level is conservative with respect to gas entrainment and NPSH and that mid-loop operation is the most severe condition. The potential of mid-loop RHR gas entrainment and vortexing is evaluated in the staff's DCD 5.4.7 SE.

In Item (4) of **RAI 695-4934 Question 06.03-93**, the staff noted that NRC Information Notice (IN) 2010-11 (ML100640465) discussed operating experience in which there was a potential for the RHR system to be inoperable due to steam voiding. The staff requested that the applicant address the design of the US-APWR for this issue, and if it was sufficient to use the forced cooling through the RHR minimum-flow recirculation method of cooling for the US-APWR to prevent steam voiding, or are other procedures required to preclude the potential for steam voiding.

The applicant provided the requested information in response to **RAI 695-4934, Question 06.03-93,** dated March 18, 2011. Countermeasures for this issue, such as appropriate procedures, will be developed during the detailed design phase to preclude the potential for steam voiding. For example, during startup operation three of four trains of the RHR system will be isolated from the RCS before entering Mode 4. One train will remain in operation as the RHR system. The other three trains will remain in standby as the CSS. The applicant provided a description of possible countermeasures to preclude the potential for steam voiding in the RHRS in response to the staff request. The applicant having provided satisfactory information in response to **RAI 695-4934, Question 06.03-93, this RAI question is resolved and closed.**

6.3.4.2.5 Failure Modes and Effects Analysis

The FMEA for the SI pumps is presented in DCD Tier 2 Table 6.3-6, "Failure Modes and Effects Analysis - Safety Injection System." The staff identified a number of concerns with the FMEA and could not determine if all failures have been considered, or if the instrumentation was also available at the RSC, and requests additional information.

The applicant responses to **RAI 391-2974, Question 06.03-45 (RAI 06.03.02.05-1), RAI 391-2974, Question 06.03-46 (RAI 06.03.02.05-2), RAI 391-2974, Question 06.03-47 (RAI 06.03.02.05-3), RAI 391-2974, Question 06.03-48 (RAI 06.03.02.05-4) and RAI 391-2974, Question 06.03-52 (RAI 06.03.02.08-2)** are addressed in Section 6.3.4.1, "Advanced Accumulators and Support Systems," of this evaluation, where the applicant's responses were found acceptable.

Long-Term Cooling and Feed Line Breaks

There was a concern with the identification of the accidents which require components of the ECCS to mitigate the consequences of an accident or maintain the plant in an acceptable state, for example to maintain long-term cooling. DCD Tier 2 TS Bases for B 3.5.4, "Refueling Water Storage Pit," stated: "*The maximum temperature ensures that the amount of cooling provided from the RWSP during the heatup phase of a feedline break is consistent with safety analysis assumptions; the minimum is an assumption in both the MSLB and inadvertent ECCS actuation analyses, although the inadvertent ECCS actuation event is typically nonlimiting.*" The staff requested the applicant provide additional clarification concerning feedline breaks and long-term cooling.

In its response to **RAI 407-3082, Question 06.03-59 (RAI 06.03.01.01-1),** dated August 5, 2009, the applicant stated the text cited above was incorrect. The SI pumps are not capable of injecting water into the RCS at the normal, at-power operating pressure. During the feedline break event the RCS temperature and pressure will increase and the ECCS would not be able to inject water into the RCS. Therefore, TS Bases for B 3.5.4 was modified to remove the statement that the maximum RWSP temperature is set by the feedline break analysis. The staff agrees with the Bases modification as the SI system plays no role in mitigating the feedline break accident and hence the maximum RWSP temperature is not set by this event. **RAI 407-3082, Question 06.03-59, is resolved and closed.**

6.3.4.2.6 Safe Shutdown

In **RAI 391-2974 Question 06.03-1 (RAI 06.03.01.02-1),** the staff requested that the applicant revise DCD Tier 2, Section 6.3.1.2, "Safe Shutdown," to identify those components of the ECCS required for safe shutdown in the event the normal systems are unavailable and summarize the functions of these components during the shutdown without normal systems. In its response to **RAI 391-2974, Question 06.03-1 (RAI 06.03.01.02-1),** dated July 27, 2009, the applicant added the following text to the DCD: "For boration and make up for compensation for shrinkage, operation of two trains of high-head injection system, each of which includes one safety injection pump and one flow control valve, is required. For letdown of reactor coolant, operation of one train of emergency letdown system including one flow control valve and one stop valve is required."

In its response to **RAI 391-2974, Question 06.03-2 (RAI 06.03.01.02-2),** dated July 27, 2009, the applicant clarified the safe shutdown mission of the ECCS components. The safe shutdown mission is to bring the plant to cold shutdown condition using only safety-related systems. The function of the ECCS during safe shutdown operation is to borate the reactor coolant by "feed and bleed," and provide make up to compensation for RCS shrinkage during safe shutdown operation.

DCD Tier 2, Section 6.3.1.2, “Safe Shutdown,” refers to DCD Tier 2 Section 5.4.7, “Residual Heat Removal System,” for the details of the safe shutdown design bases. Figure 5.4.7-4, “Residual Heat Removal System Mode Diagram,” Sheet 1 and 2, appear to be identical. Based on DCD Tier 2, Section 15.0.08, the alignment seems to be different for a LBLOCA where containment sprays, and eventually, the CS/RHR pump discharge is aligned to the RWSP. In **RAI 407-3082, Question 06.03-83 (RAI 06.03.01.02-1),** the staff asked if the valve alignment for Safe Shutdown (Sheet 2 of 4) is the same as Normal Shutdown (Sheet 1 of 4) and if all AOOs have the valve lineup given in Figure 5.4.7-4, Sheet 2. In its response to **RAI 407-3082, Question 06.03-83**, dated August 5, 2009, the applicant stated the valve alignment of the RHRS is the same for both Safe Shutdown and Normal Shutdown. However, during safe shutdown, the required cooling performance is accomplished by the operation of at least 2/4 trains. The Containment Spray System cools the RWSP water by operating the CS/RHR pumps connected to the discharge lines in the RWSP. The valve alignment to transfer the plant to cold shutdown is the same as shown in Figure 5.4.7-4, Sheet 2 for all AOOs.

The responses to **Questions 06.03-1, 06.03-2, and 06.03-83** provided the additional information sought by the staff and revised the DCD to better describe ECCS components required for safe shutdown in the event normal systems are unavailable. Therefore, the responses are acceptable, and accordingly, **RAI 391-2974, Question 06.03-1, RAI 391-2974, Question 06.03-2, and RAI 407-3082, Question 06.03-83, are resolved and closed.**

6.3.4.3 Refueling Water Storage Pit and ECC/CS Strainers

The ITAAC requirements for the RWSP is described in DCD Tier 1, 2.4.4, “Emergency Core Cooling System,” Table 2.4.4-5, “Emergency Core Cooling System Inspections, Tests, Analyses, and Acceptance Criteria.” The design commitment acceptance criterion is 7.b.iii.b: The volume of the as-built RWSP is at least 84,750 ft^3 (2400 m^3).

The TS requirements for the RWSP are provided in TS 3.5, “Emergency Core Cooling Systems (ECCS),” 3.5.4, “Refueling Water Storage Pit.”

The RWSP is located in the Containment. This structure is Seismic Category I and provides tornado missile barriers to protect the ECCS.

DCD Tier 2 Section 6.2.1 discusses the containment environmental conditions during accidents, and DCD Tier 2 Chapter 3, Section 3.11 discusses the suitability of equipment for design environmental conditions. All valves required to be actuated during ECCS operation are located to prevent vulnerability to flooding. Protection of the ECCS from missiles is discussed in DCD Tier 2 Chapter 3, Section 3.5. Protection of the ECCS against dynamic effects associated with the rupture of piping is described in Section 3.6. Protection from flooding is discussed in Section 3.4.

6.3.4.3.1 System Description

Refueling Water Storage Pit (RWSP)

The RWSP is designed to have a sufficient inventory of boric acid water for refueling and long-term core cooling during a LOCA. A nominal 2,399.8 m^3 (84,750 ft^3) of water is available in the RWSP. Sufficient submerged water level is maintained to ensure the minimum NPSH for the SI pumps. The RWSP capacity includes an allowance for instrument uncertainty and the amount of holdup volume loss within the containment. The capacity of the RWSP is optimized for a LOCA. A refueling water storage auxiliary tank containing 832.8 m^3 (29,410 ft^3) is provided separately outside the containment to ensure that the required volume for refueling operations is available. DCD Tier 2 Table 6.3-5 presents the relevant RWSP data and a detailed description of structure and capacity of RWSP is provided in DCD Tier 2 Section 6.2.2.2. The temperature during normal operation is in a range of 21.11 ºC to 48.89 ºC (70 to 120 °F). The peak temperature following a LOCA is approximately 124.4 ºC (256 °F).

The boric acid water in the RWSP is purified using the RWS. The RWS may be cross-connected to one of two SFPCS filter and demineralizer vessels to remove the solid materials and the dissolved impurities for purification. The capacity of the purification subsystem is designed to maintain the chemistry of the spent fuel pool, the refueling cavity, the refueling water storage auxiliary tank, and the RWSP. DCD Tier 2 Chapter 9, Section 9.1.3, “Spent Fuel Pit Cooling and Purification System,” discusses the SFPCS purification of the boric acid water.

The volume and temperature of the water in the RWSP are important factors in the determination of NPSH and maintenance of a proper pressure driving head to the recirculation filters. Also, during a LOCA, significant amounts of run-off from the upper containment areas into the RSWP may be captured in hold up locations throughout the containment.

The staff requests that information related to these evaluations be thoroughly discussed in the DCD.

In **RAI 391-2974, Question 06.03-24 (RAI 06.03.02.02.03-1)**, the staff asked the applicant to identify where the instrument uncertainty and holdup volume loss evaluations are documented, since it is stated in DCD Tier 2, Section 6.3.2.2.3, that “The RWSP capacity includes an allowance for instrument uncertainty and the amount of holdup volume loss within the containment.” The revised hold-up volumes due to the RWSP redesign are given in Table 3-10, “Upstream Effect Hold-up Volumes,” of MUAP-08001, Revision 7. The staff evaluation of the revised holdup areas is given in Section 6.2.2 of this SER. A level instrument uncertainty of 1.7 percent is assumed. Therefore, when water level reaches the 96 percent refill setpoint the actual water level may be 94.3 percent (96 percent-1.7 percent) as given in the response to **RAI 839-6103**, dated May 29, 2012. The nominal water level of [ ] given Table 6.2.1-33, “RWSP design Features,” reflects the 94.3 percent level. Therefore, instrument uncertainty is conservatively accounted for when determining the RWSP level and hence minimum RWSP water volume used in the LOCA analyses. The applicant’s response provided the requested information and is acceptable. Therefore, **RAI 391-2974, Question 06.03-24, is resolved and closed.**

In **RAI 407-3082 Question 06.03-61 (RAI 06.03.02.02-19)**, the staff asked the applicant to explain how TS SR 3.5.4.2, which verifies that the RWSP borated water volume, demonstrates that the required minimum volume of 2,300.18 m^3 (81,230 ft^3) is available in the RWSP.

The RWSP was redesigned to address issues with GS-191 and hence a revised response to **RAI 407-3082, Question 06.03-61**, was submitted. In the revised response to **RAI 407-3082,** Question 06.03-61, dated January 15, 2013, the applicant stated the RWSP boric acid water volume of 321,700 gallons (43,000 ft^3) (previously 329,150 gallons, 44,000 ft^3) described in TS SR 3.5.4.2 is used in the LBLOCA analysis as described in B 3.5.4, the 4th paragraph, and is converted to 43,000 ft^3 (1,218 m^3). The 43,000 ft^3 (1,218 m^3) is obtained by subtracting the ineffective pools water volume of 45,050 ft^3 (1,275 m^3) from the total of the water volume stored between the 0 – 94.3-percent level (the 94.3 percent includes a 1.7-percent measurement uncertainty) and the water volume (8,130 ft^3, 230 m^3) stored below the 0-percent water level. To ensure the water volume used in the safety analysis (43,000 ft^3, 1,218 m^3), the RWSP water level needs to be maintained at 96-percent indicated level. Therefore, the water volume described in TS SR 3.5.4.2 was revised to 597,800 gallons (80,000 ft^3), converted from the water volume stored between the 0 – 96-percent indicated water level of 79,920 ft^3 (2,263 m^3). In addition, a description of the relationship between the 321,700 gallons (43,000 ft^3) and 597,800 gallons (80,000 ft^3) was added in B 3.5.4 including a statement that the 597,800 gallons (80,000 ft^3) bounds the SI pump and CS/RHR pump NPSH requirements. The response to **RAI 407-3082, Question 06.03-61,** corrected the DCD TS SR 3.5.4.2 regarding RWSP borated water volume and is therefore acceptable. Accordingly, **RAI 407-3082, Question 06.03-61, is resolved and closed.**

In **RAI 695-4934, Question 06.03-96,** the staff requested that the applicant address three issues concerning the RWSP: (1) how temperature is controlled in the RWSP, (2) the lack of a high temperature alarm in the RWSP, and (3) where the RWSP level instrumentation and alarms are tested. In its response to **RAI 695-4934, Question 06.03-96,** dated March 17, 2011, the applicant addressed the staff issues as follow:

> Issue (1): TS 3.5.4, “Refueling Water Storage Pit,” specifies the lowest RWSP temperature limit as 32 °F, the freezing point of water. Therefore, the RWSP water temperature does not decrease below this temperature limit. The RWSP is located at the lowest part of the containment, and contains no heat source. Heat transferred from the containment air space is the only means to increase the RWSP water temperature during normal plant operation. Therefore, the RWSP water temperature during normal operation is maintained at a temperature less than or equal to the temperature of containment air space.
>
> The highest temperature limit of the RWSP TS is consistent with that of the containment air space, 120 °F (49 °C) (TS 3.6.5, “Containment Air Temperature”) therefore no specific temperature control is needed to maintain the RWSP water temperature within the limit. The staff concludes that there is no need to provide active temperature control (either by heaters or coolers) in the RWSP
>
> Issue (2): A high temperature alarm is not provided for the RWSP because the RWSP water temperature is maintained at less than or equal to the containment air temperature during normal plant operation. The staff concludes that there is no need for a high temperature alarm in the RWSP.
>
> Issue (3): The inspection and testing methods for the RWSP water level instrumentation and alarms are provided in DCD Tier 2 Chapter 14, “Verification Program”, Subsection 14.2.12.1.59, “Refueling Water Storage System Preoperational Test.” The staff confirmed the preoperational test includes the instrumentation and alarms.

The response to **RAI 695-4934, Question 06.03-96,** addressed the staff issues concerning the RWSP, and therefore, **RAI 695-4934, Question 06.03-96, is resolved and closed**.

The staff observes that the bases for SR 3.5.4.1 stated that “The SR is modified by a note that eliminates the requirement to perform this Surveillance when ambient air temperatures are within the operating limits of the RWSP. With ambient air temperatures within the band, the RWSP temperature should not exceed the limits.” The staff asked the applicant to define the meaning of “ambient air temperature” and to clarify if it is the containment temperature or the environmental temperature. A note to this effect did not appear in TS 3.5.4. The staff asked the applicant to explain or modify the TS appropriately. In its response to **RAI 407-3082, Question 06.03-80 (RAI 06.03.03.01-1)**, dated August 5, 2009, the applicant clarified that the ambient temperature referred to the containment temperature and the following note was added: “Only required to be performed when containment air temperature is < 32 °F or > 120 °F.” The response to **RAI 407-3082, Question 06.03-80,** added the requested clarification as a note to SR 3.5.4.1 of TS 3.5.4 and is therefore acceptable. Accordingly, **RAI 407-3082, Question 06.03-80 is resolved and closed.**

The purification and boration of the RWSP water is paramount in providing clean, borated water to the reactor during a LOCA. It is not clearly described in DCD Tier 2 how this is accomplished. In DCD Tier 2, Section 6.3.2.2.3, “Refueling Water Storage Pit,” the applicant states: “The boric acid water in the RWSP is purified using the refueling water storage system (RWS). The RWS may be cross-connected to one of two SFPCS filter and demineralizer vessels to remove the solid materials and the dissolved impurities for purification. The capacity of the purification subsystem is designed to maintain the chemistry of the spent fuel pool, the refueling cavity, the refueling water storage auxiliary tank, and the RWSP.”

In DCD Tier 2 Section 15.6.5.3.1.3 Post-LOCA Long Term Cooling Evaluation Model, Borated Water Source, on Page 15.6-72, MHI states that “The RWSP, accumulator, and RCS are considered as the only sources of borated water.” In **RAI 407-3082, Question 06.03-62 (RAI 06.03.02.02-20)**, the staff asked the applicant to:

1) Identify if there are any other sources of borated water that could enter the containment.

   In response to Part 1 of RAI 407-3082, Question 06.03-62, the applicant stated there is no other boric acid water source to flow into the containment than RWSP, accumulators, and the RCS.

2) If the RWS is cross-tied to the RWSP during a LOCA, an additional source of borated water is introduced. Please explain the consequences of this source of borated water and how it would impact boron concentrations in the reactor.

   In response to Part 2 of RAI 407-3082, Question 06.03-62, the applicant stated the water sources for the Refueling Water Recirculation Pumps (RWRPs) described in the DCD Tier 2 Chapter 6, Section 6.3, Figure 6.3-7 are the followings:

   a. RWSP.
   b. Refueling Cavity.
   c. Refueling Water Storage Auxiliary Tank (RWSAT).

   The refueling cavity (Item b above) is used as a water source for the RWRPs only during refueling operation; therefore, item b could not be a water source during the LOCA.

   During normal operation, RWSAT (Item c above) may be used as a water source for the RWRPs; however, isolation valves between the RWRPs outlet and RWSP are closed during RWRPs operation with the RWSAT. Therefore, this tank is not used as a water source during the LOCA.

3) If the RWS system is automatically isolated during the LOCA, identify how the additional boric acid left in the piping between the RWSP and the isolation valves is accounted for in the boric acid concentration calculation.

In response to Part 3 of RAI 407-3082, Question 06.03-62, the applicant stated during the LOCA, the RWS is isolated from RWSP by automatic closure of containment isolation valves shown in Figure 6.3-7 of DCD Tier 2 Chapter 6, Section 6.3 (the RWRPs are also automatically stopped on receiving safety injection signal). After the isolation, the boric acid water in piping between RWSP and isolation valves does not flow into the RWSP; therefore, this boric acid water is not considered in the boron concentration calculation.

The above responses to RAI 407-3082, Question 06.03-62 confirm that the only sources of borated water during a postulated LOCA are the RWSP, accumulator, and RCS; therefore, the responses are acceptable. Accordingly, **RAI 407-3082, Question 06.03-62, is resolved and closed.**

In **RAI 407-3082, Question 06.03-63 (RAI 06.03.02.02-21),** the staffed asked the applicant if a calculation has been performed to determine the maximum RWSP temperatures during a SBLOCA and LBLOCA prior to, during, and after containment spray and to identify the maximum RWSP temperatures and associated pressures for each of the different phases (prior to spray, during and when the CS/RHR discharges back to RWSP). The applicant was also asked if conservatisms and/or uncertainties were used in calculating the maximum temperatures. The applicant provided the response to **RAI 407-3082, Question 06.03-63**, dated September 28, 2009.

The maximum RWSP temperature analyses were performed for both LBLOCA and SBLOCA. The assumptions for the RWSP temperature analysis and containment maximum pressure analysis, using the GOTHIC computer program were provided. The mass and energy release analysis for the LBLOCA were calculated with the SATAN-VI, WREFLOOD and GOTHIC computer programs. The mass and energy release analysis for the SBLOCA were calculated with the M-RELAP5 computer program.

Since the containment volume was modeled as a single volume, the lag between containment spray initiation and the time when the CS/RHR discharges back to RWSP could not be identified. The calculation results are taken into account for the sump strainer integrity evaluation.

The following conservatisms were incorporated in the calculations to cover uncertainties and to be compliance with related regulations.

1. Assumptions for Maximum RWSP Temperature Analysis

   The model is similar to the containment maximum pressure evaluation in DCD Tier 2 Section 6.2.1 and comparisons between them were provided in the response.

2. Assumptions for Mass and Energy Release Analysis

   Both the LBLOCA and the SBLOCA were considered. For the LBLOCA, the method used is described in MUAP-07012-P-A and NP-A, Revision 2, “LOCA Mass and Energy

Release Analysis Code Applicability Report for US-APWR." For the SBLOCA, there are some changes from the evaluation method described in MUAP-07025-P and NP, Revision 3, "Small Break LOCA Sensitivity Analysis for US-APWR." The comparisons between them were provided in the response.

The response to RAI 407-3082, Question 06.03.63 is acceptable. It shows that the maximum RWSP temperature during postulated SBLOCA and LBLOCA conditions has been conservatively calculated. Accordingly, **RAI 407-3082, Question 06.03-63, is resolved and closed.**

ECC/CS Strainers

Four independent sets of strainers are provided inside the RWSP as part of the ECCS and CSS. ECC/CS strainers are provided to prevent debris from entering the safety systems, which are required to maintain the post-LOCA long-term cooling performance. ECC/CS strainers are designed to be consistent with RG 1.82, as discussed in DCD Tier 2 Section 6.2.2. The RWSP is located at the lowest part of the containment in order to collect containment spray water and blowdown water by gravity. A concrete structure separates the RWSP from the upper containment area. Connecting pipes that drain the collected water from the upper containment are provided in the ceiling of the RWSP. The fully submerged strainers are installed on the bottom floor of the RWSP inside containment at elevation 1.09 m (3 ft, 7 in.). Below the strainers at elevation -1.04 m (-3 ft, 5 in.) is the bottom of the RWSP sumps. DCD Tier 2 Table 6.3-5 presents relevant ECC/CS strainer data. The fully submerged strainers, in combination with the SI pump elevation, provide NPSH to ensure continuous suction availability without cavitation during all postulated events requiring the actuation of the ECCS. The strainer sizing accommodates the estimated amount of debris that could be generated in containment.

The ECC/CS strainer components are part of the ECCS. DCD Tier 2 Figures 6.2.2-8 and 6.2.2-9 show four independent sets of ECC/CS strainers located in the RWSP. The strainer design includes redundancy, and a large screen surface area to account for potential debris blockage and to maintain safety performance. The strainer design is also corrosion resistant and has a strainer hole size to minimize downstream effects.

Following evaluations, including downstream effects were performed in accordance with R.G 1.82 (Revision 3) to assure the ECC operation safely:

1. Break selection (MUAP-08001, "US-APWR Sump Strainer Performance").
2. Debris generation (MUAP-08001, "US-APWR Sump Strainer Performance").
3. Debris characteristics (MUAP-08001, "US-APWR Sump Strainer Performance").
4. Debris head loss (MUAP-08001, "US-APWR Sump Strainer Performance").
5. Net positive suction head of ECC/CS pumps (MUAP-08001, "US-APWR Sump Strainer Performance").
6. Downstream effects (MUAP-08013, "US-APWR Sump Strainer Downstream Effects").
7. Upstream effects (MUAP-08001, "US-APWR Sump Strainer Performance").
8. Chemical effects (MUAP-08001, "US-APWR Sump Strainer Performance", MUAP-08011, "US-APWR Sump Debris Chemical Effects Test Results" and MUAP-08013, "US-APWR Sump Strainer Downstream Effects").
9. Structural Analysis of the strainer (MUAP-08012, "US-APWR Sump Strainer Stress Report").

Additional design attributes are described in the US-APWR Sump Strainer Performance MUAP-08001-P, Revision 7. Staff's evaluation of the sump strainer design is documented in the DCD Chapter 6.2.2 SE. The staff's evaluation of MUAP-08013 is documented in Section 6.3.4.9 of this SE while MUAP-08011 is evaluated in the DCD Section 6.2.2 SE and MUAP-08012 in DCD Section 3 SE.

The containment design basis results for pressure and temperature were presented throughout the DCD. It is not clear how the design basis analyses were performed for debris and chemical generation for plugging of the strainers, and ultimately for the downstream effects on the core. In DCD Tier 2 Section 6.2.1.1.1, the applicant states "Figure 6.2.1-1 through Figure 6.2.1-4 are plots of containment internal pressure and temperature versus time for the most severe primary and secondary system piping failures."

In **RAI 391-2974, Question 06.03-25 (RAI 06.03.02.02.03-2)**, the staff asked the applicant to identify the most severe pipe break with respect to containment pressure and temperature and the worst case pipe break with respect to creating debris that could enter the RWSP and potentially plug the recirculation. In its response to RAI 391-2974, Question 06.03-25, dated July 29, 2009, the applicant stated the various assumptions of accident condition and results for containment analysis are summarized in DCD Tier 2 Tables 6.2.1-6 through 6.2.1-8. The worst case pipe break with respect to creating debris that could enter the RWSP and potentially plug the recirculation is the main coolant pipe break as discussed in Section 3.1 of Technical Report MUAP-08001, "US-APWR Sump Strainer Performance." The response to Question 06.03-25 is acceptable because it provided which pipe breaks were analyzed. The staff evaluated the acceptability of the pipe break selection is documented in the DCD Chapter 6.2.2 SE. Accordingly, **RAI 391-2974, Question 06.03-25, is resolved and closed.**

Instrumentation

DCD Tier 2 Figure 6.3-1 is a simplified flow diagram of the ECCS. DCD Tier 2 Figure 6.3-2 is the piping and instrumentation diagram showing system locations for all components, including system interconnections, instruments, alarms and indications. In DCD Tier 2 Chapter 7, "Instrumentation and Controls," Section 7.3, "Engineered Safety Feature Systems," the applicant discusses the instrumentation and controls, including the actuation logic, the component redundancy, the system interlocks, and the indication for the SIS.

Two wide range and two narrow range level channels are installed on the RWSP. Each channel provides level indication in the MCR and RSC, while two wide range level channels also provide the high, the below normal, and the low level alarms.

During review of Section 6.3.5.4 the staff noted that RWSP temperature instrumentation was not discussed and therefore issued **RAI 407-3082, Question 06.03-71**. In its response to **RAI 407-3082, Question 06.03-71 (RAI 06.03.05.04-1),** dated August 5, 2009, the applicant added the following statement regarding temperature measurement and indication to Section 6.3.5.4, "One temperature channel is installed on the RWSP. This channel provides temperature indication and low temperature alarm in the MCR and RSC." The staff found the DCD revision acceptable as it provided information on RWSP temperature monitoring which supports the RWSP TS temperature range. Accordingly, **RAI 407-3082, Question 06.03-71, is resolved and closed.**

Major Valves

Containment isolation is discussed in DCD Tier 2 Section 6.2.4 and controls (including interlocks) and automatic features of containment isolation valves are discussed in DCD Tier 2 Section 7.3.

There is a normally open motor-operated gate valve in each of the four CS/RHR pump suction lines from the RWSP. These valves would remain open during normal and emergency operations. The valves are remotely closed by operator action from the MCR and RSC only if a CSS had to be isolated from the RWSP to terminate a leak or during RHR cooldown operation when isolation from the RWSP is required. In the pump/valve maintenance mode, these valves are also closed. The open or closed valve position, for these valves, is indicated in the MCR and RSC. The four CS/RHR pump RWSP suction isolation valves (CSS-MOV-001A, B, C, and D) are Equipment Class 2, seismic category I.

These valves are interlocked and are allowed to open only if the two in-series RHR hot leg suction isolation valves are closed.

Operator actions are important in ensuring important equipment is available to mitigate the effects of LOCAs and should be justified, validated and thoroughly described in the DCD. In DCD Tier 2 Section 6.3.2.2.3, the applicant describes the auxiliary RWSP storage tank as being designed to ensure the required volume for refueling operations.

In **RAI 391-2974, Question 06.03-26 (RAI 06.03.02.02.03-3)**, the staff asked the applicant to provide detailed information regarding this tank and the relevant piping/pump system and to describe whether this tank is intended to be used during accident conditions (i.e., LOCAs and AOOs). If the auxiliary RWSP storage tank and associated system could be used during these events, describe the necessary operator actions and the equipment availability during these events.

In its response to RAI 391-2974, Question 06.03-26, dated July 28, 2009, the applicant stated the RWSAT is used as the water source to perform the following functions:

- Supply boric acid water to the refueling canal and the fuel inspection pit during the refueling operation.
- Supply boric acid water to the refueling canal and the cask pit during the carrying out operation for the spent fuel.
- Back up water source for the charging pumps.

The boric acid water in the RWSAT is supplied to refueling canal, fuel inspection pit, and cask pit using the Refueling Water Recirculation Pumps shown in Figure 6.3-7, DCD Tier 2 Section 6.3. The RWSAT has no safety-related function and is not used during an accident. Since pump names in DCD Tier 2 Figure 6.3-7 were not consistent with that in other sections, the pump names were revised as follows:

Refueling Water Recirculation Pump - A
Refueling Water Recirculation Pump – B

In its response to RAI 391-2974, Question 06.03-26, dated July 28, 2009, the applicant confirmed that the RWSAT has no safety-related function and is not used during accident conditions; therefore, the response is acceptable. Accordingly, **RAI 391-2974, Question 06.03-26, is resolved and closed.**

6.3.4.4 NaTB Baskets and NaTB Basket Containers - pH Controls

The ITAAC requirements for the NaTB Baskets is described in DCD Tier 1, 2.4.4, "Emergency Core Cooling System," Table 2.4.4-5, "Emergency Core Cooling System Inspections, Tests, Analyses, and Acceptance Criteria." The design commitment acceptance criterion is 7.c: A report exists and concludes that the as-built NaTB baskets contain a total calculated weight of NaTB of ≥44,100 pounds. The tops of the as-built NaTB baskets are located below plant elevation 131 ft, 6 in.

The TS requirements for the NaTB baskets are provided in TS 3.5, "Emergency Core Cooling Systems (ECCS)," 3.5.5, "pH Adjustment."

The NaTB baskets are located in the Containment. This structure is Seismic Category I and provides tornado missile barriers to protect the ECCS. DCD Tier 2 Section 6.2.1 discusses the containment environmental conditions during accidents, and DCD Tier 2 Chapter 3, Section 3.11 discusses the suitability of equipment for design environmental conditions. Protection of the ECCS from missiles is discussed in DCD Tier 2 Chapter 3, Section 3.5. Protection of the ECCS against dynamic effects associated with the rupture of piping is described in Section 3.6. Protection from flooding is discussed in Section 3.4.

6.3.4.4.1 System Description

NaTB baskets are located in the containment and are capable of maintaining the desired post-accident pH conditions in the recirculation water. The pH adjustment is capable of maintaining containment water pH at least 7.0, to enhance the retention capacity in the containment and to prevent stress corrosion cracking of the austenitic stainless steel components.

Crystalline NaTB additive is stored in the containment and is used to raise the pH of the RWSP from 4.3 to at least 7.0 during post-LOCA conditions. The chemical composition of NaTB is $Na_2B_40_7 \cdot 10\ H_20$. Sodium tetra-borate decahydrate is also known as "borax" and can be written $B_4O_7Na_2 \cdot 10\ H_20$. 84,750 ft^3 (634,000 gallons) of borated water are available in the RWSP to meet LOCA and long-term post-LOCA coolant needs. The RWSP water is borated to approximately 4,000 ppm boric acid, at a pH of approximately 4.3. Crystalline NaTB spray additive is stored in containment and is used to raise the pH of the RWSP water from 4.3 to at least 7.0, post-LOCA. This pH is consistent with the guidance of NRC BTP MTEB-6.1 for the protection of austenitic stainless steel from chloride-induced stress corrosion cracking. DCD Tier 2 Section 6.3.2.2.5 describes the design of NaTB baskets. At a pH of at least 7.0, corrosive attack of stainless steel alloys used in containment will be insignificant. Similarly, post-LOCA hydrogen generation due to material corrosion will be negligible. In addition, the generation of chemical precipitates from aluminum will be minimized.

Twenty three NaTB baskets are divided and installed into three NaTB basket containers. The NaTB Basket Containers are fully immersed and allow the optimum NaTB transfer to the RWSP.

The NaTB basket containers include nine NaTB baskets in Container A, seven NaTB baskets in Container B, and seven NaTB baskets in Container C.

The top face of each container is open to receive spray water from the CSS nozzles during an accident and, after a period-of-time, each container is filled with spray water. The NaTB in baskets is dissolved in spray water in the containers.

The solution containing NaTB is discharged from each container to the RWSP through NaTB solution transfer pipes. DCD Figure 6.3-12 shows the NaTB solution transfer piping. This piping transfers NaTB solution to the RWSP by gravity.

The water that flows into the NaTB baskets along with the dissolved NaTB and the water flow to the RWSP is an important element of the long term cooling of the containment and reactor. The staff does not find these areas to be well documented in the DCD. In DCD Tier 2, Section 6.3.2.2, the applicant states that, "The size of the NaTB transfer pipes are selected to minimize the head loss during a transfer of solution. The containerized NaTB solution overflows at the same flow rate as the spray water that flows into the container. Therefore, the NaTB dissolved in the container flows into the RWSP without losses from spilling over onto the containment operating floor. The dissolution time of the NaTB is approximately 12 hours."

In addition to component integrity, in **RAI 391-2974, Question 06.03-3 (RAI 06.03.01.03-1),** the staff asked if pH control was important from an iodine retention perspective. In its response to **RAI 391-2974, Question 06.03-3,** dated July 28, 2009, the applicant revised the DCD Tier 2, Section 6.3.1.3, wording to indicate that the function of the pH control was to also enhance the iodine retention capacity in the containment recirculation water. The staff evaluation of the adequacy of NaTB baskets to maintain proper pH control is documented in Section 6.5.2 of this SER. **RAI 391-2974, Question 06.03-3, is resolved and closed as discussed in Section 6.5.2.**

In DCD Tier 2, Section 6.3.2.2.5**,** the applicant states that "NaTB in baskets is dissolved in spray water in the containers. The solution containing NaTB is discharged from each container to the RWSP through 0.10-m (4-inch)-diameter NaTB solution transfer pipes."

In **RAI 391-2974, Question 06.03-4 (RAI 06.03.02.02.05-1)**, the staff noted that the flows from the sprays that fall onto the NaTB containers are very difficult to predict accurately, and asked the applicant to provide the calculation information that predicts the dissolution via the sprays, and to identify where it has been verified. During the aging of the plant, many characteristics of the sprays may change slightly because of maintenance activities and other effects that may cause the spray distribution onto the NaTB containers to change. The staff also asked if the calculation considered the change in spray distribution with time, whether the evaluation considered losses of NaTB into the containment environment, and whether the evaluation considered the impact of the amount and time of NaTB dissolution on the pH of water in the containment.

In its response to RAI 391-2974, Question 06.03-4, dated July 28, 2009, the applicant stated DCD Section 6.3, Figure 6.3-10 shows the containment spray pattern on the floor where the NaTB baskets are installed.

One spray pattern circle means that area inside the circle is covered by the designed spray flow from one spray nozzle. The maximum spray flow rate that flows into the NaTB basket container was calculated by using conservatively larger numbers of these spray pattern circles which cover the container. Even if this conservatively estimated spray water flows into the container, the pressure loss in the transfer piping is lower than the difference of elevation between the container and RWSP, that is, the driving force to gravity injection. Therefore, the NaTB solution in the container does not overflow from the container to the outside of RWSP. In addition, the applicant also stated the changes in the spray pattern and spray distribution with time are not considered for the following reasons:

The erosion wastage of passive components such as spray nozzles and spray piping does not occur since they are not used in the plant normal operation. In addition, the corrosion wastage does not occur since the austenitic stainless steel is used for their materials, and they are exposed in the RWSP water, which is controlled water chemistry.

The aging degradation of CS/RHR pump head is previously considered in designing spray flow rate that is in designing the resistance of flow control orifice installed on the spray header.

The staff agrees that under maximum containment spray flows the NaTB baskets will not overflow and that NaTB solution will be delivered to the RWSP via the transfer piping (also see the response to **RAI 695-4934** below). The amount and solution rate of the NaTB used as the input conditions in the pH analysis for the re-circulated water is evaluated in Section 6.5.2 of this SER. Therefore, **RAI 391-2974, Question 06.03-4, is closed, but unresolved.**

In its response to **RAI 391-2974, Question 06.03-4**, dated July 28, 2009, the applicant stated that if the driving head of the flow is greater than the corresponding pressure drop in the piping leading to the RWSP, then the basket will not overflow. In **RAI 695-4934, Question 06.03-88**, a follow-up to RAI 391-2974, Question 06.03-4, the staff requested the applicant provide additional information to support this conclusion. In addition that staff requested the applicant address the potential for debris accumulation in the basket container, the potential for overflow from the basket container and to support the claim that the NaTB baskets would be fully submerged during minimum flow conditions for the containment spray ring D.

In its response to **RAI 695-4934, Question 06.03-88, item (a)(1)**, dated May 31, 2012, the applicant provided RAI Figure 6.3.88-1, “NaTB Solution Transfer Piping Diagram,” to show the flow paths from the NaTB basket containers A, B, and C and the RWSP, with their elevations, to address the staff request for a detailed description, including elevations, of the 6” transfer flow path and where it combines with the flow paths from the other basket container. RAI Figure 6.3.88-1 also showed the flow rate in each path that was used to evaluate the piping pressure loss, as provided in the response to item (b) of the request. Thus, the applicant provided the requested information.

In response to **RAI 695-4934, Question 06.03-88, Item (a)(2)**, dated May 31, 2012, the applicant provided RAI Figure 6.3.88-2, “NaTB Solution Flow Path,” to show the piping inlets between the NaTB baskets in the container, to address the staff request concerning the potential for debris accumulation in the basket container, for use to demonstrate that debris cannot accumulate in the lower part of the basket container and block the flow to the RWSP. There are no obstacles above the piping inlet at the bottom of container. Therefore, debris flows into the pipe inlet with the NaTB solution and is discharged into the RWSP without blocking the connection to the transfer pipe. Thus, the applicant demonstrated that debris will not accumulate in the lower part of the basket container, by design.

In response to **RAI 695-4934, Question 06.03-88, items (b) and (c)**, dated May 31, 2012, the applicant provided a head loss evaluation for the NaTB flow paths, to address the staff request concerning the potential for overflow from the basket container, to demonstrate that under maximum containment spray conditions the flow into the NaTB basket container will not be large enough to cause overflow from the basket into containment and not directly back to the RWSP.

The following assumptions were used for the head loss evaluation:

(1) The flow rate into each NaTB basket container was estimated based on the spray pattern shown in DCD Tier 2 Figure 6.3-10, “Containment Spray Pattern Plan View at the NaTB Basket Installation Level.” The following assumptions are made to estimate a conservative maximum spray water flow rate:

a. All spray water from one spray nozzle flows into the container, even though only partial flow from the spray nozzle in spray ring C covers the container.
b. Four CS/RHR pumps are in operation, to maximize the flow.
c. The spray water flow into the refueling cavity was obtained by multiplying all spray water by the ratio of the refueling cavity opening and the containment cross sectional area and an additional multiplier, as a safety margin, to conservatively the increase flow rate.

(2) The piping pressure loss was calculated using the Darcy's equation from the Crane Technical Paper No. 410M, page 3-4.

(3) The gravitational driving head from the A, B, and C containers to the RWSP was evaluated as the static head differential between the elevation of the free surface in the container and the elevation of transfer pipe outlet end. The elevation of the free surface is the elevation of the horizontal region of the inverse U-pipe attached to the container outlet header.

RAI Table 6.3.88-3, “Comparison of Head Loss and Driving Head,” showed the comparison of the head loss of transfer pipe from the A, B, and C containers to RWSP and the gravitational driving head. The head loss in the transfer line from each container to RWSP is smaller than the driving head; therefore, the spray water would not overflow from the container, even though a conservative maximum flow rate is assumed. The applicant demonstrated that the spray water flow rate will not be high enough to result in overflow from the basket into containment and not directly back to the RWSP

In response to item (d) of the request, the applicant provided a discussion to supports the claim that the NaTB baskets would be fully submerged during minimum flow conditions for the containment spray ring D.

The applicant first noted that the height of NaTB basket is 3 ft -11 in, as shown in RAI Figure 6.3.88-3, and not 4 ft -11 in as shown in DCD Tier 2 Figure 6.3-11, “Containment Spray Pattern Sectional View at the NaTB Basket Installation Level.” The applicant also stated DCD Tier 2 Figure 6.3-11 would be revised to include the correct value.

The stated revision was not included in DCD Tier 2, Revision 3. **The DCD Tier 2 revision, stated in response to RAI 391-2974, Question 06.03-4, is being tracked as a Confirmatory Item.**

The applicant provided RAI Figure 6.3.88-4 to show that the transfer piping exits from the bottom of the container and rises to the horizontal portion of the U-pipe. The bottom of the U-pipe is located at an elevation higher than the top of the basket; the U-pipe is 8.625 inches in diameter; and the top of the U-pipe is approximately four inches below the top of the basket container. Spray water flowing into the container need to overflow the bottom of the U-pipe to be transferred to the RWSP. Therefore, the water level in the container is maintained at a level

4 inches higher than the top of the basket even with the minimum flow conditions for the containment spray ring D, and the baskets are completely submerged in the spray water. The applicant has demonstrated that the NaTB baskets will be fully submerged during minimum flow conditions for the containment spray ring D. The response to **RAI 695-4934, Question 06.03-88,** addressed the staff issues, and therefore, **RAI 695-4934, Question 06.03-88, is resolved and closed**.

In **RAI 391-2974, Question 06.03-23 (RAI 06.03.02.02.05-2)**, the staff asked the applicant whether there are postulated LOCA breaks that could cause debris to block the four inch lines. If so, provide the evaluation results and the associated technical references. In its response to RAI 391-2974, Question 06.03-23, dated July 28, 2009, the applicant stated there are no credible postulated LOCA breaks that could cause debris to block the four inch lines of NaTB baskets, and provided the followings clarifications:

- The postulated pipe break considered for debris blockage in the four inch lines of NaTB baskets is a primary side pipe break. There is no need to consider secondary side (i.e., main steam pipe or feed water pipe as the CS operation time for the containment pressure reduction is very short for secondary side accidents) breaks for the impact on the basket's function.
- The location of the postulated primary side pipe breaks are in the lower portions of containment inside secondary shield walls which are far from the NaTB baskets location.
- Since layers of gratings are provided inside secondary shield walls, only small/fine debris that could pass though grating is possibly blown up to containment atmosphere over the NaTB baskets.
- Therefore, there is no large debris during a LOCA which could potentially blocks the four inch lines of NaTB basket.

Based on the difference between the RCS piping and NaTB basket elevations the staff agrees that the only small/fine debris would reach the baskets. This debris would either settle in the baskets or be transported down the transfer pipes to the RWSP. Therefore, **RAI 391-2974, Question 06.03-23, is resolved and closed**.

6.3.4.5 Performance Evaluation

6.3.4.5.1 System Description

The applicant, in DCD Tier 2 Chapter 15, presents a discussion and analysis of plant AOOs, transients and PAs, and DCD Tier 2 Chapter 19 presents a PRA of more severe and even less likely accidents. In DCD Tier 2 Chapter 15 and DCD Tier 2 Section 6.2.1, the applicant describes accident analysis results that include the effects of ECCS operation. The specific events described in DCD Tier 2 Chapter 15 where the ECCS may be actuated are described in this section. In DCD Tier 2 Section 6.2.1, the applicant describes analyses that calculate maximum containment pressure and temperature from postulated accidents that release high-energy fluids into the containment. The information in DCD Tier 2 Chapter 15 and in DCD Tier 2 Section 6.2.1 indicates that the acceptance criteria are met for all events that rely on ECCS mitigation. Meeting these acceptance criteria demonstrates that the performance of the ECCS is adequate and therefore the ECCS design is acceptable. Events where actuation of the ECCS may be necessary are categorized and identified in DCD Tier 2 Tier 2 Section 6.3.3 and below:

A. Increase in Heat Removal by the Secondary System

Category A Events are non-LOCA events in which the primary protection is provided by regular monitoring of critical parameters, such as the SG level and the main steam flow from the MCR. These postulated transients could cause an automatic trip of the reactor through the Reactor Protection System. ECCS actuation would be caused by a low RCS pressure or by a high containment pressure.

A.i. Inadvertent opening of steam generator relief or safety valve

This event is an AOO. In DCD Tier 2 Section 15.1.4, the applicant provides a detailed description of the event and its results. Inadvertent opening of a steam generator relief, steam generator safety, or turbine bypass valve can cause a rapid increase in steam flow and a depressurization of the secondary system. The energy removed from the RCS by this event is sufficient to cause the RCS pressure to initiate the ECCS on low pressurizer pressure. However, the RCS pressure does not decrease below the accumulator charge pressure; therefore, the accumulators are not credited in the analysis. Only two pumps operate to inject borated water from the RWSP into the reactor vessel downcomer. This scenario is consistent with the most severe single active failure. If such a failure occurs, the remaining trains provide the functions credited in this analysis.

The time required for borated water to reach the core is determined by taking into consideration: (1) the period from the time the ECCS actuation signal is generated to the time the safety injection pumps reach full speed, and (2) the transport time for the injected water to pass through the reactor coolant piping. These delays and purge volumes are directly modeled in the MARVEL-M code. The time sequence of the event is provided in DCD Tier 2 Table 15.1.4-1, "Time Sequence of Events for Inadvertent Opening of a Steam Generator Relief or Safety Valve."

The analysis shows that the departure from nucleate boiling ratio (DNBR) remains well above the 95/95 limit. Thus, the fuel cladding temperature would not increase significantly during this transient. For this event, the RCS pressure does not challenge the RCS design pressure. Similarly, the main steam system pressure does not challenge the design pressure for the main steam system.

The staff found the discussion presented in this DCD Tier 2 section was not clear and requested the applicant make it clear that the radiological consequences are bounded by the DCD Tier 2 Section 15.1.5, "Steam System Piping Failures Inside and Outside of Containment," event.

In its response to **RAI 407-3082, Question 06.03-66 (RAI 06.03.03-3),** dated August 5, 2009, the applicant modified the DCD to state "The radiological doses for this event are described in Section 15.1.4.5 and are bounded by the Section 15.1.5 event. The response provided the clarification sought and is therefore acceptable. Accordingly, **RAI 407-3082, Question 06.03-66, is resolved and closed.**

A.ii. Steam system piping breaks inside and outside of containment

This event is a PA. In DCD Tier 2 Section 15.1.5, the applicant provides a detailed description of the event and its results. This event encompasses a spectrum of steam system piping failure sizes and locations from both power operation and hot zero power initial conditions. If the break occurs inside the containment volume, containment pressure signals are available to actuate ECCS and containment heat removal systems.

These signals and the containment systems are not used in the core response analysis presented in this section. RCS pressure decreases below the shutoff head of the SIS, resulting in the addition of borated water to the RCS. The RCS pressure does not decrease below the accumulator discharge pressure; therefore, the accumulators are not credited in the analysis.

The limiting single failure for the event initiated from hot shutdown conditions is the failure of one ECCS train. Two of the remaining trains are assumed to operate to provide the SI functions credited in this analysis. When the steam pressure in the faulted steam generator falls below the Low Main Steam Line Pressure setpoint (in any loop), the ECCS is actuated and the main steam isolation valves are closed. The ECCS signal also actuates EFWS and feedwater isolation to isolate the SGs from each other.

Only two SI trains are assumed to operate to inject borated water into the reactor vessel. The time required for borated water to reach the core is determined by taking into consideration: (1) the period from the time the ECCS actuation signal is generated to the time the safety injection pumps reach full speed and (2) the transport time for the injected water to pass through the reactor coolant piping. The time for the SI pumps to reach full speed includes time for the emergency GTGs to start for the case where offsite power is not available. ECCS signal delays, backup power start delays, and safety injection piping and purge volumes are modeled by the MARVEL-M code. The time sequence of the event is provided in DCD Tier 2 Table 15.1.5-1.

The analysis shows that the minimum DNBR remains above the 95/95 limit. Thus, the fuel cladding temperature would not increase significantly during this transient. The radiological doses are less than the guideline value of 10 CFR 50.34 and 10 percent of guideline value of 10 CFR 50.34, respectively.

The staff requested the applicant clarify the radiological consequences for this event. In its response to **RAI 407-3082, Question 06.03-67 (RAI 06.03.03-4),** dated August 5, 2009, the applicant modified the DCD to state: “The radiological doses for this event are described in Section 15.1.5.5.” This response provides the requested clarification and is therefore acceptable. Accordingly, **RAI 407-3082, Question 06.03-67, is resolved and closed.**

B. Decrease in Reactor Coolant Inventory

Category B events are LOCAs. ECCS actuation would generally be initiated by low pressurizer pressure or high containment pressure. However, it is possible that a SBLOCA with an extremely small break flow area would not result in automatic ECCS actuation.

B.i. LOCA resulting from a spectrum of postulated piping breaks within the RCPB

In DCD Tier 2 Section 15.6.5, “Loss-of-Coolant Accidents Resulting from Spectrum of Postulated Piping Breaks within the Reactor Coolant Pressure Boundary,” the applicant provides a detailed description of the large and small break analysis and results. LOCAs are accidents that would result from the loss of reactor coolant, at a rate in excess of the capability of the reactor coolant makeup system, from breaks in pipes in the reactor coolant pressure boundary up to and including a break equivalent in size to the double-ended rupture of the largest pipe in the RCS.

For this accident, the ECCS is actuated by the ECCS actuation signal due to high containment pressure. The accumulators discharge, followed by actuation of the SI pumps, and deliver borated water to the core.

Following completion of core reflood (large break) or core recovery (small break), the ECCS continues to supply borated water to the RCS for long-term cooling. In the SBLOCA, the RCS pressure may not fall below the injection pressure for the accumulators, depending on the break size. In this case, the SIS system solely provides the core reflooding function. In the event of a small break, a slow depressurization of the RCS would occur. The low RCS (pressurizer) pressure signal causes a reactor trip. A LOOP following the reactor trip is assumed in the analysis. Turbine and the RCP would trip accordingly.

The ECCS actuation signal causes the high head injection system to inject borated water to the core. With the ECCS injection, only the upper part of the core is uncovered. But then the core is recovered in a short period for the SBLOCA.

In the event of a LBLOCA, a rapid depressurization of the RCS occurs. The accumulators and the SI pumps inject borated water. The accumulators supply a large injection flow rate initially to refill the reactor vessel downcomer. The accumulator injection flow rate is then automatically switched to the small injection flow rate mode, once the accumulator water level decreases below a specified value. The SI pumps directly inject borated water from the RWSP to the reactor vessel downcomer through the DVI nozzles. As the injection flow from the SI pumps increases, the RCS pressure decreases and approaches the containment backpressure.

The calculated results for the event are presented in DCD Tier 2 Table 15.6.5-8 (Large Break) and DCD Tier 2 Table 15.6.5-10, 12, 14 (Small Break). The time sequence of the event is provided in DCD Tier 2 Table 15.6.5-6 (Large Break) and DCD Tier 2 Table 15.6.5-9, 11, 13 (Small Break).

The results of the LOCA analyses demonstrate that the acceptance criteria of 10 CFR 50.46 are satisfied. The peak containment pressure has been shown to be below the containment design pressure. The EAB and low population doses have been shown to meet 10 CFR 50.34 dose guidelines. The dose for the MCR personnel has been shown to meet the dose criteria given in GDC 19.

To clearly understand the description of the LOCA events presented in the DCD, the staff requested the applicant to provide detailed diagrams that identify elevations, flows, pumping capacities, etc. These diagrams should provide enough information along with applicable plots to adequately interpret the LOCA events.

In **RAI 407-3082, Question 06.03-70 (RAI 06.03.03-7),** the staff asked the applicant to provide Section and Plan diagrams showing locations, flows, and elevations of the important events, which include break location, ECCS injection locations and flow with respect to time for both small break and LBLOCAs.

In its response to **RAI 407-3082, Question 06.03-70**, dated August 5, 2009, the applicant provided the requested information in Figure 6.3.70-1, “Section View for Identifying SBLOCA Flow Diagram,” Figure 6.3.70-2, “Core Entrance and Exit Liquid Mass Flowrates for 7.5-inch Break,” Figure 6.3.70-3, “Core Entrance and Exit Vapor Mass Flowrates for 7.5-inch Break,” Figure 6.3.70-4, “Reactor Vessel Exit Mass Flowrate for 7.5-inch Break,” Figure 6.3.70-5, “Reactor Vessel Exit Mass Flowrate for 7.5-inch Break,” Figure 6.3.70-6, “Intact Loop SG U-

Tube Top Liquid Mass Flowrate for 7.5-inch Break," Figure 6.3.70-7, "Liquid and Vapor Discharges through the Break for 7.5-inch Break," Figure 6.3.70-8, "Accumulator and Safety Injection Mass Flowrates for 7.5-inch Break," Figure 6.3.70-9, "Core/Upper Plenum Collapsed Level for 7.5-inch Break," Figure 6.3.70-10, "SG U-Tube Top to Crossover Leg Bottom Collapsed Liquid Level for 7.5-inch Break," and Figure 6.3.70-11, "PCT at ALL Elevations for Hot Rod in Hot Assembly for 7.5-inch Break." The response provided sufficient information for the staff's evaluation and is therefore acceptable. Staff's evaluation of the US-APWR design to various LOCA break sizes is provided in the Chapter 15 (Section 15.6.5) SE. Accordingly, **RAI 407-3082, Question 06.03-70, is resolved and closed.**

The staff also identified two apparent inconsistencies in DCD Tier 2 Section 15 Table 15.6.5-1, "US-APWR Major Plant Parameter Inputs Used in the Best-Estimate Large-Break LOCA Analysis."

In its response to **RAI 391-2974, Question 06.03-55 (RAI 06.03.03.01-2),** dated July 27, 2009, the applicant addresses the discrepancy between the information provided in Table 15.6.5-1 and TS 3.5.4: the SI temperature range used in the LBLOCA analysis is 45 °F (7 ^{o}C) ≤ T_{SI} ≤ 120 °F (49 ^{o}C), while the RWSP TS SR 3.5.4.1 range is ≥ 32 °F (0 ^{o}C) and ≤ 120 °F (49 ^{o}C). The applicable range of the thermodynamic properties in WCOBRA/TRAC (M1.0) (T. Suzuta, et al., "Large Break LOCA Code Applicability Report for US-APWR," MUAP-07011-P, Revision 0, July 2007) used for the LBLOCA analysis is ≥ 280 K (44.33 °F), therefore the 45 °F (7 ^{o}C) value was used in the analysis. **RAI 391-2974, Question 06.03-55, is closed, but unresolved.**

In **RAI 695-4934, Question 06.03-94**, a follow-up to RAI 391-2974, Question 06.03-55, the staff requested that the applicant justify why the LBLOCA analysis range only goes down to 45 ºF (7 $^{\circ}$C) while T.S. SR 3.5.4.1 states a minimum temperature of 32 ºF (0 $^{\circ}$C) and to discuss the effects of low temperature on the boron concentration in the RWSP. In its response to **RAI 695-4934, Question 06.03-94**, dated March 17, 2011, the applicant provided the requested justification.

The RWSP water temperature does not have an impact on the PCT in the LBLOCA analysis because SI is conservatively assumed to start after the PCT occurs. The effects of the SI water temperature on LMO and CWO are expected to be small, and a higher SI water temperature provides more conservative results. In addition, the minimum containment pressure boundary condition in the LBLOCA analysis is evaluated with the RWSP water temperature at 32 ºF (0 ^{o}C). Therefore, the RWSP temperature range in TS SR 3.5.4.1 does not need to be revised. The applicant also stated that boron precipitation does not occur in the RWSP even at a lower temperature limit of 32 ºF (0 ^{o}C).

The staff agrees with the applicant's justification for the temperature used in the safety analysis. The current TS LCO is therefore also acceptable. Therefore the response to **RAI 695-4934, Question 06.03-94, is acceptable, and the RAI question is resolved and closed**.

In **RAI 391-2974, Question 06.03-56 (RAI 06.03.03.01-3),** the applicant was asked to justify the use of a core power less than 100 percent of rated power (98 percent ≤ Pcore ≤ 102 percent of 4451 MWt) in the ASTRUM LBLOCA analyses, as original shown in DCD Tier 2, Table15.6.5-1, "US-APWR Major Plant Parameter Inputs Used in the Best-Estimate Large break LOCA Analysis." Since that time, as now shown in DCD Tier 3, Revision 3, the rate power is now 100 percent ≤ Pcore ≤ 102 percent of 4451 MWt. Since the applicant now uses the acceptable rated power range in ASTRUM **the response to RAI 391-2974 Question 06.03-56 (RAI 06.03.03.01-3) is no longer necessary** and hence is closed and resolved.

B.ii. Radiological consequences of a steam generator tube failure

This event is a PA. In DCD Tier 2 Section 15.6.3, "Radiological Consequences of Steam Generator Tube Failure," the applicant provides a detailed description of the steam generator tube failure analysis and results.

In the SG tube failure event, the complete severance of a single SG tube is assumed. The event is assumed to take place at full power with the reactor coolant contaminated with fission products, corresponding to continuous operation with a limited number of defective fuel rods. The event leads to leakage of radioactive coolant from the RCS to the secondary system. If the pressurizer pressure decreases below the pressurizer pressure low set point, ECCS is actuated. The ECCS signal starts the SI pumps and also trips the RCPs, which coast down to natural circulation conditions. In addition, an ECCS actuation signal provides feedwater isolation by automatically tripping the main feedwater pumps and fully closing all control valves and feedwater isolation valves in the feedwater system. The core makeup from the borated SI flow (from the RWSP) provides the heat sink to remove decay heat from the reactor.

The radiological doses are less than the guideline value of 10 CFR 50.34 and 10 percent of guideline value of 10 CFR 50.34, respectively.

In its response to **RAI 407-3082, Question 06.03-68 (RAI 06.03.03-5),** dated August 5, 2009, the applicant clarified the radiological consequences discussion in DCD Tier 2 Section 6.3.3 to state that the radiological doses for this event are described in DCD Tier 2 Section 15.6.3.5 and that the time sequence of the event is provided in Table 15.6.3-1, "Time Sequence of Events for Steam Generator Tube Rupture - Radiological Dose Evaluation Input Analysis." The DCD revision is what the staff requested; hence, the response is acceptable. Accordingly **RAI 407-3082, Question 06.03-68, is resolved and closed.**

B.iii. Spectrum of rod ejection accidents

This event is a PA. A rod ejection accident also causes a SBLOCA. In DCD Tier 2 Section 15.4.8, the applicant provides a detailed description of the rod ejection analysis and results.

This accident is defined as the mechanical failure of a control rod drive mechanism pressure housing, which results in the ejection of a rod cluster control assembly (RCCA) and its drive shaft.

The consequence of this RCCA ejection is a rapid positive reactivity insertion with an increase of core power peaking, possibly leading to localized fuel rod failure. The event is analyzed under a spectrum of power levels. The time sequence of the event is provided in DCD Tier 2 Table 15.4.8-1, "Time Sequence of Events for Rod Ejection."

The RCS pressure remains well below 110 percent of the system design pressure, so the integrity of the reactor coolant pressure boundary is maintained. By meeting this criterion, the peak reactor coolant pressure also remains less than the "Service Limit C" stipulated by the ASME code. Radiological consequence is less than 25 percent of the dose guideline of 25 rem TEDE stipulated by 10 CFR 50.34.

In its response **to RAI 407-3082, Question 06.03-69 (RAI 06.03.03-6),** dated August 5, 2009, the applicant modified the DCD to state: "The radiological doses for this event are described in Section 15.4.8.5" as requested by the staff. The response is therefore acceptable. Accordingly **RAI 407-3082, Question 06.03-69, is resolved and closed.**

Operational Restrictions

In DCD Tier 2 Chapter 16, "Technical Specifications," the applicant provides system and component operating restrictions in the form of LCOs. Each LCO specifies the minimum capacities, concentrations, components, or trains and relies on redundancy to account for the component and subsystem unavailability (e.g., maintenance). The staff finds that the required test frequency and acceptance criteria to demonstrate operability are provided.

6.3.4.5.2 ECCS Performance Criteria

In DCD Tier 2 Chapter 16, "Technical Specifications," the applicant specifies the ECCS performance criteria. TS Acceptance Criteria ensure that the relevant system data (e.g., tank levels, boron concentration, flow rate, pressure) are collected, reviewed, and approved. The TS Bases section provides supporting information and rationale for each specification. The staff finds that DCD Tier 2 Chapter 15 presents relevant ECCS performance criteria.

6.3.4.5.3 Single Failure Considerations

The ECCS is designed with redundancy so that the specified safety functions are performed assuming a single failure of an active component for a short-term following an accident, and assuming either a single failure of an active component or a single failure of a passive component for a long-term following an accident. The ECCS consists of four trains. The accumulator capacity is sized such that one of four accumulators is expected to flow out of the break, with no contribution to the core re-flood. Two of four SI pump trains are required to mitigate the consequences of a LBLOCA. One train is assumed to be out of service for maintenance and one train is assumed to fail upon initiation of the SI signal. The ECCS performance, with assumed single failures, is evaluated based on the failure modes and effects analysis presented in DCD Tier 2 Table 6.3-6, "Failure Modes and Effects Analysis - Safety Injection System."

6.3.4.5.4 ECCS Flow Performance

The applicant includes a process flow diagram for the ECCS in DCD Tier 2 Figures 6.3-13 and 6.3-14. SI pump flow performance requirements are provided in DCD Tier 2 Figure 6.3-4, "Safety Injection Pump Performance Flow Requirement."

High head SI flow characteristics for minimum and maximum safeguards are provided for the system in DCD Tier 2 Figures 6.3-15, "High Head Safety Injection Flow Characteristic Curve (Minimum Safeguards)," and 6.3-16, "High Head Safety Injection Flow Characteristic Curve (Maximum Safeguards)." The applicant stated these curves were reproduced in DCD Tier 2 Figure 15.6.5-17, "Accumulator and Safety Injection Mass Flowrates for 7.5-inch Small Break LOCA," 26, "Accumulator and Safety Injection Mass Flowrates for 1-ft^2 Small Break LOCA," and 35, "Accumulator and Safety Injection Mass Flowrates for DVI-line Small Break LOCA," for the SBLOCA and in DCD Tier 2 Figure 15.6.5-7, "Accumulators and SI System Flowrates to DVI-1 and -2 for Large Break LOCA (Reference Case)," for the LBLOCA reference case, however the DCD Tier 2 Section 15.6.5 Figures provide the accumulator and safety injection mass flow rates for the specific breaks, not the safety injection flow characteristics. In its response to **RAI 407-3082, Question 06.03-65 (RAI 06.03.03-1),** dated August 5, 2009, the applicant modified the statement in DCD Tier 2, Section 6.3.3.4, to “These curves are used for the basis to evaluate the safety injection flowrate during small-break and large-break LOCAs, which are shown in Figures 15.6.5-17, 26 and 35 for the small-break LOCA and in Figure 15.6.5-7 for the large-break LOCA reference case." The response is acceptable because it revised the DCD to correctly describe the relevant Chapter 15 figures. Accordingly **RAI 407-3082, Question 06.03-65, is resolved and closed.**

The time sequences for ECCS operation, including its subsystems are presented in DCD Tier 2 Chapter 15 and Section 6.2.1. The pH of the RWSP increases when the NaTB baskets are wetted by the containment spray following a LOCA. DCD Tier 2 Section 6.5.2 contains a description of pH adjustment in the RWSP. DCD Tier 2 Section 6.2.1 also shows the initiation of the CSS. Boron precipitation in the reactor vessel is prevented by manually realigning the SIS to shift the RCS injection from the DVI line to the hot leg injection line at approximately four hours after a LOCA event.

6.3.4.5.6 Use of Dual-Function Components for ECCS

As discussed in DCD Tier 2 Section 6.3.2.2.4 above, the ECC/CS strainers are shared with the CSS. The suction pipes inside the sump pit distribute water from the RWSP to SIS, CSS, and RHRS.

The SI minimum flow and full-flow test line returns to the RWSP. The minimum flow and full-flow test line is shared with the test line piping and the CS/RHR full-flow test line piping prior to discharging into the RWSP.

The hot leg injection line is shared with the suction line to the CS/RHR pumps and emergency letdown lines. The cold leg injection line from the accumulators is shared with the return line from the RHRS.

6.3.4.5.7 Limits on Emergency Core Cooling Systems Parameters

In DCD Tier 2 Chapter 16, “Technical Specifications,” the applicant provides operating restrictions in the form of LCO. Each LCO accounts for a component or subsystem unavailability (e.g., maintenance or testing), and includes the term “operable” to account for related items such as electrical power sources, ventilation, valve lineups, and instrumentation. Acceptance criteria verify that the system data (e.g., tank levels, boron concentration, flow rate, pressure) is collected, reviewed, and approved. The staff finds that the TS Bases section provides supporting information and rationale for the applicable LCOs.

6.3.4.6 TMI Action Items, USIs, GSIs and GLs and Bulletins

6.3.4.6.1 TMI Action Plan Items

DCD Tier 2 Table 6.3-1, “Response of US-APWR to TMI Action Plan,” provides the applicant’s evaluation of the US-APWR to the TMI Action Plan items identified in RG 1.206, “Combined License Applications for Nuclear Power Plants (LWR Edition),” C.I.6.3, “Emergency Core Cooling System.”

The staff requested additional information related to two of the items to complete the issue descriptions. In its response to **RAI 391-2974 Question 06.03-5 (RAI 06.03.01.04-1),** dated July 27, 2009, the applicant added "N/A" to the column US-APWR Design in DCD Tier 2 Table 6.3-1, “Response to US-APWR to TMI Action Plan,” Item II.K.3.15, since the issue is for BWRs. In its response to **RAI 391-2974 Question 06.03-6 (RAI 06.03.01.04-2),** dated July 27, 2009, the applicant modified the issue description for US-APWR to TMI Action Plan, Item II.K.3.45 to include “…the possibility of exceeding vessel integrity limits during rapid cooldown for BWRs….," as this was omitted from the description.

The staff requested the applicant address four additional TMI Action Plan items, identified in SRP 6.3, “Emergency Core Cooling System.” In its response to **RAI 391-2974 Question 06.03-7 (RAI 06.03.01.04-3),** dated July 27, 2009, the applicant revised DCD Tier 2 Table 6.3-1 to include the US-APWR assessment for the following TMI Action Items:

1. II.K.3.16 of NUREG-0737, with regard to providing an evaluation of methods to reduce challenges and failures of RCS relief valves for BWRs.

2. II.K.3.24 of NUREG-0737, with respect to the adequacy of space cooling for long term operation of high-pressure coolant injection (HPCI) and reactor core isolation cooling (RCIC) systems for BWRs to maintain the operating environment within allowable limits.

3. II.D.3 of NUREG-0737, with respect to the requirements that RCS relief and safety valves be provided with a positive indication in the control room of flow in the discharge pipe.

4. II.F.2 of NUREG-0737, with respect to the requirement that instrumentation or controls provide an unambiguous, easy-to-interpret indication of inadequate core cooling.

The responses to **RAI 391-2974, Questions 06.03-5, 06.03-6, and 06.03-7,** dated July 28, 2009, corrected DCD Table 6.3-1 as requested; hence the responses are acceptable. Accordingly, **RAI 391-2974, Questions 06.03-5, 06.03-6, and 06.03-7, are resolved and closed.**

6.3.4.6.2 Unresolved Safety Issues

DCD Tier 2 Table 6.3-2, “Response of US-APWR to Unresolved Safety Issues,” provides the applicant’s evaluation of the US-APWR to the Unresolved Safety Issues identified in RG 1.206, “Combined License Applications for Nuclear Power Plants (LWR Edition),” C.I.6.3, “Emergency Core Cooling System.”

In its response to **RAI 391-2974, Question 06.03-8 (RAI 06.03.01.04-4),** dated July 27, 2009, the applicant modified the DCD Tier 2, Table 6.3-2 text for the evaluation of USI A-1, “Water Hammer,” to be consistent with the material provided in the DCD Tier 2 Section 6.3.2.1.1. The text under GL 86-07 was similarly revised.

In its response to RAI 391-2974, Question 06.03-8, dated July 28, 2009, the applicant discussed the full-flow testing requirements in its response to. Periodic in-service full-flow testing is conducted once every three months. TS requires that the frequency of the SI pump testing is specified in accordance with the "Inservice Testing Program."

The frequency of the periodic SI pump testing is in accordance with the ASME OM Code test requirements, as presented in DCD Tier 2 Chapter 3, Section 3.9, Table 3.9-13. “Pump IST (Sheet 1 of 7).” The response to RAI 391-2974, Question 06.03-8 revised DCD Table 6.3-2 as requested and provided the requested information regarding testing; therefore, it is acceptable. Accordingly, **RAI 391-2974 Question 06.03-8, is resolved and closed.**

The staff requested the applicant provide clarification of the use of the leak before break concept in the DCD Tier 2 Table 6.3-2 discussion for USI A-2, “Asymmetric Blowdown Loads on Reactor Primary Coolant Systems.” In its response to **RAI 391-2974 Question 06.03-9 (RAI 06.03.01.04-5),** dated July 27, 2009, the applicant provided its interpretation on the use of leak-before-break (LBB) as described in SRP 3.6.3, “Leak-Before-Break Evaluation Procedures.” **RAI 391-2974, Question 06.03-9, is closed, but unresolved** as it is addressed in the staff’s DCD 3.6.3 SE.

The implication of SRP 3.6.3, Acceptance Criteria 3 is that LBB concept is applicable not only to individual welded joints or those postulated pipe rupture locations determined from SRP Section 3.6.2 but also an entire piping system or analyzable portion thereof. LBB evaluation results of reactor coolant loop (RCL) piping are described in Technical Report MUAP-09010, Revision1, "Summary of Stress Analysis Results for Reactor Coolant Loop Piping", and LBB concept is applied to RCL piping. Therefore, the applicant concluded and the staff agreed that asymmetric blowdown loads need only be evaluated for breaks determined using LBB methodology. The LBB methodology evaluation of MUAP-09010 was evaluated in the phase 2 DCD 3.6.3 staff’s SE of June 26, 2012, and found to be acceptable.

6.3.4.6.3 Generic Safety Issues

DCD Tier 2 Table 6.3-3, “Response of US-APWR to Generic Safety Issues,” provides the applicant’s evaluation of the US-APWR to the GSIs identified in RG 1.206, “Combined License Applications for Nuclear Power Plants (LWR Edition),” C.I.6.3, “Emergency Core Cooling System.”

The staff requested the applicant provide further clarification concerning the integrity of the RHR system in the DCD Tier 2 Table 6.3-3 discussion for GSI-105, “Interfacing System LOCA at LWRS.” In its response to **RAI 391-2974, Question 06.03-10 (RAI 06.03.01.04-6),** dated July 27, 2009, the RHR design pressure was revised from “900 lb" to “900 psig,” to clarify the pressure units. The response corrected the DCD and is acceptable. Therefore, **RAI 391-2974, Question 06.03-10, is resolved and closed.**

In its response to **RAI 391-2974, Question 06.03-11 (RAI 06.03.01.04-7),** dated July 27, 2009, the applicant addressed the staff concern regarding the statement in DCD Tier 2 Table 6.3-3, “Response of US-APWR to Generic Safety Issues,” under GL-105 which states the RHR is designed to discharge the RCS inventory to the in containment RWSP if both motor operated valve should open during normal operations, since the RHR is designed to 900 psig (6.2 MPa) and the RCS operates at about 2250 psig (15.5 MPa). The applicant pointed to DCD Tier 2 5.4.7, “Residual Heat Removal System,” Section 5.4.7.1, Design Bases,” and 19.2, “Severe Accident Evaluation,” Section19.2.2.5, “Intersystem Loss-of-Coolant Accident,” for the discussions on preventing an intersystem LOCA. In summary, lines connected to the RCS have redundant isolation valves to prevent the RHRS from being exposed to RCS pressure during full power operation. Relief valves are installed to prevent over-pressurizing the RHRS if the isolation valves should leak. The relief valves are set at 470 psig (3.2 MPa). Any flow through the relief valves is directed to the in-containment RWSP. In addition, the RHRS is designed not to fail from over-pressure even if a large internal leak occurs in the redundant isolation valves. The applicant further stated the wall thickness of a 900 psi (6.2 MPa) designed pipe is sufficient to prevent pipe rupture if exposed to the RCS pressure of 2250 psi (15.5 MPa). For example, typically a schedule 80 pipe is selected for the 900 psi (6.2 MPa) design piping in the US-APWR.

For 10-inch schedule 80 piping, the piping failure pressure is above 5,000 psi (34 MPa), as shown in Table 2-18 of NUREG/CR-5603, "Pressure-Dependent Fragilities for Piping Component." Therefore, the applicant concluded this design is effective against an intersystem LOCA. The staff concludes the redundant isolation valves provide the required protection against an intersystem LOCA. Relief valves protect the CS/RHRS piping from overpressure failure. Further the applicant has shown the risk of piping failure should both isolation valves leak is low based on NUREG/CR-5603. Therefore the response to **RAI 391-2974 Question 06.03-11 is acceptable**. **RAI 391-2974, Question 06.03-11, is closed and resolved.**

DCD Tier 2 Figure 6.2.4-1, "Containment Isolation Configurations," indicated there was a single MOV inside containment for RHR system isolation. As part of **RAI 391-2974, Question 06.03-11 (RAI 06.03.01.04-7)**, the staff requested that the applicant address why a single failure would not result in a LOCA outside containment. In its response to **RAI 391-2974, Question 06.03-11 (RAI 06.03.01.04-7)**, dated July 28, 2009, the applicant stated Figure 6.2.4-1 is for the containment barrier only and the design details for the actual configuration is shown in Figure 5.4.7-2, "Residual Heat Removal System P&ID." Two motor-operated isolation valves are installed in series between the RCS and RHRS on the CS/RHR pump suction side in each train, therefore a single failure will not result in a LOCA outside the containment.

The staff requested the applicant provide further clarification concerning the sources of debris in the DCD Tier 2 Table 6.3-3 discussion for GSI-191, "Assessment of Debris Accumulation on PWR Sump Performance." In its response to **RAI 391-2974, Question 06.03-12 (RAI 06.03.01.04-8),** dated July 27, 2009, the applicant stated the design basis for the LOCA generated non-chemical debris that potentially impacts ECCS operation is discussed in the "MUAP-08001, "US-APWR Sump Strainer Performance," Revision 2. The downstream effects evaluation is discussed in Section 6.3.4.9 of this SER.

6.3.4.6.4 GLs and Bulletins

DCD Tier 2 Table 6.3-4, "Response of US-APWR to Generic Letters and Bulletins," provides the applicant's evaluation of the US-APWR to the GLs and Bulletins identified in RG 1.206, "Combined License Applications for Nuclear Power Plants (LWR Edition)," C.I.6.3, "Emergency Core Cooling System."

In its response to **RAI 391-2974, Question 06.03-13 (RAI 06.03.01.04-9),** dated July 27, 2009, the applicant provided further clarification concerning boric acid accumulation of the vessel head because the DCD Tier 2 Table 6.3-4 discussions for BL 01-01, "Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzle," and BL 02-01, "Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity," stated these bulletins are not applicable as the US-APWR does not have penetrations in the RPV head for SI.

The ISI plans to be implemented for the Reactor Vessel Closure Head relating to the boric acid leakage will be described in US-APWR DCD Tier 2 Section 5.2.4, "Inservice Inspection and Testing of the RCPB," as part of the response to **RAI 254-2075**. Design features to facilitate the

ISI of the RV Closure Head are described in US-APWR DCD Tier 2 Section 5.3.3, “Reactor Vessel Integrity,” and Section 5.3.3.7, “Inservice Surveillance.”

In its response to **RAI 391-2974, Question 06.03-13**, dated July 28, 2009, the staff requested that the applicant indicated that GLs and Bulletins BL 01- 01 and BL 02-01 are addressed in DCD Tier 2 Section 5.2.4. However, the response did not indicate DCD Tier 2 Table 6.3-4 would be revised accordingly. In its response to **RAI 695-4934, Question 06.03-95**, dated March 17, 2011, the applicant provided the proposed changes to DC Tier 2 Table 6.3-4 to indicate that ISI for the reactor vessel head is discussed in Section 5.2.4. The applicants RAI response indicated that GLs and Bulletins BL 01-01 and BL 02-01 will be addressed in DCD Section 5.2.4, “Inservice Inspection and Testing of the RCPB.”

Therefore, the responses to **RAI 391-2974, Question 06.03-13** and to **RAI 695-4934, Question 06.03-95**, are acceptable. However, **RAI 391-2974, Question 06.03-13, is closed - unresolved, and finally closure is based on updating the DCD Section 5.2.4 as requested in RAI 695-4934, Question 06.03-95, which is being tracked as a Confirmatory Item.**

In its response to **RAI 391-2974, Question 06.03-14 (RAI 06.03.01.04-10),** dated July 28, 2009, the applicant revised the issue description in the DCD Tier 2 Table 6.3-4 text to GL 89-10, “Safety-Related Motor-Operated Valve Testing and Surveillance”, Item b, to include the test "as necessary, the methods for selecting and setting all switches", as requested.

In its response to **RAI 391-2974, Question 06.03-15 (RAI 06.03.01.04-11),** dated July 28, 2009, the applicant addressed two additional GLs and two additional Bulletins as requested by the staff. DCD Tier 2 Table 6.3-4 was revised to include the following items:

1. NRC GL 2004-02: “Potential Impact of Debris Blockage on Emergency Recirculation during Design Basis Accidents at Pressurized-Water Reactors.”

2. NRC GL 2008-01: “Managing Gas Accumulation in Emergency Core Cooling, Decay Heat Removal, and Containment Spray Systems.”

3. NRC Bulletin 2003-01: “Potential Impact of Debris Blockage on Emergency Sump Recirculation at Pressurized-Water Reactors.”

NRC Bulletin 2003-02, “Leakage from Reactor Pressure Vessel Lower Head Penetrations and Reactor Coolant Pressure Boundary Integrity,” does not apply to the US-APWR because there are no RPV lower head penetrations. Accordingly, **RAI 391-2974 Questions 06.03-14 and -15, are resolved and closed.**

6.3.4.7 NPSH Requirements of the ECCS Pumps

To show compliance with GDC 35, it must be shown that the ECCS pumps will perform their intended functions during postulated accidents. The ECCS should be designed so that sufficient available NPSH 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.

In the event of a LOCA, the SI actuation signal will start up the SI pumps. The pumps take suction from the RWSP and deliver it to the DVI nozzles of the RCS. Long term recirculation cooling is maintained by the SI pumps. The water source is the RWSP throughout the event.

The available and required NPSH at the inlet of the CS/RHR and SI pumps are provided in DCD Tier 2 Table 6.2.2-1. Thus, adequate NPSH is provided to the CS/RHR and SI pumps, including margin. DCD Tier 2 Table 6.2.2-1 presents values used in the calculations described above.

Typically, the following values and analyses are provided to determine the NPSH.

- Conservatively calculated RWSP temperature.
- Maximum pump flow head-capacity curves.
- Maximum system resistances.
- Debris laden sump screen resistance.
- Reduced RWSP level to account for liquid hold up in the containment.
- RWSP liquid is at the saturation pressure corresponding to the peak calculated RWSP temperature.
- Conservative assumptions concerning pump operation.

According to DCD Tier 2, Table 6.3-5, “Safety Injection System Design Parameters,” the SI pump NPSHA value is 7.59 m (24.9 ft.) while NPSHR (including uncertainties) is 5.7 m (18.8 ft). The other SI pump parameters are also presented in DCD Tier 2, Table 6.3-5.

According to DCD Tier 2 Table 6.2.2-1 the CS/RHR pump NPSHA value is 6.37 meter (20.9 ft.) while NPSHR (including uncertainties) is 6.00 meter (19.7 ft).

The staff asked the applicant to identify the assumptions, values and head losses that were used in the NPSH calculations. All assumptions, including details of the loss coefficients used, elevations and pump values assumed in the calculations should be provided and all requirements of GDC 35, USI, Bulletins and RG 1.82 should be resolved, incorporated and identified. In its response to **RAI 391-2974, Question 06.03-31 (RAI 06.03.02.02-5)**, dated July 28, 2009, the applicant provided the DCD location of the NPSH analysis. It is summarized in DCD Tier 2 Section 6.2, “Containment Systems,” Section 6.2.2, “Containment Heat Removal Systems,” Section 6.2.2.3, “Design Evaluation.” Therefore, **RAI 391-2974, Question 06.03-31, is resolved and closed.**

An audit of the NPSH calculations for the US-APWR was performed on March 8, 2010, at the Mitsubishi Nuclear Energy Systems, Inc., office in Arlington, VA, and documented in the “March 8, 2010, Audit Report, U.S. Nuclear Regulatory Commission Audit Report Regarding the Net Positive Suction Head Calculations for the US-APWR Containment Spray/Residual Heat Removal Pumps and Safety Injection Pump,” ADAMS Accession Number ML101540249.

One additional question, **RAI 597-4590 Question 06.03-84** was identified relating to NPSH during the audit, as documented in “Request for Additional Information No. 597-4590 Revision 2, SRP Section: 06.03 - Emergency Core Cooling System - Application Section: 6.3,” dated June 8, 2010, ADAMS Accession Number ML101600721. The applicant provided responses to these RAIs in “MHI's Response to US-APWR DCD RAI No. 597-4590 Revision 2,” UAP-HF-10194, dated July 8, 2010, ADAMS Accession Number ML101930162.

Net positive suction head is the term that is usually used to describe the absolute pressure of a fluid at the inlet to a pump minus the vapor pressure of the liquid. The resultant value is known as the NPSHA.

Different pumps will have different NPSHR dependent on the impeller design, impeller diameter, inlet type, flow rate, pump speed and other factors. A pump performance curve will usually include a NPSHR graph expressed in meters or feet head so that the NPSHR for the operating condition can be established. The fluid pressure at a pump inlet will be determined by the pressure on the fluid surface, the frictional losses in the suction piping and any pressure loses within the suction piping system associated with elbows, valves, orifices and any other obstructions to flow.

The CSS and SI pumps are estimated to be six feet tall and the midpoint of the pump is three feet below the inlet of the pump. The inlet and outlet of the pumps are on top of the pump. When the staff reviewed the diagram of the pump, it appeared that the flow of water goes through the pump suction inlet at the top of the pump and travels to the centerline before it enters the impeller. The applicant assumes that the NPSH requirement should be applied at the center of the pump.

The distance from the pump inlet to the centerline is three feet and applying the applicant's assumption that the NPSHR is measured at the centerline adds three feet to the NPSHA calculated value. Based on staff experience, many NPSHA calculations are performed at the pump suction, which, for the case of the US-APWR CSS and SI pumps, is three feet above the centerline of the pump. If the NPSHR is measured at the pump inlet, the applicant's NPSHA calculation could be in error.

The staff requested the applicant provide information from the vender of the pump that states specifically where the NPSHR is measured, i.e., at the pump suction or at the pump centerline.

In its response to **RAI 597-4590, Question 06.03-84**, dated July 8, 2010, the applicant addressed the concern expressed during the audit on the end location specified for the NPSHA calculation.

The applicant stated it is defined in ANSI/HI 1.2-2008 (Hydraulic Institute Publication HI 1.1 1.2:2008 , "Rotodynamic (centrifugal) Pumps For Nomenclature And Definitions") that NPSHA is the total suction head of liquid absolute, determined at the first-stage impeller datum minus the absolute vapor pressure of the liquid at a specific rate of flow. It is also defined that the datum is the horizontal plane through the center of the impeller described by the external points of the entrance edges of the impeller blade. Although ANSI/HI does not define the position to determine NPSHR, it is general knowledge among pump manufacturers to determine NPSHR at the centerline in order to compare it with NPSHA. Some vendor information explicitly states that NPSHR is measured at the centerline. The response to RAI 597-4590, Question 06.03-84 indicates that the applicant's assumption that NPSHR is referenced to the impeller centerline is consistent with common practice. This response shows that the applicant's NPSHA calculation is correct; the response to **RAI 597-4590, Question 06.03-84, is therefore acceptable and the RAI question is resolved and closed**.

In **RAI 626-4750, Question 06.03-87**, the staff asked the applicant whether CAP was used to determine the available SI and CS pumps NPSH (NPSHA) evaluations. In DCD Tier 2, Table 6.2.2-2 the applicant addresses the use of containment accident pressure as follows:

> Post-LOCA containment pressure is not credited for the US-APWR NPSH evaluation of the ECC and containment heat removal systems.

However, in MUAP-08001-NP, Revision 2, the applicant states that

> For the minimum NPSH available calculation ... containment pressure is assumed to equal the saturation pressure corresponding to the sump water temperature.

During the March 8, 2010, NPSH audit, the staff confirmed that the NPSH calculations assume the containment pressure is equal to the saturation pressure corresponding to the sump water temperature as the post-LOCA maximum RWSP is around 256 °F (124 °C), which corresponds to a vapor pressure close to 30 psia (207 kPa).

In Section 6.2.2.4.8 of the DCD SE the staff concluded that the NPSHA was greater than the NPSHR including uncertainties associated with the required NPSH and the assumed strainer head loss. Hence pump cavitation will be minimized and the pumps will perform their safety function under worst case accident conditions. In addition, the staff also evaluated the applicant's risk assessment when using CAP in Chapter 19 of the DCD. As stated in Section 6.2.2 of this SER, the staff finds that the applicant has adequate NPSH provided to the CS/RHR and SI pumps, and therefore, **RAI 626-4750, Question 06.03-87 is resolved and closed**.

6.3.4.8 Evaluation for Recirculation Cavitation in SI and CS/RHR Pumps

Recirculation cavitation is a phenomenon that may occur in centrifugal pumps when operated at low flow rates significantly below their best efficiency flow. Under these conditions, fluid flow is reversed at the suction and/or discharge nozzles resulting in high velocity vortices between the two flow directions, resulting in cavitation. Recirculation cavitation is known to occur in "high suction energy pumps" and can cause significant vibration and damage pump nozzles, impellers, wear rings, seals, shafts, and bearings within a short time period. Recirculation cavitation is a distinctly different phenomenon from suction cavitation, which is caused by insufficient NPSHA.

The US-APWR design has four 50 percent SI pumps that will be required to mitigate a large range of small and large break LOCAs. The design flow of each pump is 1540 gpm (5.8 m^3/min) and the minimum flow is 265 gpm (1 m^3/min) through the pump minimum-flow loop. Therefore, the pumps will be required to operate at flows significantly less than their best efficiency at low flow conditions. When the pumps automatically actuate following a LOCA occurrence, they will run at these low flow conditions for a significant period of time before system pressure drops sufficiently to allow flow closer to best efficiency operation.

Similarly, the US-APWR design has four 50 percent CS/RHR pumps. Each pump has a design flow of 3000 gpm (11.3 m^3/min) and a minimum flow of 355 gpm (1.3 m^3/min) through the pump minimum-flow loop. These pumps are required to operate at the minimum-flow condition until normally closed valves are opened to initiate CS flow following a LOCA.

In **RAI 597-4590, Question 06.03-85**, the NRC staff requested the applicant to provide a description of the pump design and testing that will demonstrate the design basis capability of the SI and CS/RHR pumps under recirculation cavitation conditions. In its response to **RAI 597-4590, Question 06.03-85**, dated July 8, 2010, the applicant, in part, stated the following:

> <u>Regarding SI Pump</u>:

MHI will request time-proven pumps to vendors, which withstand significantly low-flow conditions that may encounter recirculation cavitation. MHI will request evaluations for integrity to pumps that do not have enough past records.

Regarding the RHR/CS Pump:

(1) All RHR/CS pumps are high suction energy pumps.

(2) The required CS/RHR Pump operating flow rates depend on operating pump number. CS/RHR pumps deliver borated water from RWSP to spray ring header. CS/RHR pumps and associated piping are independent but the spray ring header is common. So, the number of operating pumps is larger, the flow rate per one pump is smaller.

Therefore, the pump operating flow rates have a range from 2800 gpm (10.6 m^3/min), which is in all (four) pumps operating condition, to 3400 gpm (12.9 m^3/min), which is in two pumps operating conditions.

(3) At the minimum flow condition which differs the most from the best efficiency condition, NPSHa increases due to the decrease of pressure loss. NPSH margin (NPSHa / NPSHr) is more than four or five and the recirculation cavitation is not predicted to occur.

Regarding the applicant's response that vendors will be requested to provide SI pumps which withstand recirculation cavitation under significantly low-flow conditions, the staff initiated **RAI 867-6174, Question 06.03-103, and RAI 881-6203, Question 06.03-104,** requesting the applicant to further describe testing and ITAAC verification that will demonstrate the capability of the SI pumps under cavitation recirculation conditions. These RAIs for the SI pump are addressed below in this SE section.

Regarding the CS/RHR pumps, the applicant stated the following: "At the minimum flow condition which differs the most from the best efficiency condition, NPSHA increases due to the decrease of pressure loss. NPSH margin (NPSHA / NPSHR) is more than four or five and the recirculation cavitation is not predicted to occur." However, the staff considers recirculation cavitation a phenomenon that may occur in centrifugal pumps when operated at low flow rates significantly below their best efficiency flow, independent of NPSH margin. Therefore, the staff does not consider the applicant's response a demonstration of the design basis capability of the RHR/CS pumps under cavitation recirculation conditions. **RAI 597-4590, Question 06.03-85 is being tracked as an Open Item.**

By **RAI 867-6174, Question 06.03-103**, the NRC staff requested the applicant to describe qualification testing that will demonstrate the design-basis capability of the SI pumps for their required mission time under low flow recirculation conditions. In its response to **RAI 867-6174, Question 06.03-103**, dated January 6, 2012, the applicant stated the following:

Functional qualification and Inservice Testing Programs for safety-related pumps are described in US-APWR DCD Section 3.9.6. Section 3.9.6 was recently revised in MHI's response to RAI 801-5897 to describe that functional qualification will be in accordance with ASME QME-1-2007. MHI will confirm the capability during minimum flow rate conditions in the functional qualification and Inservice Testing Program.

US-APWR DCD Section 6.3.2.5 is revised in the attached mark-up to state that operation during minimum flow conditions is confirmed during functional qualification and Inservice Testing Program as discussed in Sections 3.9.6.1 and 3.9.6.2, respectively.

The NRC staff considers the applicant's response to **RAI 867-6174, Question 06.03-103** acceptable on the basis that the SI pumps will be functionally qualified in accordance with ASME QME-1-2007 as discussed in DCD Sections 3.9.6.1 and 3.9.6.2 to verify capability during minimum flow rate conditions**. RAI 867-6174, Question 06.03-103 is being tracked as a Confirmatory Item,** pending revision of DCD Section 6.3.2.5.

By **RAI 881-6203, Question 06.03-104**, the NRC staff requested the applicant to provide ITAAC verification in US-APWR DCD Tier 1, Table 2.4.4-5, that qualification testing and pre-operational [as-built] testing will be performed to confirm that recirculation cavitation will not occur in SI pumps when operated during low flow design-basis conditions for the required mission time.

In its response to **RAI 881-6203, Question 06.03-104**, dated March 30, 2012, the applicant stated the following in regard to qualification testing of the SI pumps to confirm that recirculation cavitation will not occur in SI pumps when operated during low flow design-basis conditions:

The applicant responses to RAI 840-6096, Question 06.02.02-80 and RAI 867-6174, Question 06.03-103, state that Safety Injection (SI) pumps will be qualified in accordance with DCD Tier 2 Section 3.9.6 and MHI Equipment Qualification Program as described in US-APWR DCD Tier 2 Section 3.11 and Technical Report MUAP-08015.

SI pump design features and performance characteristics, including design minimum flow rate, are determined by design specification and type-tested as part of the MHI Equipment Qualification Program to demonstrate actual pump capability. Pump qualification testing simulates operating plant as-built installation and use under design basis conditions, including minimum flow line flow rate. This demonstrates SI pump ability to operate for the required mission times under limiting design-basis-accident conditions. The equipment qualification data summary report (EQDSR) documents SI pump testing that assures these pumps will operate as designed without internal recirculation flow in excess of design limits or damaging cavitation that may result from such flow.

DCD Tier 1 Table 2.4.4-5 was changed to incorporate Tier 1 Table 2.4.4-5 ITAAC #15 to clarify that vendor pump testing is performed in accordance with the US-APWR Equipment Qualification Program.

The NRC staff considers the applicant's response acceptable based on the SI pump design features and performance characteristics, including design minimum flow rate, that are determined by design specification and type-tested as part of the MHI Equipment Qualification Program to demonstrate actual pump capability. The staff notes that the MHI Equipment Qualification Program specifies pump qualification in accordance with ASME QME-1-2007.

In its response to **RAI 881-6203, Question 06.03-104** dated March 30, 2012, the applicant stated the following in regard to pre-operational [as-built] testing of the SI pumps to verify as-built minimum flow rate through the minimum-flow line is established at a rate that is above the manufacturer's design minimum flow rate as verified by qualification testing:

> SI pumps are specified and purchased to provide safety injection flow to the reactor vessel at design-basis flow rates. Each pump design has a pump curve and pump minimum flow is established by the pump manufacturer for that pump design. The pump is qualified by design and by testing, including testing in accordance with ASME QME-1-2007, to operate continuously at its design minimum flow rate for the required mission time without damaging cavitation, which only occurs when the pump is operated beyond its design limits. Subsequently, the as-built minimum flow rate through the minimum-flow line is established at a rate that is above the manufacturer's design minimum flow rate but is consistent with the pump's design capacity to provide safety injection flow to the reactor vessel during a limiting design basis event. For the standard design, minimum-flow line flow is set at 265 gpm, which expected to be bounding but may vary with purchased pump design for an as-built plant. Thus, the minimum-flow line flow rate of 265 gpm is not suitable for inclusion in Tier 1 in accordance with SRP 14.3 criteria.
>
> DCD Tier 1 Table 2.4.4-5, ITAAC 7.b verifies as-built SI pump performance, including as-built SI pump minimum flow rate through the minimum-flow line. Each US-APWR SI pump has a dedicated minimum-flow line. SI pump minimum-flow lines are always open. As- built minimum flow must be established in order to measure pump differential head "at the minimum flow" and as-built minimum flow is measured at SI pump injection flow shutoff (zero flow to the reactor vessel). Consequently, the tested SI pump must, as stated in AC [Acceptance Criteria] 7.b.ii, be operating "at the minimum flow," which is the minimum-flow line flow rate. This flow rate is established as a test condition prior to taking differential header measurements. Thus, SI pump minimum flow rate, or "the minimum flow," is verified as a direct result since it is a test condition specified by the ITAAC 7.b Acceptance Criterion 7.b.ii.
>
> Closure of this ITAAC requires each SI pump to establish design minimum flow through its minimum-flow line. Once test results are collected, an analysis is performed to convert test results "at atmospheric pressure" to accident conditions to verify that the SI pumps are also capable of achieving these same results under design conditions. Existing ITAAC verify that as-built SI pumps will operate without damaging cavitation when safety injection pumps are operated during low flow design conditions for the required mission time. No additional ITAAC are needed for this purpose.

The NRC staff considers that the applicant’s response adequately describes the pre-operational [as-built] testing to verify the as-built minimum flow condition for the SI pump is above the manufacturer’s design minimum flow rate as verified by qualification testing. However, the staff does not consider ITAAC #7b in Tier 1 Table 2.4.4-5 to clearly describe the verification process that the as-built minimum flow conditions for the SI pump is above the manufacturer’s design minimum flow rate. **RAI 881-6203, Question 06.03-104, is being tracked as an Open Item.**

6.3.4.9 Technical Evaluation of LOCA Debris on ECCS Performance

Technical Report MUAP-08013-P, “US-APWR Sump Strainer Downstream Effects,” Revision 4, issued August 2012, assesses the US-APWR design with respect to the lessons learned as part of GSI-191, the component and system related concerns identified in GL 2004-02, NEI Guidance Report NEI 04-07, “Pressurized Water Reactor Sump Performance Evaluation Methodology,” Revision 0, Volume 1, issued December 2004, and WCAP-16793-NP, “Evaluation of Long-Term Cooling Considering Particulate, Fibrous and Chemical Debris in the Recirculating Fluid. The purpose of this assessment was to ensure that that these systems and components will operate as designed under post LOCA conditions. The downstream systems and components included in the assessment were the ECCS, the CSS and the reactor core. The report evaluated the effects of operating with debris-laden, post-LOCA fluid.

MUAP-08013-P addresses downstream effects that could impact safety functions associated with pumps, valves, heat exchangers, instrumentation (sensing lines and flow measuring devices), spray nozzles, and reactor vessel flow paths including long-term core coolability. Evaluation of LOCA debris on ECCS performance is documented in Section 6.2.2.4.11, “Downstream Effects – Ex-Vessel Components and Systems” of the staff’s DCD Section 6.2.2 SE.

6.3.4.9.1 Blockage at the Core Inlet

The applicant performed a series of tests to quantify the effect of fibrous and particulate debris and containment chemical effects on the head loss across the fuel assemblies of a US-APWR core during a postulated LOCA, documented in MUAP-08013-P, Revision 4, and performed in consideration of GSI-191.

The objective of these experiments was to demonstrate that there is reasonable assurance that the US-APWR can provide adequate post-LOCA long-term core cooling. The applicant used a fuel assembly that is consistent with the fuel assembly design described in the US-APWR DCD, but of shortened length. The debris loadings, the ratio of particulate to fibrous debris, the flow rate and flow direction was varied. The purpose of the tests was to select a combination of debris variables and simulated plant variables that would bound any US-APWR LOCA and to demonstrate available long-term cooling margin in the US-APWR design such that with the flow blockage caused by the containment debris transported to the reactor vessel, the head losses determined by the fuel assembly testing are bounded by the test acceptance criteria.

The applicant performed 25 different tests in three sets. The tests utilized a single mock-up assembly contained in a test section as representative of the US-APWR core, and the post-LOCA debris laden flow was simulated by introducing three types of debris: particulate, fibrous and chemical. Pressure drop over the several segments of the mock-up assembly was measured to determine the pressure drop increase due to the debris accumulation. The tests were performed using room temperature water under atmospheric pressure. The flow rate was selected as 1/257 of the US-APWR core flow rate for the applicable scenario so that the flow rate through the mock-up assembly is representative to that through the fuel assembly in the US-APWR core. The amount of debris is also determined as 1/257 of the postulated amount in the US-APWR core. Three different configurations are simulated: hot leg break, cold leg break and cold leg break after hot leg switch over (HLSO). The test loop was configured to simulate upward flow through the core for the hot leg and cold leg break tests and downward flow through the core for the HLSO tests.

Appendix H, “Conservatism Applied to HL Break Test and HLSO Test Flow Rate Conditions,” of MUAP-08013-P, Revision 4, provides an analysis of the available pressure drop for the different break scenarios. The bases for the flow rates for each of the break locations are presented along with a description of the available driving head based on the break location and flow direction. During an April 2011, audit, the staff reviewed the detailed calculations that support the summary analysis provided in Appendix H. The staff reviewed the inputs used to calculate the driving force and flow rate, which followed a conservative methodology similar to Appendix K. Based on the conservative nature of the analysis methodology and inputs, the staff concludes that the available pressure drops calculated for test criteria using the MUAP-08013-P, Revision 4, methodology are acceptable.

The first set of core inlet blockage tests is documented in MUAP-10021-P, Revision 0. All debris used for the test is prepared and is consistent with the design basis debris characteristics of the US-APWR as discussed in Section 3.3, “Debris Characteristics,” and Table 3-5, “Debris Characteristics,” and Appendix C, “Evaluation of Chemical Debris (for head loss),” of Revision 3 of MUAP-08001-P, “US APWR Sump Strainer Performance.” Nine tests were performed including three hot leg breaks, five cold leg breaks and one cold leg break after HLSO. Section 5, “Test Condition,” and Table 5-3, “Test Matrix,” of MUAP-10021 summarizes the test matrix and conditions for the first set of tests.

In **RAI 716-5527, Question 06.03-98**, issued on August 23, 2011, the staff asked for additional justification to demonstrate that the limits associated with GSI-191 in-vessel downstream effects are not violated in light of the distance between the fuel assembly and the test wall being equal to the full gap between two adjacent assemblies and in effect increasing the bypass area available around the assembly. The staff considers the full gap testing arrangement to be non-conservative because it effectively doubles the bypass area. As a result the applicant repeated the tests performed in the first set of core inlet blockage tests with a distance between the fuel assembly and the test wall equal to the half-gap between two adjacent assemblies, which was similar to a previously reviewed and approved testing methodology. The staff finds the applicant’s use of a half-gap distance and repeating all the tests acceptable and thus, **RAI 716-5527 Question, 06.03-98, is resolved and closed**.

The second set of tests in which the applicant used a half-gap distance and also updated the amount of debris for the tests, is documented in MUAP-11022-P, Revision 0. In an evaluation of the amount of debris to be applied to the tests, the applicant has applied [ ]. Both [ ] and core bypass methodologies are discussed in the “In-vessel Source Term” section below.

Using these methodologies, the fibrous debris was reduced from a maximum of [ ] per fuel assembly in the first set of tests to a maximum of [ ] per fuel assembly in the second set of tests, while the particulate was reduced from a maximum of [ ] in the first set of tests to [ ] in the second set of tests. The debris in the second set of core inlet blockage tests is consistent with the design basis debris characteristics of the US-APWR discussed in Section 3.3 and Table 3-5, and Appendix C of Revision 7 of MUAP-08001. Nine tests were performed including four hot leg breaks, four cold leg breaks and one cold leg break after HLSO. Section 5, “Test Condition,” and Table 5.3-1, “Test Matrix,” of MUAP-12004 summarizes the test matrix and conditions for the second set of tests.

The third set of core inlet blockage tests, documented in MUAP-12004-P, Revision 0, was performed to reflect design changes in ECC recirculation water return path to the RWSP. The design change resulted in an increase in chemical debris loading from [ ] in the first and second set of tests to [ ] in this third set of tests. Seven tests were performed including three hot leg breaks, three cold leg breaks and one cold leg break after HLSO.

In all three sets of tests, a pressure drop acceptance criterion was used to determine if Long Term Core Cooling (LTCC) would be maintained. Details on the method of defining the acceptance criterion for each break type are described in Appendix H, "Conservatism Applied to HL Break Test and HLSO Test Flow Rate Conditions," of MUAP-08013-P, Revision 4. This appendix describes the LOCA scenarios (hot leg break, cold leg break and cold leg break with HLSO), available pressure drop, and flow rate conditions. The pressure drop acceptance criterion, available driving force for the increase in core pressure drop due to debris accumulation, is calculated as the driving force minus the RCS pressure loss without debris loads. A separate acceptance criterion is computed for each LOCA scenario.

The test assembly, which is about 1/3 the height of a full size assembly, uses only [ ] spacers as opposed to the full size assembly which contains eleven. Specifically, the test assembly had [ ] intermediate grids versus a full fuel assembly which would have nine. In the core inlet blockage tests, the upstream grid spacers generally collected larger amounts of. [

]. The staff found the applicant's approach of using a shortened fuel assembly and the applicant's approach to accounting for the truncated portion acceptable based on the testing observations.

The staff concludes that using the design basis amount of debris satisfies the corresponding acceptance criteria, shows good repeatability, and demonstrates that sufficient driving head is available to maintain an adequate flow rate to remove decay heat during post-LOCA ECC recirculation with debris laden fluid. Therefore, the staff concludes that the test results show that the core coolability during LTCC for US-APWR is maintained even if debris build-up occurs in the core.

Calculation of Minimum Delay Time for When Debris Reaches the Core

Important boundary conditions for core blockage testing, including the acceptance criteria and debris source term for the cold-leg break LOCA, are based on the assumed time required for the debris to reach the reactor vessel after an accident. MUAP-08013-P Revision 4 Appendix E, "In-Vessel Downstream Effects Evaluation Time," provides the analysis for the estimated time required for the debris to reach the reactor vessel after an accident. It was stated in the appendix that shorter and more conservative time values were used for downstream evaluations provided in MUAP-08013-P Revision 4, including Sections 4.2.1, "Trapping Debris at Grid Spacer," 4.3.1, "Chemical Deposition on the Cladding," and Appendix G, "Boundary Conditions for Cladding Surface Temperature Evaluation."

On February 1, 2010, the staff issued **RAI 530-3989 Question 04.04-40,** requesting the applicant to provide the details of the analyses used to develop the volumes of "ineffective pools" and discuss how the initial RCS water inventory and the liquid volume of one accumulator which is assumed to spill out the break, were accounted for in the evaluation. In its response to **RAI 530-3989, Question 04.04-40**, dated March 4, 2010, the applicant stated the initial RCS inventory and the liquid volume of the accumulator were not accounted for in the calculation of the required time for debris to reach the vessel or the calculation for the post-LOCA minimum water level, to bound the LOCA scenario. The applicant stated in its RAI response that typographical errors were found in Table 3-10, "Upstream Effect Hold-up Volumes," in MUAP-08001-P Revision 2 and the water volumes for "containment spray droplets and saturated steam (including the empty spray header rings and pipes)" and "water stream on the floor (including reactor cavity floor)" should be reversed. The applicant proposed corrections to Table 3-10 in MUAP-08001-P. In its response to **RAI 530-3989, Question 04.04-40**, dated May 29, 2012, the applicant also explained why none of the "fluids in motion" contributed to filling the "ineffective pools" but only delay the time for these pools to fill and overflow into the RWSP. In addition, the physical significance of these volumes as they relate to the time delay evaluation was clarified. Most of these "fluids in motion" drain into the "ineffective pools" at first, not to the RWSP. After filling the ineffective pools, the "fluid in motion" will overflow into the RWSP. The return water to the RWSP will be started as a consequence of the "fluid in motion" including the containment spray and condensation. Therefore, "fluid in motion" is required to fill the ineffective pools and its volumes were considered in the time delay analysis. The staff found the descriptive explanation provided in the RAI response acceptable and **RAI 530-3989, Question 04.04-40, is resolved and closed.**

On August 23, 2011, the staff issued **RAI 815-5986, Question 06.03-102**, asking the applicant to justify the methodology used for calculating the delay time for debris to reach the RWSP. The RAI resulted from the following non-conservatisms: neglecting the RCS and accumulator volumes, the possibility of not filling or bypassing ineffective pools, the possibility of reduced return flow to the RWSP and the inclusion of minimum water level (margin for design basis). In its response to **RAI 815-5986, Question 06.03-102**, dated May 29, 2012, the applicant described the method of calculating the delay time which incorporates the RCS and accumulator volumes.

[

]. The purpose of the [ ] revision to the RAI response is due to a design change simplifying the recirculation flow path.

The design changes were made to allow a clear explanation and analysis of the debris transport time to the core and the hold-up volume.

[

]. The response to **RAI 815-5986, Question 06.03-102,** is included in Appendix E, "In-Vessel Downstream Effects Evaluation Time" of MUAP-08013-P, Revision 3. The staff finds the approach in the RAI acceptable, and **RAI 815-5986, Question 06.03-102, is resolved and closed**. The staff opened **Confirmatory Item CI-SRP06.03-3** to verify that the final design document incorporates the associated supplemental DCD information provided in GTR2.

On May 29, 2012, the staff issued **RAI 932-6408, Question 06.03-105,** requesting additional information regarding the flow rates into, between, and out of the reactor cavity and header compartments during a LOCA and if there is a difference between these two compartments. In its response to **RAI 932-6408, Question 06.03-105,** dated June 28, 2012 the applicant provided an analysis of the flows of water into the reactor cavity and header compartments as well as an analysis of the impact on several assumptions. The applicant concluded that the calculated level in the header compartment is approximately 1.5 in (3.8 cm) higher than the reactor cavity. Due to the difference in water level the water volume in the reactor cavity used to calculate T1 would be reduced by approximately 13.5 ft^3 ($0.38 m^3$). Since this is a small volume relative to the design basis volume of the reactor cavity and header compartment buffer areas (29,810 ft^3) and relative to the other hold-up volumes which are conservatively neglected in the debris timing analysis, the staff finds this assumption acceptable.

Another potential impact of the difference in water level is the assumption of debris split ratio per one sump strainer, however the applicant found that the debris allocation in the worst case went from 83 percent to 80 percent which was bounded in the strainer testing. The staff finds the conclusions provided in the applicant's response acceptable based on the above discussion, and **RAI 932-6408 Question 06.03-105, is resolved and closed**.

On May 29, 2012, the staff issued **RAI 932-6408 Question 06.03-106,** requesting additional information regarding the treatment of C/V drain pump room as a buffer area for the calculation of T1. The staff asked for information to support that the C/V drain pump room would fill up prior to the water level in the reactor cavity and header compartment reaching its respective overflow pipes. In its response to **RAI 932-6408 Question 06.03-106,** dated June 28, 2012 the applicant provided a response justifying their assumption. The applicant stated that the US-APWR is designed such that the C/V drain pump room is filled up earlier than other buffer areas in the case of a LOCA, based on the smaller capacity of the C/V drain pump room relative to that of the reactor cavity and the header compartment, and the lack of debris interceptors over the floor openings to the C/V drain pump room. The applicant also notes the large conservatisms in the calculation of T1. The applicant considered several cases which neglected the fill up of the C/V drain pump room, but also removed a conservatism to quantify the impacts. One of the cases neglected the volume of the C/V drain pump room, but did not take into account the steam mass as part of the inflow mass to the buffer areas; this case resulted in a greater value for T1. The staff concludes that the assumption of the C/V drain pump room filling prior to the reactor cavity and header compartment reaching their respective overflow piping is acceptable. The staff's conclusion is based on the smaller volume of the C/V drain pump room, as well as the considerable conservatisms in the applicant's analysis including those discussed in the RAI response as well as the ineffective volumes which the applicant does not credit. The staff finds the conclusions provided in the applicant's response acceptable based on the above discussion, and therefore, **RAI 932-6408 Question 06.03-106, is resolved and closed**.

The staff finds the minimum delay time assumption acceptable. The staff based this conclusion on the flow calculations which included but was not limited to the following conservatisms: a conservative calculation of the break flow including steam, and a design that required water to fill the reactor cavity and the header compartment before reaching the RWSP, assuming a delay time for debris to reach the sump from the time that debris laden water reaches the RWSP based on the minimum cross-section of the RWSP, and assuming containment spray flow from the time of the LOCA.

In-vessel Debris Source Term

The fiber only bypass tests are described in Appendix B of MUAP-08001-P (Revision 7). The bypass tests were performed to establish the maximum amount of fiber that bypasses the strainer. The fiber was introduced into the test tank in four batches. The first two batches consisted of 12.5 percent of the scaled total fiber source term. The first batch was used to replicate fiber bypass at a lesser debris load than design basis. The second batch was used to replicate fiber loading distribution of fiber to all four trains. The third batch contained an additional 25 percent of the scaled total fiber source term (50 percent scaled in total) to replicate the fiber loading on the strainer if two trains were in operation. The final batch contained an additional 50 percent of the scaled fiber source term (100 percent scaled in total) to replicate the potential fiber bypass if all plant debris accumulated to one sump. One micron filter bags were utilized to capture and measure the fiber that passed through the strainer. The bypass tests show a bypass ratio from [ ] to [ ]. The applicant then assumes a value of [ ] for use in downstream evaluations.

The staff concludes that the use of the fiber only bypass test to determine the fiber source term quantity for subsequent testing is acceptable. This conclusion is based on the bounding nature of reducing potential blockage at the strainer by removing any potential blockage from particulate or chemical debris, and by the introduction of fiber in small batches. This, combined with the overall assumption of no debris settling, leads to a bounding fiber source term used as the basis for in-vessel downstream effects.

In the case of the cold leg break LOCA only, [ ] documented in Appendix I of MUAP-08013, "Methodology for Estimate of Debris Load in Cold-leg Break LOCA," to reduce the debris load. [

]. The applicant uses a delay time of [ ] seconds for when debris will start reaching the core based on the analysis documented in Appendix E of MUAP-08013, "In-

Vessel Downstream Effects Evaluation Time," the staff's review of this time delay is discussed in the preceding "Calculation of Minimum Delay Time for When Debris Reaches the Core" section. [

]. The applicant assumes that [

]. The staff finds the reduction of debris for the cold leg break test acceptable based on [

], and assuming that [
].

The staff's review for the US-APWR highlighted the significance of certain assumptions about debris in containment to the adequacy of long-term core cooling, and a concern that the values not be revised without additional testing and analysis. Given the applicant's downstream effects testing and analysis described in Technical Report MUAP-08013-P, Revision 4, and the containment debris limits described in DCD Revision 3, Section 6.2, "Containment Systems," (as modified by GSI-191 Tracking Report dated August 2012), the staff issued **RAI 997-7033 Question 06.03-112,** requesting that the applicant evaluate applying Tier 2* designation to items associated with long-term core cooling, specifically the debris limits for blockage evaluations. **RAI 997-7033, Question 06.03-112 is being tracked as an Open Item**.

6.3.4.9.2 Trapping Debris in Fuel Assemblies

The fuel cladding surface thermal response accounting for debris clogging at fuel assembly grid spacers is considered in MUAP-08013-P Revision 4 Section 4.2.1, "Trapping Debris at Grid Spacer." Debris that bypasses the sump strainer can be trapped at grid spacers in the US-APWR fuel assemblies. The maximum diameter of an inscribed circle in the grid spacer is larger than that in the bottom nozzle, which is the narrowest gap downstream of the strainer to the core inlet.

The intermediate grid spacers have mixing vanes to increase the mixing of the primary coolant and increase the heat removal efficiency while the top and bottom grid spacers do not have mixing vanes. The springs and the dimples on the grid strap sheet create an open flow hole. The debris could also pass through the flow hole on the grid strap sheet. Section 4.2.1 in MUAP-08013-P Revision 4 states the following.

A thermal analysis was performed by the applicant to demonstrate that fuel cladding thermal response remains acceptable even with debris clogging at the grid spacers taken into account. The ABAQUS™ code, discussed below, was used to perform this steady-state heat transfer analysis applying a partial flat-plate model to represent a fuel rod with solid elements. The cladding surface temperature was evaluated by a parametric study varying the thermal conductivity and the thickness of the accumulated debris layer.

The maximum temperature was analyzed at the grid central position. For consistency, the core parameters that were used as boundary conditions in the thermal analysis, such as fuel decay heat and thermal-hydraulic conditions, were derived from the results of a WCOBRA/TRAC post-

LOCA analysis described in MUAP-08013-P, Revision 4, Appendix G, “Boundary Conditions for Cladding Surface Temperature Evaluation.” The evaluation analysis showed that the maximum temperature at the cladding surface occurred under the conditions of the worst case, which assumed 0.173 W/(m·K) (0.1 BTU/(hr·ft·°F)) for the minimum thermal conductivity and 50 mils (1.27 mm) for the maximum debris layer thickness. The predicted maximum cladding surface temperature, [ ] met the acceptable temperature limit of [ ] described in MUAP-08013-P, Revision 4, Appendix C, “Long-Term Core Cooling Acceptance Basis for GSI-191.” Based on this outcome, the applicant concluded that adequate core cooling was demonstrated even with the consideration of debris accumulation at the grid straps after a LOCA.

Describing the applied ABAQUS™ model, MUAP-08013-P, Revision 4, Section 4.2.1.2, “Methodology,” explains that the model covers one span length with a grid and includes the axial position with the maximum heat flux. The grid is located at the center of the model and the cladding extends a half span length from the grid. The model used a heat flux as a boundary condition on the inner cladding surface and modeled the radial heat transfer through the fuel cladding and the accumulated debris layer to the surrounding coolant, conservatively ignoring the longitudinal conduction. Therefore, there was no need to include the fuel pellets and the gap between the pellet and the cladding in the modeling approach. In reality, the thermal flux at the outer surface of the debris layer in contact with the coolant would decrease when the thickness of the debris layer increases due to debris accumulating on the cladding surface. Thus, the applicant concluded that the flat plate model with a constant heat flux based on the inner cladding surface provided conservative results. Based on the review of the applied model and obtained results, the staff found necessary to issue a series of RAI questions as described in the following:

On February 1, 2010, the staff issued **RAI 530-3989, Question 04.04-24**, asking the applicant to explain if the use of ABAQUS™ for such a type of heat transfer analysis has been reviewed by the NRC staff. In its response to **RAI 530-3989, Question 04.04-24,** dated March 4, 2010, the applicant addressed the use of the ABAQUS™ code for this evaluation. The applicant explained that ABAQUS™ is a commercially available Finite Element Analysis (FEA) computer code, which is applicable to structural and heat transfer analyses in many different engineering fields, such as mechanical, construction, chemical, etc. The applicant considers using ABAQUS™ for the heat transfer analysis to be consistent with standard industry practice, and the results are appropriate. The staff considers the conclusions provided by the applicant in the RAI response acceptable based on the clear description of the modeling approach and applied boundary conditions, and therefore, **RAI 530-3989 Question 04.04-24, is closed**.

On February 1, 2010, the staff issued **RAI 530-3989, Question 04.04-25**, asking the applicant to clarify the basis for the heat transfer coefficient applied at the interface with surrounding fluid in the analysis. In its response to **RAI 530-3989, Question 04.04-25**, dated March 4, 2010, the applicant addressed the basis for the uniform heat transfer coefficient used in the evaluation.

The evaluation uses the thermal hydraulic conditions at the maximum heat flux axial position. These conditions are used to determine the power conditions. Based on the WCOBRA/TRAC analysis, the lowest value of the heat transfer coefficient at this axial position is used for the evaluation. The staff agrees with the applicant’s statement in the RAI response that using both the maximum power and the minimum heat transfer gives conservative results. Thus, **RAI 530-3989, Question 04.04-25, is resolved and closed**.

On February 1, 2010, the staff issued **RAI 530-3989, Question 04.04-27**, asking the applicant if there was evidence through testing that the debris would form and accumulate in a uniform manner on the cladding surfaces. In addition, the applicant was asked to explain the impact of a non-uniform debris formation on the final results. In its response to **RAI 530-3989, Question 04.04-27**, dated March 4, 2010, the applicant stated that MHI has no evidence through testing that the debris forms and accumulates in a uniform manner. The applicant explained that the analysis gives the results of a parametric evaluation of the effects of both the thermal conductivity and the thickness of the debris layer on the cladding outer surface temperature and does not specifically address the process of debris formation, such as uniform or non-uniform accumulation of the debris. Non-uniform accumulation of the debris impacts both the value of the thermal conductivity and thickness of the debris layer. Compared with non-uniform debris formation, the assumption of a uniform debris layer with both low thermal conductivity and high thickness will give conservative results for the effects of the debris layer on the outer surface temperature of the cladding. The applicant concluded that non-uniform debris formation has no impact on the final results in the analysis. A further conservatism in this parametric study is that axial heat transfer is ignored, which could occur at both the upper and bottom horizontal surface of the debris at the grid. Considering the results from the parametric evaluation provided in Table 4.2.1-1, "Cladding Metal Surface Temp. vs Debris Thickness," of MUAP-08013-P, Revision 4, and taking into account the explained conservatism in the performed analysis, the staff finds the response to **RAI 530-3989 Question 04.04-27 acceptable and the RAI is considered closed and resolved**.

On February 1, 2010, the staff issued **RAI 530-3989, Question 04.04-28**. Part (a) asked the applicant to explain how it was determined that the conductivity of the matted debris, formed around the fuel, was equivalent to that of crud 0.87 W/(m·K) (0.5 BTU/(hr·ft·°F)) plus or minus 0.69 W/(m·K) (0.4 BTU/(hr·ft·°F)). The applicant provided its response to this RAI question on March 4, 2010. It was explained that the fibrous debris material is glass, which typically has a thermal conductivity of 1.0 W/(m·K) (0.58 BTU/(hr·ft·°F)). The thermal conductivity of the debris is based on the moisture content ratio of the debris. The thermal conductivity of the moisture included in the debris is 0.67 W/(m·K) (0.39 BTU/(hr·ft·°F)). For debris containing 50 percent glass and 50 percent moisture, the thermal conductivity is 0.85 W/(m·K) (0.49 BTU/(hr·ft·°F)), calculated as an average value of the glass and moisture thermal conductivities. In **RAI 530-3989, Question 04.04-28 Part (b)**, the applicant was asked to justify why the conductivity of the debris could not be lower than 0.17 W/(m·K) (0.1 BTU/(hr·ft·°F)). The applicant provided its response on March 4, 2010. The estimated thermal conductivity of dry NUKON is 0.040 W/(m·K) (0.023 BTU/(hr·ft·°F)). The thermal conductivity of NUKON, the thermal insulator, in the wet condition would be higher than that in the dry condition since the thermal conductivity of moisture is larger than that of the NUKON.

The minimum value of 0.17 W/(m·K) (0.1 BTU/(hr·ft·°F)) for the debris thermal conductivity assumed in the parametric study corresponds to a moisture fraction in the NUKON insulator of about 20 percent.

Finally, in **RAI 530-3989, Question 04.04-28 Part (c)**, the applicant was asked what the rise of the final cladding temperature would be if the assumed conductivity was found to be less than 0.17 W/(m·K) (0.1 BTU/(hr·ft·°F)) and amounting, for example, to about 0.087 W/(m·K) (0.05 BTU/(hr·ft·°F)). In its response to RAI 530-3989, Question 04.04-28 Part (c), the applicant acknowledged the final cladding temperature would be higher if the debris conductivity was less than 0.17 W/(m·K) (0.1 BTU/(hr·ft·°F)). The applicant concluded that the thermal conductivity in the parametric study, based on the thermal conductivity minimum value of 0.17 W/(m·K) (0.1 BTU/(hr·ft·°F)), was conservative for wet debris. The staff finds this response to **RAI 530-3989, Question 04.04-28,** acceptable and in line with industry assumptions including those assumptions used by WCAP-16793 and the AP1000 design, and **the RAI is resolved and closed**.

On February 1, 2010, the staff issued **RAI 530-3989, Question 04.04-29**, asking the applicant to explain how it was determined that the maximum thickness of the debris amounted to 50 mils (1.27 mm or 0.05 in) for the US-APWR fuel. In addition, the staff requested the applicant provide a justification and references to data, if available, to demonstrate that the thickest debris formation that can be trapped on the cladding was less than 50 mils (1.27 mm or 0.05 in) for the US-APWR fuel. In its response to **RAI 530-3989, Question 04.04-29**, dated March 4, 2010, the applicant described how it was determined that the maximum thickness of the debris amounted to 50 mils (1.27 mm or 0.05 in) for the US-APWR fuel. The structures in the grid spacer have the potential to trap debris. The grid spacer spring and dimple separate two adjacent fuel rods by 122 mils (3.10 mm or 0.122 in). The debris is assumed to accumulate on each fuel rod and therefore about 50 mils (1.27 mm or 0.05 in) are available for debris to accumulate on each fuel rod, accounting for the thickness of the grid strap sheet. The applicant also noted that this clearance also includes the grid metal structure, such as the spring and dimple. The debris could accumulate around the fuel rods in the grid spacer, but would not block the coolant flow through the grid spacer. The staff finds this response to **RAI 530-3989, Question 04.04-29,** acceptable based on the conservative analysis and assumptions described above, and **this RAI is resolved and closed**.

On February 1, 2010, the staff issued **RAI 530-3989, Question 04.04-30**, asking the applicant to clarify the statement “The analyses have no heat barrier between each material, such as cladding and accumulating debris.” The question also pointed out that the actual composition of the barrier that forms adjacent to the cladding could be difficult to evaluate and asked for a justification that a heat transfer barrier does not form between the fuel and the debris formation. In addition, it was asked if there were testing results indicating how the debris is formed adjacent to the US-APWR fuel. In its response to RAI 530-3989, Question 04.04-30, dated March 4, 2010, the applicant explained that the analysis treated the debris layer as a heat transfer barrier. It was also clarified that the statement meant that the heat transfer analysis assumed no thermal contact resistance at the cladding-debris interface. The applicant stated that including a thermal contact resistance at the cladding-debris interface in the analysis would be overly conservative as the range of values for the debris thermal conductivity used in the parametric study was conservatively chosen. The applicant also clarified that it did not have test results indicating how the debris is formed adjacent to the fuel. With regard to **RAI 530-3989, Question 04.04-30**, the staff agrees that the debris thermal conductivity range used in the parametric study in MUAP-08013-P, Revision 3, Section 4.2.1, “Trapping Debris at Grid Spacer,” was rather broad to account for associated uncertainties with regard to this parameter. The applied debris thermal conductivity values are documented in Table 4.2.1-1, “Cladding Metal Surface Temp. vs Debris Thickness,” of MUAP-08013-P, Revision 4.

In addition, taking into account that the followup question to **RAI 530-3989, Question 04.04-28,** addresses the determination of the limiting thermal conductivity value, the staff agrees that there is no need to include additional uncertainties associated with contact resistance.

Therefore, the response is considered acceptable, and **RAI 530-3989, Question 04.04-30, is resolved and closed**.

The staff concludes that the applicant has acceptably demonstrated that core cooling will be maintained in the presence of debris trapping in the fuel assemblies.

6.3.4.9.3 Chemical Effects on Fuel Rods

During the long term cooling recirculation phase following a LOCA, the injected coolant will contain chemical impurities, which are dissolved or suspended in the recirculating water. The continuous supply of such products from chemical reactions occurring in the containment and transported into the reactor core can cause precipitation on the fuel cladding surfaces. As a result of such deposits, the fuel cladding temperature can increase.

The impact from these chemical effects on the fuel rod thermal response is considered in MUAP-08013-P Revision 4 Section 4.3, “Chemical Effects on Fuel Rods.” The concentration of chemical effect products due to core boiling during long term cooling and the plate-out of deposits on the fuel rods are assessed in this section.

The applicant performed chemical effects testing to obtain experimental data on the corrosion products that may form in a post-LOCA environment for the US-APWR under simulated plant conditions documented in MUAP-08011-P. The testing provided information with regard to the compositions, characteristic properties, and yields of corrosion products from chemical reactions that can take place in the containment building under representative post-LOCA conditions. Using such chemical effects test data, predictions for the fuel cladding temperature with deposits of chemical impurities were performed and presented in the report. In the applied methodology, the deposition of chemical impurities present in the reactor coolant on the cladding surface was calculated first and the impact from such deposits on the fuel temperature was assessed next. It was shown that the scale gradually increases during the LOCA. However, after boiling termination, the LOCA scale growth rate decreases. The LOCA scale thickness was predicted as approximately [ ] at 30 days (720 hours) after the LOCA. Therefore, the applicant concluded that the deposited LOCA scale will not block the coolant flow path and has no influence on fuel cladding cooling. The fuel cladding temperature was found to gradually decrease because of the reducing decay heat. This effect was found as being larger than the temperature increase effect from the thermal resistance of scale formed during the LOCA. The fuel cladding temperature was shown to be maintained lower than the temperature acceptance criterion of [ ] during the entire evaluation period. The applicant concluded that the structural integrity of fuel cladding is maintained because scale formed during the LOCA would not affect coolability and the fuel cladding temperature would be maintained lower than the acceptance criterion limit. Based on the review of the applied methodology and provided results, the staff issued a series of RAI questions, which are described below.

On February 1, 2010, the staff issued **RAI 530-3989, Question 04.04-31**, asking the applicant to provide test data, if available, showing that the epoxy coatings in the US-APWR are resistant to chemical reactions at temperatures up to 171.1 °C (340 °F). In its response to **RAI 530-3989, Question 04.04-31**, dated March 4, 2010, the applicant stated it does not have test data that shows epoxy coatings are chemically resistant to highly acidic and caustic environments at high temperature at this stage. However, the applicant explained in the response that the standard US-APWR design will utilize only DBA epoxy coatings, which were stated as being resistant to chemical reactions at high temperature in the provided RAI response.

The response referred to NUREG-CR/6914, “Integrated Chemical Effects Project: Consolidated Data Report,” Appendix C, “Test Plan,” Section 5.2.7, “Protective Coatings,” with regard to expected utilization of DBA epoxy coatings, tested in accordance with ASTM D 3911, in light-water nuclear power reactor plants. The response also clarified that ASTM D 3911, “Test Method for Evaluating Coatings Used in Light-Water Nuclear Power Plants at Simulated Design

Basis Accident (DBA) Conditions," identifies test parameters, conditions, and procedures related to high temperatures and highly acidic and caustic environments. Thus, the RAI response concluded that the DBA epoxy coatings in the standard US-APWR design would show chemical resistance to highly acidic and caustic environments at high temperatures. In conclusion, the RAI response stated that "the chemical effects test plan was submitted to the NRC for advance review to ensure the test plan was adequate for implementation," and referred to **RAI 45-876, Question 06.02.02-01,** for further details. **RAI 45-876, Question 06.02.02-01,** relates to coatings and chemical effects testing and references MHI Technical Report MUAP-08006-P Revision 0, "US-APWR Sump Debris Chemical Effects Test Plan." **RAI 45-876, Question 06.02.02-01,** is discussed in Section 6.2.2 of this SER. Based on the additional information provided in the applicant's response, the staff considers the response acceptable, and **RAI 530-3989, Question 04.04-31, is resolved and closed**.

On February 1, 2010, the staff issued **RAI 530-3989, Question 04.04-32**, asking the applicant if the chemical plate-out process was evaluated for long-term boiling conditions corresponding to high inlet blockage cases presented in Section 4.1.2, "Evaluation of Blockage at the Core Inlet," MUAP-08013-P Revision 0. In this RAI question, the staff pointed out that such high-blockage cases should be evaluated for plate-out deposits since local boiling conditions in the core can differ from conditions normally expected when debris core blockage is not considered. The applicant was also asked what the effect of additional plate-out would be on the fuel cladding temperature, if any. In its response to **RAI 530-3989, Question 04.04-32**, dated March 4, 2010, the applicant stated that the conditions considered in MUAP-08013-P, Revision 0, Section 4.1.2, were not taken into account when the chemical plate-out process was evaluated in MUAP-08013-P, Revision 0, Section 4.3.1, "Chemical Deposition on the Cladding." The applicant explained in the RAI response that the chemical plate-out rate is proportional to the chemical concentration in the coolant and the heat flux at the surface of fuel rod. These two parameters would not be affected by inlet blockage since sufficient coolant remains in the core even if high-blockage conditions are assumed. The staff agrees with the explanation provided in the RAI response, and **RAI 530-3989, Question 04.04-32, is resolved and closed**. It is noted that Section 4.1.2 was removed from MUAP-08013-P in Revision 2 as the content of this section was replaced by MUAP-11022 "US-APWR Additional Core Inlet Blockage Test."

On February 1, 2010, the staff issued **RAI 530-3989, Question 04.04-33**, asking the applicant if the computer program OLI StreamAnalyzer™ was reviewed and approved by the NRC staff for use in this type of evaluation and to provide a reference for the computer program. In addition, **RAI 530-3989 Question 04.04-33** also asked if Al and Zn products were considered in determining predictions of the impact from chemical deposits. In particular, the staff asked the applicant to provide the concentrations of Al and Zn in the containment sump water, to explain how the amounts of Al and Zn in the US-APWR containment were determined, and how the transport of the corresponding products to the core was evaluated. In **RAI 530-3989, Question 04.04-33**, the staff referred to other materials, typically found in the containment and susceptible to chemical reactions including carbon steel, copper, and non-metallic materials, such as paints, thermal insulation (for example, Cal-Sil and fiberglass), and concrete and asked if these additional materials were considered as part of the evaluation of chemical deposits and debris blockage in the core.

In its response to **RAI 530-3989, Question 04.04-33**, dated March 4, 2010, the applicant stated that the qualification of StreamAnalyzer™ for thermodynamic modeling of debris components under post-LOCA conditions was reported in NUREG/CR-6873, "Corrosion Rate Measurements and Chemical Speciation of Corrosion Products Using Thermodynamic Modeling of Debris Components to Support GSI-191." The applicant also explained that the likely precipitate species, considered in the core in Section 4.3.1.3.4, were simply used as a guide when selecting the deposit density and thermal conductivity whereas the deposition rate itself was set equal to the steaming rate times the impurity concentration quantified from the dissolution experiment. Regarding the concentrations of Al and Zn in the US-APWR containment sump water, the applicant responded that these quantities were determined based on the recirculation test results and autoclave test results documented in MUAP-08011-P (R0), “US-APWR Sump Debris Chemical Effects Test Results.” It was also stated that the results were shown in MUAP-08013-P Revision 0 Figure 4.3.1-3, “Impurity Concentrations Trend at Core Inlet.” The amounts of Al and Zn were accounted for by multiplying the volume of the sump water by the corresponding concentrations. During transport from the RWSP to the core, it was assumed that all chemical products were transported without any concentration reduction.

The RAI response also stated that additional materials were listed in MUAP-08006-P Revision 1 Table 3.1-2, “Sump Debris Sources Information” and clarified that materials such as carbon steel, copper, thermal insulation (Nukon™), and concrete were also considered in the debris source. Based on the applicant providing the references which were reviewed in Section 6.2.2 of this SER the staff found the response acceptable, and **RAI 530-3989, Question 04.04-33, is resolved and closed.**

On February 1, 2010, the staff issued **RAI 530-3989, Question 04.04-34**, asking the applicant to provide references for the chemical effect tests that represented the recirculated sump chemical concentrations after a LOCA in the US-APWR as considered in MUAP-08013-P Revision 0 Section 4.3.1.4.2, “Chemical Condition.” In its response to **RAI 530-3989, Question 04.04-33**, dated March 4, 2010, the applicant referred to MUAP-08006-P Revision 1, “US-APWR Sump Debris Chemical Effects Test Plan,” Section 3.1.3, “Test Conditions.” The applicant also explained that this report was considered in the responses to **RAI 45-876,** issued July 31, 2008, and in **RAI 85-1445,** issued October 16, 2008. In addition, it was stated that testing was witnessed by NRC staff on September 17, 2009. Based on the applicant providing the references which were reviewed in Section 6.2.2 of this SER, the staff finds the response acceptable, and **RAI 530-3989, Question 04.04-34, is resolved and closed**.

On February 1, 2010, the staff issued **RAI 530-3989, Question 04.04-35**, asking the applicant if the 50 mils (1.27 mm or 0.05 in) of debris build-up on the cladding, identified in MUAP-08013-P Revision 0 Section 4.2.1.3, “Inputs,” was considered applicable for the location where the long-term chemical deposition could begin accumulating. The staff raised the concern that additional chemical build-up can impede the heat transfer and cause the cladding surface temperature to exceed the acceptable temperature limit and asked the applicant provide a justification as of why this situation could not occur for the US-APWR fuel. In its response to **RAI 530-3989, Question 04.04-35,** dated March 4, 2010, the applicant explained that the parametric study performed with the assumed values for the debris thickness and thermal conductivity was believed to provide conservative result as discussed in the response to **RAI 530-3989, Question 04.04-29**. Therefore, the applicant concluded that the cladding surface temperature would not exceed the acceptable temperature limit even if additional plate out by chemical debris was added to the initial maximum debris thickness of 50 mils (1.27 mm or 0.05 in) because the cladding temperature rapidly decreases during the evaluation period.

In this regard, the RAI response referenced the predicted fuel cladding temperature shown in MUAP-08013-P Revision 0 Figure 4.3.1-4, “LOCA Scale Thickness and Fuel Cladding Temperature.” The staff agrees with this conclusion as the declining heat flux from decay heat becomes a controlling factor for predicting the cladding temperature as more prolonged time periods are considered in order to allow for additional plate-outs to develop on fuel surfaces.

Therefore, the staff considers the provided response acceptable, and **RAI 530-3989, Question 04.04-35, is resolved and closed**.

The staff finds that the applicant that acceptably demonstrated that chemical effects on the fuel rods would not inhibit adequate core cooling. The staff bases this finding on the applicant's analysis which calculates that approximately 390 microns (0.39 mm or 0.015 in) at 30 days (720 hours) after the LOCA which is less than the acceptance criteria and much less than the spacing between fuel pins.

6.3.4.9.4 In-Core Effects

Boric Acid Precipitation

Based on SECY-12-0093 the staff will consider the effects of boron precipitation as part of the in-vessel effects testing, however the review is being addressed in Section 15.6.5 of the staff's review. Section 4.4.1, "Chemical Effect on Boric Acid Precipitation Evaluation" of MUAP-08013-P Revision 4 addresses possible impacts from debris in the core coolant and resulting blockages of core flow paths on the boric acid precipitation process in the core region. US-APWR DCD Tier 2 Revision 3 Section 15.6.5, "Loss-of-Coolant Accidents Resulting from Spectrum of Postulated Piping Breaks within the Reactor Coolant Pressure Boundary," describes the evaluation model and presents the analysis results for boric acid precipitation during long term cooling following a LOCA. According to the US-APWR procedures for boric acid control, the operators are instructed to switch operating DVI lines over to the hot leg injection line (simultaneous reactor vessel and hot leg injection) approximately four hours following a postulated LBLOCA to prevent the boric acid concentration from reaching the precipitation limit. The switchover time is described in US-APWR DCD Tier 2 Revision 3 Section 15.6.5.3, "Core and System Performance."

The issue of boron precipitation is reviewed as part of the SE of US-APWR DCD Tier 2 Revision 3 Section 15.6.5. On March 17, 2011, the staff issued **RAI 719-5352, Question 15.6.5-89**, asking the applicant to demonstrate that fuel blockage at the core inlet will not preclude or adversely impact coolant mixing between the lower plenum and the core and to show that fuel blockage by debris in the top core area will not interfere with downwards coolant penetration into the core region during the core flushing process. The applicant was also asked to discuss effects of fuel blockage by debris in the reactor coolant on the US-APWR boric acid precipitation evaluation. If fluid mixing between the reactor lower plenum and adjacent core regions has been credited in the precipitation analysis, demonstrate that fuel blockage at the core inlet will not preclude or adversely impact coolant mixing between the lower plenum and the core; and show that fuel blockage by debris in the top core area will not interfere with downwards coolant penetration into the core region during the core flushing process. In its response to **RAI 719-5352, Question 15.6.5-89,** dated May 25, 2011 the applicant explained that the limiting scenario for boric acid precipitation is a cold-leg break when the core inlet flow matches the core boil-off rate so that the core practically experiences stagnant flow conditions. The response also explained that the possibility of the core inlet blockage due to debris is discussed in MUAP-080013-P, Revision 0, Section 4.4.1.

The applicant performed several sets of core inlet blockage tests, documented in MUAP-10021-P (R0), "US-APWR Core Inlet Blockage Test," MUAP-11022-P (R0), "US-APWR Additional Core Inlet Blockage Test," and MUAP-12004-P (R0), "US-APWR Core Inlet Blockage Test for Test Conditions with Design Changes in Recirculation Water Flow Path to Refueling

Water Storage Pit." The core inlet blockage tests documentation do not discuss boric acid precipitation.

Therefore, the staff deferred the final acceptance **of RAI 719-5352, Question 15.6.5-89** until the final MUAP-080013-P revision is issued and core blockage test results are taken into account by the applicant. Currently, **RAI 719-5352, Question 15.6.5-89 is being tracked as an Open Item** and a follow-up RAI question was issued as **RAI 861-6062, Question 15.06.05-93** on October 31, 2011. This open item will track the closure of boron precipitation as it relates to core inlet blockage.

As a result of the open item the staff is unable to finalize its conclusion with regard to the effect of Boric Acid Precipitation in the presence of debris generated from a LOCA.

Fuel Swelling and Blockage

The evaluation of swelling and rupture of the fuel rod cladding during design basis LOCAs is required under 10CFR50.46(b)(4). Following a LBLOCA, some of the fuel rods in the core may swell and rupture leaving sharp edges at the rupture locations and a reduced channel flow area. Debris may collect in these channels and at the rough edges of the rupture locations. This blockage might produce hot spots above the blockage location.

Debris that flows in the core coolant path will pass thought the bottom nozzle and, based on the expected location of the swelling and rupture, through several grids. The accumulation of significant debris at the localized rupture location is not easily evaluated, unlike other places where fibers are more likely to gather before the HLSO. Additionally, there would only be a limited number of fuel rod cladding ruptures in the reactor core and rupture is most likely to occur in the highest power fuel rods in the highest power assemblies. Therefore, the applicant concluded there is little possibility that significant blockage will occur due to fuel swelling and fuel rupture in the LBLOCA scenario.

The staff agrees with the process as identified by the applicant to describe how blockage could potentially occur. The staff also notes that no large scale planar balloon-burst is predicted to occur as a result of a LBLOCA, as covered in Section 15.6.5 of DCD Revision 3. Therefore the staff concludes that failed fuel rods as a result of balloon-burst during a LBLOCA will not lead to core blockage to such an extent that core cooling is challenged.

Hot-Leg Injection

The US-APWR design uses ECCS switch over to hot leg injection approximately four hours following the occurrence of a postulated LBLOCA. By that point in time, the RWSP coolant is expected to have been circulated through the ECCS and CSS several times. As a result, particulate and fibrous debris initially present in the coolant is expected to be depleted either due to accumulation on the RWSP strainer surfaces or by sedimentation and trapping in low velocity regions. Therefore, the applicant concluded that the amount of debris entering the core during the hot leg injection mode will be small and the core cooling will not be significantly affected by the debris. In addition, the core flow rate would be maintained high enough to remove decay heat since the core power at HLSO decreases to around one third of that at the time core quench is completed.

For CL breaks during HLSO mode, coolant is injected into the RV from the HL piping and flows down through the core and goes out the break. Debris entrained in the coolant injected from the

HL piping may accumulate in the core, however, core coolability will be maintained since the core flow rate required for decay heat removal decreases and debris is expected to decrease at this switchover time.

To confirm the effect on core coolability during HLSO with CL break with downward flow, three core blockage tests were performed simulating HLSO with CL break as described in MUAP-10021-P, MUAP-11022-P, and MUAP-12004-P. This test was conducted with the maximum debris loads and demonstrated that the pressure drop measured is less than what is required to maintain core flow in HLSO mode. Based on the Core Inlet Blockage testing discussed in Section 6.3.4.9.1 of this SE the staff concludes that the applicant has acceptably demonstrated that sufficient flow will reach the core to remove core decay heat and core coolability is maintained in the event of downstream debris build up.

6.3.5 Combined License Information Items

There are no COL items associated with Section 6.3 given in DCD Tier 2 Section 1.8, "Interfaces for Standard Design," Table 1.8-2, "Compilation of All Combined License Applicant Items for Chapters 1-19.": No additional COL information items need to be included for this area of review.

6.3.6 Conclusions

As a result of the open and/or confirmatory items the staff is unable to finalize its conclusion on Section 06.03 related to containment heat removal system and the effects of accident generated and latent debris.

6.4 Habitability Systems

6.4.1 Introduction

The US-APWR DCD Chapter 6.4 indicates that the MCR habitability systems allow operators to remain safely inside the control room envelope (CRE) and take the actions necessary to manage and control the plant under normal and abnormal plant conditions, including a LOCA. The MCR habitability systems protect operators against a postulated release of radioactive material, natural phenomenon induced missiles, radioactive shine, smoke, and toxic gases. These systems include the following:

- MCR HVAC system (Chapter 9, Section 9.4.1).
- MCR emergency filtration system (part of MCR HVAC system).
- Radiation monitoring system (Chapter 7).
- Radiation shielding (Chapter 12).
- Lighting system (Chapter 9, Section 9.5.3).
- Fire protection system (Chapter 9, Section 9.5.1).

The CRE includes the MCR and is served by the MCR HVAC system during normal and abnormal conditions including MCR smoke purge operations. Personnel occupying the CRE are protected from the respiratory effects and eye irritation of smoke.

6.4.2 Summary of Application

DCD Tier 1: US-APWR DCD Tier 1 addresses the MCR HVAC system and MCR emergency filtration system in Section 2.7.5.1 and Tables 2.7.5.1-1, 2.7.5.1-2, and 2.7.5.1-3. The radiation monitoring system is covered by Section 2.7.6.6 and Tables 2.7.6.6-1 and 2.7.6.6-2; and Section 2.7.6.13 and Tables 2.7.6.13-1 and 2.7.6.13-3. Radiation shielding is addressed in Sections 2.2 and 2.8 and Table 2.8-1. The lighting system is discussed in Section 2.6.6 and Table 2.6.6-1. The fire protection system is covered in Section 2.7.6.9 and Table 2.7.6.9-1 and 2.7.6.9-2.

DCD Tier 2: The applicant has provided a DCD Tier 2 description in Sections 6.0.4 and 6.4, summarized here in part, as follows:

> The application describes that the control room habitability system is the ESF that allows operators to remain safely inside the CRE while taking the necessary actions to manage and control unusual, unsafe, or abnormal plant conditions, including a LOCA. Further, the application identifies that the control room habitability system protects the operators against postulated releases of radioactive material, toxic gases, and smoke, and enables the operators to occupy the CRE safely and for an extended time.
>
> The CRE boundary is shown in Figure 6.4-1. The MCR habitability systems protect operators against a postulated release of radioactive material, natural phenomenon induced missiles, radioactive shine, smoke, and toxic gases. The MCR habitability systems enable operators and technical staff to occupy the CRE safely for the duration of accidents analyzed in Chapter 15, "Transient and Accident Analyses." The application defines the MCR habitability systems as:
>
> - MCR HVAC system (Chapter 9, Section 9.4.1).
> - MCR emergency filtration system (Part of MCR HVAC system).
> - Radiation monitoring system (Chapter 7).
> - Radiation shielding (Chapter 12).
> - Lighting system (Chapter 9, Section 9.5.3).
> - Fire protection system (Chapter 9, Section 9.5.1).
>
> Further, the application states that the CRE includes the MCR and is served by the MCR HVAC system during normal and abnormal conditions, as well as control room smoke purge operations, as described in Chapter 9, Section 9.4.1. Personnel occupying the CRE are protected from the respiratory effects and eye irritation of smoke.
>
> US-APWR DCD Tier 2 Section 6.4 provides the habitability systems design. The safety-related habitability systems support the MCR, which is located in the R/B. The applicant's habitability systems are acceptable if they meet the codes, standards, and regulatory guidance commensurate with the safety function to be performed. Section 6.4.4 of this SER provides the SE of the habitability systems.
>
> TS in US-APWR DCD Chapter 16 address the MCR HVAC system and MCR emergency filtration system in Sections 3.7.10 and B.3.7.10. Sections 3.3.2 and B.3.3.2 addresses MCR radiation and isolation from radioactivity. US-APWR DCD Chapter 14 addresses MCR HVAC system preoperational tests (including Habitability) in Section 14.2.12.1.101.
>
> <u>Design Basis for the CRE</u>

This habitability system provides for MCR operators to remain safely inside the CRE and take actions necessary to manage and control the plant under normal and abnormal plant conditions, including a LOCA. The CRE is contained within a heavy wall, floor, and ceiling that provides shielding for protection against radiation, radiation particles and gasses, smoke, and toxic gasses. The CRE as shown in Figure 6.4-1 “Control Room Envelope” contains the following facility areas:

- Large display panel and operator console.
- Shift technical advisor and supervisor consoles.
- Computer printer and hard copier.
- Operator’s desk.
- Data management and industrial television consoles.
- Diverse human system interface panel.
- Offices for the shift supervisor and clerk.
- Tagging room.
- Operator’s area including a kitchen and restroom.

The design bases for habitability in the CRE include:

- The CRE contains food, water, medical supplies, and sanitary facilities accessible and sufficient to support the physical needs of five plant staff members for six days. The CRE contains the information resources (e.g., technical reference material, monitors, displays, and communications), access to plant monitoring and controls necessary to manage the postulated accidents in Chapter 15.

- Two 100 percent capacity MCR EFUs, including fans, are provided. Each MCR EFUis capable of meeting the control room access and occupancy requirements of Criterion 19 of Appendix A to 10 CFR 50, “General Design Criteria 19,” including the requirements for radiation protection. Either MCR EFU is capable of establishing and maintaining the design positive pressure in the CRE with respect to the surrounding areas to minimize unfiltered in-leakage during emergency operation in the pressurization mode.

- The design of the MCR EFUs is based on ensuring that the radiation dose (TEDE) to MCR operators is well below 10 CFR 50, Appendix A “General Design Criteria 19” requirements (five Roentgen equivalent man [rem] TEDE) while occupying the CRE for the duration of the most severe Chapter 15 accident. The MCR emergency filtration design basis ensures the protection of control room personnel and equipment with an environment satisfactory for extended performance.

- As noted in Chapter 3, the MCR HVAC system is designed to Equipment Class 3, Seismic Category I standards. The CRE is an area of the control room complex in the power block. Accordingly, the CRE is, by definition, the same equipment class and seismic category (e.g., Equipment Class 3, Seismic Category I) as the MCR.

ITAAC: The ITAAC associated with DCD Tier 2 Section 6.4 are given in DCD Tier 1, Section 2.7.5.1.

TS: The TS associated with DCD Tier 2, Section 6.4 are given in DCD Tier 2, Chapter 16, Sections 3.3.2, B3.3.2, 3.7.10, B3.7.10, 5.5.11, and 5.5.20.

Topical Reports: There are no topical reports for this area of review.

Technical Reports: There are no technical reports for this area of review.

US-APWR Interface Issues identified in the DCD: There are no US-APWR interface issues for this area of review.

Site Interface Requirements identified in the DCD: There are no site interface requirements for this area of review.

Cross-cutting Requirements (TMI, USI/GSI, Op Ex): None for this area of review.

RTNSS: There is no RTNSS for this area of review.

10 CFR 20.1406: There are no 10 CFR 20.1406 requirements for this area of review.

CDI: There is no CDI for this area of review.

6.4.3 Regulatory Basis

The relevant requirements of the Commission's regulations for this area of review, and the associated acceptance criteria, are given in Section 6.4, “Habitability Systems,” of NUREG-0800 and are summarized below. Review interfaces with other SRP sections can be found in Section 6.4 of NUREG-0800. These regulations include:

1. GDC 4, “Environmental and dynamic effects design bases,” as it relates to SSCs important to safety being designed to accommodate the effects of and to be compatible with the environmental conditions associated with postulated accidents.

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 maintaining the nuclear power unit in a safe condition under accident conditions and providing adequate radiation protection. A control room shall be provided from which actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions, including LOCAs.

4. 10 CFR 50.34(f)(2)(xxviii), as it relates to evaluations and design provisions to preclude certain control room habitability problems.

5. 10 CFR 52.47(b)(1) which requires that a DC application contain the proposed ITAAC that are necessary and sufficient to provide reasonable assurance that if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the DC is built and will operate in accordance with the DC provisions of the Atomic Energy Act of 1954, and NRC regulations.

SRP 6.4 Acceptance criteria adequate to meet the above requirements include:

1. Control Room Emergency Zone

   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.

   The control room emergency zone should conform to the guidelines of RG 1.196, May 2003, "Control Room Habitability at Light Water Nuclear Power Reactors," and RG 1.197, May 2003, "Demonstrating Control Room Envelope Integrity at Nuclear Power Reactors."

2. Ventilation System Criteria. The ventilation system should include the following design features:

   A. Isolation dampers used to isolate the control zone from adjacent zones or the outside should be low leakage dampers or valves. The degree of leaktightness should be documented in the SAR.

   B. Single failure of an active component should not result in loss of the system's functional performance. All the components of the control room emergency filter train should be considered active components. See Appendix A to this SRP for criteria regarding valve or damper repair.

3. Pressurization Systems. Ventilation systems that will pressurize the control room during a radiation emergency should meet the following criteria:

   A. Systems having pressurization rates of greater than or equal to 0.5 volume changes per hour should be subject to periodic verification (every 18 months) that the makeup is + 10 percent of design value. During plant construction or after any modification to the control room that might significantly affect its capability to maintain a positive pressure, measurements should be taken to verify that the control room emergency zone is pressurized to at least to the value used in the accident analysis relative to all surrounding air spaces while applying makeup air at the design rate.

B. Systems having pressurization rates of less than 0.5 and equal to or greater than 0.25 volume changes per hour should have identical testing requirements as indicated in acceptance criteria 1 above. In addition, at the CP, COL, or standard DC stage, an analysis should be provided (based on the planned leaktight design features) that ensures the feasibility of maintaining the tested differential pressure with the design makeup airflow rate.

C. Systems having pressurization rates of less than 0.25 volume changes per hour should meet all the criteria for acceptance criteria 2 above, except that periodic verification of control room pressurization (every 18 months) should be specified.

4. Emergency Standby Atmosphere Filtration System. Iodine removal for this system should be in accordance with the guidelines of RG 1.52. For new applications, the system should also conform to ASME Code AG-1, "Code on Nuclear Air and Gas Treatment," including the AG-1a-92 Addenda (Reference 14). Protection of control room personnel from releases of chlorine or other toxic gases is addressed in RG 1.78 as discussed in the criteria below.

5. Relative Location of Source and Control Room. The control room inlets should be located with consideration of the potential release points of radioactive material and toxic gases. Specific criteria as to radiation and toxic gas sources are as follows:

A. Radiation sources. As a general rule the control room ventilation inlets should be separated from the major potential release points by at least 31 meters (100 feet) laterally and by 16 meters (50 feet) vertically. However, the actual minimum distances should be based on the dose analyses (Reference 9).

B. Toxic gases. The minimum distance between the toxic gas source and the control room is dependent upon the amount and type of the gas in question, the container size, and the available control room protection provisions. The acceptance criteria for the control room habitability system are provided in the regulatory positions of RG 1.78 with respect to postulated hazardous chemical releases in general.

6. Radiation Hazards

A. Applicants for and holders of construction permits and operating licenses under 10 CFR Part 50 who apply on or after January 10, 1997, applicants for DCs under 10 CFR Part 52 who apply on or after January 10, 1997, applicants for and holders of COLs under 10 CFR Part 52 who do not reference a standard DC, or holders of operating licenses using an alternative source term under 10 CFR 50.67, shall meet the requirements of GDC 19, except that with regard to control room access and occupancy, adequate radiation protection shall be provided to ensure that radiation exposures shall not exceed 0.05 Sv (5 rem) TEDE as defined in 10 CFR 50.2 for the duration of the accident.

7. Toxic Gas Hazards.

Three exposure categories are defined: protective action exposure (two minutes or less), short-term exposure (between two minutes and one hour), and long-term exposure (one hour or greater). Because the physiological effects can vary widely from one toxic gas to another, the following general restrictions should be used as guidance: there should be no chronic effects from exposure; acute effects, if any, should be reversible within a short period of time (several minutes) without benefit of any measures other than the use of self-contained breathing apparatus. The allowable limits should be established on the basis that the operators should be capable of carrying out their duties with a minimum of interference caused by the gas and subsequent protective measures. The limits for the three categories normally are set as follows:

A. Protective action limit (two minutes or less): Use a limit that will ensure that the operators will quickly recover after breathing apparatus is in place. In determining this limit, it should be assumed that the concentration increases linearly with time from zero to two minutes and that the limit is attained at two minutes.

B. Short-term limit (two minutes to one hour): Use a limit that will ensure that the operators will not suffer incapacitating effects after a one hour exposure.

C. Long-term limit (one hour or greater): Use a limit assigned for occupational exposure (40-hour week).

The protective action limit is used to determine the acceptability of emergency zone protection provisions during the time personnel are in the process of fitting themselves with self-contained breathing apparatus. The other limits are used to determine whether the concentrations with breathing apparatus in place are applicable. They are also used in those cases where the toxic levels are such that emergency zone isolation without use of protective gear is sufficient. Self-contained breathing apparatus for the control room personnel (at least 5 individuals) should be on hand. A six hour onsite bottled air supply should be available with unlimited offsite replenishment capability from nearby location(s). As an example of appropriate limits, the following are the three levels for chlorine gas:

protective action:	15 ppm by volume
short-term:	4 ppm by volume
long-term:	1 ppm by volume

RG 1.78 provides a partial list of protective action levels for other toxic gases.

6.4.4 Technical Evaluation

The control room ventilation system and control building layout and structures, as described in the applicant's DCD, are reviewed to ensure that plant operators are adequately protected against the effects of accidental releases of toxic and radioactive gases and to assure conformance with the requirements of GDC 4, 5, and 19, and of 10 CFR 50.34 (f)(2)(xxviii), 10 CFR 52.47(b)(1), and 10 CFR 52.80(a). Additionally, review is performed to ensure that the control room can be maintained as the backup center from which technical support center personnel can safely operate in the case of an accident. These objectives are accomplished by the following:

- Confirming that all critical areas serviced by the control room emergency ventilation system are accessible and those areas not requiring access are excluded.

- Confirming that control room design includes analysis for the expected capacity and includes methods coping with rising CO2 levels or other habitability issues.

- Confirming that control room ventilation design flow rates and filter efficiencies can mitigate buildup of radioactive or toxic gases.

- Confirming that layout and physical location of the control room would not promote contamination from hazardous airborne materials.

- Confirming that radiation streaming into the control room is minimized or shielded against.

- Confirming that independent analyses are performed to determine the radiation doses and toxic gas concentrations.

- Reviewing proposed ITAAC to ensure that all SSCs in this area of review are identified and addressed as appropriate in accordance with Section 14.3 of NUREG-0800.

The staff reviewed the habitability systems in US-APWR DCD Tiers 1 and 2 information related to DCD Section 6.4 in accordance with SRP Section 6.4, “Control Room Habitability System.” The applicant’s habitability systems are acceptable if they meet the codes, standards, and regulatory guidance commensurate with the safety function to be performed. This will ensure conformance with applicable requirements of 10 CFR 50, Appendix A, and that the relevant requirements of GDCs 4, 5, and 19, 10 CFR 50.34(f)(2)(xxviii) and 10 CFR 52.47(b)(1) are met. Discussed below are the technical evaluations of these requirements.

6.4.4.1 Summary of RAI Exchange

As part of the staff’s evaluation during the DCD review process, 24 RAIs were prepared and sent to the applicant. Many of the RAI questions are given below, along with the applicant’s responses (ML082670703, letter dated September 16, 2008, and ML091700682, letter dated June 17, 2009), and the staff evaluation for disposition of the RAIs. A few of the RAIs including, **RAI 46-895, Question 06-04 (Numbers 06.04-1, 06.04-3, 06.04-13, 06.04-14, 06.04-22, and 06.04-23) were closed without discussion in the SER** as they were below the level of detail required necessary to be included in the technical evaluation. The **staff omitted discussion of RAI 46-895, Question 06-04 (06.04-18) due to redundancy of the issue.**

There were several follow-up RAIs initiated from the twenty-four questions associated with **RAI 46-895**. These follow-up RAIs are identified (with ADAMS accession numbers) in the following SER discussion associated with the original **RAI 46-895** issue. The staff also issued **RAI 827-5812, Question 09.04.01-28 (ML12240A183); RAI 927-6460, Question 06.04-16; RAI 442-3378, Question 09.04.01-10 (ML092650173) and RAI 473-3801, Question 06.04-9 (ML093210471)** as salient concerns later in the review process.

6.4.4.2 Conformance with the Regulatory Basis and Guides

GDC 4 – "Environmental and dynamic effects design bases"

The staff reviewed the habitability system to ensure that the relevant requirements of GDC 4 are met by accommodating the effects of and being compatible with the environmental conditions associated with postulated accidents.

The staff evaluated the applicant's DCD Section 6.4 conformance with GDC 4 and the relevant guidance of SRP 6.4. The staff initiated **RAI 46-895, Question 06.04-8 and -11,** to address these requirements. **RAI 46-895, Question 06.04-8** addressed the concern about the MCR and EFU and AHU rooms being flooded by a deluge of water released within the building from above due to a failure in the main steam/feedwater pipe, EFWP, or from fire fighting activities. **RAI 46-895, Question 06.04-11** addressed a concern about whether or not the MCR smoke-purge mode of operation was considered an abnormal condition.

In **RAI 46-895, Question 06.04-8** (ML082670703), the staff cited SRP 6.4, Section III.5.C and requested additional information about pressurized piping that might cause a pressure gradient and impact safety-related SSCs in plant areas below should the piping fail. In particular, the staff noted that DCD Figure 6.4-6 displays the "Main Control Room Emergency Filtration System Sectional View." This figure displays the main steam and feed water valve (MSFV) room directly above the EFU and AHU rooms, which in turn is located directly above the MCR. The staff asked the applicant to provide additional information on the assessment for potential steam/water boundary failure leaks from the MSFV room migrating to the EFU and AHU rooms and MCR room below. The applicant responded that DCD Subsection 3.6.1.2.3.2 shows no impact to the MCR by any of the effects of a postulated break of main steam piping. The applicant indicated that a watertight door is installed on the access opening to the MS/FW piping room and the floor drain in the MS/FW piping room is directly connected to the turbine building sump.

Subsequently, the staff and applicant exchanged a series of RAIs and responses pertaining to the potential of internal flooding impacting the MCR HVAC system EFUs and AHUs, the RSC, Class 1E I&C and UPS room doors and the MCR entrances. The exchanges identified that besides the MSFV room other principle sources of internal flood threats to the MCR HVAC system and control room envelope include the EFWPs and water from firefighting operations. The exchanges included **RAI 338-2325, Question 06.04-8** (ML091700682) and **RAI 501-4004, Question 06.04-10** (ML100260131). The applicant indicated that the EFWPs are Seismic Category I and the pit man-way hatch has a watertight door for preventing water flow-out caused by earthquake sloshing.

The staff noted that in DCD Section 3.4.1.5, "Evaluation of Internal Flooding," the analysis indicated that the corridor area in front of the MCR and Class 1E I&C and UPS room doors could be vulnerable to floodwater up to nearly three feet. The applicant indicated that the maximum flooding event assumed for the floor locations of the MCR and AHU is water discharged from firefighting activities in corridors. The design of the MCR watertight doors are includes a specification to resist firefighting water. The staff expressed concerns about MCR egress/ingress via watertight doors during both normal and abnormal plant operations. The staff also inquired about how these doors would be arranged with respect to the control room envelope boundary and whether the doors would impact tracer gas testing. This residual concern was the subject of **RAI 559-4387, Question 06.04-11**.

On May 20, 2010 (ML101450224), the applicant responded with an alternate remedy. The applicant proposed to use barriers to prevent the possibility of fire fighting water accumulating outside the CRE doors. In its response, the applicant provided a revision to DCD Subsection 3.4.1.5.2.2 "NRCA" and DCD Figure 3K-5 to indicate the use of flood barriers at each MCR vestibule. In the same RAI response, the applicant provided resolutions of two other staff concerns pertaining to internal flooding. In particular, the applicant's review of the floor design of MCR emergency filter units determined that the units have a steel frame base installed on the top of the concrete foundations. The additional height of this base results in a total of 1.5 feet between the floor level and the filtration units. Therefore the applicant concluded that, when considering the steel frame base units, the current design has sufficient margin (i.e., 0.63 feet above the postulated flood level) to protect against the postulated flooding. The applicant provided a revision to DCD Subsection 3.4.1.5.2.2 to reflect this design provision. In addition, the applicant indicated that the design of the MCR penetrations prevent water from flowing into the MCR by applying appropriate sealing features. With respect to the issue of HVAC duct work, the applicant responded that the HVAC ducts coming from the MCR AHUs and the filter train units would be routed horizontally above the postulated flooding level. The vertical HVAC ducts penetrate the MCR ceiling and would be welded to embedded sleeves for penetration isolation. The HVAC duct sections of concern and the embedded sleeves would be designed to withstand the hydrostatic load of flooding. The applicant provided a revision to DCD Subsection 3.4.1.5.2.2 to reflect these design provisions. The staff verified that DCD Revision 3 Subsection 3.4.1.5.2.2 contained these changes. Since these changes satisfy the requirements of GDC4, the staff found these resolutions acceptable.

However, the staff noted that DCD Revision 3 Figure 3K-5, "Location of Watertight Doors and Flood Barrier Walls R/B Plan View Elevation 25'-3," did not reflect the use of flood barriers at the two entrances to the MCR in lieu of the use of water tight doors. **RAI 841-6055, Question 03.04.01-30,** was initiated to address this residual issue. Accordingly, the staff holds the following RAI series as an **Open Item: RAI 46-895, Question 06-04 (06.04-8); RAI 338-2325, Question 06.04-8; RAI 501-4004, Question 06.04-10; RAI 559-4387, Question 06.04-11; and RAI 841-6055, Question 03.04.01-30.**

In **RAI 46-895, Question 06-04 (06.04-11)**, the staff invoked the SRP 9.4.1 "Control Room Area Ventilation System" Section III.1 guidance that a review of normal and emergency operations and the ambient temperature limits for the areas serviced is required. The staff observed that DCD Table 9.4-1, sheet one, does not display the smoke-purge mode of operation as an abnormal condition like LOOP or SBO. The staff requested additional information about the habitability systems smoke-purge mode of operation concerning the issue.

In its response to **RAI 46-895, Question 06-04 (06.04-11)**, (ML082670703), the applicant stated in part that "…The smoke purge portion of the MCR HVAC system outside the CRE and downstream of the safety-related isolation damper at the wall of the CRE does not serve any safety-related function and has no safety design bases. Therefore, DCD Table 9.4-1 does not have to include smoke purge mode of operation under abnormal conditions."

The applicant agreed to revise DCD Section 6.4.2, “System Design,” to describe the function of the smoke-purge mode portion of the system. Based on the proposed amendment, the staff found the applicant’s response acceptable. The staff verified that DCD Section 6.4.2, Revision 3 contained the amendment. The staff **closed RAI 46-895, Question 06-04 (06.04-11).**

The staff’s GDC 4 Concluding Statement

Due to the existence of the open item associated with **RAI 841-6055, Question 03.04.01-30**, the staff cannot conclude that the US-APWR plant’s Habitability Systems satisfies the guidance of SRP 6.4 and the requirements of 10 CFR 50 Appendix A Criterion 4.

GDC 5 – "Sharing of structures, systems and components"

The staff reviewed the habitability system to ensure that the relevant requirements of GDC 5 are met. Since the design application is for a single plant, GDC 5 does not apply to this system.

GDC 19 – “Control room”

The staff reviewed the habitability system to ensure that DCD Section 6.4 addressed the relevant requirements of GDC 19. In particular, the staff’s reviewed Section 6.4 to ensure the maintenance of a suitable environment in the control room for occupancy during normal and accident conditions by meeting the guidelines of RG 1.78. Since toxic gas threats to MCR habitability relate to site characteristics, this specific part of the evaluation will be completed at the COL stage of review.

6.4.4.3 Design of the Emergency Filter Units for the MCR HVAC System

The staff evaluated the design of the EFUs as described in DCD Sections 6.4 “Habitability Systems” and 9.4.1”Main Control Room Heating, Ventilation and Air Conditioning System” for conformance with the guidance of SRP 6.4. **RAI 46-895, Questions 06-04 (06.04-4, -5 and -16)** addressed this SRP guidance. **RAI 46-895, Question 06-04 (06.04-4)** addressed the MCR EFU design for: (1) electric heating coil capacity; (2) fan airflow rate; and (3) HVAC system isolation dampers closure time. **RAI 46-895, Question 06-04 (06.04-5)** pertained to the design of the charcoal bed adsorbers and the beds’ potential to ignite and burn due to radioactive material buildup during a LOCA. **RAI 46-895, Question 06-04 (06.04-16)** inquired about the EFU design airflow rate and the flow-rate capacity of the HEPA filters.

In **RAI 46-895, Question 06-04 (06.04-4)**, the staff invoked the review guidance of SRP 6.4, Sections II and III. The staff requested additional details about calculations, including assumptions and margins used in the design of DCD Section 6.4 SSCs. In particular, the staff inquired about calculations used to establish the MCR EFUs electric heating-coil capacity, fan airflow rate, and HVAC system isolation dampers closure time as described in DCD Table 6.4-1 “Main Control Room Emergency Filtration System – Equipment Specifications”. The staff requested this additional information to ensure adequate humidity control of the airflow to the EFU adsorbent beds and to ensure that the component design parameters were consistent with the dose analyses.

In its response to **RAI 46-895, Question 06-04 (06.04-4**), the applicant stated that the MCR EFU electric heating-coil is designed to reduce relative humidity of filtration air from 100 percent RH to 70 percent RH. The results of dose analysis show that the MCR EFU fan airflow rate is sufficient. The damper closure-time parameter comes from existing plant experience and becomes the requirement for the isolation dampers of the MCR HVAC system.

After review of the parameters used in the MCR EFU design calculation, the staff found the information presented for fan airflow rates and electric heating-coil capacity acceptable. However, the staff did question the absence of a limiting closing time for the HVAC system isolation dampers. As displayed in DCD Table 3.2-2 “Classification of Mechanical and Fluid Systems, Components, and Equipment**”** these dampers are Equipment Class 3, Seismic I components and are safety related. The staff in its review of DCD Chapter 15 could not locate a closure time for the MCR isolation dampers. The staff found that this stood in contrast with the applicant’s response, which indicated that this parameter directly affected dose analysis.

The staff found the applicant's overall response incomplete and requested further information in **RAI 338-2325, Question 06.04-4,** for the required stroke times, leakage criteria, and code or standard to which the dampers were being designed.

In its response **to RAI 338-2325, Question 06.04-4**, the applicant stated that the MCR HVAC system isolation dampers are designed to close with a maximum stroke time of 10 seconds. DCD Table 6.4-1 lists this stroke time. The maximum opening time is also required to be 10 seconds. The applicable code for the MCR HVAC system isolation dampers is ASME AG-1-2003. ASME AG-1-2003, Section DA defines the leakage class for seat and frame of dampers. Maximum frame leakage rate-Class A (Article DA 5000, paragraph DA-5130 and Appendix DA-1). Maximum seat leakage rate-Class 0, Bubble Method (Article DA 5000, paragraph DA-5141). The staff found the applicant's response acceptable because Tier 2 Table 15.6.5-5 “US-APWR Major Input Parameters Used in the MCR and TSC Consequence Analysis for the LOCA” lists the “Time delay to switch from normal operation to emergency CRE air filtration mode (s)” as 180 seconds. Since the dose analysis for CRE habitability assumes 180 seconds for system re-alignment and since 10 seconds fits well within this limiting time bounds, the staff **closed RAI 46-895, Question 06-04 (06.04-4) and RAI 338-2325, Question 06.04-4.**

In **RAI 46-895, Question 06-04 (06.04-5),** the staff cited that Criterion 19 of 10 CFR 50 Appendix A requires provision of a control room from which actions can be taken to maintain the plant in the safe state under accident conditions, including LOCAs. The staff requested that the applicant provide additional information about the design of the charcoal adsorber components. In particular, the staff requested additional information as to how the adsorber design addresses and prevents the potential for the charcoal to ignite and burn, particularly when exposed to radioactive material during a LOCA.

The applicant’s response, dated September 16, 2008 (ML082670703), referenced RG 1.52, Section 4.10 and read in part that:

> …In US-APWR, ignition and burning of the charcoal bed is mitigated by a fixed water spray system designed into the charcoal filter unit enclosure. The water spray is directed at the charcoal bed(s).
>
> Subsection 9A.3.52 and 9A.3.53 of DCD Revision 1 reads ‘*A fixed water suppression system and automatic fire detection is provided for the charcoal filter in MCR filter unit.’* If an MCR emergency filtration unit starts alarming due to a fire scenario, redundancy of the MCR EFUs allows the affected unit to be isolated and the other unit to take over operation. …

The staff found the applicant's RAI response acceptable since the design of the fixed water suppression system satisfies the single-failure criterion in that a redundant adsorber bed can take over emergency filtration operation. The staff verified that for Fire Areas FA2-405 and FA2-406 Revision 3 of the DCD contains the passage: "*A fixed water suppression system and automatic fire detection is provided for the charcoal filter in MCR filter unit.*" The applicant's response fully resolved the staff's concerns of **RAI 46-895, Question 06-04 (06.04-5)**. Based on this the staff **closed RAI 46-895, Question 06-04 (06.04-5).**

In **RAI 46-895, Question 06-04 (06.04-16),** the staff asked the applicant about an apparent mismatch for flow rates listed in DCD Table 6.4-1. The staff noted that DCD Table 6.4-1 listed a flow rate of 2,000 scfm for the HEPA filter and a design airflow rate of 3,600 scfm for the fans of the MCR EFUs.

In its responses to **RAI 46-895, Question 06-04 (06.04-16)**, dated September 16, 2008, and a subsequent **RAI 338-2325** response, dated June 17, 2009, the applicant explained that the two HEPA filters (i.e. each flow rated at 2000 cfm) are installed in parallel in the MCR EFU HEPA filter assembly for a total capacity of 4000 cfm. Therefore, the 3600 cfm supplied by the MCR EFU fan does not exceed the HEPA filters combined airflow capacity. The applicant agreed to revise DCD Table 6.4-1 with a note to explain the mismatch between HEPA filter flow rate and the fan design airflow rate. Based on the review of this amendment to the DCD, the staff found the applicant's response to **RAI 338-2325, Question 06.04-5** acceptable. The staff verified that Revision 3 of the US-APWR DCD Table 6.4-1 contained the change of the latter RAI response. Based on this, the staff **closed RAI 46-895, Question 06-04 (06.04-16), and RAI 338-2325, Question 06.04-5.**

6.4.4.4 Control Room Envelope Integrity

The staff evaluated DCD Section 6.4 for conformance with SRP 6.4 guidance with respect to ensuring CRE integrity during all modes of plant operations. **RAI 46-895, Question Numbers 06-04 (06.04-2 and -6),** pertained to this SRP guidance. RAI **46-895, Question 06-04 (06.04-2),** addressed the need for conformance to RGs 1.196 and 1.197. **RAI 46-895, Question 06-04 (06.04-6),** addressed design provisions for maintaining door use accessibility and control room occupancy in the MCR during all MCR modes of operation. **RAI 473-3801, Question 06.04-9,** concerned unfiltered air in-leakage to the CRE, particularly due from ingress and egress to the CRE through the vestibule.

The staff noted that DCD Section 6.4 does not reference or address RGs 1.196 and 1.197 in accordance with SRP 6.4, SRP Acceptance Criteria Section 1.E. In **RAI 46-895, Question 06-04 (06.04-2),** the staff requested that the applicant provide additional information to clarify where DCD Section 6.4 addresses these two regulatory guides.

In its response to **RAI 46-895, Question 06-04 (06.04-2),** dated September 16, 2008, the applicant agreed that the DCD needed to document compliance with RG 1.196 and RG 1.197. The applicant amended DCD Sections 6.4.1, "Design Basis," 6.4.8 "References," and 1.9.1 (i.e., Table 1.9.1-1, "US-APWR Conformance with Division 1 Regulatory Guides") with changes that captured within the DCD a US-APWRs licensing basis commitment to the guidance of RG 1.196 and RG 1.197. The "Status" column for each RG 1.196 and RG 1.197 in Table 1.9.1-1 now reads, "Conformance with no exceptions identified." While the applicant's response did not document in significant detail how and where compliance was in evidence in the DCD, other RAI exchanges as captured in this SER section provided the staff with assurance of compliance to the guidance of RG 1.196 and RG 1.197. Accordingly, the staff found the applicant's response to **RAI 46-895, Question 06-04 (06.04-2)** acceptable. The staff verified that the applicant incorporated into Revision 3 of the US-APWR DCD, the concise changes of the RAI response. Based on this, **the staff closed RAI 46-895, Question 06-04 (06.04-2).**

In **RAI 46-895, Question 06-04 (06.04-6),** the staff inquired about the design of the two entrances into the CRE as displayed in DCD Figure 6.4-1, "Control Room Envelope." The staff noted that no airlock, vestibule, or other design detail was shown at the doors to maintain a proper air balance and control room environment for accessibility and occupancy when the doors are opened, particularly during periods of emergency pressurization, emergency isolation, and smoke purging operation modes. The staff cited that SRP Section III.3.C addresses system layout diagrams and single failure criteria to mitigate the quantity of unfiltered air that enters the control room and to mitigate the effects on the air balance. The staff requested that the applicant provide additional details about design and the procedural provisions employed to maintain door use accessibility and control room occupancy that meets SRP 6.4 Acceptance Criteria during all MCR operating modes.

In its response to RAI 46-895, dated September 16, 2008, the applicant responded that:

> "The MCR of US-APWR has a vestibule at the two access doors to the CRE. DCD Figure 1.2-18 shows a short corridor at both entrances to the CRE. A door is installed at the end of the short corridor leading to the main corridor. Therefore, the short corridors will serve as vestibules.
>
> In access by the MCR personnel, the air balance of the MCR will be maintained by this vestibule during the emergency pressurization mode of operation.
>
> During the emergency isolation mode of operation, when there is no positive pressure in the CRE, the access doors will be administratively controlled to prevent there being opened during the event.
>
> The CRE may be under negative pressure during the smoke purge mode of operation since it is necessary to purge smoke in the CRE. Therefore, there are no concerns about the door being opened during the smoke purge mode of operation."

In its response to RAI 46-895, dated September 16, 2008, the applicant included an amendment to DCD Figure 6.4-1 that displays the vestibule and doors at each access point to the CRE. The staff confirmed that US-APWR DCD, Revision 3, contained this amendment. As a result of follow-up **RAI 559-4387, Question 06.04-12**, the applicant also included in DCD Revision 3 Section 6.4.2 words that ensure the CRE access doors will be administratively controlled during the emergency isolation mode of operation. Based on these DCD changes, the staff found the applicant's composite response complete and acceptable. The **staff closed both RAI 46-895, Question 06-04 (06.04-6) and RAI 559-4387, Question 06.04-12.**

In **RAI 473-3801, Question 06.04-9,** the staff requested that the applicant provide additional information about unfiltered air in-leakage to the CRE. The staff noted that DCD 6.4.2.3 "Leaktightness" reads that a total system leakage of ~120 cfm in the pressurization mode will be confirmed by ASTM E-741 testing. In contrast, the staff observed that the total control room unfiltered in-leakage assumed in the DBA dose analyses is 120 cfm. DCD Section 15.6.5.5.1.2 "Main Control Room Consequence Model" and Tables 14.3-1f, "Radiological Analysis Key Design Features" state that the assumed control room unfiltered in-leakage of 120 cfm is due to ingress/egress through the vestibule entrance. The staff requested that the applicant clarify how much of the 120 cfm of control room unfiltered in-leakage is attributed to in-leakage through the CRE and subject to the ASTM E-741 testing in the TS 5.5.20 “Control Room Envelope Habitability Program” and how much (i.e. of the 120 cfm) is an analysis assumption for vestibule ingress/egress.

In its response to **RAI 473-3801, Question 06.04-9,** dated November 13, 2009, the applicant agreed to revise the DCD to reflect that the total amount of unfiltered in-leakage into control room is 120 cfm. This total value includes the in-leakage through the CRE and unexpected in-leakage through the CRE such as through ingress to and egress from doors. The applicant assumes consistent with RG 1.197 that 10 cfm is the leakage due to ingress to and egress from CRE. In summary, the applicant agreed to revise DCD Section 6.4.2.3 to remove the contradiction from the DCD.

The staff found the applicant’s response acceptable since the DCD amendment would fully resolve the staff’s concern. The staff verified that Revision 3 DCD Section 6.4.2.3 contained the amendment. Based on this, **the staff closed RAI 473-3801, Question 06.04-9.**

6.4.4.5 Control Room Atmosphere (Habitability)

The staff evaluated DCD Section 6.4 for conformance with SRP 6.4 guidance with respect to ensuring the CRE’s atmosphere remains habitable during all modes of plant operations. **RAI 46-895, Question 06-04 (06.04-9, -10, -15, and -19),** addressed this SRP guidance. With **RAI Question 06-04 (06.04-9)**, the staff inquired about how the gaseous fire suppression material used beneath the MCR floor (i.e. housing plant cables) satisfied RG 1.189, “Fire Protection for Nuclear Power Plants.” **RAI Question 06-04 (06.04-10)** addressed a concern about iodine protection factors and zone isolations as covered in SRP 6.4 Section III.3.D and E. **RAI Question 06-04 (06.04-15)** inquired about the CRE capacity in terms of the number of people an isolated MCR can accommodate for an extended period of time before the carbon dioxide levels became excessive. **RAI Question 06-04 (06.04-19)** addressed the concern about the newer chiller refrigerants, which are much more toxic and chemically and thermally reactive than the older refrigerants, finding pathways to the CRE.

In **RAI 46-895, Question 06-04 (06.04-9),** the staff noted that DCD Section 6.4.2.4, “Interaction with Other Zones and Pressure-Containing Equipment,” reads: “There are no pressure-containing tanks or piping systems in the CRE that could on failure, transfer or introduce hazardous material into the CRE (with exception of installed gaseous fire suppression in the cable spreading area below the floor).” In apparent conflict with this excerpt, DCD Section 9.5.1, “Fire Protection Program,” Table 9.5.1-1, “US-APWR Fire Protection Program Conformance with RG 1.189,” Position Number 6.1.2, under remarks states: “The MCR raised-floor compartment is also provided with a fire suppression system that discharges an environmentally friendly clean fire extinguishing agent that does not present a hazard to MCR personnel.”

The staff asked the applicant to provide additional information to clarify whether or not the MCR cable spreading area beneath the floor uses hazardous fire suppression material. The staff also requested that the applicant provide an accurate assessment of the fire suppression material. In particular, the staff inquired about how the material meets the guidance of RG 1.189 for use in the MCR envelope (MCRE) area.

In its response to **RAI 46-895, Question 06-04**, dated September 16, 2008, the applicant stated that: DCD Subsection 9.5.1.2.5 states, "Halon and carbon dioxide total flooding systems are not used; however, a clean agent gaseous fire suppression agent in conjunction with very early warning fire detection is used for selected areas with heavy cable fire loading." The applicant went on to note that DCD Subsection 9.5.1.2.5, "Automatic Extinguishing Systems," further states:

> The US-APWR employs several gaseous fire suppression systems in select critical plant areas with heavy fire loading or raised-floor compartments where access for fire fighting may be difficult. For each area where a total flooding gaseous fire suppression system is identified, an environmentally-friendly fire suppression clean agent is used (Novec® 1230 fluid in a 5.6% concentration for cable raised-floor areas, or equal). In conjunction with the gaseous system, an air aspirating, very early warning fire detection system (VESDA® or equal) is used to provide notification of a fire. Such an early notification provides a defense-in-depth fire protection approach for these areas which helps assure adequate fire safety for the areas.

In addition, the applicant noted that DCD Section 9.5.1.2, "System Description," provides references in Table 9.5.1-1, which is a point-by-point comparison of the conformance of the US-APWR fire protection program with the guidelines of RG 1.189, Revision 1.

To eliminate the point of confusion, the applicant agreed to amend the DCD by removing the words "with the exception of installed gaseous fire suppression in the cable-spreading area below the floor" from the last sentence of DCD Section 6.4.2.4. The staff verified that DCD Revision 3 included this change. The applicant made changes to DCD Section 6.4.2.4 that fully resolved the staff's concern. The staff found the applicant's response to **RAI 46-895, Question 06-04 (06.04-9),** acceptable since the amended DCD conforms to the guidance of RG 1.189 with respect to the subject issue. Based on this, the staff **closed RAI 46-895, Question 06-04 (06.04-9).**

In **RAI 46-895, Question 06-04 (06.04-10),** the staff noted, that DCD Section 6.4 does not define or discuss iodine protection factors (IPF) and zone isolation as covered in SRP 6.4 Sections III.3.D and III.3.E.i, ii, and iii. The staff requested that the applicant provide additional information about this absent information.

In its response to **RAI 46-895, Question 06-04 (06.04-10)**, the applicant stated that

> "As noted in SRP 6.4 Section III.3.D, the IPF methodology described in the "Murphy-Campe paper" (Reference 9 of SRP 6.4) models a steady-state control room condition. However, as noted in DCD Section 15.6.5, the time-dependent analysis program RADTRAD was used for control room evaluation of a LOCA and, per SRP 6.4 Section III.3.D, IPF methodology does not apply.

With regards to zone isolation, US-APWR utilizes an SRP 6.4 Section III.3.E.iii, "Zone isolation with filtered recirculated air and a positive pressure" type of system. Associated input parameters used in the MCR Consequence Analysis for the LOCA have been provided in Table 15.6.5-5, Chapter 15."

After review of SRP 6.4 Section III.3.E.iii and the parameters of Table 15.6.5-5, the staff concurred with the applicant's response. Because this is consistent with the guidance, the staff found it acceptable, and **closed RAI 46-895, Question 06-04 (06.04-10).**

In RAI **46-895, Question 06-04 (06.04-15),** the staff invoked the guidance of SRP 6.4 Section I.2, and requested additional information regarding the capacity of an isolated CRE in terms of the number of people it can accommodate for an extended period of time before carbon dioxide levels become excessive.

In its response to RAI 46-895, Question 06-04 (06.04-15), the applicant cited SRP 6.4 Section III.2, which reads, "A control room designed with complete isolation capability from the outside air to provide radiation and toxic gas protection is reviewed to determine if the buildup of carbon dioxide could present a problem. The air inside a 2830 m^3 (100,000 $feet^3$) control room would support five persons for at least 6 days. Thus, carbon dioxide buildup in an isolated emergency zone is not normally considered a limiting problem." The applicant noted that the CRE volume of US-APWR is approximately 140,000 ft^3 (3965 m^3), as stated in DCD Table 15.6.5-5. Based on this, the applicant concluded that the carbon dioxide buildup in an emergency isolation mode is not a limiting problem for the US-APWR.

The applicant agreed to amend DCD Section 6.4.1, "Design Bases," with the following sentence, "The CRE volume is approximately 140,000 ft^3, which exceeds 100,000 ft^3. The air inside the CRE can support five persons for at least 6 days. Therefore, the carbon dioxide buildup in emergency isolation mode is not considered a limiting problem."

This amendment is contained in DCD Revision 3 and fully resolves the staff concern since the DCD complies with the guidance of SRP 6.4 Section I.2. Based on this, the staff **closed RAI 46-895, Question 06-04 (06.04-15).**

In **RAI 46-895, Question 06-04 (06.04-19),** the staff invoked the review guidance per SRP 6.4 Section III.5.C. In particular, Section III.5 reads: "…the organization responsible for ventilation and air filtration for its toxic gas estimates for use in the control room habitability analysis. There are three basic categories: Radioactive sources, toxic gases such as chlorine, and gases with the potential for being released inside confined areas adjacent to the control room."

The staff requested that the applicant provide additional information with respect to gases (e.g., fire fighting materials, carbon dioxide, chiller refrigerants, etc.) with the potential for release inside confined areas adjacent to the CRE. In particular, the staff requested that the applicant provide additional information as to whether the existing design will house any sources of gases in the areas adjacent to the CRE.

In its response to **RAI 46-895, Question 06-04 (06.04-19)**, (ML082670703), the applicant stated that:

> "There is no asphyxiation hazard associated with the control room atmosphere due to a potential release of refrigerants in areas adjacent to the control room, because of the remote location and structural barriers between the refrigerant and the control room air inlets. Essential chiller units are located on B1 F in the Power Source Building. The non-essential chiller units are located on 3F in the Auxiliary Building. There are no refrigeration units used in the control room equipment."

The applicant agreed to amend the last sentence of first paragraph Section 6.4.4.2, "Toxic Gas Protection," as follows: "*The designated storage areas of hazardous chemicals as recommended by RG 1.78 are sited at distances greater than 330 feet from the MCR or the fresh air inlets shown in Figures 6.4-5 and 6.4-6. <u>There is no asphyxiation hazard associated with the MCR atmosphere in areas adjacent to the CRE</u>.*"

The staff found the applicant's response to **RAI 46-895, Question 06-04 (06.04-19)** as incomplete and requested additional information with **RAI 338-2325, Question 06.04-6**. The staff noted that chillers that use the new HCFC and HFC refrigerants are of a particular concern. The new refrigerants can be toxic. In the event of a large release of the new refrigerants that may result from operator error or a chiller refrigerant pressure boundary leak, the danger to personnel due to potential asphyxiation from air displaced by the refrigerant, refrigerant toxicity, and potential chemical reactions can be detrimental. The staff noted that HCFC and HFC refrigerants breakdown when exposed to heat and can create hydrofluoric and hydrochloric acid fumes when combined with water or moist air. The staff requested that the applicant conduct further review and analysis to address all the above issues.

In its response to RAI 338-2325, dated June 17, 2009, the applicant indicated that the Equipment Room in the Auxiliary Building containing the Non-Essential chiller units would be designed in accordance with ANSI/ASHRAE Standard 15, "Safety Standard for Refrigeration Systems" and that each equipment room would contain a refrigerant detector. ANSI/ASHRAE Standard 15 requires that the refrigerating system is protected by a pressure-relief device to safely relieve pressure buildup due to fire or other abnormal conditions and are piped to the outside of the building. The applicant also noted that doors between the Auxiliary Building and the R/B that lead to the control room would have weather stripping around them and sweeps at the threshold that would minimize air passing around the door.

The staff notes that DCD Revision 3, Subsections 6.4.4.2, "Toxic Gas Protection," 9.2.7.2.1, "Essential Chilled Water System," and 9.2.7.2.2, "Non-Essential Chilled Water System," contain the above described attributes contained in the applicant's first two responses.

In **RAI 917-6272, Question 06.04-15,** the staff identified to the applicant three examples of nuclear plant operating experiences where uncontrolled release of refrigerants caused operating plants to declare unusual events. In addition, the staff made a comparison of the US-APWR design to the guidance provided by ASHRAE Standard 15 and identified four areas where additional information was required to make a regulatory finding. The four areas of concern pertained to (1) Non-Essential Chiller Machinery Room Isolation; (2) Essential Chiller and Non-Essential Chiller Machinery Rooms Ventilation; (3) Refrigerant Type and Amounts; and (4) Chiller Room Fire Hazard's Analysis.

While the applicant's response to RAI 917-6272, Question 06.04-15, dated June 4, 2012, did resolve some technical issues, it did not resolve all staff concerns. The staff issued **RAI 955-6585, Question 06.04-17** and requested additional information and DCD enhancement for concerns (1) through (3).

Accordingly, the staff holds as an **Open Item -- RAI 46-895, Question 06.04-19; RAI 338-2325, Question 06.04-6; RAI 559-4387, Question 06.04-13; RAI 917-6272, Question 06.04-15; and RAI 955-6585, Question 06.04-17.**

6.4.4.6 Remote Shutdown Capability

The staff evaluated DCD Section 6.4 for conformance with the GDC 19 requirement that mandates that equipment including instrumentation and controls exists at appropriate locations outside the control room for attaining and maintaining hot shutdown of the reactor and ultimately attaining cold shutdown of the reactor. In particular, **RAI 46-895, Question 06-04 (06.04-24),** addressed this requirement.

In **RAI 46-895, Question 06-04 (06.04-24),** the staff asked the applicant for additional information about the location of remote shutdown console relative to the location of the MCR and MCR EFU rooms. Specifically, the staff requested that the applicant provide additional information as to why the RSC is not remote and is located very close to the MCR and the MCR EFU facilities and how this arrangement satisfies the requirements of Criterion 19 of 10 CFR 50 Appendix A. The staff noted that the RSC, which is two levels above the CRE and one level above the MCR EFU and AHU Rooms, provides backup reactor control and shutdown capability.

In its response to **RAI 46-895, Question 06-04,** dated September 16, 2008, the applicant stated that:

> "The requirements for the RSC are detailed in Section 5.4 of RG 1.189, as the basis for RSC facility provisions. Under that scenario, the RSC is required for shutdown of the plant only in the event of a fire in the MCR with no other coincident event, including any radiological events.
>
> The RSC complies with GDC 19 as it houses equipment and controls necessary for continued safe shutdown of the reactor, at an alternate location away from the MCR. Section 6.16, "Alternative/Dedicated Shutdown Panels" of the RG 1.189, which describes the specific requirement for RSC about separation from MCR, reads '*Barriers having a minimum fire rating of 3 hours should separate panels providing alternative/dedicated shutdown capability from the control room complex.*' The MCR is enclosed by 3-hour-rated fire-barriers and the RSC is two levels above the MCR. Therefore, the RSC is unaffected by a fire in the MCR.
>
> Fire and Radiological events are permitted to be mutually exclusive (See RG 1.53 -Application of Single-Failure Criterion to Safety Systems)."

After comparing the applicant's response to **RAI 46-895, Question 06-04 (06.04-24),** to the guidance of RG 1.189 "Fire Protection For Nuclear Power Plants" the staff concluded that the design of the US-APWR MCR and the location of the remote shutdown console satisfies the guidance RG 1.189. Based on this, the staff found the applicant's response acceptable and **closed RAI 46-895, Question 06.04-24.**

6.4.4.7 Control Room Habitability Dose Analyses

GDC 19 requires that the control room be designed to provide adequate radiation protection permitting personnel to access and occupy the control room under accident conditions to ensure that radiation exposures will not exceed 0.05 Sv (5 rem) TEDE for the duration of the accident. The applicant has performed radiological consequences analyses that show that the design of the MCRE and MCR HVAC systems are adequate and the MCR doses for each of the DBAs are within the requirements of GDC 19.

The applicant's control room habitability dose analyses considered all sources of radiation that are expected to cause exposure to MCR personnel during a DBA. The sources include intake and infiltration of airborne radioactive material within the radioactive plume released from the facility as well as direct radiation from the external radioactive plume, radioactive material in the containment, and radioactive material in the MCR EFU. The applicant's analyses calculated a combined TEDE from these sources equal to or less than 0.05 Sv (5 rem) for each of the DBAs evaluated in Chapter 15 of the US-APWR FSAR. This set of DBAs is consistent with the guidance on performing DBA radiological consequences analyses for light-water power reactor facilities given in RG 1.183.

The following DCD, Tier 2 sections discuss the DBA dose analyses for the US-APWR:

Design Basis Accident	DCD Section
Main Steam Line Break outside containment (MSLB)	15.1.5.5
Reactor Coolant Pump Rotor Seizure (LRA)	15.3.3.5
Rod Ejection Accident (REA)	15.4.8.5
Failure of Small Lines Carrying Coolant Outside Containment	15.6.2.5
Steam Generator Tube Rupture (SGTR)	15.6.3.5
Loss of Coolant Accident (LOCA)	15.6.5.5
Fuel Handling Accident (FHA)	15.7.4

The control room dose calculated by MHI for each DBA is reported in the associated DCD section. US-APWR DCD Tier 2, Table 15.0-17 gives a summary of the applicant's calculated doses for each of the DBAs, including the dose in the MCR. In each case, the dose in the MCR was less than 0.05 Sv (5 rem) TEDE, therefore meeting the GDC 19 criterion.

The applicant described the modeling of the MCR in the LOCA dose analysis in DCD, Tier 2, Section 15.6.5.5.1.2. The control room dose analyses considered the effect of the MCR HVAC system emergency isolation, pressurization and filtration, unfiltered inleakage through the MCRE and through ingress/egress, and direct dose. The MCR isolation signal causes the MCR HVAC system to change to the emergency pressurization mode upon receipt of an ECCS actuation signal or high MCR outside air intake radiation. The MCR outside air intake radiation monitor setpoint is set to initiate the MCR HVAC emergency pressurization mode in order to limit the intake of contaminated air to the CRE for accidents that do not cause the ECCS to actuate.

MHI also evaluated the direct radiation shine dose to the control room operators for the LOCA and added this value to the LOCA dose from inhalation and immersion within the radioactive material in the MCR air. The applicant's modeling of the direct dose is discussed in Sections 6.4 and 12.3 of the US-APWR DCD. The calculated LOCA direct dose is less than 0.0002 Sv (0.02 rem) TEDE. The applicant added the MCR direct dose calculated for the LOCA to the inhalation and immersion dose for each of the remaining DBAs. The staff agrees that the direct radiation shine dose to the control room operator for the DBA LOCA would be bounding for the other DBAs, due to the higher radioactive material concentration in the containment, airborne plume, and captured in the recirculation filters.

To verify the applicant's assessment of the MCR radiological habitability, the staff performed independent radiological consequence calculations using the dose analysis computer code RADTRAD. The staff modeled each of the DBAs with the MCR emergency filtration system in operation as described in the US-APWR FSAR, Tier 2, Section 6.4. The staff also performed separate analyses to confirm the applicant's direct dose analysis. The staff used the following information in its analyses:

- Reactor accident source terms based on NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants," and RG 1.183, "Alternative Radiological Source Terms for Evaluating Design-Basis Accidents at Nuclear Power Reactors" as applied to the US-APWR design described in FSAR, Tier 2.

- Design reference control room χ/Q values, control room structure dimensions, ventilation system design flow rates and filter efficiencies, and unfiltered in-leakage rates provided by MHI in the US-APWR FSAR, Tier 2.

- Control room occupancy factors referenced in Section 6.4 of the SRP.

The staff's independently calculated TEDE in the control room for each of DBAs in Chapter 15 of the US-APWR FSAR confirm the applicant's analysis results and conclusion that the dose requirements of GDC 19 are met for the US-APWR MCR. Section 15.0.3 of this report discusses further the staff's review of the applicant's analysis of control room radiological habitability and the staff's independent confirmatory radiological consequence analyses.

Therefore, based on the staff's review of the applicant's description of the control room dose analyses in the US-APWR FSAR, Section 6.4 and Chapter 15, as confirmed through independent analysis, the staff finds that the MCR structure and MCR emergency filtration system, as described in FSAR Tier 2, Section 6.4, is capable of mitigating the dose in the MCR following DBAs to meet the dose criterion specified in GDC 19.

<u>Staff's GDC 19 Summation Statement</u>

Due to the existence of the open item associated with **RAI 955-6585, Question 06.04-17,** the staff cannot conclude that the US-APWR plant's Habitability Systems satisfies the guidance of SRP 6.4 and the requirements of 10 CFR 50 Appendix A Criterion 19.

10 CFR 50.34(f)(2)(xxviii) – "Provisions To Preclude Certain MCR Habitability Problems"

The staff reviewed the habitability system to ensure that the relevant requirements of 10 CFR 50.34(f)(2)(xxviii) are met and to determine if there may be potential pathways for radioactivity and radiation under accident conditions that need to be addressed by design provisions. In particular, **RAI 46-895, Questions 06-04 (06.04-7 and 06.04-20) and RAI 927-6460, Question 06.04-16** addressed these requirements. **RAI 49-895, Question 06-04 (06.04-7),** addressed the ventilation layout and functional design for zone isolation and maintaining positive pressure in the MCR. **RAI 46-895, Question 06-04 (06.04-20),** addressed CRE testing to ensure that areas adjacent to the CRE would maintain airflow away from the MCRE. **RAI 927-6460, Question 06.04-16,** questioned the absence of discussion and/or depiction on plant drawings pertaining to the MCR AHU drain lines. These four drain lines represent a potential source of unfiltered air in-leakage to the CRE.

In **RAI 46-895, Question 06-04 (06.04-7),** the staff cited the review guidance of SRP 6.4 Section III.3.A.ii and iii. In particular, the staff noted that the applicant's DCD should reflect ventilation layout and functional design, including zone isolation and maintaining positive pressure in the control room. The staff noted that DCD Figure 6.4-5, "Main Control Room Emergency Filtration System Plan View," shows intake airflow through the MCR EFU, which applies to only the emergency pressurization mode of operation. The staff found that this figure is inconsistent with plant design in that the airflow for the normal and emergency isolation operation modes for the MCR ventilation system does not pass through the MCR EFUs.

In its response to **RAI 46-895, Question 06-04**, dated September 16, 2008, the applicant concurred with the staff observation and agreed to revise DCD Figure 6.4-5 with a clarifying note stating that the airflow configuration applies only to the emergency pressurization mode. Based on the applicant's decision to revise Figure 6.4-5, the staff found the applicant's response to **RAI 46-895, Question 06-04 (06.04-7)** acceptable. The staff verified that Revision 3 of the US-APWR DCD, contained the change of the RAI response. Based on this **the staff closed RAI 46-895, Question 06-04 (06.04-7).**

In **RAI 46-895, Question 06-04 (06.04-20),** the staff cited SRP 6.4 Section III.5.C.ii since it prompts the staff to verify that the ventilation zones adjacent to the emergency zone are configured and balanced to preclude airflow toward the emergency zone. The staff could find no evidence in its review of DCD Section 9.4, "Air Conditioning, Heating, Cooling, and Ventilation Systems," that this guidance is being invoked in the "Inspection and Testing Requirements" DCD sections for the applicable HVAC system that provide ventilation to the areas adjacent to the CRE. In its response to **RAI 46-895, Question 06-04,** the applicant agreed to amend Chapters 6, 9, and 14 to include balancing and testing requirements to ensure that airflow will be directed away from CRE boundaries. After review of the applicant's response, the staff concluded that the proposed DCD amendments did not provide sufficient resolution and DCD clarity. Subsequently, the staff issued follow-up **RAI 338-2325, Question 06.04-7**.

In its response to RAI **338-2325, Question 06.04-7**, dated June 17, 2009, the applicant agreed to revise DCD Section 6.4.2.4 to describe the HVAC systems that provide heating, cooling, and ventilation to the adjacent areas of the CRE. The applicant also agreed to revise DCD Section 6.4.2.4 to provide clarification. Furthermore, the applicant agreed to revise the wording of DCD Subsection 14.2.12.1.101, "MCR HVAC System Preoperational Test (including MCR Habitability)," under "D Acceptance Criteria" to clarify the direction of airflow from the CRE with respect to the adjacent areas.

As a result, the staff found the applicant's total response acceptable. The staffs review of US-APWR DCD, Revision 3 verified that the requisite changes (i.e. as captured in the two applicant responses) to DCD Chapters 6 and 14 had been accurately incorporated into the DCD. The staff **closed both RAI 46-895, Question 06-04 (06.04-20) and RAI 338-2325, Question 06.04-7.**

In **RAI 927-6460, Question 06.04-16**, the staff documented that neither DCD Tier 2 Figure 9.4.1-1 "MCR HVAC System Flow Diagram" nor Figure 9.3.3-1 "Equipment and Floor Drain System Flow Schematic Radiological Controlled Area" display the equipment drain lines for collecting condensate from the four MCR AHU cooling coils. The staff noted that the portions of the equipment drain lines that connect to the AHUs would have to be Safety Related and Seismic Category I since they form part of the CRE boundary and could affect MCR habitability. The staff posited that since these equipment drain lines tie into the four AHUs just upstream of the respective AHU fan, the drain lines could be below atmospheric pressure and provide unfiltered paths for radioactive containments into the CRE in a post-accident plant environment. The staff noted that SER Section 9.4.1 discusses the need for adequate sizing of these drain lines in **RAI 883-6063, Question 09.04.01-32**.

The staff requested additional information about how the design of the four AHU equipment drain lines will satisfy the requirements of GDC 2, GDC 4 and GDC19. The staff also requested that the applicant appropriately amend the DCD (e.g. Tier 2 Sections 6.4, 9.3.3 and 9.4.1) to capture the applicant's response. R**AI 927-6460, Question 06.04-16 is being tracked as an Open Item.**

10 CFR 52.47(b)(1) – Inspections, Tests, Analyses, And Acceptance Criteria (ITAAC)

The requirements contained in 10 CFR 52.47(b)(1) do apply to the habitability system and will be addressed here and in SER Section 14.3.7. **RAI 46-895, Question 06-04 (06.04-17)** addressed a concern with respect to surveillance frequency. SRP 6.4 guidance intends for measurement of the CRE pressure relative to all external areas adjacent to the CRE boundary to be on an 18-month frequency, whereas the DCD stated a 24-month frequency. **RAI 827-5812, Question 09.04.01-28,** questioned a DCD Revision 3 design conflict for emergency filter unit fresh air make up flow rates to the CRE during the emergency pressurization mode of operation.

In **RAI 46-895, Question 06-04 (06.04-17)**, the staff requested that the applicant explain the basis for the difference in the unfiltered air in-leakage test frequencies specified in DCD Chapter 16, SR 3.7.10.4 (i.e., 24 months on a staggered test basis) as compared to guidance of every 18 months contained in SRP 6.4, Sections 3.A, B, and C. In addition, the staff requested that the applicant include within Tier 1 or Tier 2 a documented value for the CRE volume.

The applicant responded in part that "DCD Chapter 16 SR 3.7.10.4 would be revised to be consistent with TSTF-448 Revision 3. The applicant also agreed to revise the surveillance frequency of SR 3.7.10.4 to read "*In accordance with the Control Room Envelope Habitability Program*." In addition, the applicant agreed to add a CRE Habitability Program to Section 5.5.20 of Chapter 16.

The applicant also agreed to amend DCD Section 6.4.1 to include the value for the CRE volume (i.e., 140,000 ft^3).

The staff reviewed the applicant's changes to the DCD for Chapter 16 TS 3.7.10, SR 3.7.10.4 and 5.5.20 against the guidance of TSTF 448, Revision 3. In particular, the staff reviewed Section 5.5.20 "Control Room Habitability Program" in DCD Revision 3 Chapter 16. The staff notes that program element "c" reads in part "Requirements for (i) determining the unfiltered air in-leakage past the CRE boundary into the CRE in accordance with the testing methods and at the Frequencies specified in Sections C.1 and C.2 of Regulatory Guide 1.197". Regulatory position C.1 and Figure 1 of RG 1.197 permits applicant CRE integrity testing to be conducted as in-frequently as once every six years if certain criteria are met. Since SRP 6.4 endorses RG 1.197, the staff concluded that the surveillance requirements of Chapter 16 SR 3.7.10.4 are consistent with the regulatory guidance. The staff also verified that Revision 3 DCD Section 6.4.1 contains the value for the CRE volume. Based on the above DCD amendments the staff found the applicant's response acceptable and **closed RAI 46-895, Question 06-04 (06.04-17).**

In **RAI 827-5812, Question 09.04.01-28,** the staff identified that Revision 3 of the DCD created a new area of conflict. This conflict pertains to the air intake flow rates found in the DCD to the MCR EFUs. In particular, DCD Tier 2 Section 6.4.2.3 indicates that the makeup (outside air ventilation) flow rate during emergency pressurization mode is equal to or less than 1,200 ft^3/min (34 m^3/min). In contrast, DCD Tier 2 Subsection 9.4.1.2.2.1 "Pressurization Mode" indicates that the make-up design airflow rate is less than 600 ft^3/min in the emergency pressurization mode. The staff also noted that the "Acceptance Criteria" of DCD Tier 1 Table 2.7.5.1-3 "Main Control Room HVAC System Inspections, Tests, Analyses, and Acceptance Criteria", Item 4.b.ii reads "*The as-built MCR HVAC system provides filtered air intake flow of <1200 cfm, filtered air recirculation flow of >2400 cfm, and maintains positive pressure in the as-built CRE in the emergency pressurization mode.*" The staff requested that the applicant provide additional information and DCD clarification for these conflicting flow rates.

The applicant initially responded on October 7, 2011. The applicant subsequently provided an amended response on August 24, 2012 after the staff requested additional clarification. The amended response provided proposed comprehensive changes to the DCD for the staff's concerns pertaining to Tier 2 Sections 6.4.2.3, 9.4.1.2.2.1 and B 3.7.10, "Main Control Room HVAC System (MCRVS)," and to the "Acceptance Criteria" associated with Tier 1 Table 2.7.5.1-3 ITAAC "Design Commitment" 4.b and 4.c. These Tier 1 and Tier 2 changes ensure that the total unfiltered CRE in-leakage will be less than or equal to 120 cfm in the most non-conservative system flow configuration while operating in the emergency pressurization mode. This 120 cfm in-leakage value includes an assumed value of 10 cfm for CRE ingress/egress. These changes are consistent with the guidance of SRP 6.4, RG 1.196 and RG 1.197. Based on this compliance the staff found the applicant's response acceptable. **RAI 827-5812, Question 09.04.01-28 is being tracked as a Confirmatory Item.**

6.4.4.8 SRP Section 6.4 – Other Guidance

The staff evaluated DCD Section 6.4 for design conformance with other miscellaneous guidance as captured in SRP 6.4. The staff initiated **RAI 46-895, Question 06-04 (06.04-12 and -21)** and **RAI 442-3378, Question 09.04.01-10,** to ensure conformance with this guidance. In **RAI 46-895, Question 06.04-12**, the staff noted that the DCD lacked FMEA for both the Habitability system and the MCR HVAC system. In **RAI Question 06.04-21**, the staff inquired as to whether the MCR can be maintained as the backup center from which Technical Support Center personnel can safely operate in the case of an accident. In **RAI 442-3378, Question 09.04.01-10,** the staff asked the applicant to qualify as a design standard the use of ASME AG-1-2003 instead of ASME AG-1-1997.

In **RAI 46-895, Question 06-04 (06.04-12)**, the staff noted that SRP 9.4.1 Sections III.1, 3, and 4 make reference to the use of a FMEA. The FMEA confirms that the essential safety-related portions of the system are capable of functioning in spite of the failure of any active component, in the event of an earthquake, during LOOP, or a concurrent single active failure. The staff found that neither DCD Section 6.4 nor DCD Section 9.4.1 contains any reference to a FMEA. The staff requested that the applicant provide detailed information about the FMEA in DCD Sections 6.4 and 9.4.1.

In its response to **RAI 46-895, Question 06-04**, dated September 16, 2008, the applicant agreed to add a FMEA for the safety related active components of the MCR HVAC system to the DCD. The staff verified that Revision 3 of the DCD included a comprehensive FMEA (i.e. Table 9.4.1-2) for the active components of the MCR HVAC system.

The staff review of US-APWR DCD, Revision 2, indicated that the applicant did not include a FMEA for DCD Section 6.4. The applicant noted in their response, dated September 20, 2012, to the follow-up **RAI 559-4387, Question 06.04-11**, that the US-APWR will utilize barriers to protect the MCR from postulated flooding outside of the CRE, instead of relying on the water tight doors. Other penetrations involve passive components, which are not subject to wear by frequent "use" by plant personnel. Therefore, the applicant concluded that that a specific FMEA will not be required for the CRE. The staff notes that SRP 6.4 does not prompt the applicant to provide a FMEA for the passive SSCs that comprise the CRE. The staff also notes that SRP 9.4.1 only provides direction to perform an FMEA on active safety related components. Therefore, based on the lack of FMEA guidance in SRP 6.4 and the FMEA guidance contained in SRP 9,4,1, the staff determined the applicant's composite RAI responses to be acceptable. Based on this, **the staff closed RAI 46-895, Question 06-04 (06.04-12) and RAI 559-4387, Question 06.04-11.**

In **RAI 46-895, Question 06-04 (06.04-21)** the staff cited the following excerpt from SRP 6.4 Section I: "*Additionally,SRP 6.4 Section I requires that a review is performed to ensure that the control room can be maintained as the backup center from which Technical Support Center personnel can safely operate in the case of an accident.*" In its review of DCD Sections 6.4 and 9.4.1, the staff found insufficient evidence to conclude that the design of the CRE habitability systems and the MCR HVAC system addressed this guidance. The staff requested that the applicant provide additional information about how the design of the CRE and the MCR HVAC system satisfied this SRP excerpt.

The applicant responded to **RAI 46-895, Question 06-04 (06.04-21),** dated September 16, 2008. Additionally another technical Office, Office of Nuclear Security and Incident Response, during the review of the emergency preparedness generated a similar question during its review of DCD Section 13.3, "Emergency Planning," subsequent to the response received for **RAI 46-895, Question 06-04 (06.04-21)**.

In its response to **RAI 108-1515, Question 13.03-2,** dated December 25, 2008, the applicant provided a comprehensive response. The most significant details of the applicant's response follow:

> In the unlikely event that the Technical Support Center (TSC) becomes uninhabitable, the size and equipment available in the MCR will be sufficient to absorb the plant management function of the TSC.
>
> To meet NRC requirements, the normal required complement of operations personnel in the MCR with actual controls responsibilities is expected to be four (4). Allowing for some transience of equipment operators in and out of the MCR, and also for clerical and other personnel, the total number of people in the MCR at any given time could reach ten (10) during a typical day-shift mode. The MCR has a total floor area of approximately 2250 square feet, and an adjacent support room of similar size that contains an operator area, shift supervisor's office, clerical space, kitchen, and restrooms.
>
> Section 2.6 of NUREG-0696 states in part that, "If the TSC becomes uninhabitable, the TSC plant management function shall be transferred to the control room." Consistent with this Section, MHI intends that, for the US-APWR, the plant management function would be transferred to the MCR should the TSC become uninhabitable. While the ultimate details of this contingency would be part of a licensee's emergency plan and are beyond the scope of a standard design, MHI estimates that, in terms of manpower, the "plant management function" would consist of three (3) senior licensee plant management personnel, and the five (5) NRC personnel. MHI would anticipate that the additional seventeen (17) licensee personnel, representing the technical support function of

the TSC, would be transferred to the Emergency Operations Facility (EOF) or possibly the plant simulator facility, at the discretion of the licensee and depending on the configuration and capabilities of a particular site.

MHI believes that the strategy described above would accomplish NRC's goal of avoiding over-crowding of the MCR, while at the same time preserving all of the required functions of the TSC.

After comparing the applicant's response to **RAI 108-1515** to the SRP 6.4 guidance invoked in Question 06.04-21, the staff concluded that the design of the US-APWR MCR has the capability of satisfying the SRP and NUREG guidance since the MCR has the capability to assimilate the TSC plant management function from the TSC. Based on this, the staff **closed RAI 46-895, Question 06-04 (06.04-21).**

In **RAI 442-3378, Question 09.04.01-10** the staff asked the applicant to qualify as a design standard the use of ASME AG-1-2003 instead of ASME AG-1-1997. **RAI 442-3378, Question 09.04.01-10**, in particular, applies to DCD Section 6.4, "Habitability Systems," (i.e., Section 6.4.5 and Tables 6.4-1 and 6.4-2). Specifically, the staff questioned the use of a more recent version of ASME AG-1 than had been endorsed by the staff. Revision 3 of RG 1.52 endorses ASME AG-1-1997 while the applicant's DCD referenced AG-1-2003 and AG-1a-2000. In its responses to **RAI 442-3378, Question 09.04.01-10**, dated September 18, 2009, and December 09, 2009, (ML092650173 and ML093480146, respectively), the applicant made a tabular comparison of the endorsed version versus the newer version. The applicant concluded that the use of the 2003, edition of the Code, rather than the 1997, edition referenced in the NRC guidance documents, is justified. The staff conducted an independent side-by-side comparison of the two AG-1 Codes to confirm that the use of ASME AG-1-2003 edition is an acceptable alternative to ASME AG-1-1997 for the MCR HVAC system design and testing. For the substantive areas of difference, the applicant provided a comprehensive technical justification (as applicable) for the use of AG-1-2003 and AG-1a-2000, in lieu of AG-1-1997, in the design of the US-APWR DCD ventilation subsections. Based on this review, the staff concluded that the applicant had provided sufficient justification to demonstrate that the proposed alternatives to the SRP acceptance criteria provide acceptable methods of compliance with the NRC guidance of RG 1.52. Based on this conclusion, the staff **closed both RAI 442-3378, Question 9.4.1-10, and RAI 484-3850, Question 09.04.01-15**.

6.4.5 Combined License Information Items

The following is a list of COL item numbers and descriptions pertinent to this section from Table 1.8 2 of the DCD:

Table 6.4-1
US-APWR Combined License Information Items

Item No.	Description	Section
6.4(1)	The COL Applicant is responsible to provide details of specific toxic chemicals of mobile and stationary sources within the requirements of RG 1.78 and evaluate the control room habitability based on the recommendation of RG 1.78	6.4
6.4(2)	The COL Applicant is responsible to discuss the automatic actions and manual actions for the MCR HVAC system in the event of postulated toxic gas release.	6.4

Table 6.4-1 US-APWR Combined License Information Items		
Item No.	**Description**	**Section**
6.4(5)	The number, locations, sensitivity, range, type, and design of the toxic gas detectors are COL items. Depending on proximity to nearby industrial, transportation, and military facilities, and the nature of the activities in the surrounding area, as well as specific chemicals onsite, the COL Applicant	6.4

COL information items not identified in Table 1.8 2 of the DCD: None required, but additional COL items associated with the noted open items may be necessary.

6.4.6 Conclusions

As set forth in SER Sections 6.4.2, 6.4.3, 6.4.4, and 6.4.5, the US-APWR habitability systems serve safety-related functions and thus have a safety design bases. The staff evaluated the habitability systems for the US-APWR standard plant design in accordance with guidance referred to in the technical evaluation section of this SER. Based on this review, the staff concludes that sufficient information has been provided by the applicant in: (a) US-APWR DCD Tier 1 Sections 2.7.5.1, 2.7.6.6, 2.7.6.13, 2.8; and (b) DCD Tier 2 Sections 6.4 and 9.4.1. In addition, the staff compared the design information in the DCD application to the relevant NRC regulations, acceptance criteria defined in NUREG-0800 SRP Section 6.4, and other NRC RGs. In conclusion, the US-APWR DCD for the habitability systems is acceptable and meets the requirements of 10 CFR Part 50, Appendix A, GDC 4, 5, 19 and 10 CFR 50.34(f)(2)(xxviii) and 10 CFR 52.47(b)(1) and the guidelines of SRP Section 6.4 with exception of the RAIs listed as open items in the body of this SER.

The staff recommends that the status of the following RAI question numbers remain US-APWR DCD Open Items until follow-up questions and concerns are resolved and closed:

- **Open Item: RAI 46-895, Question 06.04-8; RAI 338-2325 Question 06.04-8; RAI 501-4004, Question 06.04-10; NRC 559-4387, Question 06.04-11; and RAI 841-6055, Question 03.04.01-30**.

- **Open Item: RAI 46-895 Question 06.04-19; RAI 338-2325, Question 06.04-6; RAI 559-4387, Question 06.04-13; RAI 917-6272, Question 06.04-15; and RAI 955-6585, Question 06.04-17**.

- **Open Item: RAI 927-6460, Question 06.04-16.**

In addition, **RAI 827-5812, Question 09.04.01-28 is being tracked as a Confirmatory Item**.

Responses to the other staff RAI questions captured in the SER Section 6.4.4 and the applicant's subsequent revisions to the DCD have resolved the subject RAI concerns. The staff has closed these RAI questions. Accordingly, the staff concludes that the applicant meets the relevant requirements of 10 CFR Part 50, Appendix A, GDC 4, 5, and 19, and 10 CFR 50.34(f)(2)(xxviii) and 10 CFR 52.47(b)(1) contingent upon the closure of the above open items. DCD Section 6.4 is acceptable subject to the resolution and closure of the open items listed above and normal progression of the existing and potential COL items described in SER Section 6.4.5.

6.5 Fission Product Removal and Control Systems

The fission product removal systems are ESFs that remove fission products that are released from the reactor core as a result of postulated accidents and become airborne. The containment controls the leakage of fission products from the containment to ensure that the leakage fraction that may reach the environment is below limits. The US-APWR fission product removal (three systems) and control (containment) systems are as follows:

- MCR HVAC system (includes the MCR emergency filtration system).
- Annulus emergency exhaust system.
- CSS.
- Containment.

The fission product removal effects under accident conditions are shown in Table 6.5-1, “Summary of Fission Product Removal and Control Mechanisms.” The annulus emergency exhaust system is separate and distinct from the MCR HVAC system, which is described in Section 6.4 above. The CSS for containment cooling is described in Section 6.2.2.

6.5.1 ESF Filter Systems (Related to NUREG-0800, Section 6.5.1, “ESF Atmosphere Cleanup Systems”)

6.5.1.1 Introduction

The ESF atmosphere cleanup systems are designed for fission product removal in post accident environments. These systems include primary systems, such as in-containment recirculation, and secondary systems, such as standby gas treatment systems and emergency or post-accident air-cleaning systems for the fuel -handling building, control room, shield building, and areas containing ESF components.

6.5.1.2 Summary of Application

DCD Tier 1: The Tier 1 information associated with this section is found in DCD Tier 1, Section 2.7.5.1, “Main Control Room HVAC System,” and Section 2.7.5.2, “Engineered Safety Features Ventilation System (ESFVS).”

DCD Tier 2: The applicant has provided a DCD Tier 2 description for the ESF Atmospheric Cleanup Systems in Sections 6.0.5, “Fission Product Removal and Control Systems,” 6.2.1, “Containment Functional Design,” 6.2.2, “Containment Heat Removal Systems,” 6.5.1, “ESF Filter Systems,” 6.4, “Habitability Systems,” 9.4.1, “Main Control Room Heating, Ventilation and Air Conditioning System,” and 9.4.5, “Engineered Safety Feature Ventilation System,” summarized here in part, as follows:

> The application describes the fission product removal systems as ESFs that remove fission products that are released from the reactor core as a result of postulated DBAs and become airborne. These systems, sometimes referred to as “atmosphere cleanup,” ensure that fission products are confined to prevent an unintended release to the environment. The containment reduces leakage of fission products from the

containment to ensure that the leakage fraction that may reach the environment is below limits. The US-APWR fission product removal (three systems) and control (containment) systems are as follows:

- MCR HVAC system.
- Annulus emergency exhaust system.
- CSS (Section 6.5.2).
- Containment vessel (Section 6.5.3).

DCD Section 6.4 indicates that the MCR habitability systems allow operators to remain safely inside the CRE and take the actions necessary to manage and control the plant under normal and abnormal plant conditions, including a LOCA. The MCR habitability systems protect operators against a postulated release of radioactive material, natural phenomenon induced missiles, radioactive shine, smoke, and toxic gases. These systems include the following:

- MCR HVAC (Chapter 9, Section 9.4.1).
- MCR emergency filtration system (the ESF portion of MCR HVAC system).
- Radiation monitoring system (Chapter 7).
- Radiation shielding (Chapter 12).
- Lighting system (Chapter 9, Section 9.5.3).
- Fire protection system (Chapter 9, Section 9.5.1).

The CRE includes the MCR and is served by the MCR HVAC system during normal and abnormal conditions as well as MCR smoke purge operations. Personnel occupying the CRE are protected from the respiratory effects and eye irritation of smoke.

The US-APWR DCD Section 9.4.1 indicates that the control room area ventilation system is designed to provide proper environment in the MCR and other areas within the MCRE as defined in DCD Section 9.4.1 "Main Control Room Heating, Ventilation and Air Conditioning System". This MCR HVAC system enables MCR personnel to remain safely inside the MCRE and take actions necessary to manage and control the plant under normal and abnormal plant conditions, including a LOCA and SBO. The MCRE consists of the MCR, operator area, shift supervisor office, clerk room, tagging room, toilet and kitchen.

The MCR HVAC system is shown in Figure 9.4.1-1 "MCR HVAC System Flow Diagram" and system equipment and components design data are presented in Table 9.4.1-1 "Equipment Design Data". The MCR HVAC system consists of two redundant 100 percent ESF EFUs and four 50 percent capacity AHUs, two 100 percent toilet/kitchen exhaust fans, one 100 percent smoke purge fan, ductwork, associated dampers and I&C. The AHUs are connected to a common overhead air distribution ductwork system. The MCR HVAC system is capable of operating in the normal, emergency pressurization, emergency isolation, and emergency smoke purge operation modes. The MRC HVAC system, with exception of the toilet/kitchen exhaust and smoke purge subsystems, is classified as safety-related, Seismic Category 1.

The US-APWR DCD Section 9.4.5 indicates the ESF ventilation system is designed to provide the proper environmental conditions within plant areas that house ESF equipment. The system function is to support and assure the safe and continuous operation of the ESF equipment during normal and emergency operating conditions. The ESF ventilation system includes:

- Annulus Emergency Exhaust System.
- Class 1E Electrical Room HVAC System.
- Safeguard Component Area HVAC System.
- EFW Pump Area HVAC System.
- Safety Related Component Area HVAC System.

The application further explains that the annulus emergency exhaust system is separate and distinct from the control room habitability system, which is presented in Section 6.4.

ITAAC: The ITAAC associated with DCD Tier 2, Section 6.5.1 are given in DCD Tier 1, Section 2.7.5.1 and Section 2.7.5.2.

TS: The TS associated with DCD Tier 2, Section 6.5.1 are given in DCD Tier 2, Chapter 16, Sections 3.3.2 “Engineered Safety Feature Actuation System (ESFAS) Instrumentation”, B.3.3.2, 3.6.6 “Containment Spray System”, B3.6.6, Sections 3.7.10 “Main Control Room HVAC System (MCRVS)”, B.3.7.10, 3.7.11 “Annulus Emergency Exhaust System”, B 3.7.11, 5.5.11 “Ventilation Filter Testing Program (VFTP)” and 5.5.20 “Control Room Envelope (CRE) Habitability Program”.

Topical Reports: There are no topical reports for this area of review.

Technical Reports: There are no technical reports for this area of review.

US-APWR Interface Issues identified in the DCD: There are no US-APWR interface issues for this area of review.

Site Interface Requirements identified in the DCD: There are no site interface requirements for this area of review.

Cross-cutting Requirements (TMI, USI/GSI, Op Ex): None for this area of review.

RTNSS: There is no RTNSS for this area of review.

10 CFR 20.1406: There are no 10 CFR 20.1406 requirements for this area of review.

CDI: There is no CDI for this area of review.

6.5.1.3 Regulatory Basis

The relevant requirements of the Commission's regulations for this area of review, and the associated acceptance criteria, are given in Section 6.4, “Habitability Systems,” of NUREG-0800 and are summarized below. Review interfaces with other SRP sections can be found in Section 6.4 of NUREG-0800. These regulations include:

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 the 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 AOOs, and from postulated accidents.

7. 10 CFR 52.47(b) (1), which requires that a DC application contain the proposed ITAAC that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the DC is built and will operate in accordance with the DC, the provisions of the Atomic Energy Act of 1954, and the NRC's regulations.

Acceptance criteria adequate to meet the above requirements include:

1. Relevant aspects of the requirements listed above are met by use of the regulatory positions of RG 1.52 as to the design, testing, and maintenance of ESF atmosphere cleanup system air filtration and adsorption units.

6.5.1.4 Technical Evaluation

The staff reviewed the ESF filter systems for the US APWR in accordance with Section 6.5.1 of the SRP. These systems include those discussed in the US-APWR DCD: (1) Section 6.4 “Habitability Systems“; (2) Section 9.4.1 “Main Control Room Heating, Ventilation and Air Conditioning System”; and (3) Section 9.4.5 “Engineered Safety Feature Ventilation System”. SRP 6.4, SRP 9.4.1 and SRP 9.4.5 each contain a review interface with SRP 6.5.1. As a consequence, during their reviews per SRP 6.4, SRP 9.4.1 and SRP 9.4.5 the staff considered the implications and requirements of SRP 6.5.1 in detail during the development of the SER.

The US-APWR DBA radiological consequences analyses in DCD, Tier 2, Section 15, were reviewed and it was verified that accident dose sources originate in the primary containment, annulus, containment penetration areas, and safeguard component areas, which have atmospheres that are treated by the annulus emergency exhaust ESF filter system. Additionally, the ventilation supply and recirculation flow to the MCR is also treated by an ESF filter system, the MCR emergency filtration system. The fission product removal capability of one 100 percent capacity train of the annulus emergency exhaust ESF filter system was credited in the DBA radiological consequences analyses for the LOCA and control rod ejection accident, in accordance with the guidance in RG 1.183 and SRP 6.5.1. The fission product removal capability of one 100 percent capacity train of the MCR emergency filtration system was credited in the DBA radiological consequences analysis for each DBA evaluated, in accordance with the guidance in RG 1.183 and SRP 6.5.1. The applicant's DBA radiological consequences in DCD, Tier 2, Chapter 15, which take credit for the ESF filtration system discussed above, show that the offsite, control room, and technical support center doses for each DBA meet the applicable regulatory dose criteria in RG 1.183, and therefore also meet the requirements of 10 CFR 52.47(a)(2), GDC 19, and Paragraph IV.E.8 of Appendix E to 10 CFR Part 50, respectively. The applicant's control room dose results for each of the DBAs are listed in DCD, Tier 2, Table 15.0-17, "Summary of Calculated Doses for Events with a Radiological Release." Detailed review of the DBA radiological consequences analyses is discussed in SER Section 15.0.3.

6.5.1.5 Combined License Information Items

No COL information items are identified in Tier 1 DCD Table 1.8-2 of the DCD.

6.5.1.6 Conclusions

The staff reviewed the ESF filter systems for the US-APWR in accordance with Section 6.5.1 of the SRP. These systems include US APWR DCD: (1) Section 6.4, "Habitability Systems"; (2) Section 9.4.1, "Main Control Room Heating, Ventilation and Air Conditioning System"; and (3) Section 9.4.5, "Engineered Safety Feature Ventilation System." The open items and NRC confirmatory items against these three sections are identified in the staff's SER for these sections.

The staff finds that the applicant has conformed to the guidance on credit for ESF fission product removal systems in RG 1.183 in order to calculate the DBA radiological releases and expected MCR HVAC inlet conditions that would result from accidents in the proposed DCD. The DBA radiological consequences analyses meet the regulatory requirements of 10 CFR 52.47(a)(2), GDC 19, and Paragraph IV.E.8 of Appendix E to 10 CFR Part 50 with respect to the offsite, control room habitability, and technical support center habitability doses, respectively. Therefore, the staff finds that the proposed design and operation of the ESF filter systems will provide adequate fission product removal in post accident environments.

As set forth above in Sections 6.5.1.3 through 6.5.1.5 of this report, the staff's SE of the US-APWR ESF Filter Systems is provided in Sections 6.4, 9.4.1, and 9.4.5 of this SER. Accordingly, the staff's conclusion that the applicant meets the relevant requirements of 10 CFR Part 50, Appendix A, GDC 19, 41, 42, 43, 61, 64 and 10 CFR 52.47(b)(1) is predicated on the conclusions of SER Sections 6.4, 9.4.1 and 9.4.5.

6.5.2 Containment Spray System (CSS)

6.5.2.1 Introduction

In the event of a design-basis LOCA or REA there is an assumed release of radionuclides to the containment atmosphere. This release would consist of fission products, including noble gases, particulates, and a small amount of elemental and organic iodine. The CSS is an automatically actuated, dual function ESF for heat removal and fission product removal in the primary containment. This system mitigates the DBAs that release fission products into the containment.

6.5.2.2 Summary of application

DCD Tier 1: The Tier 1 information for this section is found in the DCD Tier 1, Section 2.11.3, "Containment Spray System."

DCD Tier 2: The applicant has provided a DCD Tier 2 description in Section 6.5.2, summarized here in part, as follows:

> The fission product removal feature of the containment spray system is accomplished by increasing the pH of the RWSP from its normal value of approximately 4.3, to a post-DBA pH of at least 7.0. The RWSP is the ESF source for borated water for containment spray and the ECCS; there is no automatic switchover to a borated ESF coolant source external to the containment.
>
> Radioactive iodine is the primary concern in evaluating and mitigating the potential radiological consequences of a DBA. Without an outside agent to reduce precipitation, radioactive iodine deposit on components in the containment, or may leak from the containment. The containment spray enhances iodine retention over an extended time period to allow decay of the longest lived radioactive iodine isotope (Iodine-131, with half-life of eight days).
>
> The containment spray system is started as follows:
>
> - In a DBA, elevated containment pressure actuates the containment spray system automatically.
>
> - If high radiation is detected in the containment, the MCR operator manually starts the containment spray function
>
> Crystalline NaTB is used to raise the pH of the RWSP from 4.3 to at least 7.0, after containment spray actuation. Twenty three pre-positioned baskets of NaTB are stored in the amounts and at the locations shown in Figure 6.3-8. The NaTB baskets are discussed in Section 6.3. The basket locations ensure full wetting and dissolution of the NaTB.
>
> CSS pumps, piping and valves are described in Section 6.2.2.
>
> Following a DBA, the containment pressure approaches atmospheric pressure. When the containment pressure is reduced sufficiently and the operator determines that containment spray is no longer required, the operator terminates containment spray.
>
> DCD Section 6.2.2, "Containment Heat Removal Systems," contains subsections on all major components, the heat removal function, and design evaluation calculations. Physical attributes of the system (flow rates, droplet diameters, etc.) are given in Tables 6.2.1-4, "Initial Conditions for Maximum Containment Pressure Analytical Model," 6.2.1-5, "Engineered Safety Feature Systems Information," 6.2.2-1, "Input Values Employed in CSS Evaluation Calculations," and 6.5-4, "Containment Sprayed/Unsprayed Volume." The fission-product removal process is described in DCD Section 6.5.2, "Containment Spray Systems," without many details or calculations. Details of the fission-product removal calculations for DBA are provided in DCD Appendix 15A "Evaluation Models

and Parameters for Analysis of Radiological Consequences of Accidents." Two key components of the system are the RWSP, and baskets in containment containing NaTB. These are described in DCD Sections 6.3.2.2.3, "Refueling Water Storage Pit," and 6.3.2.2.5, "NaTB Baskets and NaTB Basket Container," respectively. Spray patterns and system geometry are illustrated in a number of figures; those used in this analysis are Figures 6.2.1-5, "Containment Sectional View," 6.2.1-9, "Outline of Paths that Solutions from the ECCS and CSS would follow in the Containment to the RWSP," 6.2.1-10, "Volume of Ineffective Water," 6.2.1-11, "RWSP Water Levels," 6.2.2-5, "Containment Spray System Spray Ring Elevations," and 6.3-11, "Containment Spray Pattern Sectional View at the NaTB Basket Installation Level." The applicant describes testing and monitoring of the NaTB in DCD Chapter 16, TSs 3.5.5 and B 3.5.5, and has provided a few extra details of the system performance in an additional proprietary Technical Report MUAP-08001, "US-APWR Sump Strainer Performance," (Reference 1), which is mentioned in DCD 6.2.2.2.6, "ECC/CS Strainers."

ITAAC: The ITAACs associated with DCD Tier 2, Section 6.5.2 are given in DCD Tier 1 table 2.4.5-5, "Residual Heat Removal System Inspections, Tests, Analyses, and Acceptance Criteria."

TS: The TS associated with DCD Tier 2, Section 6.5.2 are given in DCD Tier 2, Chapter 16, Section 3.6.6, "Containment Spray System."

Topical Reports: There are no topical reports for this area of review.

Technical Reports: MHI Technical Report MUAP-08001-P, "US-APWR Sump Strainer Performance," Revision 1, September 2008.

US-APWR Interface Issues identified in the DCD: There are no US-APWR interface issues for this area of review.

Site Interface Requirements identified in the DCD: There are no site interface requirements for this area of review.

Cross-cutting Requirements (TMI, USI/GSI, Op Ex): None for this area of review.

RTNSS: There is no RTNSS for this area of review.

10 CFR 20.1406: There are no 10 CFR 20.1406 requirements for this area of review.

CDI: There is no CDI for this area of review.

6.5.2.3 Regulatory Basis

The relevant requirements of the Commission's regulations for this area of review, and the associated acceptance criteria, are given in Section 6.5.2 of NUREG-0800, "Containment Spray as a Fission Product Cleanup System," and are summarized below. Review interfaces with other SRP sections can be found in Section 6.5.2.I of NUREG-0800.

1. GDC 41, as it relates to containment atmosphere cleanup systems designed to control fission product releases to the reactor containment following postulated accidents.

2. GDC 42, as it relates to containment atmosphere cleanup systems designed to permit appropriate periodic inspections.

3. GDC 43, as it relates to containment atmosphere cleanup systems designed for appropriate periodic functional testing.

4. 10 CFR 52.47(b)(1), which requires that a DC application include the proposed ITAAC that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the DC is built and will operate in accordance with the DC, the provisions of the Atomic Energy Act of 1954, and the NRC's regulations.

6.5.2.4 Technical Evaluation

The staff performed its review using the guidance of SRP 6.5.2. These criteria also cite ANSI/ANS 56.5, “PWR and BWR Containment Spray System Design Criteria,” as a standard for system design (but not for pH control chemical). The SRP Section 6.5.2 contains a number of detailed acceptance criteria that should be met concerning system performance and the calculation of fission product removal from the containment atmosphere.

The CSS is an ESF which serves to remove both heat and fission products from the containment atmosphere. The subject of heat removal is discussed in Section 6.2.2 of this SER, while fission product removal is the focus of this review. This feature of the CSS is intended to mitigate the effects of DBAs, in which iodine transport and speciation is of paramount concern. The pH is an important consideration for iodine retention in water. Therefore, this review must consider pH in spray and sump solutions to assure that the assumed release of iodine to the atmosphere is appropriately modeled. Additionally, the staff performs an independent calculation of pH in conjunction with the review of the DBA radiological consequences analyses in Section 15.0.3 of this SER.

The RWSP is a large tank of borated water in the bottom of the containment. A minimum of 81,230 ft^3 (2300 m^3) is maintained at room temperature (70-120 °F, or 21-49 °C) with a nominal boric acid concentration of 4000 ppm. Water in the RWSP can be processed through the refueling water storage system to remove impurities and regulate the boric acid chemistry.

Containment sprays draw water from the RWSP and send it through CSS piping and out the spray nozzles near the top of the containment airspace. As described in DCD Section 6.5.2.2, approximately 60 percent of the total containment free volume is sprayed. Unsprayed regions are mostly comprised of rooms that are covered or sheltered physically from the main containment airspace. While these regions do not receive direct spray contact, they are open to convective mixing. As illustrated in DCD Figures 6.3-10, “Containment Spray Pattern Plan View at the NaTB Basket Installation Level,” and 6.3-11, Containment Spray Pattern Sectional View at the NaTB Basket Installation Level,” the sprays reach nearly all of the main containment airspace, including regions near the wall.

Spray water drains to the lower reaches of the containment where some of it returns to the RWSP through drain pipes. Some of the spray water near the periphery of the containment will fall into baskets containing NaTB. This water will dissolve the NaTB and drain back to the RWSP.

A minimum of 44,100 lb. (20,000 kg) of NaTB is stored in three containers along the containment wall, which are in an effective location to receive abundant spray water. This mineral dissolves readily and forms a strongly basic solution. Thus, as water cycles through the sprays several times, the pH of the RWSP will gradually rise until all the NaTB is dissolved. This material is important in accident mitigation, therefore the applicant has listed TS and surveillance requirements to assure both the quantity and quality of stored NaTB (DCD Chapter 16, SR 3.5.5 and B 3.5.5).

Initially, the RWSP water has a pH of about 4.5 (4000 ppm boric acid), according to the calculational procedure used in the EPRI PWR Primary Water Chemistry Guidelines. The DCD mentions an initial pH of 4.3 (Sections 6.5.2.1, “Design Bases,” 6.5.2.2, “System Design (for Fission Product Removal),” and 6.3.2.2.5, “NaTB Baskets and NaTB Basket Containers”), which rises above 7 over the course of 12 hours during a DBA. This timing is in apparent conflict with the SRP recommendation that pH be raised above seven before sprays actuate. The applicant was asked to explain the timing and supply details of these pH calculations. Initially, in its response to **RAI 234-2040, Question 06.05.02-1**, dated March 24, 2009, the applicant maintained that the release of CsOH from the fuel-cladding gap of the damaged fuel rods would raise the water pH above 7 very early in a DBA. Since the staff was unable to corroborate this calculation, the applicant supplied complete details of their pH calculation in response to **RAI 416-2912, Question 06.05.02-5**, dated August 28, 2009. On review, the staff discovered that the applicant’s calculation failed to account for the buffering effect of polyborate species, and was therefore invalid. In its response to **RAI 460-3484, Question 06.05.02-7**, dated November 13, 2009, the applicant presented a revised calculation that included all relevant factors. However, the revised pH did not rise above 7 until 12 hours after spray actuation (15 hours after accident initiation). In addition, the pH barely exceeded 7 at 100 °C (212 °F), suggesting the possibility that it may be less than 7 for lower temperatures.

The SRP states explicitly that the pH must be greater than 7 to avoid re-evolution of iodine from sump water. Since the applicant’s DBA sequence assumed retention of iodine for the period before pH exceeded 7, they were asked to explain and defend this element of the sequence, or to recalculate the DBA releases without iodine retention. The applicant’s explanation, given as response to **RAI 517-4088, Question 06.05.02-8**, dated February 25, 2010, stated that precedent from licensing actions at other plants should excuse the consideration of iodine re-evolution during the first phase (roughly the first day) of a DBA. Due to the peculiarities of individual plants, the staff does not rely on the relevance of these other licensing decisions without further justification.

In its response to **RAI 715-5262, Question 06.05.02-9**, dated May 31, 2012, the applicant furnished a calculation using more realistic parameters than used in the DBA sequence calculation in the DCD. This revised calculation estimated iodine revolatilization using equilibrium partitioning between water and gas spaces and is based on the actual computed pH as it rises from 4.3 to 7. The amount of iodine revolatilized reaches a peak of [ ] at about three hr after spray actuation, then declines steadily to negligible levels within one additional hour. (This result has been verified with a similar calculation performed by the staff.) The applicant calculated effects that this revolatilization had on release and dose calculations, including additional iodine released. They also showed, in their RAI response sensitivity analysis, that the dose results using realistic parameters and revolatilization were very similar (even slightly lower) than the DBA doses calculated in the DCD using conservative parameters with no consideration for iodine revolatilization. Based on this comparison, the staff considers that the applicant has satisfactorily addressed the issue, and that the requirements of SRP 6.5.2 have been met. Therefore, **RAI 517-4088, Question 06.05.02-8; RAI 715-5262, Question 06.05.02-9;** and **RAI 794-5871, Questions 06.05.02-10, -11, -12 and -13** are resolved and closed.

Some of the spray water falls into containment regions that do not drain effectively into the RWSP. These are described in the DCD as "ineffective pools." While the RWSP itself may gradually increase in pH to levels that provide acceptable iodine retention, the ineffective pools may not. In fact, their pH is uncertain, because they do not participate in the continual dissolution of NaTB that occurs as water cycles through sprays and RWSP. This situation is complicated by the large quantity of water that can end up in ineffective pools. The DCD states that 297,000 gal (1124 m^3) of water can accumulate in the containment in ineffective pools (DCD Figure 6.2.1-10, "Volume of Ineffective Water," also Table 3-10 of MUAP-08001, "Upstream Effect Hold-up Volumes"), which is nearly half of the total capacity of the RWSP. Because this appeared not to meet the SRP guidelines, the staff requested the applicant to clarify this matter. In its response to **RAI 234-2040, Question 06.05.02-2**, dated March 24, 2009, the applicant noted that about 35 percent of spray water will distribute to areas that do not immediately drain to the RWSP, and therefore are not part of the main circulation pattern that dissolves NaTB. These regions are identified primarily as three containment locations: the reactor cavity, the containment reactor coolant drain pump room, and the containment recirculation air distribution chamber. The last volume is interconnected to the RWSP by piping, and should have a pH similar to that of the RWSP. The first two volumes are calculated to be filled completely within the first few hours of a DBA, after which their surfaces will be exposed to (and covered by) spray water that drains back to the RWSP. Thus, these volumes do not directly mix with the recirculating spray water whose pH is lowered through continual dissolution of NaTB. However, the upper reaches of these volumes are eventually covered with the recirculating water. While this contact may gradually lower the pH of the entire volume through diffusion or low-grade mixing, more importantly it establishes a high-pH barrier that iodine would have to penetrate in order to revolatilize. The applicant has described this process in response to **RAI 416-2912, Question 06.05.02-6**, dated August 18, 2009, and furnished calculations that indicate negligible revolatilization would occur. Thus, the ineffective pools mentioned in the DCD do not appear to be major repositories of iodine that could easily be revolatilized.

The importance of pH control in containment water is to ensure retention of fission product iodine. Many forms of iodine are scrubbed from the containment atmosphere by spray water and deposited in the RWSP. Under neutral or basic conditions iodine will stay dissolved in water, primarily as iodide (I^-) and, to a lesser extent, iodate (IO_3^-). However, acidic solutions can produce more volatile species (chiefly I_2), which can evaporate from the water and re-enter the containment airspace. In this form, it is much more likely to be released outside of containment through venting or leakage.

Assessing the consequences of a DBA requires calculation of fission product removal from the containment airspace by containment sprays. The acceptable methods for such calculation are described in SRP Section 6.5.2 (III.4) and constitute first-order rate equations. The applicant does not include noble gasses or organic iodides consistent with the guidelines in the SRP. The applicant does include removal of elemental iodine (I_2) by natural deposition onto wetted wall surfaces which is also consistent with the model in the SRP. (The calculation itself, including parameter values used, is described in DCD Appendix 15A, Sect. 15A.1.2.1.) The applicant credits removal of particulate iodine by the sprays using a model given in SRP 6.5.2 for particulates. However, the applicant does not include any details of the calculation or parameters involved, and the staff requested the applicant supply this information. In its response to **RAI 234-2040, Question 06.05.02-4**, dated March 24, 2009, the applicant supplied values for these parameters that the staff finds are reasonable.

Additionally, the applicant credits removal of particulate iodine through natural processes using a model by Powers, that is consistent with the models in the SRP, and also follows guidance in RG 1.183, Appendix A, position 3.2, with respect to credit for aerosol natural deposition in the containment. The staff's review of aerosol removal through natural processes is discussed in more detail in Section 15.0.3 of this SER.

The removal of iodine by rate processes is not unlimited, as there is a maximum amount of iodine that can be dissolved in a given volume of water. This equilibrium is established for each species between the two phases (water and air). It is commonly measured using the Henry's Law constant, i.e., the ratio between equilibrium concentrations (C^*) in liquid and gas:

$$H = C_l^* \Big/ C_g^* \ .$$

Sometimes the inverse of *H* is called the partition coefficient; to maintain consistency with the SRP, the phrase "partition coefficient" is identical to *H*. The decontamination factor (*DF*) is the ratio of activity before and after some decontamination process. In this case, it is a ratio of total activity to activity remaining in the gas phase after some has dissolved in water, and can be derived from the partition coefficient using the equation:

$$DF = \frac{n_g + n_l}{n_g} = 1 + \frac{HV_l}{V_g}$$

where n_l, V_l, n_g, V_g represent moles and volumes of liquid and gas, respectively. The equilibrium partitioning between air and water for I_2 depends strongly on temperature and solution pH. The analysis must assume sufficient mixing by recirculation so that an equilibrium value is a reasonable estimate of the end result of transient dissolution and revaporization processes. The applicant did not describe a calculation of equilibrium partitioning, although they did note in DCD Section 6.5.2.3.3, "Iodine Decontamination Factor (DF)," that all I_2 removal ceases after a DF of 200 has occurred and matches the maximum value allowed by the SRP (III.4.D). The staff requested additional information regarding the applicant's calculation of equilibrium partitioning. In its response to **RAI 234-2040, Question 06.05.02-3**, dated April 22, 2009, the applicant provided detailed calculations describing the partitioning of iodine species between the liquid and gas phases. The applicant used references that the staff finds technically acceptable for the parameters involved and included all likely solution components. The calculation assumed a pH equal to 7, which is satisfactory after the pH has indeed risen to this level. Nonetheless, the analysis discussed in the responses to RAIs 715-5262 and 794-5871 showed that the more realistic dose calculation, which includes the impact of iodine re-evolution from the pools in containment, estimates offsite and control room doses which are bounded by the more conservative DBA dose analysis in the DCD. Therefore, the staff finds that the applicant has acceptably shown that the DBA dose analyses in DCD Chapter 15 bound the effects of potential iodine re-evolution from the sump water during time periods when the pH is not greater than 7.

6.5.2.5 Combined License Information Items

There are no COL information items for this area of review identified in DCD Section 6.5.2 or Table 1.8-2.

6.5.2.6 Conclusions

The applicant has described the operation of the containment sprays, including the control of pH in containment water and the removal of fission products by the sprays. The applicant has furnished a number of additional calculations that address the issue of iodine revolatilization, so as to ensure that regulatory requirements are satisfied. The staff therefore concludes, based on the information supplied by the applicant, that the requirements will be met.

6.5.3 Fission Product Control Systems

6.5.3.1 Introduction

The release of fission products following a postulated DBA is mitigated by several US-APWR design features. The major features that provide this function include primary and partial secondary containment structures and ventilation systems. This section describes those features that prevent or limit the release of fission products from primary containment. The DBA radiological dose 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 DCD Tier 2, Chapter 15 for each of the DBAs that credit such features.

The description of the fission product control systems and structures is reviewed to: (a) provide a basis for developing the mathematical model for design basis LOCA dose computations, (b) verify that the values of certain key parameters are within pre-established limits, (c) confirm the applicability of important modeling assumptions, and (d) verify the functional capability of ventilation systems used to control fission product releases. The parameters that must be established for use in the calculation of the radiological consequences of accidents in Chapter 15 of the SER and the systems whose functions must be reviewed are outlined below.

6.5.3.2 Summary of Application

DCD Tier 1: The Tier 1 information associated with this section is found in DCD Tier 1, Sections 2.4.5, 2.4.6, 2.11.2, and 2.11.3.

DCD Tier 2: The applicant has provided a DCD Tier 2 description in Section 6.5.3, summarized here in part, as follows:

> The US-APWR does not require a containment purge system. The removal of iodine and particulates by containment spray reduces fission product leakage to the environment below the guidelines. The analysis presented in Chapter 15 details the radiological consequences of the US-APWR design following a DBA that releases radioactive material into the containment. The inservice leakage rate test program detailed in Section 6.2.6 monitors and protects the assumed containment leakage rate.
>
> The application also explains that the following systems or structures are not applicable to the fission product control function for the US-APWR:
>
> - ESF Hydrogen Purge System.

- Secondary Containment.
- Ice Condenser.
- Pressure Suppression Pool.

ITAAC: The ITAAC associated with DCD Tier 2, Section 6.5.3 are given in DCD Tier 1, Section 2.2.

TS: The TS associated with DCD Tier 2, Section 6.5.3 are given in DCD Tier 2, Chapter 16, Sections 3.5.5 and B3.5.5.

Topical Reports: There are no topical reports for this area of review.

Technical Reports: There are no technical reports for this area of review.

US-APWR Interface Issues identified in the DCD: There are no US-APWR interface issues for this area of review.

Site Interface Requirements identified in the DCD: There are no site interface requirements for this area of review.

Cross-cutting Requirements (TMI, USI/GSI, Op Ex): None for this area of review.

RTNSS: There is no RTNSS for this area of review.

10 CFR 20.1406: There are no 10 CFR 20.1406 requirements for this area of review.

CDI: There is no CDI for this area of review.

6.5.3.3 Regulatory Basis

The relevant requirements of the Commission regulations for this area of review, and the associated acceptance criteria, are given in Section 6.5.3 of NUREG-0800 and are summarized below. Review interfaces with other SRP sections can be found in Section 6.5.3 of NUREG-0800.

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 52.47(b)(1), which requires that a DC application include the proposed ITAAC that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the DC is built and will operate in accordance with the DC, the provisions of the Atomic Energy Act of 1954, and the NRC's regulations.

Acceptance criteria adequate to meet the above requirements include:

1. GDC 41 requires that fission product control systems be provided in the reactor containment to reduce the concentration of fission products released to the environment after postulated accidents. In the primary containment, fission product control systems include spray, filtration systems, and pressure suppression devices; in the secondary containment, such systems include fission product removal and holdup systems. The function of the fission product control systems is to mitigate the radiological offsite consequences of postulated accidents by decreasing the concentration of fission products available for release to the environment. The review and evaluation of these fission product control systems is the subject of Section 6.5.3 of NUREG 0800. 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. Meeting the requirements imposed by GDC 41 provides assurance that offsite radiation doses resulting from postulated accidents will be within the doses specified in 10 CFR Part 52.47(a)(2).

2. GDC 42 requires that the fission product control systems be designed to permit periodic inspections of important components. The fission product control systems are provided to ensure that offsite radiation doses resulting from postulated accidents are within the doses specified in 10 CFR Part 52.47(a)(2).

   The ability to perform periodic inspection is essential for ensuring that components of the systems will function as designed. Testing and inspection criteria for fission product control systems are outlined in RG 1.52, Regulatory Positions C.5 and C.6. Meeting the requirements imposed by GDC 42 provides assurance that offsite radiation doses resulting from postulated accidents will be within the doses specified in 10 CFR Part 52.47(a)(2).

3. GDC 43 requires that fission product control systems be designed to permit periodic functional testing of important components. Fission product control systems are provided to ensure that offsite radiation doses resulting from postulated accidents are within the doses specified in 10 CFR Part 52.47(a)(2).

Periodic functional testing is essential for ensuring that components of the systems will function as designed. Testing and inspection criteria for fission product control systems are outlined in RG 1.52, Regulatory Positions C.5 and C.6. Meeting the requirements imposed by GDC 43 provides assurance that offsite radiation doses resulting from postulated accidents will be within the doses specified in 10 CFR Part 52.47(a)(2).

6.5.3.4 Technical Evaluation

The US-APWR design includes a primary containment consisting of a pre-stressed, post-tensioned concrete structure, as described in DCD, Tier 2, Chapter 3, Section 3.8.1. The US-APWR design also includes a partial secondary containment to collect and filter containment releases into the penetration areas and the safeguards component areas. These features are discussed below. Review of the fission product mitigating function of the ventilation systems and ESF filter systems is discussed in SER Section 6.5.1.

GDC 41 requires that fission product control systems be provided in the reactor containment to reduce the concentration of fission products released to the environment after postulated accidents. The US-APWR design provides a primary containment, an ESF containment spray system and ESF annulus emergency filtration system for reduction of fission products released from the containment.

GDC 42 requires that fission product control systems be designed to permit periodic inspections of important components. GDC 43 requires that fission product control systems be designed to permit periodic functional testing of important components. The US-APWR TS include testing and inspection programs for the primary containment structures, containment sprays and ESF atmosphere cleanup systems.

The staff's discussion of its review in Sections 6.2, 6.5.1 and 6.5.2 of this SER includes the US-APWR design conformance to GDCs 41, 42 and 43 and with the ITAAC requirements of 10 CFR 52.47(b)(1).

6.5.3.4.1 Primary Containment

US-APWR TS program 5.5.16, "Containment Leakage Rate Testing Program" states that the maximum allowable containment leakage rate, L_a, at Pa (57.5 psig), shall be 0.10 percent of containment air weight per day. The DBA radiological consequences analysis containment leakage assumption based on 0.15 percent per day bounds the TS testing requirements and is acceptable. The US-APWR design does not include an ESF hydrogen purge system, but the DBA radiological consequences analysis for the LOCA does analyze the impact of the non-safety purge release prior to purge isolation. The staff's review of the primary containment is discussed in Section 6.2 of this SER.

The applicant's DBA radiological consequences analyses take credit for radioactive material removal through natural processes and by containment spray. The staff's review of radioactive material removal through natural processes and containment spray is discussed in Section 15.0.3 of this SER. The staff's review of the containment spray system is discussed in Section 6.5.2 of this SER.

6.5.3.4.2 Secondary Containment

DCD 6.5.3.2 states that the US-APWR design includes a partial secondary containment. In the DBA LOCA and REA analyses in DCD Chapter 15, credit is taken for collection and filtration in the penetration areas of 50 percent of the containment leakage. Although a secondary containment in the manner of a structure that completely surrounds the primary containment is not part of the US-APWR design, the NRC staff considers the LOCA and REA analyses are treating the penetration areas as a partial dual containment, given that the analyses credit fission product removal by filtration systems in those areas. SRP Section 6.5.3, "Fission Product Control Systems and Structures," in addition to the primary and secondary containments also discusses other fission product control structures for collection and control of post-accident releases. The SRP acceptance criteria for secondary containments states that partial dual containments should meet the same basic criteria as secondary containments, in order for credit for fission product removal to be found acceptable by the NRC staff. The DCD Chapter 15 DBA radiological consequences analyses for the LOCA and control rod ejection accident take credit for removal of particulate iodine by filtration of the containment leakage to the penetration areas and safeguards component areas by the annulus emergency exhaust system. The staff's review of the DBA radiological consequences analysis modeling of the collection and filtration of the primary containment leakage in the penetration areas is discussed in Section 15.0.3 of this SER. The staff's review of the annulus emergency filtration system is discussed in Section 6.5.1 of this SER.

6.5.3.5 Combined License Information Items

There are no COL information items identified for DCD Section 6.5.3.

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 and containment spray. The US-APWR fission product and removal control incorporates passive removal by natural processes in addition to the active removal by ESF systems. The DCD discusses the performance capability of each system used for fission product control, including operation following a DBA.

As described above, and in conjunction with the review described in SER Sections 6.2, 6.5.1 and 6.5.2, the US-APWR design is in compliance with applicable GDC of 10 CFR 50, Appendix A, and meets the recommendations in SRP 6.5.3. The review of the DCD sections indicates that the design of fission product control systems and structures is acceptable and meets the requirements of GDCs 41, 42, and 43 and applicable requirements of 10 CFR 52.47.

6.6 Inservice Inspection and Testing of Class 2 and 3 Components

6.6.1 Introduction

ISI programs are based on the requirements of 10 CFR 50.55a, which requires that 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 Boiler and Pressure Vessel Code. ISI includes preservice examinations prior to initial plant startup as required by Subsubarticles IWC-2200 and IWD-2200 of Section XI of the ASME Code.

6.6.2 Summary of Application

DCD Tier 1: The Tier 1 information associated with this section is found in DCD Tier 1, Section 2.3, "Piping Systems and Components."

DCD Tier 2: The applicant has provided a DCD Tier 2 description in Section 6.6, summarized here in part, as follows:

> Regular and periodic examinations, tests, and inspections of pressure retaining components and supports are required by 10 CFR 50.55a(g). This section discusses the ISI program to address these requirements. This section includes preservice and inservice examinations and system pressure tests. The COL applicant is responsible for the preparation of a preservice inspection program (non-destructive baseline examination) and an ISI program for ASME Code Section III Class 2 and 3 systems, components (pumps and valves), piping, and supports.

ITAAC: The ITAAC associated with DCD Tier 2, Section 6.6 are given in DCD Tier 1, Section 2.2, “Structural and Systems Engineering.”

TS: The TS associated with DCD Tier 2, Section 6.6 are given in DCD Tier 2, Chapter 16, Section 5.5.8, “Inservice Testing Program.”

Topical Reports: There are no topical reports for this area of review.

Technical Reports: There are no technical reports for this area of review.

US-APWR Interface Issues identified in the DCD: There are no US-APWR interface issues for this area of review.

Site Interface Requirements identified in the DCD: There are no site interface requirements for this area of review.

Cross-cutting Requirements (TMI, USI/GSI, Op Ex): None for this area of review.

RTNSS: There is no RTNSS for this area of review.

10 CFR 20.1406: There are no 10 CFR 20.1406 requirements for this area of review.

CDI: There is no CDI for this area of review.

6.6.3 Regulatory Basis

The relevant requirements of the Commission regulations for this area of review, and the associated acceptance criteria, are given in Section 6.6 of NUREG-0800 and are summarized below. Review interfaces with other SRP sections can be found in Section 6.6 of NUREG-0800.

1. 10 CFR 50.55a, 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.

2. GDC 36, “Inspection of emergency core cooling system,” found in Appendix A to 10 CFR Part 50, as it pertains to designing the ECCS to permit appropriate periodic inspection of important safety components, such as spray rings in the reactor pressure vessel.

3. GDC 37, “Testing of emergency core cooling system,” found in Appendix A to 10 CFR Part 50, as it pertains to designing the ECCS to permit appropriate testing to assure structural integrity, leak tightness, and the operability of the system.

4. GDC 39, “Inspection of containment heat removal system,” found in Appendix A to 10 CFR Part 50, 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.

5. GDC 40, “Testing of containment heat removal system,” found in Appendix A to 10 CFR Part 50, as it pertains to designing the containment heat removal system to permit appropriate pressure and functional testing.

6. GDC 42, “Inspection of containment atmosphere cleanup systems,” found in Appendix A to 10 CFR Part 50, as it pertains to designing the containment atmospheric clean up system to permit appropriate inspection of components such as filter frames and ducts.

7. GDC 43, “Testing of containment atmosphere cleanup systems,” found in Appendix A to 10 CFR Part 50, 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.

8. GDC 45, “Inspection of cooling water system,” found in Appendix A to 10 CFR Part 50, as it pertains to designing the cooling water system to permit appropriate periodic inspection of important components, such as HX.

9. GDC 46, “Testing of cooling water system,” found in Appendix A to 10 CFR Part 50, as it pertains to designing the cooling water system to permit appropriate pressure and functional testing to assure structural and leaktight integrity of it components.

10. 10 CFR 52.47(b)(1), which requires that a DC application include the proposed ITAAC that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the DC is built and will operate in accordance with the DC, the provisions of the Atomic Energy Act of 1954, and the NRC's regulations.

Acceptance criteria adequate to meet the above requirements include:

1. Components Subject to Inspection - 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 Article NCA 2000 of Section III of the ASME Code.

2. Accessibility - The design and arrangement of Class 2 and 3 systems should include allowances for adequate clearances to conduct the examinations specified in Articles IWC 2000 and IWD 2000 at the frequency specified.

3. Examination Categories and Methods - The examination categories and requirements specified in the SAR are acceptable if they are in agreement with the rules of Articles IWA 2000, IWC 2000, and IWD 2000.

4. Inspection Intervals - The ISI program schedule provided in the SAR is acceptable if the required examinations and pressure tests are specified for completion during each ten year interval, hereinafter designated as the “inspection interval,” and as required by ASME Code Section XI, Articles IWA 2000, IWC 2000, and IWD 2000.

5. Evaluation of Examination Results - The methods for evaluation of examination results are reviewed for compliance with Articles IWC 3000 and IWD 3000 in the ASME Code.

6. System Pressure Tests - The program provided in the SAR for Class 2 and 3 system pressure testing is acceptable if it meets the criteria of ASME Code Section XI, Articles IWC 5000 and IWD 5000.

7. Augmented ISI to Protect Against Postulated Piping Failures - The augmented ISI program for high energy fluid system piping between containment isolation valves is acceptable if it specifies certain requirements.

8. Code Exemptions - The exemptions from Code examination requirements identified by the applicant are acceptable if they have been permitted by Subject IWC 1220 or IWD 1220 of Section XI of the ASME Code.

9. Relief Requests - Request for relief from the ASME Code Section XI examination requirements that are found to be impractical due to the limitations of design, geometry, or materials of construction of components are evaluated in accordance with 10 CFR 50.55a.

10. Code Cases - The exemptions from Code examination requirements identified by the applicant or licensee are acceptable if they have been permitted by appropriate ASME Code cases.

11. Operational Programs - For COL reviews, the description of the operational program and proposed implementation milestones for the preservice inspection and ISI and testing programs for Class 2 and 3 components are reviewed in accordance with the requirements of 10 CFR 50.55a.

6.6.4 Technical Evaluation

The staff reviewed US-APWR DCD, Tier 2, Section 6.6, “Inservice Inspection of Class 2 and 3 Components,” in accordance with Section 6.6, “Inservice Inspection of Class 2 and 3 Components,” of the SRP. The ASME Code of record (edition) for the design of the US-APWR is the 2001 edition with the 2003 Addenda of the ASME Boiler and Pressure Vessel Code, as stated under DCD Section 5.2.1.1, “Compliance with 10 CFR 50, Section 50.55a.”

6.6.4.1 Components Subject to Inspection

The NRC staff reviewed the applicant’s definition of ASME Code Class 2 and 3 components and systems subject to an ISI program to ensure that it is in agreement with the NRC quality group classification system or the definitions in Article NCA-2000 of Section III of the ASME Code. In addition, the NRC staff reviewed any exceptions to the testing of components that are different from the ASME XI requirements.

US-APWR DCD, Section 6.6.1, “Components Subject to Examination,” states that DCD Section 3.2, “Classification of Structures, Systems, and Components,” identifies the ASME Code Section III Class 2 and 3 components as corresponding Quality Group B and C components. Class 2 and 3 pressure-retaining components and supports subject to examination include pressure vessels, piping, pumps, valves, and their bolting. Preservice and inservice examinations, tests and inspections are performed in accordance with ASME Code Section XI including associated Mandatory Appendices, Table IWC-2500-1 for Class 2 components and Table IWD-2500-1 for Class 3 components. In addition, the applicant states that the specific edition and addenda of the ASME Code used to determine the requirements for the inspection plan for the initial and subsequent inspection intervals are to be delineated in the ISI and IST program. However, US-APWR DCD, Section 5.2.4.1, "Inservice Inspection and Testing Program," states, "The 2001 edition with the 2003 Addenda of the ASME Code is used to determine the requirements for the initial and subsequent inspection intervals in the ISI and IST programs." Based on the inconsistency between the two DCD sections, the staff issued **RAI 232-2114, Question 06.06-1**.

In its response to **RAI 232-2114, Question 06.06-1**, dated March 26, 2009, the applicant stated that DCD, Section 6.6.10 will be revised to reflect that the rules for ISI of nuclear power plant components will be in accordance with ASME Section XI, 2001 Edition with the 2003 Addenda. Based on the proposed changes, the staff concludes that the ISI program for Class 2 and 3 components is consistent with the Class 1 program, and is therefore, acceptable. The staff has also confirmed that the proposed changes have been included in Section 6.6.10 of the US-APWR DCD. Therefore, **RAI 232-2114, Question 06.06-1, is resolved and closed.**

Section III of the ASME Code presents the construction requirements for Class 2 and 3 components, and ASME Section XI defines the preservice and inservice (ISI) examinations requirements. The design follows the ASME Code Section III as required by 10 CFR 50.55a. The 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 NRC staff reviewed the DCD's description of the design and arrangement of Class 2 and 3 systems to ensure that it includes allowances for adequate clearances to conduct the examinations specified in 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 Subarticle IWA-1500. Special design considerations should also be given to those systems that are intended to be examined during normal reactor operation or other operational configurations that might cause Code examination methods to be ineffectual.

US-APWR DCD, Section 6.6.2, "Accessibility," states that the physical arrangement of ASME Code Class 2 and 3 components is designed to allow personnel and equipment access "to the extent practical" to perform the required inservice examinations specified by the ASME Code Section XI and mandatory appendices. Section 6.6.3, "Examination Techniques and Procedures," states, "To the maximum extent possible, sufficient radial clearances are provided around pipe or component welds requiring volumetric or surface examination for inservice inspection." The DCD also states that removable hangers are provided, "as necessary and practical," to facilitate inservice inspection. However, the US-APWR DCD states that the piping arrangement allows for adequate separation of piping welds so that space is available to perform ISI, and modules fabricated offsite are designed and engineered to provide access for ISI and maintenance activities. The staff notes that the phrases "to the extent practical" and "to the maximum extent possible" are inconsistent with a design that enables the performance of PSI/ISI examinations by eliminating interferences due to design, geometry, or materials of construction. 10 CFR 50.55a(g)(3)(i) and (3)(ii) require that for a boiling or pressurized water-cooled nuclear power facility whose construction permit under this part, or design certification, design approval, combined license, or manufacturing license under Part 52 of this chapter, was issued on or after July 1, 1974, components (including supports) classified as Class 1, 2, and 3 must be designed and be provided with access to enable the performance of inservice examination and must meet the preservice examination requirements set forth in the editions and addenda of Section XI of the ASME Code incorporated by reference. Based on the above discussion, the staff issued **RAI 233-2115, Question 06.06-2**.

In its response to **RAI 233-2115, Question 06.06-2**, dated April 17, 2009, the applicant stated that the phrase “to the extent practical” plus the phrase “to the maximum extent practical” will be removed from the DCD, Sections 6.6.2 and 6.6.3. In addition, the applicant proposed to add the statement that for a limited number of austenitic welds, where two-sided access for UT examinations is difficult or not possible, an inspection method that complies with the performance demonstration requirements of ASME Section XI, Appendix VIII and 10 CFR 50.55a(b)(2)(xv)(A)(2) will be provided. The staff concludes from the applicant’s response that the proposed changes assure that the ASME Code required volume will be obtained during inservice examinations. Furthermore, pipe hangers and supports are positioned to accommodate weld inspection. The design incorporates permanent and temporary platforms to provide access, along with sufficient lighting and power to perform the required inspections. The design incorporates access to enable the conduct of inservice examinations required under ASME Section XI, consequently, the staff concludes that the US-APWR design meets the accessibility acceptance criteria established under IWA-1500 and the regulations under 10 CFR 50.55a(g)(3)(ii), 10 CFR 50.55a(b)(2)(xv)(A)(2) and is therefore, acceptable. The staff has also confirmed that the proposed changes described in the RAI response have been made to Section 6.6.2 and Section 6.6.3 of the US-APWR DCD. Therefore, **RAI 233-2115, Question 06.06-2, is resolved and closed.**

6.6.4.3 Examination Categories and Methods

The NRC staff reviewed the examination categories and methods specified in the DCD for compliance 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 IWC-2000 and IWD-2000 must be examined, at least to the extent specified. The requirements of Article IWC-2000 and IWD-2000 also list the methods of examination for the components and parts of the pressure-retaining boundary.

The NRC staff reviewed the applicant’s examination techniques and procedures used for preservice inspection or ISI of the system to verify that they meet the following criteria:

- The methods, techniques, and procedures for visual, surface, or volumetric examination are in accordance with Article IWC-2000 and IWD-2000.

- Alternative examination methods, combinations of methods, or newly developed techniques to those above are acceptable provided that the results are equivalent or superior. The acceptance standards are given in Articles IWC-3000 and IWD-3000.

- The methods, procedures, and requirements regarding qualification of personnel performing ultrasonic examination reflect the requirements provided in Appendix VII, “Qualification of Nondestructive Examination Personnel for Ultrasonic Examination,” to Division 1 of ASME Code, Section XI.

- 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.

US-APWR DCD, Tier 2, Section 6.6.3, “Examination Techniques and Procedures,” discusses examination techniques, categories, and methods. The visual, surface, and volumetric examination techniques and procedures agree with the requirements of Articles IWA-2000, IWC-2000, and IWD-2000. The examination acceptance standards are in accordance with Articles IWC-3000 and IWD-3000. The 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 DCD also states that qualification of ultrasonic procedures, equipment, and personnel will be in accordance with Appendix VII and VIII of the ASME Code. The DCD lists the pertinent component categories for Class 2 and 3 components. The staff compared the listing of categories with the categories for Class 2 and 3 components under IWC-2500 and IWD-2500, and found that the listed categories meet Code requirements.

On the basis that the examination methods and categories applied to Class 2 and 3 components comply with the requirements of ASME Code, Section XI, the staff finds examination categories and methods for the EPR design of Class 2 and 3 components to be acceptable.

6.6.4.4 Inspection Intervals

The required examinations and pressure tests must be completed during each inspection interval. In addition, the scheduling of the program must comply with the provisions of Article IWA-2000, IWC-2000, and IWD-2000 as related to inspection intervals of the ASME Code, Section XI. US-APWR DCD Tier 2, Section 6.6.4, “Inspection Intervals,” discusses inspection intervals. It states that the intervals are established as defined in Articles IWC-2400 and IWD-2400 of the ASME Code, Section XI. The inspection intervals specified for the US-APWR Class 2 and 3 components are consistent with the definitions in Section XI of the ASME Code, and therefore, are acceptable.

6.6.4.5 Evaluation of Examination Results

The NRC staff reviewed the DCD to ensure that standards for examination evaluation in the program for flaw evaluation comply with the requirements of ASME Code, Section XI, Article IWC-3000, and IWD-3000 “Acceptance Standards.” The staff also reviewed the DCD to ensure that the proposed program regarding repairs of unacceptable indications or replacement of components containing unacceptable indications meets 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.”

US-APWR DCD 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 US-APWR Class 2 and 3 components meets the requirements of ASME Code, Section II, and is therefore, acceptable.

6.6.4.6 System Pressure Tests

The NRC staff reviewed the DCD to ensure that pressure-retaining Code Class 1 component leakage and hydrostatic pressure test program are in accordance with the requirements of Section XI, Articles IWC-5000 and IWD-5000. The US-APWR DCD Tier 2, Section 6.6.7, "System Pressure Tests," states that Class 2 and 3 systems and components are pressure tested in accordance with Articles IWC-5000 and IWD-5000. It also states that system leakage testing may be performed in accordance with IWC-5220 and IWD-5220. It further defines a system leakage test as one requiring the segment of the system to be tested to be in service at system pressure that it will be performing its normal operating functions. This definition is consistent with the definition of a system leakage test in IWA-5000 and, is therefore, acceptable. Based on the applicant's use of the appropriate sections of the ASME Code for system pressure tests, the staff concludes the methodology meets 10 CFR 50.55a and is acceptable.

6.6.4.7 Augmented ISI to Protect Against Postulated Piping Failures

The NRC staff reviewed the DCD to ensure that the augmented ISI program for high-energy fluid system piping between containment isolation valves meet the following criteria:

- 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.

The US-APWR DCD, Tier 2, Section 6.6.8, "Augmented ISI to Protect against Postulated Piping Failures," states that "as noted in Section 6.6.2, the design and installed arrangement of US-APWR Class 2 and 3 components provide clearance adequate to conduct Code-required examinations. The COL applicants are required to have administrative programs that ensure plant design translates accurately into the construction phase." The applicant did not discuss in sufficient detail necessary for the staff to determine if the acceptance criteria were met. Based on the above, the staff issued **RAI 258-2116, Question 06.06-4**.

In its response to **RAI 258-2116, Question 06.06-4**, dated April 17, 2009, the applicant stated that portions of pressure retaining welds of Class 2 piping subject to examination is defined in accordance with ASME Section XI, IWC-2000, Examination Category C-F. The applicant also stated that the DCD will be changed to incorporate the above as described in its response to **RAI 233-2115, Question 06.06-2**. Based on its acceptable response to **RAI 233-2115, Question 06.06-2**, the proposed changes assure access to enable the performance of the required inservice examinations. Hand holes, ports, or removable sections of guard pipe are provided as necessary to enable examination. 100 percent volumetric examination of the circumferential and longitudinal welds during the inspection interval is performed in accordance with the requirements of IWC-2000 for Examination Category C-F welds. The staff concludes the US-APWR design incorporates the access provisions, 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. **RAI 258-2116, Question 06.06-4 is resolved and closed.**

6.6.4.8 Code Exemptions, Relief Requests and Code Cases

The NRC staff reviewed the DCD to ensure that exemptions from Code examination requirements identified by the applicant are consistent with those permitted by IWC-1220 or IWD-1220 of ASME Section XI. The US-APWR DCD Tier 2, does not address exemptions, relief requests and code cases for Class 2 and 3 components. The applicant stated that exemptions, relief requests to the Code and Code cases invoked is the responsibility of the COL applicant. The staff could not determine if the ASME Code requirements were met, therefore, **RAI 241-2118, Question 06.06-3** was issued.

In its response **RAI 241-2118, Question 06.06-3**, dated April 17, 2009, the applicant stated that the exact number of welds is still under development at this time, but for each type of weld, the accessibility requirements of ASME Code Section XI are applied to the design. Also, the applicant stated that there are no exemptions to the examination requirements for Class 2 and 3 components other than those allowed under IWC-1220 and IWD-1220. The applicant proposed to revise the fourth paragraph of Section 6.6.1 to state that exemptions include components as defined in ASME Section XI, IWC-1220 or IWD-1220 for Class 2 and 3 components, and that there are no additional exemptions expected. The proposed changes also state that no relief requests are necessary for PSI and first interval ISI examinations for Class 2 and 3 components, and that approved code cases that are listed in RG 1.147, "Inservice Inspection Code Case Acceptability, ASME Section XI, Division 1," may be used. The staff finds that the applicant's response is acceptable because proposed changes are in compliance with ASME Section XI with respect to exemptions and that the use of staff approved code cases as defined under RG 1.147 is in compliance with 10 CFR 50.55a. The staff has also confirmed that the proposed changes described in the RAI response have been included in Section 6.6.1 of the US-APWR DCD. Therefore, **RAI 241-2118, Question 06.06-3, is resolved and closed**.

6.6.4.9 General Design Criteria

GDCs 36, 37, 39, 40, 42, 43, 45, and 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 SER, the staff concluded that accessibility by design enabled the performance of inservice examinations and pressure testing, thereby meeting IWA-1500 of the ASME Code 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, 37, 39, 40, 42, 43, 45, and 46, and are therefore, acceptable.

RAI 232-2114, Question 06.06-1, RAI 233-2115, Question 06.06-2, RAI 241-2118, Question 06.06-3, and RAI 258-2116, Question 06.06-4, are all resolved and closed.

6.6.5 Combined License Information Items

The following is a list of COL item numbers and descriptions from Table 1.8-2 of the DCD.

Table 6.6.5-1
U.S. APWR Combined License Information Items

Item No.	Description	Section
6.6(1)	The COL Applicant is responsible for the preparation of a preservice inspection program (non-destructive baseline examination) and an inservice inspection program for ASME Code Section III Class 2 and 3 systems, components (pumps and valves), piping, and supports in accordance with 10 CFR50.55a(g), including selection of specific examination techniques and preparing appropriate inspection procedures.	6.6
6.6(2)	The COL Applicant is responsible for preparing an augmented inservice inspection program for high-energy fluid system piping.	6.6

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 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's ISI program description complies with the rules published in 10 CFR 50.55a and Section XI of the ASME Code, 2001 Edition through the 2003 Addenda. The ISI program will consist of a preservice and ISI plan(s). The staff concludes that the ISI program is acceptable and meets the inspection and pressure testing requirements of GDCs 36, 37, 39, 40, 42, 43, 45, and 46, and 10 CFR 50.55a. This conclusion is based on the applicant's meeting the requirements of the ASME Boiler and Pressure Vessel Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components."