CHAPTER 3 – DESIGN of STRUCTURES, SYSTEMS, COMPONENTS, and EQUIPMENT

3.0 Design of Structures, Systems, Components, and Equipment

Chapter 3 of this safety evaluation report (SER) summarizes the results of the review by the staff of the U.S. Nuclear Regulatory Commission (NRC), hereinafter referred to as the staff, of Chapter 3 of Mitsubishi Heavy Industries, Ltd. (MHI) (hereinafter referred to as the applicant) MUAP-DC003, Design Control Document (DCD), Revision 3 (March 2011), for the design certification (DC) of its United States - Advanced Pressurized Water Reactor (US-APWR).

3.1 Conformance with NRC General Design Criteria

DCD Tier 2, Section 3.1, "Conformance with NRC General Design Criteria," addresses how the US-APWR design conforms to the general design criteria (GDC) of Appendix A, "General Design Criteria," to Part 50, "Domestic Licensing of Production and Utilization Facilities," of Chapter 1, "Nuclear Regulatory Commission," of Title 10, "Energy," of the *Code of Federal Regulations* (10 CFR Part 50, Appendix A).

Chapter 3 of NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, LWR [light-water reactor] Edition," begins with Section 3.2.1, "Seismic Classification;" therefore, the staff did not perform a specific review of DCD Tier 2, Section 3.1. In general, conformance to the applicable GDCs is discussed in each individual section of this chapter.

3.1.5 Combined License Information Items

The following is a list of Combined License (COL) item numbers and descriptions from Table 1.8-2, "Compilation of All Combined License Applicant Items for Chapters 1-19," of the DCD related to conformance with NRC GDC.

Table 3.1-1
US-APWR Combined License Information Items

Item No.	Description	Section
3.1(1)	The COL Applicant is to provide a design that allows for the appropriate inspections and layout features of the ESWS [Essential Service Water System].	3.1.7

The staff determined the above listing to be complete COL Information Item 3.1(1) is described in DCD Tier 2, Section 3.1.4.16, "Criterion 45 – Inspection of Cooling Water System," The staff finds COL Information Item 3.1(1) acceptable since a portion of the ESWS is site-specific. The staff notes that the applicant chose to include COL Information 3.1(1) in DCD Tier 2, Section 3.1 as a special case. Alternatively it could have been included in DCD Tier 2, Section 9.2.1, "Essential Service Water System," which describes design criteria for the ESWS. Also, the list adequately describes actions necessary for the COL applicant or licensee. No additional COL information items need to be included in DCD Tier 2, Table 1.8-2 or Section 3.1.

3.2 Classification of Structures, Systems, Components and Equipment

In Section 3.2, "Classification of Structures, Systems, and Components," of the DCD, the applicant states that SSCs in the US-APWR are classified by safety classification, seismic category, quality group (QG), and codes and standards. As broadly defined in NRC regulations, SSCs important to safety are those that provide reasonable assurance that the NRC-licensed facility can be operated without undue risk to the health and safety of the public. As defined in 10 CFR 50.49(b), important-to-safety SSCs comprise (b)(1) safety-related SSCs, (b)(2) certain nonsafety-related SSCs, whose failure could impact a safety function, and (b)(3) certain post-accident monitoring (PAM) equipment as delineated in the NRC Regulatory Guide (RG) 1.97, "Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants." As defined in 10 CFR 50.2, 10 CFR 50.49(b)(1), 10 CFR 50.65(b)(1), and 10 CFR 21.3 (as basic components), safety-related SSCs are those that are required to remain functional during and following design-basis events (DBEs), to assure (1) the integrity of the reactor coolant pressure boundary (RCPB), (2) the capability to shut down the reactor and maintain it in a safe shutdown condition, and (3) to prevent or mitigate the consequences of accidents that could result in the offsite release of radioactivity with potential offsite radiation exposures comparable to the guidance exposures of 10 CFR 50.34(a)(1) or 10 CFR 100.11. This section presents a review of the methodology used in the categorization of SSCs in the US APWR.

3.2.1 Seismic Classification

3.2.1.1 Introduction

SSCs important to safety must be designed to withstand the effects of earthquakes without loss of capability to perform their safety functions. The earthquake against which these plant features are designed is defined as the safe shutdown earthquake (SSE). The SSE is based upon an evaluation of the maximum earthquake potential and is the earthquake that produces the maximum vibratory ground motion for which SSCs important to safety must be designed to remain functional.

According to RG 1.29, "Seismic Design Classification," Position C.1, certain listed SSCs of a nuclear power plant, including their foundations and supports, are designated as seismic Category I and must be designed to withstand the effects of the SSE and remain functional. In addition, according to Position C.2, those portions of SSCs of which continued function is not required but of which failure could reduce the functioning of any seismic Category I plant features to an unacceptable safety level or could result in incapacitating injury to occupants of the control room should be designed and constructed so that the SSE would not cause such failure. Although not labeled in RG 1.29, this category of seismic classification has historically been referred to as seismic Category II.

The objective of the staff review is to determine that SSCs important to safety have been appropriately categorized and designed to withstand the effects of earthquakes without loss of capability to perform their safety functions. As stated in Section 3.2 above, the term "important to safety" and "safety-related" have different definitions. The staff does not consider the use of these two terms to be synonymous. However, it is recognized that all safety-related SSCs are seismic Category I.

3.2.1.2 Summary of Application

DCD Tier 1: The Tier 1 information associated with buildings and structures is found in Section 2.2.1 "Building Structures Design Description," and Table 2.2-1, "Seismic Classification of Structures," and Table 2.2-4, "Structural and Systems Engineering Inspections, Tests, Analyses, and Acceptance Criteria." The Tier 1 information associated with seismic classification and qualification of electrical and mechanical components is addressed in Sections 2.4, "Reactor Systems," 2.5, "Instrumentation and Controls," 2.6, "Electrical Systems," 2.7, "Plant Systems," and 2.11, "Containment Systems," of the Tier 1 DCD. The specific sections are referenced below in the discussion of inspections, tests, analyses, and acceptance criteria (ITAAC).

DCD Tier 2: The applicant has provided a Tier 2 system description in Section 3.2.1, "Seismic Classification," summarized here in part, as follows:

To meet the NRC seismic requirements with regard to the design for earthquakes, DCD Tier 2, Table 1.9.1-1, "US-APWR Conformance with Division 1 Regulatory Guides," indicates that SSCs are seismically classified in accordance with RG 1.29, Revision 4, March 2007, with no exceptions. DCD Tier 2, Table 1.9.1-1 and Section 3.2.1 identify that the US-APWR is consistent with RG 1.143, "Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants," Revision 2, November 2001, with no exceptions for RWMS seismic design and the regulatory requirements in GDC 61, "Fuel Storage and Handling and Radioactivity Control," as it relates to the design of radioactive waste systems and other systems that may contain radioactivity. The DCD also states that safety-related instrument sensing lines meet the seismic design recommendations contained in RG 1.151, "Instrument Sensing Lines," Revision 0, July 1983; and the fire protection systems meet the seismic design recommendations contained in RG 1.189, "Fire Protection for Nuclear Power Plants," Revision 1, March 2007.

In DCD Tier 2, Section 3.2.1.1, "Definitions," the applicant identifies three seismic categories: seismic Category I, seismic Category II, and non-seismic, described as follows:

> Seismic Category I applies to safety-related SSCs (including their foundations and supports) that must remain functional and/or retain their pressure integrity in the event of an SSE. This category includes SSCs designated as seismic Category I in accordance with RG 1.29. These SSCs are designed to withstand the effects of the SSE and maintain their structural integrity (including pressure integrity) and their specified design functions.
>
> Seismic Category II applies to SSCs, which perform no safety-related function, and whose continued function is not required, but whose structural failure or interaction could degrade the functioning or integrity of a seismic Category I SSC to an unacceptable level, or could result in incapacitating injury to occupants of the control room. Seismic Category II SSCs are designed so that the SSE could not cause unacceptable structural interaction or failure with seismic Category I SSCs. For fluid systems, this requires an adequate level of pressure boundary integrity to prevent seismically-induced flooding that may cause adverse effects on safety-related SSCs.
>
> SSCs that are not classified as seismic Category I or seismic Category II are classified as Non-Seismic (NS). NS SSCs have no safety-related function or nuclear safety design requirements. The applicant states the NS SSCs are primarily located outside of

safety-related buildings or segregated from seismic Category I SSCs so that the failure of their structural integrity would not impact the seismic Category I SSCs and cause adverse system interactions. If it is determined that a SSC would cause an adverse impact on a seismic Category I SSC, then it is designed and/or mounted in accordance with seismic Category II requirements to withstand an SSE event so that it could not fail and cause an adverse impact or interaction with the seismic Category I SSC.

DCD Tier 2, Table 3.2-3, "Comparison of Various Requirements to Equipment Class" provides the relationship between different equipment classes, RG 1.29 seismic design requirements and the associated quality assurance (QA) requirements. DCD Tier 2, Table 3.2-2, "Classification of Mechanical and Fluid Systems, Components, and Equipment," provides a list of mechanical and fluid systems, components, and equipment and their designated seismic category, equipment class, QA classification, and design codes and standards. The equipment classification is shown on the piping and instrumentation diagrams (P&IDs) included in various sections. The seismic classification is identified in DCD Tier 2, Table 3.2-2 and can be determined by the equipment class on the P&IDs, or as noted in DCD Tier 2, Table 3.2-2.

The US-APWR SSCs that are identified in DCD Tier 2, Section 3.2.1, Table 3.2-2, Table 3.2-4, "Seismic Classification of Buildings and Structures," and Appendix 3D, "Equipment Qualification List - Safety and Important to Safety Electrical and Mechanical Equipment," Table 3D-2, "US-APWR Environmental Qualification Equipment List," include both pressure boundary components of fluid systems and non-pressure boundary items such as structures, electrical items, instrumentation and reactor vessel (RV) internals.

The applicant states that the site-independent seismic design of the US-APWR sets the operating-basis earthquake (OBE) ground motion at one-third (1/3) of the SSE (discussed further in Section 3.7.1.1, "Design Ground Motion)," eliminating the requirement for performing explicit design analysis for OBE loads. DCD Tier 2, Table 3.2-1, "Non-Safety Components Required for Normal Shutdown," presents a list of nonsafety-related components required for normal plant shutdown.

The applicant states that some SSCs required for operation (excluding electrical features) do not need to be designed to seismic Category I standards. Examples include those portions of seismic Category I systems such as vent lines, drain lines, fill lines and test lines on the downstream side of isolation valves and those portions of the system not required to perform a safety function.

For the seismic design and classification of safety-related instrumentation sensing lines, the applicant uses RG 1.151, Revision 0, July 1983, as guidance, stating that seismic classification is in accordance with Positions C.2 and C.3.

For seismic design and classification of radioactive waste management SSCs, the applicant uses RG 1.143, Revision 2, as guidance.

For seismic design and classification of fire protection systems, the applicant uses RG 1.189, Revision 1, March 2007, as guidance to establish the design requirements of fire protection systems to meet the requirements of GDC 2, "Design Bases for Protection Against Natural Phenomena," as it relates to designing these SSCs to withstand earthquakes. RG 1.189 is used to identify portions of fire protection SSCs requiring some level of seismic design consideration. The applicant uses RG 1.189, Positions 3.2.1, 6.1.1.2, and 7.1 to provide guidance for seismic classification of fire protection systems.

DCD Tier 2, Section 3.2.1.3, “Classification of Building Structures,” addresses seismic classification of buildings and structures. DCD Tier 2, Table 3.2-4, “Seismic Classification of Buildings and Structures,” presents the applicant’s seismic classification of buildings and structures, except for some minor non-seismic buildings and structures in the plant yard.

ITAAC: As noted in the summary of DCD Tier 1, a table is included below of all the ITAAC that pertain to seismic classification and qualification of electrical and mechanical equipment.

Table 3.2.1-1
US-APWR ITAAC for Seismic Classification and Qualification of Electrical and Mechanical Components

DCD Section Number	DCD Table Number	ITAAC Number(s)
2.4.1	2.4.1-2	8
2.4.2	2.4.2-5	7.a.i, 7.a.ii, 7.a.iii
2.4.4	2.4.4-5	5.a.i, 5.a.ii, 5.a.iii
2.4.5	2.4.5-5	5.a.i, 5.a.ii, 5.a.iii
2.4.6	2.4.6-5	5.i, 5.ii, 5.iii
2.5.1	2.5.1-5	5.i, 5.ii, 5.iii
2.6.1	2.6.1-3	6.a
2.6.2	2.6.2-2	2
2.6.3	2.6.3-3	3
2.6.4	2.6.4-1	6, 20
2.6.6	2.6.6-1	5
2.6.8	2.6.8-1	2
2.7.1.2	2.7.1.2-5	5.a.i, 5.a.ii, 5.a.iii
2.7.1.9	2.7.1.9-5	5.a.i, 5.a.ii, 5.a.iii
2.7.1.10	2.7.1.10-3	5.a.i, 5.a.ii, 5.a.iii
2.7.1.11	2.7.1.11-5	5.a.i, 5.a.ii, 5.a.iii
2.7.3.1	2.7.3.1-5	5.a.i, 5.a.ii, 5.a.iii
2.7.3.3	2.7.3.3-5	5.a.i, 5.a.ii, 5.a.iii
2.7.3.5	2.7.3.5-5	5.a.i, 5.a.ii, 5.a.iii
2.7.5.1	2.7.5.1-3	2.a, 2.b, 2.c
2.7.5.2	2.7.5.2-3	2.a, 2.b, 2.c
2.7.5.4	2.7.5.4-2	2.a, 2.b, 2.c
2.7.6.1	2.7.6.1-1	1.a, 1.b, 1.c
2.7.6.2	2.7.6.2-1	1.a, 1.b, 1.c
2.7.6.3	2.7.6.3-5	5.a, 5.b, 5.c
2.7.6.4	2.7.6.4-2	2
2.7.6.5	2.7.6.5-1	2
2.7.6.6	2.7.6.6-2	2.a, 2.b
2.7.6.7	2.7.6.7-3	5.a, 5.b

Table 3.2.1-1
US-APWR ITAAC for Seismic Classification and Qualification of Electrical and Mechanical Components

DCD Section Number	DCD Table Number	ITAAC Number(s)
2.7.6.9	2.7.6.9-2	7
2.7.6.13	2.7.6.13-3	2.i, 2.ii
2.11.2	2.11.2-2	5.a.i, 5.a.ii, 5.a.iii
2.11.3	2.11.3-5	5.a.i, 5.a.ii, 5.a.iii

Technical Specifications: 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 for this area of review.

Cross-cutting Requirements (Three Mile Island [TMI], Unresolved Safety Issue [USI]/Generic Safety Issue [GSI], Operating Experience [Op Ex]): There are no cross-cutting requirements for this area of review.

US-APWR Interface Issues Identified in the DCD: See DCD Tier 2, Table 1.8-2, "Compilation of All Combined License Applicant Items for Chapters 1-19."

Site Interface Issues Identified in the DCD: See DCD Tier 2, Table 1.8-1, "Significant Site Specific Interfaces with the Standard US-APWR Design."

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

3.2.1.3 Regulatory Basis

The relevant requirements of the Commission's regulations for this area of review, and the associated acceptance criteria, are given in Section 3.2.1, "Seismic Classification," Revision 2, March 2007, 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 also can be found in Section 3.2.1 of NUREG-0800.

1. GDC 1, "Quality Standards and Records," and the pertinent QA requirements of 10 CFR Part 50, Appendix B, as they relate to applying QA requirements to activities affecting the safety-related functions of SSCs designated as seismic Category I, commensurate with their importance to safety.

2. GDC 2, as it relates to the requirements that SSCs important to safety shall be designed to withstand the effects of earthquakes without loss of capability to perform necessary safety functions.

3. GDC 61, as it relates to the design of radioactive waste systems, and other systems that may contain radioactivity, to assure adequate safety under normal and postulated accident conditions.

4. 10 CFR Part 100, Appendix A, “Seismic and Geologic Siting Criteria for Nuclear Power Plants,” and 10 CFR Part 50, Appendix S, “Earthquake Engineering Criteria for Nuclear Power Plants,” as they relate to certain SSCs being designed to withstand the SSE and remain functional.

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 has been constructed 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 2, 10 CFR Part 100, Appendix A, and 10 CFR Part 50, Appendix S, regarding seismic design classification, are met by using guidance provided in RG 1.29. This guide describes an acceptable method of identifying and classifying those plant features that should be designed to withstand the effects of the SSE.

2. RG 1.143, Revision 2, November 2001, provides guidance used to establish the seismic design requirements of radioactive waste management SSCs to meet the requirements of GDC 2 and GDC 61, as they relate to designing these SSCs to withstand earthquakes. The guide identifies several radioactive waste SSCs requiring some level of seismic design consideration.

3. RG 1.151, Revision 0, July 1983, as it relates to guidance with regard to seismic design requirements and classification of safety-related instrumentation sensing lines.

4. RG 1.155, “Station Blackout,” Revision 0, August 1988, as it relates to the seismic classification of nonsafety-related risk-significant components.

5. RG 1.189, Revision 1, March 2007, provides guidance used to establish the design requirements of fire protection to meet the requirements of GDC 2, as it relates to designing these SSCs to withstand earthquakes. This guide identifies portions of fire protection SSCs requiring some level of seismic design consideration.

6. Memo to the NRC Staff, “Standard Definitions for Commonly-Used Safety Classification Terms,” (Harold Denton) dated November 20, 1981.

3.2.1.4 Technical Evaluation

3.2.1.4.1 Classification Criteria

The staff reviewed the criteria identified in DCD Tier 2, Section 3.2.1 used to select the appropriate seismic classification in DCD Tier 2, Table 3.2-2 for principal components. The staff determined that the classification criteria for seismic Category I, seismic Category II and non-seismic SSCs is essentially consistent with RG 1.29, Revision 4, and SRP Section 3.2.1. One difference in terminology is that RG 1.29 does not use the term seismic Category II. However, the basic methodology between RG 1.29 and the DCD definition of seismic Category II is consistent, in that SSCs are to be seismically analyzed if their failure could adversely affect seismic Category I SSCs.

However, the staff determined that additional information was needed to clarify the seismic classification criteria and the application of those criteria to the classification of important-to-safety SSCs. DCD Tier 2, Revision 1, Section 3.2.1 defines non-seismic SSCs that must maintain their structural integrity and are designated as seismic Category II. DCD Tier 2, Revision 1, Subsection 3.2.1.1.3, "Non-Seismic," defines non-seismic as those SSCs not classified as seismic Category I or seismic Category II. Since non-seismic is not the same as seismic Category II, in Request for Additional Information (**RAI) 287-2041, Question 03.02.01-1**, the staff requested the applicant to clarify why the term non-seismic is used to define seismic Category II in DCD Tier 2, Section 3.2.1.

In its response to **RAI 287-2041, Question 03.02.01-1,** dated May 8, 2009, the applicant stated that the wording in DCD Tier 2 Section 3.2.1 will be clarified in a revision to the DCD to read: "These SSCs that must maintain their structural integrity to prevent unacceptable structural interaction or failure with seismic Category I SSCs are designated as seismic Category II." The staff found the response acceptable since the applicant clarified the use of the term seismic Category II. The staff confirmed that the proposed change has been incorporated in DCD Revision 2. Accordingly, **RAI 287-2041, Question 03.02.01-1 is resolved**.

3.2.1.4.2 Site-Specific Structures, Systems, and Components

In DCD Tier 2, Section 3.2.3, "Combined License Information," COL Information Items 3.2(4) and 3.2(5) specify that the COL applicant is to identify site-specific SSCs. SRP Section 3.2.1 identifies plant features of the ultimate heat sink (UHS) including dams, ponds, and cooling towers to be seismic Category I. In **RAI 287-2041, Question 03.02.01-2**, the staff requested the applicant to identify which site-specific SSCs are to be classified in the COL application (COLA) and if these UHS plant features are site-specific SSCs.

In its response to **RAI 287-2041, Question 03.02.01-2**, dated May 8, 2009, the applicant referenced DCD Tier 2, Section 3.2.3, COL Information Item 3.2(4), that requires the COL applicant to identify site-specific safety-related SSCs designed to seismic Category I requirements. The response clarified that the UHS is one such SSC which is site-specific. Based on the selection of the type of UHS, the extent of the safety-related boundary and seismic Category I requirements will vary. The response further clarified that DCD Tier 2, Section 3.2.3, COL Information Item 3.2(5) requires the COL applicant to identify the equipment classification and seismic category of the site-specific safety-related and nonsafety-related fluid systems and components. Per this COL information item, the COL applicant will need to identify UHS make-up water, blowdown, and chemical injection systems and components. Also, if required, water inventory transfer system between the cooling towers (bays) will need to be identified. This COL information item is broad based and includes all site-specific fluid systems and components such as the circulating water system.

The response included a change to DCD Tier 2, Table 3.2-4, that adds a Note 4 to clarify that UHS related structures (UHSRS) include but are not limited to dams, ponds, or cooling towers (including the cooling tower enclosure and the pump house). The specific features of the UHSRS are site dependent and not part of the US-APWR standard plant. The UHSRS are seismic Category I structures selected based on site-specific conditions and meteorological data. The staff reviewed the changes and found them to be acceptable since the applicant clarified the COL applicant's responsibilities with regard to the UHSRS. The staff confirmed that this DCD change was included in DCD Revision 2. Accordingly, **RAI 287-2041, Question 03.02.01-2 is resolved**. Sections 3.2.1.4.13 and 3.2.1.4.15 of this report further discuss COL information items and scope of site-specific SSCs.

3.2.1.4.3 Activities Affecting Safety-Related Functions

SRP Section 3.2.1.I.3 allows the use of tables to identify those SSCs that are designated seismic Category I. SRP Section 3.2.1.I.3 also states that the tables should identify all activities affecting the safety-related functions of these seismic Category I plant features that are required to meet GDC 1 and 10 CFR Part 50, Appendix B, requirements. DCD Tier 2, Table 3.2-2, identifies which SSCs are subjected to the 10 CFR Part 50, Appendix B, QA requirements; however, DCD Tier 2, Table 3D-2, "US-APWR Environmental Qualification Equipment List," does not identify what QA requirements apply to seismic Category I or II SSCs. In **RAI 287-2041, Question 03.02.01-3**, the staff requested the applicant to explain how the DCD tables specified above identify all activities affecting safety-related functions.

In its response to **RAI 287-2041, Question 03.02.01-3,** dated May 21, 2009, the applicant stated that, in accordance with the guidance of RG 1.29, Positions C.1 and C.4, 10 CFR Part 50, Appendix B, requirements will be applied to electrical, mechanical, and instrumentation and control (I&C) equipment that are classified as seismic Category I and II. DCD Tier 2, Section 3D.1.6, "Determination of Seismic Requirements," will be revised to indicate that 10 CFR Part 50, Appendix B, requirements will be applied to seismic Category I and II electrical, I&C, and mechanical equipment contained in DCD Tier 2, Table 3D-2. The staff found the response acceptable since the applicant clarified that 10 CFR Part 50, Appendix B, requirements will be applied according to the guidance in RG 1.29. The staff confirmed that the proposed change has been incorporated in DCD Revision 2. Accordingly, **RAI 287-2041, Question 03.02.01-3 is resolved**.

3.2.1.4.4 Auditable Information

10 CFR 52.47, identifies that the Commission will require, prior to DC, that information normally contained in certain procurement specifications and construction and installation specifications be completed and available for audit. In **RAI 287-2041, Question 03.02.01-4**, the staff requested the applicant to clarify if the design-basis information on seismic classification for all important to safety SSCs within the scope of the DCD, including structures, is included in specifications and if this information is now available for audit.

In its response to **RAI 287-2041, Question 03.02.01-4,** dated May 21, 2009, the applicant stated that the design-basis information on seismic classification is addressed in design specifications. Design specifications for important to safety SSCs will be prepared by the applicant during the DCD review phase. In addition, for risk-significant, unique-fabrication-required, and/or newly designed SSCs, as-designed stress reports for important to safety SSCs will be prepared by the applicant during the DCD review phase. For the remaining low risk and/or more conventional SSCs, design specifications will be prepared by the applicant during the DCD review phase. Numerous American Society of Mechanical Engineers (ASME) qualified vendors are available to manufacture these SSCs. Stress reports will be prepared by those vendors prior to material procurement. The design specification will be met by those experienced vendors. The applicant's latest design completion plan for US-APWR SSCs is summarized in Table 4, "Design Completion Plan for PSCs in the US-APWR DCD," of the applicant's letter, "Revised Design Completion Plan for US-APWR Piping Systems and

Components," dated December 7, 2012. In this table, the schedule of the available information is presented. This information will be available for NRC audit.

Other than design specifications and stress reports, the response did not identify any other design-basis information on seismic classification. Therefore, **RAI 287-2041, Question 03.02.01-4,** was closed as unresolved and in follow-up **RAI 580-4584, Question 03.02.02-16**, the staff requested the applicant to clarify if any additional design-basis documents will be available for the audit to establish the basis for the individual quality group (QG) classifications to enable the classifications to be validated. The staff notes that design-basis documents typically address both seismic qualification and QG classification, so one follow-up RAI was issued to address audits for both Section 3.2.1 and 3.2.2. For example, supporting design-basis documents that supplement the design specifications and can be used to verify the safety function for individual classifications may include documents such as a "Q-List," detailed P&IDs, system summary documents, and procurement specifications.

In its response to **RAI 580-4584, Question 03.02.02-16,** dated July 21, 2010, the applicant stated that it agrees to make additional documents used to establish the basis for QG designation for individual SSCs available for audit by the NRC so that the classifications can be validated. Additional documents should include the Q-List (or its equivalent), detailed P&IDs, system specifications, and equipment procurement specifications for those SSCs within the scope of the DCD.

In the applicant's latest Design Completion Plan dated December 7, 2012, which is discussed above, the applicant stated that design specifications of all risk-significant ASME Class 1, 2, and 3 piping systems and components (PSCs), except valves and orifices, that have been previously prepared, will be revised to incorporate the revised design load for the PSCs. The applicant proposed that the design specification audit be held after the design specifications are revised. Source design documents referenced by the design specifications and related to the bases of design transients and details of load conditions will also be available at that time. The staff finds the applicant's response, as augmented by the Design Completion Plan, acceptable as it clarifies that appropriate documents will be available for audit. Accordingly, **RAI 580-4584, Question 03.02.02-16 is resolved**. To track the need for the applicant to make the documents available for the audit, the staff issued **RAI 1015-7054, Question 03.09.03-31.** Pending the staff completing and documenting the audit, **RAI 1015-7054, Question 03.09.03-31, is being tracked as an Open Item**.

3.2.1.4.5 Application of Risk Insights

Based on initial reviews of the DCD and the probabilistic risk assessment (PRA), the staff identified the leak detection system (LDS) for the RCPB as having a high importance. DCD Tier 2, Subsection 3.1.2.5.1, "Discussion," states: "Instrumentation is provided to detect significant leakage from the RCPB with indication in the MCR (see Section 5.2)." However, DCD Tier 2, Section 5.2.5.5, "Safety Evaluation," states: "Leak detection monitoring has no safety-related function." DCD Tier 2, Table 17.4-1, "Risk-significant SSCs," identifies risk-significant SSCs, but the LDS does not appear to be included. In **RAI 287-2041, Question 03.02.01-5**, the staff requested the applicant to clarify the risk significance of the LDS and if the RCPB LDS belongs under the Phase 1 D-RAP as discussed in DCD Tier 2, Section 17.4. Also, if this system does belong under the Phase 1 D-RAP, the applicant was requested to discuss where in DCD Tier 2, Section 3.2, "Classification of Structures, Systems, and Components," Section 17.4.7.1, "SSCs Identification," and Table 17.4-1 this system has been identified and if augmented requirements such as a graded approach is to be applied to the seismic design and QA. In its response to

RAI 287-2041, Question 03.02.01-5, dated May 8, 2009, the applicant stated that the RCPB LDS is not listed in DCD Tier 2, Table 17.4-1. The risk significance of SSCs in the LDS was not considered since the system has a small effect on the probability of a large break LOCA. The RCPB LDS, which is nonsafety-related but has the important function of monitoring RCPB integrity is designed to be qualified in accordance with RG 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems," Revision 1, May 2008.

DCD Tier 2, Section 5.2.5, "Reactor Coolant Pressure Boundary (RCPB) Leakage Detection," provides a discussion of the leak monitoring system (LMS). The LMS consists of several subsystems focused on the identification of both identified and unidentified leakage from various SSCs. DCD Tier 2, Section 5.2.5.1, "Design Bases," states the LMS is designed in accordance with the requirements of GDC 30, "Quality of Reactor Coolant Pressure Boundary," and guidance of RG 1.45 and RG 1.29. DCD Tier 2 Section 5.2.5.5 states the LMS has no safety function but the containment airborne particulate radioactivity monitor subsystem of the LMS is seismic Category I. There are some non-specific seismic qualification statements in DCD Tier 2, Section 5.2.5.5 for the containment airborne gaseous radioactivity monitor, the containment air cooler condensate flow rate monitoring subsystem, and the containment sump level and flow monitoring subsystem of the LMS. DCD Tier 2, Table 1.9.1-1 also indicates that no exceptions to RG 1.45 and RG 1.29 are identified.

Since the response identified that the LMS has no safety function, in **RAI 581-4582, Question 03.02.01-17,** the staff requested the applicant to (1) describe what criteria of RG 1.29 are being met by the LMS design and function; (2) provide additional information regarding the seismic classification of (a) the containment airborne gaseous radioactivity monitor, (b) the containment air cooler condensate flow rate monitoring subsystem, and (c) the containment sump level and flow monitoring subsystem with regard to the statement that these three subsystems are "qualified for seismic events not requiring a plant shutdown;" and (3) clarify how the seismic Category I classification of the containment airborne particulate radioactivity monitor subsystem satisfies any supplemental design requirements for the high risk-significant LMS.

In its response to **RAI 581-4582, Question 03.02.01-17,** dated July 21, 2010, the applicant included the following additional information:

> Containment sump level and flow monitoring system, while not required to perform a safety function, does have TS limits and surveillance requirements and is qualified for a SSE. Containment airborne gaseous radioactivity and condensate flow rate from air coolers, while not required to perform a safety function, do have TS limits and surveillance requirements and are qualified to perform their intended function following seismic events that do not require plant shutdown. This exceeds the guidance of RG 1.45 Position 2.4 which states "At least one of the leakage monitoring systems required by the plant Technical Specifications **...**should be capable of performing its function(s) following any seismic event that does not require plant shutdown."

As noted above and described in DCD Tier 2, Subsections 5.2.5.4.1.1, "Containment Sump Level and Flow Monitoring System," 5.2.5.4.1.3, "Containment Airborne Gaseous Radioactivity Monitor," and 5.2.5.4.1.4, "Containment Air Cooler Condensate Flow Rate Monitoring System," the containment sump level and flow monitoring system and the containment airborne gaseous radioactivity monitor are qualified for an SSE, and the containment air cooler condensate flow rate monitoring system is qualified to perform intended function following seismic events that do not require plant shutdown. Qualification is in accordance with the program description in DCD Tier 2, Section 3.10, "Seismic and Dynamic Qualification of Mechanical and Electrical

Equipment," and the applicant's Technical Report MUAP-08015, "US-APWR Equipment Qualification Program."

The RAI response further stated that the seismic Category I classification of the containment airborne particulate radioactivity monitor and containment sump level and flow monitoring system satisfy design requirements that supplement and exceed the guidance of RG 1.45. Furthermore, the containment air cooler condensate flow rate monitoring system and containment airborne gaseous radioactivity monitor are designed to perform their function following a seismic event not requiring a shutdown, which satisfies the design guidance of RG 1.45. The former two LMSs are designed to perform their function with diversity following an SSE. The latter two LMSs are designed to perform their function with diversity following a seismic event not requiring a shutdown. These four LMSs are designed to be subjected to TS limits and surveillance and are consistent with the risk significance of the LMS.

The staff reviewed and found response to be acceptable since the applicant clarified that important portions of the LMS are classified as seismic Category I, but the LDS is not considered risk-significant and is designed to be consistent with RG 1.45. The scope of SSCs included in DCD tier 2, Table 17.4-1 is reviewed in Chapter 17 of this report. Accordingly, **RAI 287-2041, Question 03.02.01-5** and **RAI 581-4582, Question 03.02.01-17, are resolved**.

3.2.1.4.6 Quality Assurance Requirements for Seismic Category II Structures, Systems, and Components

RG 1.29 states, in Regulatory Position C.2, that SSCs whose continued function is not required but whose failure could reduce the functioning of SSCs required to function should be designed and constructed so that the SSE would not cause failure. In addition, RG 1.29 states, in Regulatory Position C.4, that pertinent QA requirements of 10 CFR Part 50, Appendix B, should be applied to these SSCs. DCD Tier 2, Section 3.2.1.1.3 states that "For non-seismic items located in the proximity of safety-related SSCs that are upgraded to seismic Category II, the pre-assigned equipment class remains unchanged." This could result in a lower level of QA requirements than is required for seismic Category II, which would not be acceptable under RG 1.29. For example, DCD Tier 2, Table 3.2-2, sheet 41 states that the prestressed concrete containment vessel (PCCV) polar crane is Equipment Class 5, seismic Category II, but indicates that 10 CFR Part 50, Appendix B, is not applicable. In **RAI 287-2041, Question 03.02.01-6**, the staff requested the applicant to clarify how the non-seismic items upgraded to seismic Category II meet the pertinent QA requirements in 10 CFR Part 50, Appendix B.

In its response to **RAI 287-2041, Question 03.02.01-6,** dated May 21, 2009, the applicant references the responses to **RAI 287-2041, Question 03.02.01-3**, dated May 21, 2009, and **RAI 276-2043, Question 03.02.02-3**, dated May 8, 2009, regarding the application of 10 CFR Part 50, Appendix B, QA requirements for seismic Category I and II SSCs. The response to **RAI 287-2041, Question 03.02.01-6** states that seismic Category II SSCs are defined as SSCs which perform no safety-related function, and whose continued function is not required, but whose structural failure or interaction could degrade the functioning or integrity of a seismic Category I SSC to an unacceptable level, or could result in incapacitating injury to occupants of the control room. In addition, SSCs classified as seismic Category II will meet the pertinent QA requirements of 10 CFR Part 50, Appendix B, in accordance with the seismic design guidance of RG 1.29, Revision 4, Regulatory Position C.4 as described in DCD Tier 2, Subsection 3.2.1.1.2, "Seismic Category II." A new Note 4 will be added to DCD Tier 2, Table 3.2-2 that indicates that SSCs classified as seismic Category II will meet the pertinent requirements of 10 CFR Part 50, Appendix B. The response states that Equipment Class 4 or 5 designations will remain unchanged for SSCs classified as seismic Category II.

The applicant reviewed DCD Tier 2, Table 3.2-2 to identify additional inconsistencies between the application of 10 CFR Part 50, Appendix B, requirements, seismic category, and equipment classification. The applicant identified additional changes to DCD Tier 2 Table 3.2-2, which are included in the response to **RAI 287-2041, Question 03.02.01-6**. The response also states that DCD Tier 2, Table 3.2-2 has also been revised to include changes as described in the responses to **RAI 200-1983, Questions 09.01.04-1 and 09.01.04-2**, dated September 23, 2009, that provided additional fuel handling equipment, revised the equipment classification, and seismic classification. For fuel handling equipment that was classified as seismic Category II, a new Note 4 has been added to indicate that SSCs classified as seismic Category II will meet the pertinent requirements of 10 CFR Part 50, Appendix B. DCD Tier 2, Table 3.2-2 Note 4 now includes the above QA requirement for seismic Category II SSCs designated as Equipment Class 4 or 5.

The staff reviewed the response and found that the DCD changes to clarify that 10 CFR Part 50, Appendix B apply to seismic Category II SSCs are acceptable, and the DCD has been revised to include a new note to the end of DCD Tier 2, Table 3.2-2 referencing DCD Tier 2 Section 3.2.1.1.2. Therefore, **RAI 287-2041, Question 03.02.01-6 is resolved**.

3.2.1.4.7 Seismic Guidance per RG 1.143

In DCD Tier 2, Table 3.2-2, the Equipment Class 6 (radioactive waste management systems (RWMS) components) systems and components refer to RG 1.143, Revision 2, for the seismic category, which is in accordance with SRP Section 3.2.1. DCD Tier 2, Table 1.9.1-1 identifies that the US-APWR conforms RG 1.143 with no exceptions. However, RG 1.143 defines three hazard levels with different seismic design guidance for each level. In **RAI 287-2041, Question 03.02.01-8**, the staff requested the applicant to identify the specific hazard level for the Equipment Class 6 RWMS and components so that seismic requirements are identified.

In its response to **RAI 287-2041, Question 03.02.01-8,** dated May 8, 2009, the applicant stated that component hazard classifications are performed separately in accordance with the guidelines provided in RG 1.143, Figure 2, “Flowchart of Safety Classification Process,” and the limits in 10 CFR Part 71, Appendix A, “Determination of A1 and A2.” The component hazard classifications are incorporated in the equipment datasheets for the purpose of equipment specifications. As an example, the waste holdup tanks have a hazard classification of “IIc,” and the foundation anchorage is designed with the applicable Standard Uniform Building Code, (UBC), in accordance with the guidelines in RG 1.143. The response cites conformance with RG 1.143 in the DCD and the basis for the seismic classification and specific hazard level can be validated during the verification of design-basis information. The staff found the response acceptable because it explained the applicant’s approach to conformance with RG 1.143. Accordingly, **RAI 287-2041, Question 03.02.01-8 is resolved**.

3.2.1.4.8 Equipment Class 4 Discrepancy

RG 1.29 states in Regulatory Position C.1.I that the spent fuel storage pool structure, including the fuel racks should be seismic Category I. DCD Tier 2, Table 3.2-2 specifies that the new fuel storage rack and spent fuel storage rack are Equipment Class 4 and seismic Category I. However, DCD Tier 2, Section 3.2.2.4, “Equipment Class 4,” states that Equipment Class 4

SSCs are classified as non-seismic or seismic Category II. In **RAI 287-2041, Question 03.02.01-9**, the staff requested the applicant to clarify the basis for this discrepancy.

In its response to **RAI 287-2041, Question 03.02.01-9,** dated May 21, 2009, the applicant stated that the spent fuel storage pool structure, including the fuel racks, will be seismic Category I in accordance with RG 1.29. However, the new fuel storage rack and spent fuel storage rack will be revised from Equipment Class 4 to Equipment Class 3 in DCD Tier 2, Table 3.2-2. This classification is consistent with DCD Tier 2, Section 3.2.2.3, "Equipment Class 3," which states that Equipment Class 3 SSCs are classified as seismic Category I. The equipment classification of the new fuel storage rack and spent fuel storage rack will be revised from Equipment Class 4, QG D to Equipment Class 3, QG C, and the QA requirements of 10 CFR Part 50, Appendix B, will be applied. The staff found the response acceptable since the applicant addressed the equipment classification discrepancy. The staff confirmed that the proposed change has been incorporated in DCD Revision 2. Accordingly, **RAI 287-2041, Question 03.02.01-9 is resolved**.

3.2.1.4.9 Quality Assurance for Seismic Category I Structures, Systems, and Components

DCD Tier 2, Table 3.2-2 states the new and spent fuel storage racks will be seismic Category I but will not be subjected to 10 CFR Part 50, Appendix B, requirements. DCD Tier 2 Subsection, 3.2.1.1.1, "Seismic Category I," states that seismic Category I SSCs will meet the QA requirements of 10 CFR Part 50, Appendix B. In **RAI 287-2041, Question 03.02.01-10**, the staff requested the applicant to provide justification for these exceptions to the criteria in DCD Tier 2, Section 3.2.1.1.1.

In its response to **RAI 287-2041, Question 03.02.01-10,** dated May 21, 2009, the applicant referenced DCD Tier 2, Section 3.2.1.1.1, which states seismic Category I SSCs meet the QA requirements of 10 CFR Part 50, Appendix B. DCD Tier 2, Table 3.2-2 will be revised to indicate that the seismic Category I new and spent fuel storage racks are to be subjected to 10 CFR Part 50, Appendix B, QA requirements. The staff found the response acceptable since it corrected the discrepancy regarding QA requirements. The staff confirmed that the proposed change has been incorporated in DCD Revision 2. Accordingly, **RAI 287-2041, Question 03.02.01-10 is resolved**.

3.2.1.4.10 List of Structures, Systems, and Components required for Continued Operation

Appendix S to 10 CFR Part 50, Section IV(a)(2)(i)(B)(*l*), states that SSCs necessary for continued operation without undue risk to the health and safety of the public must remain functional and within applicable stress, strain, and deformation limits when subjected to the effects of the OBE ground motion in combination with normal operating loads. SRP Section 3.2.1 states that, if the applicant has set the OBE ground motion to the value one-third of the SSE Ground Motion, then the applicant should also provide a list of SSCs necessary for continued operation that must remain functional without undue risk to the health and safety of the public and within applicable stress, strain and deformation, during and following the OBE. DCD Tier 2, Table 3.2-1, includes a list of nonsafety-related components needed for normal shutdown. In **RAI 287-2041, Question 03.02.01-11**, the staff requested the applicant to clarify why the list of SSCs included in DCD Tier 2, Table 3.2-1, are only SSCs needed for shutdown rather than all SSCs needed for continued operation during and following OBE. Also the applicant was requested to explain how these nonsafety-related SSCs are classified such that they will be seismically qualified for OBE to remain functional.

In its response to **RAI 287-2041, Question 03.02.01-11,** dated May 21, 2009, the applicant stated that all US-APWR SSCs that are necessary for continued operation without undue risk to the health and safety of the public will remain functional and within applicable stress, strain, and deformation limits when subjected to the effects of OBE ground motion in combination with normal operating loads. These SSCs are those that are classified as Equipment Class 1, 2 and 3 in DCD Tier 2, Table 3.2-2 and designated as seismic Category I. They are also classified as Institute of Electrical and Electronic Engineers (IEEE) Class 1E, as appropriate given their required safety-related functions. As a result, these SSCs will maintain their structural integrity and they will also remain capable for performing all required functions following the SSE as well as the OBE. The staff found the applicant's response acceptable because the applicant has identified the SSCs needed for continued operation and because, based on DCD Tier 2 Table 1.9.1-1, pre-earthquake planning as addressed in RG 1.166, "Pre-earthquake Planning and Immediate Nuclear Power Plant Operator Postearthquake Actions," March 1997, is considered to be a site-specific operational program. Accordingly, **RAI 287-2041, Question 03.02.01-11, is resolved**.

3.2.1.4.11 Safety-Related Versus Important to Safety

GDC 2 states the SSCs important to safety shall be designed to withstand the effects of natural phenomena, including earthquakes. DCD Tier 2 Section 3.1.1.2.1, "Discussion," uses the term "safety-related" and DCD Tier 2, Section 3.2.1 uses both terms, "safety-related" and "important to safety," to identify the SSCs that must be designed to satisfy the requirements of GDC 2. The staff refers to definitions in: 1) 10 CFR Part 50 and 2) memo to NRC Staff, "Standard Definitions for Commonly-Used Safety Classification Terms," regarding application of these terms. In **RAI 287-2041, Question 03.02.01-12**, the staff requested the applicant to clarify the application of the terms "safety-related" and "important to safety" to the seismic classification of SSCs and compliance with GDC 2. Also the staff requested the applicant to clarify to what extent those SSCs that are important to safety that are not considered safety-related are seismically classified so that they are designed to withstand earthquakes.

In its response to **RAI 287-2041, Question 03.02.01-12,** dated May 21, 2009, the applicant referenced its response to **RAI 287-2041, Question 03.02.02-7**, dated May 8, 2009, and explained its application of the terms safety-related and important to safety to the US-APWR and compliance with GDC 2. The response to **RAI 287-2041, Question 03.02.01-12**, clarified the seismic classifications of the SSCs and their relationships to safety-related and important to safety SSCs. The response stated that safety-related SSCs are seismic Category I SSCs (RG 1.29, Position C.1) and nonsafety-related SSCs are either seismic Category II or non-seismic (RG 1.29, Positions C.2 and C.4). The response indicated that important to safety SSCs include safety-related SSCs and additional nonsafety-related SSCs, and referred to DCD Tier 2, Table 3.2-3 for the definition of the requirements. The response also indicated that the fire protection systems are designed to RG 1.189 (RG 1.29 Position C.5), RWMS designed to RG 1.143, and safety-related instrumentation sensing lines designed to RG 1.151. The response indicated that no changes are required to DCD Tier 2 Section 3.2.1.

The staff reviewed the response and found that the response: 1) does not address seismic requirements for risk-significant nonsafety-related SSCs that are important to safety or 2) include a DCD revision to replace the term "safety-related" with the more comprehensive term "important to safety" in satisfying GDC 2. The guidance in the memorandum from Denton clarifies that important to safety SSCs that require special treatment are not limited to safety-related SSCs. Supplemental seismic requirements for important to safety SSCs depend on the safety function and the reliability and integrity assumed in the PRA in response to an earthquake. Enclosure 3 to the applicant letter, “Additional Information for Design Completion Plan of US-APWR Piping Systems and Components,” dated July 14, 2008, and the applicant’s response to **RAI 150-1635, Question 17.04-19,** dated May 4, 2009, identify a list of risk-significant SSCs, but it is not clear if seismic requirements are applied to all nonsafety-related, risk-significant SSCs that are considered important to safety. For example, QG D piping in the refueling water storage system is identified as risk-significant, but it is classified as non-seismic. Important to safety SSCs are to include not only safety-related SSCs, but also nonsafety-related SSCs that are risk-significant. As a result, **RAI 287-2041, Question 03.02.01-12,** is closed as unresolved.

In follow-up **RAI 581-4582, Question 03.02.01-15**, the staff requested the applicant to clarify if all risk-significant nonsafety-related SSCs are classified as seismic Category I or II such that they are designed to withstand earthquakes consistent with GDC 2. If that is the case, the applicant was requested to identify those nonsafety-related or risk-significant SSCs that are designed to withstand earthquakes and confirm that the seismic classification is consistent with assumptions used in the PRA. Also, the term “safety-related” should be replaced with the term “important to safety” in DCD Tier 2, Sections 3.2.1 and 3.1.1.2, “Criterion 2 – Design Bases for Protection Against Natural Phenomena,” in order to satisfy GDC 2.

In its response to **RAI 581-4582, Question 03.02.01-15,** dated July 21, 2010, the applicant did not address changing the terminology in DCD Tier 2 Sections 3.2.1 and 3.1.1.2 to satisfy GDC 2, but stated that the applicant understands that the staff’s request on what seismic design requirements are applied to risk-significant, nonsafety-related SSCs that are important to safety. In addition, the applicant understands that the staff indicated that the DCD Tier 2, Section 3.2, is not clear which seismic requirements are applied to all nonsafety-related, risk-significant SSCs that are considered important to safety. The Harold Denton memorandum more importantly specified that by GDC 2, SSCs important to safety must be designed to withstand the effects of natural phenomena, like earthquakes, without the loss of the capability to perform their safety functions. For the earthquake, SSCs important to safety need to be designed with specific seismic design requirements. Also, by GDC 1, SSCs important to safety must meet quality standards commensurate with the importance of their safety functions to be performed, including recognized quality codes, standards and design criteria, such as pertinent QA requirements from 10 CFR Part 50, Appendix B.

DCD Tier 2, Section 17.4, discusses the D-RAP and what is considered in designating an SSC to be risk-significant. SSCs are identified as risk-significant based on the performance of importance analyses, seismic margin analysis, PRA results, engineering judgment, operational experience feedback and meetings conducted by the expert panel, and are treated by the D-RAP. DCD Tier 2, Table 17.4-1 lists risk-significant SSCs. One rationale for risk-significance includes the seismic event (SM); for which possible failure modes include functional failure by seismic hazard (FS) and structural failure by seismic hazard (SS). The response concluded that, taking these discussions into consideration, the DCD will be revised to state that the nonsafety-related SSCs that are listed in DCD Tier 2, Table 17.4-1 for SM in the column "Rationale" will be categorized as seismic Category II and classified as Equipment Class 5. In DCD Tier 2, Revision 3, Section 3.2.2.5, “Other Equipment Classes,” has been revised to include a reference to DCD Tier 2, Section 17.4, for Equipment Class 5, risk-significant SSCs and a requirement for the COL applicant to apply DCD methods of equipment classification and seismic classification of risk-significant nonsafety-related SSCs based on their safety role assumed in the PRA and treatment by the D-RAP. However, DCD Tier 2, Section 3.2.1, regarding seismic classification was not revised to clarify the criteria to classify risk-significant nonsafety-related SSCs. The staff recognizes that a minimum of seismic Category II is commonly applied to SSCs that are not considered safety-related. However, seismic requirements that pertain to nonsafety-related risk-significant SSCs should be determined by the process to define risk-significant components and their degree of functionality credited after a seismic event.

Consistent with NRC Interim Staff Guidance DC/COL-ISG-018," Interim Staff Guidance on Standard Review Plan, Section 17.4, Reliability Assurance Program," one purpose of the D-RAP is to provide reasonable assurance that a reactor is designed, constructed and operated consistently with key assumptions and risk insights for within-scope SSCs. It is understood that the scope, basis and adequacy of the seismic Category II classification for risk-significant SSCs will be determined as part of the D-RAP and PRA process. The RAI response and DCD Tier 2 Section 3.2.2.5 revision provides some degree of recognition that seismic requirements for risk-significant SSCs are to consider PRA assumptions in regard to seismic classification. However, since DCD Tier 2, Section 3.2.1 was not revised, the process to apply seismic requirements to risk-significant nonsafety-related SSCs is not entirely clear. Since the RAI response did not revise DCD Tier 2, Section 3.2.1 or adequately address the application of the terms safety-related compared to important to safety in order to satisfy GDC 2, **RAI 581-4582, Question 03.02.01-15**, was closed as unresolved.

In follow-up **RAI 684-5294 Question 03.02.01-19**, the staff requested the applicant to specifically clarify if the SSCs in DCD Tier 2, Table 17.4-1 classified as SM in the column "Rationale," combined with other seismic Category II SSCs, represent the complete list of nonsafety-related or risk-significant SSCs that are important to safety and designed to withstand earthquakes consistent with GDC 2. In addition, the staff requested the applicant to replace the term "safety-related" with the term "important to safety" for clarity in complying with GDC 2.

In its response to **RAI 684-5294, Question 03.02.01-19**, dated July 27, 2012, the applicant referred to the response to **RAI 667-5235, Question 03.02.02-17,** dated July 27, 2012, in which the applicant significantly revised DCD Tier 2, Sections 3.1.1, "Overall Requirements," 3.2.1, and 3.2.2, "System Quality Group Classification," and Table 3.2-2 to address the equipment classification methodology and to reflect the design process for risk-significant SSCs. Specifically, the applicant's response was provided in three discussions: 1) Compliance with GDC 1, 2) Identification of risk-significant SSCs, and 3) Application of a RTNSS process. Detailed discussion on **RAI 667-5235, Question 03.02.02-17** is documented in Sections 3.2.2.4.10 and 3.2.2.4.14 of this safety evaluation.

In summary, the staff found the response regarding the QA program (QAP), identification of risk-significant SSCs, as well as the replacement of the term "safety-related" to "important to safety" in DCD Tier 2 Section 3.1.1.1.1 acceptable. However, such replacement of these two terms was not revised in DCD Tier 2, Sections 3.2.1 and 3.2.2. Under the current revision of the DCD to address GDC 1, DCD Tier 2, Sections 3.2.1 and 3.2.2 still state that "safety-related SSCs be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed." This deviates from the current regulation in GDC 1, in which states "Structures, systems and components important to safety shall be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed." Therefore, while the staff considers part of the response to **RAI 667-5235, Question 03.02.02-17**, dated July 27, 2012, acceptable, the issue with the use of the terms "safety-related" and "important to safety" remains the same. As a result, **RAI 684-5294, Question 03.02.01-19**, is closed as unresolved. In follow-up **RAI 1012-7049, Question 03.02.01-23**, the staff requested the applicant to replace the term "safety-related” with the more comprehensive term “important to safety” in DCD Tier 2 Section 3.2.1 and Section 3.2.2.

In its response to **RAI 1012-7049, Question 03.02.01-23,** dated April 11, 2013, the applicant clarified the application of GDC 1 by replacing the term “safety-related” with the term “important to safety” in DCD Tier 2, Sections 3.2.1 and 3.2.2 and providing corresponding markups. The staff found the response acceptable as it conforms to GDC 1. Since the applicant has identified DCD changes, **RAI 1012-7049, Question 03.02.01-23 is tracked as Confirmatory Item**..

3.2.1.4.12 Inspections, Tests, Analyses and Acceptance Criteria

DCD Tier 1, Chapter 2, and DCD Tier 2 Section 14.3 describe various ITAAC to confirm the ability of safety-related seismic Category I SSCs to withstand a design-basis seismic event. It is not clear if there is a proposed ITAAC to address nonsafety-related seismic Category II SSCs. In **RAI 287-2041, Question 03.02.01-13**, the staff requested the applicant to identify if there is an ITAAC to address seismic Category II SSCs or explain why an ITAAC is not required. In addition, in **RAI 287-2041, Question 03.02.01-14**, the staff requested the applicant to clarify if there is an ITAAC for nonsafety-related, but important to safety SSCs, such as nonsafety-related RV internals and seismic Category II SSCs.

In its response to **RAI 287-2041 Question 03.02.01-13**, dated May 8, 2009, the applicant stated that its response to **RAI 156-1877, Question 14.03-2**, dated February 5, 2009, concurred with the need for ITAAC for nonseismic Category I (i.e., seismic Category II and non-seismic) structures to verify their failure will not impair the ability of nearby safety-related SSCs to perform their safety-related functions. SRP Section 14.3.2, “Structural and Systems Engineering - Inspections, Tests, Analyses, and Acceptance Criteria,” issued March 2007, Acceptance Criterion II.6 states in part that for non-seismic SSCs, the need for ITAAC to verify that their failure will not impair the ability of near-by safety-related SSCs to perform the safety-related functions should be assessed based on the specific design. The SRP also acknowledges that in certain cases due to details of the final design and as-built configuration, non-seismic to seismic interactions may not be evaluated until the plant is constructed. General ITAAC has been included in DCD Tier 1 to verify the functional arrangement of SSCs. The acceptance criteria for these ITAAC include the inspection/verification of as-built SSCs. Therefore, ITAAC that verify the as-built plant may also include verification that SSCs are designed and constructed to avoid adverse seismic Category II interactions with seismic Category I SSCs. Furthermore, in its response to **RAI 287-2041, Question 03.02.01-14**, dated May 21, 2009, the applicant stated that the DCD is consistent with the provision of Section IV.2.4.A of Appendix A to SRP Section 14.3, “Inspections, Tests, Analyses, and Acceptance Criteria,” issued March 2007, “The level of detail in Tier 1 is governed by a graded approach to the SSCs of the design, based on the safety significance of the functions they perform.” Therefore, safety-related SSCs are described with a relatively greater amount of detail than nonsafety-related SSCs in DCD Tier 1. Those nonsafety-related SSCs that are subjected to special regulatory treatment (e.g. risk-significant, severe accident design features, fire protection, and anticipated transient without scram (ATWS)) are addressed in their respective Tier 1 sections.

In addition, the applicant also stated that DCD Tier 1, Section 2.4.1, “Reactor System,” adequately addresses the reactor fuel, reactor pressure vessel (RPV) and reactor internals. The core support structures are identified as Class CS (Core Support) in DCD Tier 1, Table 2.4.1-1, “Equipment Key Attributes.” The core support structures are subjected to the ITAAC for the reactor system in DCD Tier 1, Table 2.4.1-2, “Reactor System Inspections, Tests, Analyses, and Acceptance Criteria," to address design and construction in accordance with ASME Code, Section III requirements. ASME Code, Section III, Subsection NG includes jurisdictional boundaries that distinguish core support structures from other RV internal structures. ASME Code, Section III, Subsection NG further requires certification that the construction of all internal structures is such as not to adversely affect the integrity of the core support structures. Thus, the ASME-related ITAAC for the reactor system provide assurance that RV internal structures will not adversely affect the core support structures.

The staff understands that the level of detail in DCD Tier 1 is governed by a graded approach to the SSCs of the design based on the safety significance of the functions they perform and therefore, safety-related SSCs are described with a greater amount of detail than nonsafety-related SSCs. The staff also agrees that the existing ITAAC in place for the reactor system adequately address the construction of RV internal structures in such a way as to not adversely affect the integrity of the core support structures. However, a generic ITAAC on a system level for all nonseismic Category I SSCs to not adversely interact with seismic Category I SSCs is still needed to ensure that during or following an SSE, the seismic Category I SSCs will not be impaired from performing their safety functions due to interaction with nonseismic Category I SSCs.

DCD Tier 2, Subsection 3.2.1.1.2, states that seismic Category II applies to SSCs which perform no safety-related function, and whose continued function is not required, but whose structural failure or interaction could degrade the functioning or integrity of a seismic Category I SSC to an unacceptable level, or could result in incapacitating injury to occupants of the control room. Seismic Category II SSCs are designed so that the SSE could not cause unacceptable structural interaction or failure with seismic Category I SSCs. DCD Tier 2, Subsection 3.2.1.1.3 also identifies that non-seismic SSCs are primarily located outside of safety-related buildings or segregated from seismic Category I SSCs so that the failure of their structural integrity would not impact the seismic Category I SSCs and cause adverse system interactions. If it is determined that an SSC would cause an adverse impact on a seismic Category I SSC, then it is designed and/or mounted in accordance with seismic Category II requirements to withstand an SSE event so that it could not fail and cause an adverse impact or interaction with the seismic Category I SSC. Further, DCD Tier 2, Section 3.7.2.8,"Interaction of Non-seismic Category I Structures with seismic Category I Structures," identifies per COL Information Item 3.7(9) that the COL applicant is to assure that the design or location of any site-specific seismic Category I SSCs, such as pipe tunnels or duct banks, will not expose those SSCs to possible impact due to the failure or collapse of nonseismic Category I structures, or with any other SSCs that could potentially impact, such as heavy haul route loads, transmission towers, nonsafety-related storage tanks, etc. Alternately, site-specific seismic Category I SSCs are designed for impact loads due to postulated failure of the nonseismic Category I SSCs. As a result, **RAI 287-2014, Question 03.02.01-13**, was closed as unresolved. In follow-up **RAI 581-4582, Question 03.02.01-18**, the staff requested the applicant to clarify which specific ITAAC is used to verify completion of a system interaction review.

In its response to **RAI 581-4582, Question 03.02.01-18,** dated July 21, 2010, the applicant referenced its response to **RAI 580-4584, Question 03.02.02-15**, dated July 21, 2010, for ITAAC verifying system interaction and stated that the US-APWR does not include a generic ITAAC verifying system interaction between nonsafety-related and safety-related SSCs. The applicant stated that SRP Section 14.3.2 recognizes that seismic interaction between safety-related and nonsafety-related SSCs cannot be evaluated until the plant has been constructed. In this case, the COLA would require a description of a process for verifying acceptable seismic interaction between safety-related and nonsafety-related SSCs. However, SRP Section 14.3

identifies that some nonsafety-related SSCs that are important to safety are a special case and should have an ITAAC to verify the design of the system. The applicant further stated that an ITAAC to ensure nonseismic Category I SSCs would not interact with seismic Category I SSCs during or following an SSE is included for structures, light load handling system, and the overhead heavy load handling system.

The staff recognizes that SRP Section 14.3.2 states that seismic interaction between safety-related and nonsafety-related SSCs cannot be evaluated until the plant has been constructed. However, a design verification process needs to be in place to ensure that nonseismic Category I SSCs will not interact with seismic Category I SSCs for all systems in the plant. Therefore, **RAI 581-4582, Question 03.02.01-18**, was closed as unresolved. In follow up **RAI 684-5294, Question 03.02.01-20**, the staff requested the applicant to include an ITAAC for system interaction to ensure that nonseismic Category I SSCs will not interact with seismic Category I SSCs during or following an SSE.

In its response to **RAI 684-5294, Question 03.02.01-20**, dated July 27, 2012, the applicant referenced its response to **RAI 667-5235, Question 03.02.02-18**, dated July 27, 2012, which in turn referenced the response to **RAI 571-4365, Question 09.02.02-48**, dated July 29, 2011. In this response, the applicant agreed to include an ITAAC to ensure the seismic Category II systems and components will not impair the ability of seismic Category I SSCs to perform their design-basis safety function during or following an SSE in DCD Tier 1, Table 2.2.4, “Structures and Systems Engineering Inspections, Tests, Analyses, and Acceptance Criteria.” Along with the ITAAC for structures, these two ITAAC (Numbers 23.a and 23.b in the response to **RAI 571-4365, Question 09.02.02-48**), ensure by analyses and inspections that all seismic Category II SSCs will not impair the ability of seismic Category I SSCs to perform their design-basis safety functions during or following an SSE. With the understanding in DCD Tier 2, Section 3.2.1.1.3 that non-seismic SSCs are primarily located outside of safety-related buildings or segregated from seismic Category I SSCs so that the failure of their structural integrity would not impact the seismic Category I SSCs and cause adverse system interactions, the staff found this response acceptable because the proposed ITAAC are sufficient to ensure that all seismic Category II structures, systems, and components will not impair the ability of seismic Category I SSCs from performing their design basic safety functions during or following an SSE. Therefore, **RAI 684-5294, Question 03.02.01-20, is being tracked as Confirmatory Item** until the proposed ITAAC is incorporated in the next revision of the DCD.

3.2.1.4.13 Combined License Information Items

DCD Tier 2, Section 3.2.3, COL Information Item 3.2(4) refers only to safety-related systems and components that are to be identified by the COL applicant. The COL applicant should identify all site-specific SSCs including nonsafety-related SSCs that are not included in the DCD. For example, if the applicant adds a nonsafety-related site-specific SSC that should be seismic Category II, then that SSC should be included in the COLA. In **RAI 581-4582, Question 03.02.01-16**, the staff requested the applicant to provide additional information to explain how the COL applicant will be required to identify the seismic classification for all site-specific SSCs, including nonsafety-related SSCs.

In its response to **RAI 581-4582, Question 03.02.01-16,** dated July 21, 2010, the applicant stated that a new COL Item will be added to ensure the inclusion of the equipment classification and seismic classification of risk-significant, nonsafety-related SSCs as follows:

COL 3.2(6) The COL Applicant is to apply DCD methods of equipment classification and seismic categorization of risk-significant, nonsafety-related SSCs based on their safety role assumed in the PRA and treatment by the D-RAP.

Although not referenced in the response, COL Information Item 17.4(1) in DCD Tier 2, Section 17.4.9, "Combined License Information," addresses development of the D-RAP by the COL applicant. DCD Tier 1, Section 2.13, "Design Reliability Assurance Program," includes an ITAAC for the D-RAP. In addition, COL Information Item 3.2(5) in DCD Tier 2, Section 3.2.3 includes a provision for the COL applicant to identify the equipment class, seismic classification and applicable industry codes and standards of site-specific, safety-related and nonsafety-related fluid systems, components and equipment. Since ITAAC are included for the D-RAP and with the inclusion of the new COL Information Item 3.2(6), the staff found the response acceptable. Accordingly, **RAI 581-4582, Question 03.02.01-16, is resolved**.

3.2.1.4.14 Scope – Reactor Vessel Internals and Permanent Cavity Seal

10 CFR 50.34(f)(3)(ii) requires that the applicant ensure that the QA list required by Criterion II of 10 CFR Part 50, Appendix B, includes all SSCs important to safety. It is not clear that all SSCs that are within scope of the DCD and are not site-specific are included within scope of DCD Tier 2, Table 3.2-2. Certain potentially important to safety mechanical items such as nonsafety-related RV internals and the RV insulation do not appear to be specifically identified in DCD Tier 2, Table 3.2-2. In **RAI 287-2041 Question 03.02.01-14**, the staff requested the applicant to review the entire scope of SSCs that are not site-specific and include any missing items in DCD Tier 2, Table 3.2-2. Also the staff requested the applicant to clarify if there is an ITAAC for such nonsafety-related, but important to safety SSCs such as nonsafety-related RV internals and seismic Category II SSCs.

In its response to **RAI 287-2041, Question 03.02.01-14**, dated May 21, 2009, the applicant stated that RV internals were inadvertently omitted from Revision 1 of DCD Tier 2, Table 3.2-2. The applicant committed to revise DCD Tier 2, Table 3.2-2 to include reactor internals, the reactor coolant system (RCS) insulation, and other SSCs not specifically mentioned in DCD Tier 2, Revision 1, Table 3.2-2. DCD Tier 2, Revision 2, Table 3.2-2, includes missing SSCs identified by the applicant. However, the staff is concerned that other SSCs, such as the RV refueling cavity seal, are not specifically seismically classified in DCD Tier 2, Table 3.2-2 and, therefore, the scope of SSCs included in DCD Tier 2, Table 3.2-2, may still not be complete. Therefore, **RAI 287-2014, Question 03.02.01-14**, was closed as unresolved.

The staff understands that the seismic classification and other design issues associated with the refueling cavity seal are addressed in **RAI 507-3993, Question 09.01.04-16,** and its associated response, dated February 15, 2010, shows that the refueling cavity seal is classified as seismic Category II. RG 1.29 specifies that those systems, other than RWMS, which contain or may contain radioactive material are designated as seismic Category I if their failure could result in doses exceeding 0.5 rem (0.005 sievert). Further, RG 1.13, "Spent Fuel Storage Facility Design-Basis," Revision 2, issued March 2007, specifically identifies that the spent fuel storage facility, including all structures and equipment necessary to maintain minimum storage levels necessary for radiation shielding, should be designed to seismic Category I requirements. DCD Tier 2, Table 1.9.1-1 states that the applicant conforms with no exceptions to RG 1.29 and RG 1.13. DCD Tier 2, Section 9.1.3, "Spent Fuel Pit Cooling and Purification System," states that the spent fuel pit (SFP) cooling and purification system (SFPCS) cooling portions are safety-related, seismic Category I and the SFPCS conforms to RG 1.13. In follow-up **RAI 723-5382, Question 03.02.01-21**, the staff requested the applicant to further review the entire scope of SSCs that are not site-specific and include any missing items in DCD Tier 2, Table 3.2-2, including the refueling cavity seal or otherwise explaining where the refueling cavity seal is seismically classified**.**

In its response to **RAI 723-5382, Question 03.02.01-21,** dated April 21, 2011, the applicant referenced its response to **RAI 507-3993, Question 09.01.04-16**, dated February 15, 2010, in which the applicant revised the DCD so that DCD Tier 2, Table 3.2-2, includes the permanent cavity seal (PCS), also known as the RV refueling cavity seal. The applicant also referenced **RAI 724-5524, Question 03.02.02-20,** for the seismic classification and QG classification for the PCS. In its response to **RAI 724-5524, Question 03.02.02-20**, dated April, 21, 2011, the applicant determined that the PCS is a seismic Category I, safety-related, mechanical component that is assigned as QG C. All other SSCs that have a function to maintain minimum water levels for radiation shielding, specifically the fuel transfer tube, cask pit, fuel inspection pit, and refueling cavity drain piping are included in DCD Tier 2, Table 3.2-2, as seismic Category I. The staff found the response acceptable because the PCS is appropriately classified as seismic Category I, QG C with full 10 CFR 50 Appendix B requirements. Therefore, **RAI 723-5382, Question 03.02.01-21, is being tracked as a Confirmatory Item** until DCD Tier 2, Table 3.2-2, has been revised to show that the PCS is classified as seismic Category I.

3.2.1.4.15 Scope – Vent Stack and Essential Service Water Intake Screen

In **RAI 287-2041, Question 03.02.01-14** and **RAI 723-5382, Question 03.02.01-21**, the staff identified examples of SSCs that did not appear to be seismically classified in DCD Tier 2, Table 3.2-2. Although the applicant's responses to these questions proposed a revision to include certain missing components, the responses have not addressed the request to review the entire scope of SSCs that are not site-specific and completeness in scope of DCD Tier 2, Table 3.2-2. Two additional examples of SSCs that should be seismically classified are the vent stack and essential service water (ESW) intake screens. In **RAI 813-5935, Question 03.02.01-22**, the staff requested the applicant to clarify if DCD Tier 2, Table 3.2-2, has been reviewed for completeness of nonsite-specific SSCs and to include any additional SSCs and their classification. The staff also requested the applicant to identify SSCs that are site-specific to be classified by the COL applicant.

In its response to **RAI 813-5935, Question 03.02.01-22**, dated July 27, 2012, the applicant stated that based on the scope of SRP Section 3.2.1, which includes structures, dams, ponds, cooling towers and reactor internals, fluid systems important to safety that are identified in RG 1.29, safety-related instrument sensing lines that are identified in RG 1.151, ventilation systems, standby diesel generator auxiliary systems, fuel handling systems and cranes, DCD Tier 2, Table 3.2-2 as revised by the response to **RAI 667-5235, Question 03.02.02-17**, is complete in describing the seismic classification of nonsite-specific systems and components within the scope of SRP Section 3.2.1.

The applicant also revised DCD Tier 2, Table 3.2-4, to include the plant vent stack as a seismic Category II structure. In addition, the applicant stated that the ESWS is a safety-related system, and the components of the ESWS, including the intake screens, are classified as seismic Category I to ensure their functional capability. However, there are portions of the ESWS, including the ESWS intake screens, which are site-specific and are dependent on the site-specific design of the UHS. Therefore, these site-specific components are not added to the DCD Tier 2, Table 3.2-2. There are existing COL requirements to seismically classify these site-specific SSCs. The staff found the applicant's response acceptable because it stated that the seismic classification for nonsite-specific systems and components is complete in DCD Tier

2, Table 3.2-2. The plant vent stack and ESWS intake screens are also adequately addressed. In addition, the applicant identified and included, but not limited to, a list of site-specific SSCs that interface with the standard plant design. Therefore, all issues are resolved and **RAI 813-5935, Question 03.02.01-22, is being tracked as Confirmatory Item** until these proposed changes are revised in the next revision of the DCD.

3.2.1.5 Combined License Information Items

The following is a list of COL item numbers and descriptions from Table 1.8-2, "Compilation of All Combined License Applicant Items for Chapters 1-19," of the DCD related to seismic classification.

Table 3.2.1-2
US-APWR Combined License Information Items

Item No.	Description	Section
3.2(4)	The COL applicant is to identify the site-specific, safety-related systems and components that are designed to withstand the effects of earthquakes without loss of capability to perform their safety function; and those site-specific, safety-related fluid systems or portions thereof; as well as the applicable industry codes and standards for pressure-retaining components.	3.2.1 3.2.3 3.2.3 Table 3.2-2
3.2(5)	The COL applicant is to identify the equipment class and seismic category of the site-specific, safety-related and nonsafety-related fluid systems, components (including pressure-retaining), and equipment as well as the applicable industry codes and standards.	3.2.1 3.2.3 Table 3.2-2
3.2(6)	The COL Applicant is to apply DCD methods of equipment classification and seismic categorization of risk-significant, nonsafety-related SSCs based on their safety role assumed in the PRA and treatment by the D-RAP.	3.2.1 3.2.2 3.2.3

The staff determined the above listing to be complete. Also, the list adequately describes actions necessary for the COL applicant or licensee. No additional COL information items need to be included in DCD Tier 2, Table 1.8-2 or Section 3.2.3, for seismic classification consideration.

3.2.1.6 Conclusions

As a result of the open item for **RAI 1015-7054, Question 03.09.03-31**, the staff is unable to finalize its conclusions on Section 3.2.1 related to seismic classification, in accordance with NRC regulations.

3.2.2 System Quality Group Classification (Safety Classification)

3.2.2.1 Introduction

The objective of the staff review in this section is to determine that SSCs important to safety have been identified and appropriately categorized and that appropriate codes and standards for design, erection, fabrication and testing have been selected commensurate with their importance to safety.

3.2.2.2 Summary of Application

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

DCD Tier 2: The applicant has provided a DCD Tier 2 description in Section 3.2.2, summarized here in part, as follows:

> DCD Tier 2, Section 3.2.2, identifies the criteria used to classify US-APWR mechanical equipment and fluid systems. The safety-related US-APWR fluid systems and components that are classified as QG A, B or C are identified in DCD Tier 2, Table 3.2-2, “Classification of Mechanical and Fluid Systems, Components, and Equipment.” Nonsafety-related fluid systems that do not fall within QG A, B or C are also included in this table as either QG D or NA for not applicable. DCD Tier 2, Table 3.2-3, “Comparison of Various Requirements to Equipment Class,” summarizes the relationship between ASME Code, Section III Class, equipment class and QG. In addition to the QG for pressure-retaining components, these tables identify various Equipment Classes and QA requirements as either 10 CFR Part 50, Appendix B, applicable or not. The applicable chapters on various fluid systems together with simplified P&IDs found in other sections of the DCD Tier 2 also identify applicable codes and industry standards as well as quality and equipment classifications for fluid systems. DCD Tier 2, “Table 1.9.2-3, US-APWR Conformance with Standard Review Plan Chapter 3 Design of Structures, Systems, Components, and Equipment,” identifies that there are no differences with SRP Section 3.2.2, “System Quality Group Classification,” Revision 2, issued March 2007, and DCD Tier 2, Table 1.9.1-1, “US-APWR Conformance with Division 1 Regulatory Guides,” identifies no exceptions to RG 1.26, “Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants,” Revision 4, issued March 2007, RG 1.143, “Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants,” Revision 2, issued November 2001, and RG 1.151, “Revision 0, issued July 1983, for the US-APWR. DCD Tier 2, Section 3.2.2 further identifies that RG 1.26 is used to meet regulatory requirements by classifying safety-related fluid systems and components and applying corresponding quality codes and standards.

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

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 for this area of review.

Cross-cutting Requirements (TMI, USI/ GSI, Op Ex): There are no cross-cutting requirements for this area of review.

US-APWR Interface Issues Identified in the DCD: See DCD Tier 2 Table 1.8-2, “Compilation of All Combined License Applicant Items for Chapters 1-19.”

Site Interface Issues Identified in the DCD: See DCD Tier 2, Table 1.8-1, "Significant Site Specific Interfaces with the Standard US-APWR Design."

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

3.2.2.3 Regulatory Basis

The relevant requirements of the Commission's regulations for this area of review, and the associated acceptance criteria, are given in Section 3.2.2, "System Quality Group Classification," Revision 2, issued March 2007, of NUREG-0800, the SRP, and are summarized below. Review interfaces with other SRP sections also can be found in Section 3.2.2 of NUREG-0800.

1. GDC 1 and 10 CFR 50.55a, as they relate to SSCs important to safety being designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety function to be performed.

2. 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 has been constructed 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 provides an acceptable method of meeting the requirements of GDC 1 and 10 CFR 50.55a. This guide describes an acceptable method for determining quality standards for QG B, C, and D, water- and steam-containing components important to safety in light-water-cooled nuclear power plants.

2. RG 1.143, as it relates to the classification and application of quality standards for RWMS.

3. RG 1.151, as it relates to the classification and application of quality standards for instrument sensing lines.

4. Components that are part of the RCPB must meet the requirements for ASME Code Class 1 components in ASME Boiler and Pressure Vessel (B&PV) Code, Section III. QG A standards are required for pressure-containing components of the RCPB that comply with ASME B&PV Code, Section III, Class 1. QG B and QG C must meet the requirements for ASME Code Class 2 and Class 3, respectively.

3.2.2.4 Technical Evaluation

3.2.2.4.1 Quality Group A

QG A standards are required for pressure-containing components of the RCPB that comply with ASME B&PV Code, Section III, Class 1. The staff reviewed DCD Tier 2, Tables 3.2-2, 3.2-3, and the simplified P&IDs in accordance with SRP Section 3.2.2. Section 5.2 of this report discusses the conformance of ASME Code Class 1, RCPB components to the requirements of 10 CFR 50.55a. RG 1.26 designates these RCPB components as QG A. The staff determined that the applicant has properly classified RCPB components consistent with 10 CFR 50.55a.

3.2.2.4.2 Quality Group B

QG B standards are required for pressure-containing components of the RCPB that comply with ASME B&PV Code, Section III, Class 2. Based on its review of the information in DCD Tier 2, Section 3.2.2, Tables 3.2-2, 3.2-3, and the applicable simplified P&IDs, the staff has determined that the classifications for QG B SSCs of the US-APWR are consistent with the guidelines in SRP Section 3.2.2 and RG 1.26, and are in conformance with GDC 1, and therefore are acceptable.

3.2.2.4.3 Quality Group C

QG C standards are required for pressure-containing components of the RCPB that comply with ASME B&PV Code, Section III, Class 3. Based on its review of the information in DCD Tier 2, Section 3.2.2, Tables 3.2-2, 3.2-3, and the applicable simplified P&IDs, the staff has determined that the classifications for QG C SSCs of the US-APWR are consistent with the guidelines in SRP Section 3.2.2 and RG 1.26 and in conformance with GDC 1, and therefore they are acceptable. To make this finding, the staff requested that the applicant clarify classification criteria in the DCD to conform to the SRP Section 3.2.2 and RG 1.26 guidance as follows.

DCD Tier 2, Section 3.2.2.3, "Equipment Class 3," Revision 1, stated that Equipment Class 3 is equivalent to RG 1.26, NRC QG C. However, the applicant only partially addressed the guidelines in RG 1.26, Section 2, QG C, paragraph (d). In **RAI 276-2043, Question 03.02.02-2**, the staff requested the applicant to discuss where the requirements regarding single component failures for systems located in seismic Category I structures are addressed or why these requirements are not applicable. In its response to **RAI 276-2043, Question 03.02.02-2**, dated April 24, 2009, the applicant modified DCD Tier 2, Section 3.2.2.3 to fully incorporate all positions of RG 1.26 for QG C SSCs. The staff found the response acceptable since the applicant conforms to RG 1.26 regarding QG C. The staff reviewed the DCD Revision 2 and confirmed that the changes were acceptable. Accordingly, **RAI 276-2043, Question 03.02.02-2, is resolved**.

3.2.2.4.4 Quality Group D

QG D systems are those that contain or may contain radioactivity that are not included in QGs A, B or C. Equipment Class 4 and Equipment Class 8 include nonsafety-related systems and components that contain radioactive material. Based on its review of the information in DCD Tier 2, Section 3.2.2, Tables 3.2-2, 3.2-3, and the applicable simplified P&IDs, the staff has determined that the classifications for QG D SSCs of the US-APWR are, in general, consistent with the guidelines in SRP Section 3.2.2 and RG 1.26 and in conformance with GDC 1, and therefore they are acceptable. However, systems such as the circulating water system (CWS) are not classified as QG D consistent with the guidance in SRP Section 10.4.5, "Circulating Water System," Revision 3, issued September 2007. In **RAI 580-4584, Question 03.02.02-13**, the staff requested the applicant to explain why the CWS is not considered QG D. The staff also requested the applicant to verify that all systems that may contain radioactivity are appropriately classified consistent with QGs A, B, C, or D and designed consistent with the guidance provided in RG 1.143.

In its response to **RAI 580-4584, Question 03.02.02-13**, dated July 22, 2010, the applicant stated that the US-APWR CWS is not safety-related and it does not have any safety-related components or functions. The CWS is not a RWMS. The system is not expected to be radioactively contaminated during normal plant operations and, even under transient conditions involving radioactive contamination of the secondary side of the plant, the CWS is not expected to contain measurable amounts of radioactive materials. Therefore, the QG of US-APWR CWS components is not classified as QGs B, C, or D components consistent with the guidance in RG 1.26 and the guidance provided in RG 1.143 does not apply. Moreover, leakage due to CWS failure cannot reach safety-related equipment located in seismic Category I plant structures as described in DCD Tier 2, Section 3.4.1.3, "Flood Protection from Internal Sources." Therefore, the CWS components equipment are classified as "Class 9," consistent with the definition provided in DCD Tier 2, Section 3.2.2, with a QG of "N/A" as identified in the DCD Tier 2, Table 3.2-2.

Since the applicant identified that the CWS is not expected to contain measurable amounts of radioactive material, staff concludes that the CWS need not be QG D. All issues associated with the DCD Tier 2, Section 3.2.2 QG classification of CWS are addressed. The staff noted that the applicant did not address other nonsafety-related systems outside the scope of RG 1.143 that may contain radioactive material. As a result, **RAI 580-4584, Question 03.02.02-13,** was closed as unresolved.

In follow-up **RAI 667-5235, Question 03.02.02-19**, the staff requested the applicant to clarify if other nonsafety-related systems that may contain radioactivity that are not classified as QG A, B or C, are either radwaste systems consistent with RG 1.143 or are classified as QG D consistent with RG 1.26. In its response to **RAI 667-5235, Question 03.02.02-19**, dated July 27, 2012, the applicant clarified that as per definition in DCD Tier 2, Section 3.2.2, and Table 3.2-2, water-containing and steam-containing nonsafety-related SSCs that contain or may contain radioactivity are classified as QG D consistent with RG 1.26 (i.e., Equipment Classes 4 or 8). Nonsafety-related SSCs within the scope of RG 1.143 are classified as Equipment Class 6. The staff found this response to be acceptable since it clarified how systems that may contain radioactivity are classified. Accordingly, **RAI 667-5235, Question 03.02.02-19, are resolved**.

3.2.2.4.5 Graded QA Approach

The staff reviewed the DCD to ensure appropriate criteria, quality standards and QA criteria was applied to fluid systems, mechanical components and their supports important to safety. The DCD states conformance with RG 1.26 and SRP Section 3.2.2 regarding QG designations for pressure-retaining fluid systems and components, including their supports in order to comply with GDC 1. QGs identified in RG 1.26 are based on their importance to safety. Each QG designates applicable codes, and standards that are combined with associated QA standards to ensure compliance with GDC 1. Quality standards and measures applicable to safety-related QGs are based on application of the ASME Code, Section III, but appropriate quality standards and criteria applicable to nonsafety-related SSCs that are important to safety (special treatment) is to be defined by the designer, based on the applicant's design process, such as a risk-informed approach or the RTNSS process, and the SSCs' particular safety functions.

A risk-informed classification methodology has been applied on a trial basis for operating reactors, but this approach has not been applied to new reactor designs. RG 1.26 includes four basic QGs for systems and equipment depending on their safety function. QA controls associated with the RAP in SRP Section 17.4, "Reliability Assurance Program (RAP)," issued

March 2007, are addressed in the DC/COL- ISG-018, "Interim Staff Guidance on Standard Review Plan, Section 17.4, "Reliability Assurance Program," for implementing QA controls during the DC phase as part of the D-RAP.

DCD Tier 2, Section 3.2.2.4, "Equipment Class 4," states for Equipment Class 4 components, the guidance of RG 1.26, QG D will be applied. DCD Tier 2, Table 3.2-3 contains Footnote 4 that states a graded approach to QA of Equipment Class 4 components will be applied. This footnote references a description of a graded approach as provided in DCD Tier 2, Chapter 17, "Quality Assurance and Reliability Assurance," but no description of "graded approach" was found. In **RAI 276-2043, Question 03.02.02-3**, the staff requested the applicant to describe how this proposed graded approach is consistent with the guidance in RG 1.26 for QG D components and any applicable special treatment requirements. This same Footnote 4 is also referenced to attribute a graded approach to Equipment Class 5 components. Per DCD Tier 2, Section 3.2.2.5, "Other Equipment Classes," components in Equipment Class 5 are nonsafety-related components that are not part of the RWMS, are not within the purview of RG 1.26, and to which 10 CFR Part 50, Appendix B, requirement do not apply. In **RAI 276-2043, Question 03.02.02-3**, the staff also requested the applicant to provide further information on how this "graded approach" is to be applied to nonsafety-related components classified as Equipment Class 5. In its response to **RAI 276-2043, Question 03.02.02-3**, dated May 8, 2009, the applicant proposed to revise DCD Tier 2, Table 3.2-3, in Revision 2 to remove references to a graded approach and to clarify that seismic Category II, Equipment Class 4 and 5 SSCs will meet the pertinent requirements of 10 CFR Part 50, Appendix B. DCD Tier 2, Table 3.2-2, however, still shows that 10 CFR Part 50 Appendix B, is not applicable to seismic Category II SSCs and this concern is further reviewed in **RAI 287-2041, Question 03.02.01-6 and RAI 287-2041, Question 03.02.01-7**. Since the DCD has been revised to delete reference to a graded approach without adequately describing how risk-significant, nonsafety-related SSCs will be classified so that they have appropriate special treatment applied, additional information was needed. As a result, **RAI 276-2043, Question 03.02.02-3** was closed as unresolved. In follow-up **RAI 580-4584, Question 03.02.02-10**, the staff requested the applicant to clarify what unique classification and special treatment, including QA, applies to nonsafety-related risk-significant systems and components to distinguish them from nonsafety-related SSCs that have no risk significance.

In its response to **RAI 580-4584, Question 03.02.02-10,** dated July 22, 2010, the applicant references the D-RAP discussed in DCD Tier 2, Section 17.4, "Reliability Assurance Program," for the process to identify and design risk-significant SSCs. The applicant will revise the description in DCD Tier 2, Section 3.2.2.5 to include the risk-significant, nonsafety-related SSCs that are identified with the Seismic Event Rationale in DCD Tier 2, Table 17.4-1, "Risk-significant SSCs," and a new COL Information Item, 3.2(6), is to be included for the COL applicant to apply the D-RAP process. The response stated that this process will be applied to the final US-APWR design to ensure that all appropriate nonsafety-related, risk-significant SSCs within the scope of the DCD are captured. These SSCs will be provided with special treatment requirements to ensure reliability assumed in the PRA in accordance with the D-RAP.

The staff finds the response to **RAI 580-4584, Question 03.02.02-10,** acceptable as follows. The guidance in SRP Section 17.1, "Quality Assurance During the Design and Construction Phases," Revision 2, issued July 1981, states that the QA program identifies the extent QA controls are involved in applying a graded approach to certain SSCs in accordance with their importance to safety. Although reference to a graded approach was deleted in DCD Tier 2, Table 3.2-3, the applicant's response to **RAI 580-4584, Question 03.02.02-10**, clarifies that the PRA process and D-RAP are applied for risk-significant SSCs. The staff's concern, that a graded QAP to apply pertinent elements of the 10 CFR Part 50, Appendix B, program to seismic Category II SSCs and risk-significant systems, is resolved by application of the PRA process and D-RAP combined with the QAP described in Topical Report PQD-HD-19005, "Quality Assurance Program (QAP) Description For Design Certification of the US-APWR," Revision 3, issued September 2009, and DCD Tier 2, Chapter 17 that includes QA requirements for both safety-related and nonsafety-related systems. Application of the PRA and D-RAP to identify risk-significant SSCs and QA requirements based on safety significance is consistent with DC/COL ISG-018. The staff confirmed that DCD, Revision 3 incorporates the changes including the new COL Information Item 3.2(6). The staff's remaining concern associated with the process to apply appropriate quality standards to risk-significant, nonsafety-related SSCs, including the applicant's process to apply the terms important to safety and safety-related, is addressed by **RAI 667-5235, Question 03.02.02-17**. Accordingly, **RAI 580-4584, Question 03.02.02-10, is resolved**.

3.2.2.4.6 Conformance with RG 1.143

QG classification of structures, such as the radwaste facility, is excluded from the scope of the review in SRP Section 3.2.2. However, the QG classification of fluid systems within the radwaste building is within scope of the review. For the liquid radwaste system, DCD Tier 2, Table 3.2-2 identifies these systems as Equipment Class 6 with no corresponding QG. Although systems containing radioactive fluids are normally QG C or QG D, no QG need be specified since RG 1.143 designates appropriate codes and standards. However, DCD Tier 2, Section 3.2.2.5, Equipment Class 6, states that codes and standards defined in RG 1.143, Table 1, "Codes and Standards for the Design of SSC in Radwaste Facilities," are applied to Equipment Class 6 components, implying only codes and standards in Table 1 are being used. This same language also appears in DCD Tier 2, Chapter 11, "Radioactive Waste Management." In **RAI 276-2043, Question 03.02.02-4**, the staff requested the applicant to clarify if Equipment Class 6 components will be subject to all appropriate design guidance of RG 1.143 or only those specified in Table 1 of RG 1.143. In its response to **RAI 276-2043, Question 03.02.02-4**, dated April 24, 2009, the applicant clarified DCD Tier 2, Sections 3.2, "Classification of Structures, Systems, and Components," 11.2, "Liquid Waste Management System," and 11.3, "Gaseous Waste Management System," to stipulate that Equipment Class 6 components are to be designed to be compliant with the applicable codes, standards, and guidance provided in RG 1.143 and has removed references to Table 1 of RG 1.143. The staff found the response acceptable since the applicant clarified that Equipment Class 6 components conform to RG 1.143. The staff confirmed that the proposed DCD changes were incorporated into DCD Revision 3. The staff reviewed the changes and found them to be acceptable. Accordingly, **RAI 276-2043, Question 03.02.02-4, is resolved**.

3.2.2.4.7 Codes and Standards

The requirements of 10 CFR 50.55a identify the codes and standards to be used by applicants. In addition, SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs," issued April 2, 1983, and the associated SRM, issued July 21, 1983, states that the NRC will review evolutionary plant designs using the newest codes and standards endorsed by the NRC. Unapproved revisions to codes and standards referenced in the applications will be reviewed on a case by case basis. In DCD Tier 2, Section 3.2.4, "References," editions of codes and standards are either omitted or are not current. In **RAI 276-2043, Question 03.02.02-5,** the staff requested the applicant to clarify how this requirement is fulfilled. In its response to **RAI 276-2043, Question 03.02.02-5**, dated May 8, 2009, the applicant identified that DCD Tier 2, Section 3.2.4 has been revised to update or add the current editions of codes and standards and referenced the list of codes and standards in the applicant's response to **RAI 276-2043, Question 03.02.02-6**, dated May 8, 2009, that will be applied or updated during development of detailed design specifications. Certain codes and standards listed, such as American National Standards Institute/American Nuclear Society (ANSI/ANS) 51.1-1988, have not been endorsed by the NRC and the NRC does not use this document as a basis for acceptability. The RAI response does not explain the applicant's process to apply the latest codes and standards endorsed by the NRC. As a result, **RAI 276-2043, Question 03.02.02-5,** was closed as unresolved and additional information was requested in follow-up **RAI 580-4584, Question 03.02.02-11,** discussed below.

DCD Tier 2, Table 3.2-3 Note 5 states that codes and standards defined in design-basis are applied. Equipment Class 5 references Note 5 for the codes and standards. In **RAI 276-2043, Question 03.02.02-6**, the staff requested the applicant to clarify what codes and standards, including their editions, will be used when Note 5 is referenced. In its response to **RAI 276-2043, Question 03.02.02-6**, dated May 8, 2009, the applicant included a list of codes and standards and identified those additional codes and standards or later editions will be applied during the development of detailed design specifications. Since the DCD was not revised, **RAI 276-2043, Question 03.02.02-6,** was closed as unresolved and additional information was requested in **RAI 580-4584, Question 03.02.02-11,** as discussed below.

DCD Tier 2, Table 3.2-2 identifies Equipment Class 5 for two systems and RG 1.26 is referenced for applicable codes and standards. RG 1.26 does not identify codes and standards for Equipment Class 5 systems that are not included in one of the four QGs. In **RAI 276-2043, Question 03.02.02-9**, the staff requested the applicant to identify the applicable codes and standards for Equipment Class 5. In its response to **RAI 276-2043, Question 03.02.02-9**, dated April 24, 2009, the applicant identified that, for the two Equipment Class 5 systems in question, codes and standards are as defined in the design-basis. The staff considers this acceptable since RG 1.26 does not identify codes and standards for Equipment Class 5, however, codes and standards for Equipment Class 5 should be defined. For example, DCD Tier 2, Revision 2, Table 3.2-2 identified Equipment Class 5 for several SSCs of the essential chilled water system and RG 1.26 is referenced for applicable codes and standards. However, the DCD was not updated to include these codes, standards, and the applicant's response to **RAI 276-2043, Question 03.02.02-9**, refers to the design-basis for codes and standards applicable to Equipment Class 5, rather than identifying the specific codes and standards. Since the RAI response did not resolve this issue for mechanical components, **RAI 276-2043, Question 03.02.02-9,** was closed as unresolved and additional information was requested in **RAI 580-4584, Question 03.02.02-11,** as discussed below.

The applicant should identify all editions of codes and standards in the DCD and clarify if the COL applicant should update the editions used for procurement. The COL applicant can also be expected to identify applicable codes and standards in the COLA if different from the DCD.

As noted above, **RAI 580-4584, Question 03.02.02-11,** is a follow-up to **RAI 276-2043, Questions 03.02.02-5, 03.02.02-6, and 03.02.02-9.** In **RAI 580-4584, Question 03.02.02-11,** the staff requested the applicant to identify the process, including update of the DCD, that is to be used to verify the latest editions of codes and standards endorsed by the NRC will be applied to the design and procurement of mechanical components.

In its response to **RAI 580-4584, Question 03.02.02-11**, dated July 22, 2010, the applicant stated that it will review the latest version of codes and standards in support of design and procurement efforts. The response also clarified that the latest codes and standards, when endorsed by the NRC, will be applied to the design and procurement of mechanical equipment within scope of the US-APWR DCD. Since the applicant has clarified that the latest codes and standards, when endorsed by the NRC, will be applied to the design and procurement of mechanical equipment within scope of the US-APWR DCD, this approach is consistent with SECY-93-087. Although the RAI responses pertaining to codes and standards do not fully address updating the DCD to apply the latest editions of the entire list of codes and standards included in the RAI response or those endorsed by the NRC, DCD Tier 2, Section 3.2.4 has been revised to update those specific editions of codes and standards referenced in RG 1.26, such as ASME B31.1, that will be applied to design and procurement of Equipment Class 4 and 5 systems and components. Specific editions of codes and standards are reviewed for acceptability in other sections of this report. DCD Tier 2, Section 3.2.4 also references 10 CFR 50.55a for the ASME Code, Section III endorsed by the NRC. Compliance with 10 CFR 50.55(a) is evaluated in Section 5.2.1.1 of this report. Therefore, the updated code editions that the applicant applies to the design and procurement for both safety-related and nonsafety-related systems and components that are important to safety are now defined in the DCD. The staff finds the response acceptable since the applicant clarified the issues associated with defining editions of such codes and standards for mechanical components. Accordingly, **RAI 580-4584, Question 03.02.02-11, are resolved**.

The discussion of the previous RAIs in this section describes the applicant's approach to identifying codes and standards and the process for applying the latest editions of these codes for nonsafety-related Equipment Class 5. However, regarding specific codes and standards applied to various components listed in DCD Tier 2, Table 3.2-2, there appears to be a discrepancy regarding DCD Tier 2, Table 3.2-2, Note 3, Item (5), and the codes and standards designated in DCD Tier 2, Table 3.2-3. For various components, DCD Tier 2, Table 3.2-2, Note 3, Item (5), references codes and standards as those defined in the design-basis. Since DCD Tier 2, Table 3.2-3 identifies that ASME Code, Section III Class 1, 2 and 3 applies to the corresponding Equipment Class 1, 2, and 3 and QG A, B, and C, it is understood that the ASME Code, Section III is applied to QG A, B, and C pressure-retaining components and their supports. For example, any component designated QG A, B or C would be expected to be constructed to ASME Code, Section III, but in DCD Tier 2, Table 3.2-2 various equipment Class 3, QG C emergency gas turbine auxiliary system components reference Note 3, Item (5) for codes and standards rather than Note 3, Item (3) for ASME Code, Section III, Class 3. It is presumed that DCD Tier 2, Table 3.2-2 Note 3, Item (5) is used only to signify that the code of record referenced in the DCD will be documented for each pressure-retaining component in design-basis documents. Further, staff understands that DCD Tier 2, Table 3.2-3 Note 5 is only applied to nonsafety-related Equipment Class 5 components and, as such, does not specifically apply to safety-related Equipment Class 1, 2, or 3 and the reference to codes and standards in the design-basis is not intended to negate the ASME Code classification.

Also, since DCD Tier 2, Table 3.2-2, Note 3, Item (5) does not identify the specific code, it is not clear what specific codes and standards apply to various safety-related and nonsafety-related systems and components, including Equipment Class 2 fuel assemblies and Equipment Class 5 risk-significant systems and components.

In addition, it is not clear that Tier 1 includes all the risk-significant components included in Tier 2, including the ASME Code Class. Since Tier 1 information is to be based on Tier 2 information, Tier 1 information should be consistent with Tier 2. For example, it is not clear where the ASME Code classification of the PCS or heating, ventilation, and air conditioning (HVAC) systems is included in Tier 1.

Therefore, to resolve these apparent discrepancies, in **RAI 914-6365, Question 03.02.02-22**, the staff requested the applicant to clarify the application of DCD Tier 2, Table 3.2-2 Note 3, Item (5) and DCD Tier 2, Table 3.2-3 Note 5 to define specific codes and standards used to establish the individual component design-basis and licensing basis for pressure-retaining components and their supports. The entire DCD Tier 2, Table 3.2-2 codes and standards column including the notes and DCD Tier 1 should be reviewed and revised as necessary for clarity and consistency concerning ASME Code class and the designated codes and standards.

In its response to **RAI 914-6365, Question 03.02.02-22**, dated November 2, 2012, the applicant stated that as part of the applicant's response to **RAI 667-5235, 03.02.02-17**, DCD Tier 2, Tables 3.2-2 and 3.2-3 have been revised to clarify equipment classifications, the applicant's QA classifications, and to provide consistency in application of RG 1.26 QG classifications. The change has included specifying nonsafety-related risk-significant SSCs in DCD Tier 2, Table 3.2-2. As included in the revised DCD Tier 2 markups provided in the applicants response to **RAI 667-5235, 03.02.02-17**, the descriptions of the equipment classes have been revised, in part, to clarify that some Equipment Class 2 and 3 components have design-basis codes that are different than ASME Code Section III. The Equipment Class 2 and 3 descriptions now state for example, "Other non-ASME Code, Section III Equipment Class [X] components are designed to meet codes and standards described in the design bases description of the applicable system...." This change makes these Class descriptions consistent with DCD Tier 2, Table 3.2-3, Note 1, which reads, "Items not covered by the ASME Code are designed to other applicable codes and standards."

US-APWR applies a QA classification not only for pressure-retaining components and their supports, but also for non-pressure retaining components. RG 1.26 QG classifications are only applied to those components and their supports within the scope of RG 1.26. Therefore, there are some components that are described as Equipment Class 3, N/A for RG 1.26 QG classification, and seismic Category I. For these components, ASME Code, Section III is not listed as the applicable code in DCD Tier 2, Table 3.2-2 because they are non-pressure retaining components. Instead, applicable codes and standards are specified in the design-basis (i.e., corresponding DCD Tier 2 sections). Changes were made in the response to **RAI 667-5235, Question 03.02.02-17,** to ensure that this approach was consistently applied. Additionally, Note 13 was added to DCD Tier 2, Table 3.2-3 for Equipment Class 2 and 3 components in the RG 1.26 QG column that reads "N/A for items not covered by RG 1.26 (Reference 3.2-13) Table 1." As a result, the discrepancy regarding DCD Tier 2, Table 3.2-2 Note 3, Item (5) and the codes and standards designated in DCD Tier 2, Table 3.2-3 was corrected.

The response to **RAI 914-6365, Question 03.02.02-22,** states that for example, the design bases for emergency gas turbine auxiliary system components have been addressed in DCD Tier 2, Sections 9.5.4, "Gas Turbine Generator Fuel Oil Storage and Transfer System," 9.5.6, "Gas Turbine Generator Fuel Oil Storage and Transfer System," 9.5.7, "Gas Turbine Lubrication System," and 9.5.8, "GTG Combustion Air Intake, Turbine Exhaust, Room Air Supply, and Air Exhaust Systems." Those components are designed in accordance with either ASME Code, Section III Class 3 requirements or manufacturer standards as specified in DCD Tier 2. For fuel assemblies, DCD Tier 2, Section 4.2.1.5, "Fuel Assembly," has addressed the design bases and applicability of ASME Code, Section III requirements. Thus, the appropriate design-basis for safety-related SSCs and certain nonsafety-related SSCs have been defined in corresponding DCD Tier 2 sections, in accordance with SRP acceptance criteria, and application of DCD Tier 2, Table 3.2-2 Note 3 Item (5) is appropriate for these SSCs. However, the applicant revised several Tier 2 descriptions to clarify the design bases of several SSCs, including the alternate AC gas turbine generators (GTGs) and the essential chilled water system.

The response to **RAI 914-6365, Question 03.02.02-22,** includes a Table 1 that provides a cross-reference between systems, including HVAC system, listed in DCD Tier 2 Table 3.2-2 and location of the design information in DCD Tier 1 for ASME Code, Section III Classes 1, 2 and 3 PSC and nonsafety-related risk-significant SSCs. Risk-significant SSCs are listed in DCD Tier 2, Table 17.4-1. If either or both of the ASME Code, Section III PSC and the risk-significant SSCs are not addressed in Tier 1, Table 1 in the response to **RAI 914-6365, Question 03.02.02-22,** addresses where in Tier 1 it will be added or why this information is not required in Tier 1.

The applicant revised the Tier 1 and Tier 2 descriptions as shown in the markups to the response with the intent that (1) DCD Tier 2 provides clear descriptions of design bases codes and standards for nonsafety-related risk-significant SSCs and Equipment Class 2 and 3 components that are not ASME Code, Section III, and (2) DCD Tier 1 ITAAC exist for nonsafety-related risk-significant SSCs and the Tier 1 information is consistent with the corresponding DCD Tier 2 information. This latter change includes a correction of Tier 1 information to address missing ASME Code, Section III components in the HVAC systems such as DCD Tier 1, Tables 2.7.5.1-1, “Main Control Room HVAC System Equipment Characteristics,” and 2.7.5.2-1, “Engineered Safety Features Ventilation System Equipment Characteristics.”

The staff finds the applicant’s response acceptable as the applicant has clarified all of the staff concerns, including clarifying the codes and standards applied to ASME Code, Section III components and risk-significant components. In addition, the applicant has provided corresponding DCD markups to address both DCD Tier 1 and Tier 2 changes. Accordingly, **RAI 914-6365, Question 03.02.02-22, is being tracked as a Confirmatory Item** pending revision of the DCD.

3.2.2.4.8 Nonmetallic Piping

For systems such as the ESWS that may be installed below grade, it is not clear if these systems may use nonmetallic piping. In **RAI 580-4584, Question 03.02.02-12**, the staff requested the applicant to clarify if any nonmetallic piping or nonmetallic lined piping will be used in ASME class piping systems or nonsafety-related but important-to-safety piping systems and, if so, to identify the applications and appropriate codes and standards, including quality requirements.

In its responses to **RAI 580-4584, Question 03.02.02-12**, dated July 22, 2010, the applicant clarified that, for underground piping systems, lined piping or nonmetallic piping is not used in any ASME class piping or any piping listed as Seismic Event Rational in DCD Tier 2, Table 17.4-1 other than ESWS piping. The applicant proposed to revise DCD Tier 2, Subsection 9.2.1.2.2.5, “Piping,” to change the ESWS piping materials from epoxy lined carbon steel to carbon steel with lining material according to site soil and ESWS water chemistry requirements. Use of piping lining is reviewed in Chapter 9 of this report. This change is acceptable because nonmetallic piping is not used for any pressure boundary applications in important to safety systems. The applicant also proposed to revise DCD Tier 2, Table 3.2-2 to show QG, codes and standards and seismic category of ESWS piping. The staff finds applicant response acceptable as the applicant has clarified the use nonmetallic piping or nonmetallic lined piping and identified the corresponding codes and standards. The staff confirmed that DCD Revision 3 incorporated the changes. Accordingly, **RAI 580-4584, Question 03.02.02-12,** is resolved.

3.2.2.4.9 Refueling Seals

In **RAI 724-5524, Question 03.02.02-20**, the staff identified a concern that the scope of SSCs included in DCD Tier 2, Table 3.2-2 may not be entirely complete and previous RAI responses to questions concerning Sections 3.2.1 and 9.1.4 of the application did not address the completeness of this table. This concern was based on the response to design issues associated with the PCS addressed in **RAI 507-3993, Question 09.01.04-16**, which stated that the classification for the PCS is Equipment Class 4 with QG D and seismic Category II. In **RAI 724-5524, Question 03.02.02-20**, the staff requested the applicant to justify these classifications as follows.

- If the PCS is considered a structural component, QG should not apply.
- If the PCS is considered a mechanical component, a description of the extent of certification and stamping should be provided, including an explanation as to why this component is classified as QG D rather than QG C. If designed to ASME Code, Section III, QG C would be more consistent with the classification of other mechanical components in the refueling system.
- Describe the safety function and the basis for the designation as safety-related, important to safety or nonsafety-related.
- Since the seal is classified as seismic Category II rather than seismic Category I, explain why the seal would remain functional during and after a seismic event.
- If the seal is considered safety-related, the basis for the classification as QG D should be described.
- If the seal is defined as nonsafety-related, but is important to safety concerning the risk to health and safety of the public, describe the evaluation of risk-significance.
- If the seal is not postulated to fail, justify why a single failure (rupture or crack) is not postulated to occur.
- If the seal could fail or leak as a postulated passive failure during refueling operations, explain why the seal failure will not result in excess off-site doses.

In its response to **RAI 724-5524, Question 03.02.02-20**, dated April 21, 2011, the applicant stated that the concern from the applicant's response to **RAI 507-3993, Question 09.01.04-16,** for design issues associated with the PCS is addressed in the applicant's response to **RAI 633-4857, Question 09.01.04-21**, dated October 21, 2010, which states that the PCS is considered a mechanical component. The seal performs a safety function to ensure the water level of the reactor cavity is maintained so the stored fuel and fuel in transit remain adequately submerged and exposure limits specified in 10 CFR 50.34(a)(1) and 10 CFR 100.11 are not exceeded. Based on this safety function, the applicant will revise DCD Tier 2, Table 3.2.2 such that the seal will be classified as a safety-related, Equipment Class 3, QG C, and seismic Category I SSC. As such, the PCS will be designed in conformance with appropriate codes and standards selected according to this classification and the QAP.

The RAI response has not specifically addressed the completeness in scope of SSCs included in DCD Tier 2, Table 3.2-2, of which the PCS was an example. The completeness of DCD Tier 2, Table 3.2-2 is addressed in Section 3.2.1 of this report in the discussion of **RAI 813-5935, Question 03.02.01-22**.

Moreover, because the PCS was reclassified to QG C, it was unclear to the staff whether the PCS was designed using the ASME Code, Section III, Class 3 rules. The response to **RAI 887-6261, Question 09.01.04-23**, regarding applicable codes and standards for the PCS, stated that since the PCS is not a pressure-retaining component for the RCS, ASME Code, Section III does not apply to the seal itself. Moreover, based on the indirect relationship with the RV, this response also stated that the ASME Code, Section III is not applicable to the PCS, including the connecting weld between the seal and the seal ledge. The response to **RAI 887-6261, Question 09.01.04-23,** references DCD Tier 2, in Table 3.2-3, Note 1, which applies to components not designed to the ASME Code, and references DCD Tier 2, Table 3.2-2, Note 3, Item (5), which corresponds to codes and standards as defined in the design bases.

Since the revised classification of the PCS is Equipment Class 3 and QG C in the response to **RAI 724-5524, Question 03.02.02-20,** and the corresponding revision to DCD Tier 2, Table 3.2-2, the staff does not understand which codes and standards other than the ASME Code, Section III, Class 3 can be applied to the PCS and its connecting welds, in accordance with RG 1.26. Therefore, **RAI 724-5524 Question 03.02.02-20,** was closed as unresolved, and additional information was requested in follow-up **RAI 914-6365, Question 03.02.02-21,** in which the staff requested the applicant to:

(a) Clarify this apparent discrepancy between the Equipment Class 3 and the associated ASME Code, Section III, Class 3 in DCD Tier 2, Table 3.2-3, the QG C, and the use of other codes and standards than ASME Code, Section III, Class 3 for the PCS design.
(b) Identify which codes and standards are being used for the design and construction (e.g., welding) of this component.
(c) Discuss the need to include in Tier 1 the PCS construction code and to provide an ITAAC for its verification.

In its response to **RAI 914-6365, Question 03.02.02-21**, dated October 9, 2012, the applicant stated that as part of its response to **RAI 667-5235, Question 03.02.02-17**, DCD Tier 2, Table 3.2-2 was revised to clarify applicant QA classifications, and to provide consistency in application of RG 1.26 QG classifications. The PCS, which is a non-pressure retaining component, was revised in the response to **RAI 667-5235, Question 03.02.02-17,** to be QG N/A.

The stress limits of ASME Code, Section III, Subsection ND, are used for the design of the PCS. However, since ASME Code, Section III is intended to be applied to design of pressure-retaining components and their supports, for the reasons described in the response to **RAI 887-6261, Question 09.01.04-23**; ASME Code certification is not required. In addition, material selection, fabrication and examination of the PCS are in accordance with requirements of ASME Code Section II, Section IX and Section V. The applicant proposed to revise DCD Tier 2, Subsection 9.1.4.2.1.13 to add a description of the codes applied to the PCS.

The applicant proposed to revise DCD Tier 1 Section 2.7.6.4, “Light Load Handling System,” to add the PCS as a safety-related, seismic Category I component. Being seismic Category I, the

PCS is subject to the ITAAC 2.a in DCD Tier 1, Table 2.7.6.4-2, "Light Load Handling System Inspections, Tests, Analyses, and Acceptance Criteria," which will verify functionality of the PCS under seismic design-basis loads. Given that ASME Code certification is not required, no additional description about the PCS construction code and corresponding ITAAC for its verification are needed.

The staff finds applicant response acceptable as the applicant has clarified all of the staff concerns, including clarifying the QG classification and the codes and standards applied to the PCS, and has committed to revise the DCD accordingly. Accordingly, **RAI 914-6365, Question 03.02.02-21, is being tracked as a Confirmatory Item** pending revision of the DCD.

The response to **RAI 914-6365, Question 03.02.02-21,** clarified that the PCS is a not a pressure retaining component, and as a result QG and the ASME Code, Section III do not apply, but the design is to ASME Code, Section III ND with no Code certification or ITAAC required. The response to **RAI 724-5524, Question 03.02.02-20** further defined the PCS as Equipment Class 3, QG N/A, QA Class "Q" and seismic Category I with codes and standards as defined in the design bases. DCD Tier 2, Table 3.2-3 shows Equipment Class 3 as QG C. Although the staff concurs that a QG is not required for a component that is not pressure retaining or supporting a pressure-retaining component, the ASME Code, Section III does include attachments, and the boundary for attachments is to be defined in ASME design specifications. Further, there should be an ITAAC or other verification method to ensure the correct classification and integrity of the attachment weld. Therefore, in **RAI 976-6934, Question 03.02.02-24**, the staff requested the applicant to:

(a) Clarify the ASME Code Class jurisdictional boundary for the attachment to the ASME. Section III Class 1 RPV.
(b) Confirm that an ITAAC or other verification applies to the attachment weld to the RPV.
(c) Clarify in DCD Tier 2, Table 3.2-2 and Table 3.2-3 that the design of the Equipment Class 3 PCS is to ASME Code, Section III Class 3 without ASME Code certification (with reference to DCD Tier 2 Section 9.1.4.2.1.13).

In its response to **RAI 976-6934, Question 03.02.02-24**, dated January 15, 2013, the applicant stated that the details of the PCS and its attachment to the Seal Ledge and RV were discussed in the applicant's response to **RAI 887-6261, Question 09.01.04-23**. The applicant repeated the discussion from the response.

> The detail configuration and materials of the Permanent Cavity Seal (PCS) around the Seal Ledge (SL) are described below. The SL is a ring type configuration attached to outside of the RV and is made of stainless steel. The SL is supported by the RV [Reactor Vessel] at the bottom, welded to the PCS and the RV flange at the top, and bolted to the PCS. The welding at the top of the SL are continuous welds and used for sealing the cavity water.
>
> There are partial penetration continuous welds and bolts connecting the PCS, the SL, and the RV. The first connecting weld (between the SL and the RV) and the second connecting weld (between the PCS and the SL) are both used for sealing but not for structural support. The SL is supported by the RV at the bottom, and is designed so that the weld is within the cladding of the RV, which does not affect the structural integrity of the RV. The PCS is bolted and seal welded to the SL since it can only be installed at site. The weld between the SL and the RV is to the RV

cladding and is considered as a non-ASME weld because the weld is the first connecting weld of a welded non-pressure retaining, non-structural attachment to the RV. The welding will be performed in accordance with ASME NB-4430. (Reference ASME III, NB-1132.2 (c) and (e)).

(a) Both the PCS and SL are non-pressure retaining and non-structural items. The PCS is not attached to the RV but to the SL which, in turn, is attached to the cladding of the RV. The SL to RV connecting weld is considered as a non-ASME weld because the SL is a non-pressure retaining and non-structural attachment. The boundaries of jurisdiction are defined in ASME Code, Section III, NB-1132.2, "Jurisdictional Boundary." In accordance with NB-1132.2(e), stating that "The first connecting weld of a welded non-structural attachment to a component shall be considered part of the attachment. At or within 2t from the pressure-retaining portion of the component, the first connecting weld shall conform to NB-4430," the SL and its connecting weld to the RV is not a part of the vessel, but treated as an attachment to the vessel. Therefore, in accordance with NB-1 132.2(e) the SL to RV connecting weld is considered part of the attachment (SL) and the weld will conform to ASME NB-4430. (Please refer to the figure at the end of the response).

(b) The SL to RV weld is part of the SL but is within a distance of 2*t* of the RV [where *t* is the nominal thickness of the RV]. Hence, the rules of ASME Code, Section III will be applied to the SL to RV connecting weld in order to verify its fabrication, installation and inspection, and as stated above, will conform to the requirements of NB-4430. The design and installation specifications and procedures will ensure that the requirements of NB-4435(a) are complied with; including (1) The welding procedure and welders have been qualified in accordance with NB-4321; (2) The material is identified and is compatible with the material to which it is attached; (3) The welding material is identified and compatible with the material joined; and (4) the welds are post-weld heat treated when required by NB-4620. Per SRP 14.3.3, Acceptance Criteria 2.B these non-pressure retaining, non-structural welds, no ITAAC are required. Instead, they are to be included in Tier 2. DCD Subsection 9.1.4.2.1.13 identifies the codes to be used for the material selection, fabrication and examination of the PCS and SL.

(c) Refer to response to RAI 978-6931 for a discussion of design-basis codes and standards including Equipment Classes 2 and 3 non-ASME Code, Section III components. For the PCS, reference is made to DCD Subsection 9.1.4.2.1.13.

A sketch of the PCS/seal ledge/RV showing the ASME Code, Section III boundary of jurisdiction is shown in the RAI response.

The staff finds applicant response acceptable as the applicant has clarified all of the staff concerns, including the jurisdictional boundary, ITAAC, and QG classification of the PCS. Accordingly, **RAI 976-6934, Question 03.02.02-24, is resolved**.

3.2.2.4.10 Compliance with Regulations

GDC 1 states in part that SSCs important to safety shall be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed. DCD Tier 2, Subsection 3.1.1.1.1, "Discussion," uses the term "safety-related" and DCD Tier 2, Section 3.2.2, uses both terms, "important to safety" and "safety-related" to identify the SSCs that must be designed to satisfy the requirements of GDC 1. In **RAI 276-2043, Question 03.02.02-7**, the staff requested the applicant to clarify the application of the terms "safety-related" and "important to safety" to the QG classification of SSCs and compliance with GDC 1. The staff referred the applicant to definitions in: 1) 10 CFR Part 50, 2) memo to NRC Staff, "Standard Definitions for Commonly-Used Safety Classification Terms," and 3) ANS 58.14 regarding application of these terms. The staff also requested the applicant to clarify to what extent those SSCs that are important to safety that are not considered safety-related are classified so that they are designed to appropriate quality standards.

In its response to **RAI 276-2043, Question 03.02.02-7**, dated May 8, 2010, the applicant referred to its response to **RAI 287-2041, Question 03.02.01-12**, dated May 21, 2009, in regard to the terms safety-related and important to safety and states that additional quality standards are applied to nonsafety-related SSCs commensurate with the functions of the SSC and contribution to safety. The response clarifies that QAP controls will also be applied to nonsafety-related SSCs designated as Equipment Class 4 or 5. The response further explains that safety-related and certain nonsafety-related, but important to safety SSCs, such as fire protection and waste management systems, have been assigned a QG classification along with the QA requirements in accordance with GDC 1. Based on this response, that additional quality controls will be applied to certain nonsafety-related SSCs designated Equipment Class 4 or 5, the applicant recognizes that certain nonsafety-related SSCs important to safety require augmented quality controls and the staff finds this aspect of the response to **RAI 276-2043, Question 03.02.02-7** acceptable. However the response to **RAI 276-2043, Question 03.02.02-7,** does not adequately address to what extent those SSCs that are risk-significant but are not considered safety-related are classified so that they are designed to appropriate quality standards. Therefore, additional clarification is needed regarding the application of the term important to safety and special treatment for risk-significant SSCs. As a result, **RAI 276-2043, Question 03.02.02-7,** was closed as unresolved.

In follow-up **RAI 580-4584, Question 03.02.02-14**, the staff requested the applicant to provide clarification of how risk-significant, nonsafety-related SSCs that are important to safety are identified and classified with respect to the determination of appropriate quality standards and any special treatment. The staff also requested the applicant to replace the term "safety-related" with the term "important to safety" in DCD Tier 2, Sections 3.2.2 and 3.1.1.1, "Criterion 1 – Quality Standards and Records," in order to be consistent with GDC 1.

In its response to **RAI 580-4584, Question 03.02.02-14**, dated July 22, 2010, the applicant stated that based on the PRA process described in DCD Tier 2, Section 17.4, "Reliability Assurance Program," the risk-significant, nonsafety-related SSCs that are listed in DCD Tier 2, Table 17.4-1 for Seismic Event Rational column will be categorized as seismic Category II and classified as Equipment Class 5 and references the applicant's response to **RAI 580-4584, Question 03.02.02-10,** concerning changes made to the DCD.

Although the applicant's response to **RAI 580-4584, Question 03.02.02-14,** appropriately references the PRA process to identify risk-significant SSCs and PRA for consideration of seismic events, the staff is concerned that the response does not address changing the term "safety-related" to "important to safety" in DCD Tier 2, Subsection 3.1.1.1 and has not clearly explained the process to apply the terms "important to safety" and "safety-related" to the classification process to satisfy GDC 1. The applicant's process is still unclear regarding application of the terms safety-related and important to safety to satisfy GDC 1. As a result, **RAI 580-4584, Question 03.02.02-14,** was closed as unresolved.

In follow-up **RAI 667-5235, Question 03.02.02-17**, the staff requested the applicant to replace the term "safety-related" with the term "important to safety" in DCD Tier 2, Section 3.1.1.1 and the applicant should clearly explain the process to apply the terms "important to safety" and "safety-related" to the classification process to satisfy GDC 1. Other than for seismic Category II SSCs, no QA requirements or any special treatment requirements are identified in DCD Tier 2, Tables 3.2-2 and 3.2-3 for Equipment Class 4 and 5 components that may be risk-significant. The applicant may alternatively choose to reference another section of the DCD or a separate topical report to describe this process in detail.

In its response to **RAI 667-5235, Question 03.02.02-17**, dated July 27, 2012, the applicant provided detail discussions regarding compliance with GDC 1 and identification of risk significant SSCs in the US-APWR to explain how it identified and treated the important-to-safety SSCs.

The applicant revised the DCD Tier 2, Subsection 3.1.1.1.1 to more clearly explain the process of applying QA requirements to SSCs including the application of QA requirements to nonsafety-related SSCs. As part of this change, the term "safety-related" will be replaced with the term "important to safety." The applicant also revised the DCD Tier 2, Table 3.2-2 to identify the applicable scope of QA controls applied to each specific SSC by the "Quality Assurance Classification," supplemented by use of "Equipment Classifications" 1 through 10 and explanatory notes. The "Quality Assurance Classification" identifies SSCs that meet the full 10 CFR Part 50, Appendix B, QA requirements (designated "Q"), SSCs that meet augmented QA requirements (designated "A"), and SSCs that do not need to meet either 10 CFR Part 50, Appendix B, or augmented quality requirements (designated "N"). Explanatory notes are provided for additional clarification and basis.

The applicant also revised DCD Tier 2, Section 3.2, "Classification of Structures, Systems, and Components," including DCD Tier 2, Table 3.2-2, as necessary to properly reflect the identification and treatment of both safety-related and nonsafety-related SSCs with regard to quality requirements. The applicant notes that markups to Equipment Classes 1 through 3 address **RAI 914-6365, Question 03.02.02-23,** but are included for completeness. The applicant also revised as necessary other sections of the DCD that use the term "important to safety" when referring to SSC classification, except DCD Tier 2, Section 3.11 and DCD Tier 2, Appendix 3D. The applicant stated that markups for Section 3.11 and Appendix 3D will be included in its response to **RAI 805-5915, Question 3.11-41**.

The staff evaluated this response and determined that, although the response addresses the issues, revises the DCD compliance with GCD 1 to include the term important to safety and submits a proposed revision to DCD Tier 2, Table 3.2-2, the response is not entirely adequate to address the classification process to identify important to safety systems and components and their special treatment to satisfy GDC 1. An important part of this process is to identify the basis for the classification in terms of the specific safety function performed by the individual component. If the DCD is changing, the detailed design may not be complete. Since the detailed design may not be complete and the response included a significant revision to DCD Tier 2, Table 3.2-2, staff is concerned that the QG classification of components and their basis may not yet be finalized or verified. As a result, **RAI 667-5235, Question 03.02.02-17,** was closed as unresolved and in follow-up **RAI 978-6931, Question 03.02.02-25,** the staff requested the applicant to establish the basis for the QG classifications in terms of specific safety functions and any special treatment. In its response to **RAI 978-6931, Question 03.02.02-25**, dated February 25, 2013, the applicant provided the basis for equipment and QG classification and has also described the process for verifying QG classification information with the design of the plant. The applicant's response is evaluated in Section 3.2.2.4.14 of this report.

3.2.2.4.11 Auditable Information

10 CFR 52.47 identifies that the Commission will require, prior to DC, that information normally contained in certain procurement specifications and construction and installation specifications be completed and available for audit. In **RAI 276-2043, Question 03.02.02-8**, the staff requested the applicant to clarify if the design-basis information on QG classification for all important to safety SSCs within the scope of the DCD, including structures, is included in specifications and if this information is available for audit. In its response to **RAI 276-2043, Question 03.02.02-8**, dated May 8, 2009, the applicant identified that this information will be available for audit and referenced Enclosure 3 to its letter, "Additional Information for Design Completion Plan of US-APWR Piping Systems and Components," dated July 14, 2008, for the detailed completion plan.

Other than design specifications and stress reports, the response did not identify any other design-basis information on QG classification. Therefore, **RAI 276-2043, Question 03.02.02-8,** was closed as unresolved and in follow-up **RAI 580-4584, Question 03.02.02-16**, the staff requested the applicant to clarify if any additional design-basis documents will be available for audit to establish the basis for the individual QG classifications to enable the classifications to be validated. For example, supporting design-basis documents that supplement the design specifications and can be used to verify the safety function for individual classifications may include documents such as a "Q-list," detailed P&IDs, system summary documents, a RTNSS requirements document, and procurement specifications.

In its response to **RAI 580-4584, Question 03.02.02-16**, dated July 22, 2010, the applicant agreed to make additional documents used to establish the basis for QG designation for individual SSCs available for audit by the NRC so that the classifications can be validated. The additional documents include the "Q-List" (or its equivalent), detailed P&IDs, system specifications, and equipment procurement specifications for those SSCs within the scope of the DCD. The staff finds the applicant's response acceptable as it clarifies that appropriate documents will be available for audit. Accordingly, **RAI 580-4584, Question 03.02.02-16, is resolved**. As discussed in Section 3.2.1.4.4 of this report, the applicant is planning on making the documents available for audit. To track the need for the applicant make the documents available for audit, the staff issued **RAI 1015-7054, Question 03.09.03-31.** Pending the staff completing and documenting the audit, **RAI 1015-7054, Question 03.09.03-31 is being tracked as an Open Item**.

3.2.2.4.12 Inspection, Test, Analysis, and Acceptance Criteria

ASME Code, Section III Class' is included in Tier 1 ITAAC design descriptions. DCD Tier 2, Section 14.3.4.4, "ITAAC for Reactor Systems," identifies that ITAAC are specified to verify ASME Code classifications. DCD Tier 2, Table 3.2-3 correlates ASME Code, Section III Class 1, 2 and 3 to NRC QG A, B and C, respectively. As a result, it is understood the as-built QG classifications A, B, and C will be verified for safety-related pressure retaining systems, components and their supports. Based on the changes to the QAP described in DCD Tier 2, Chapter 17, it is also understood that there are to be ITAAC for nonsafety-related risk-significant

SSCs included in D-RAP. However, it is not clear which specific ITAAC are to be used to verify as-built classification of ASME Code Class and nonsafety-related QG D systems and components that may be important to safety. In **RAI 580-4584, Question 03.02.02-15**, the staff requested the applicant to confirm that QG classifications for all important to safety SSCs, including those considered risk-significant that are not safety-related, will be verified and to identify the specific ITAAC used for these verifications.

In its response to **RAI 580-4584, Question 03.02.02-15**, dated July 22, 2010, the applicant clarified that DCD Tier 2, Section 14.3, 'Inspections, Tests, Analyses, and Acceptance Criteria," identifies the key design features, based on the guidance in SRP Section 14.3, issued March 2007. The response noted that SRP Section 14.3 does not identify seismic Category II as criteria for safety significant design features and does not require a generic seismic Category II ITAAC although some nonsafety-related SSCs were determined to require an ITAAC to verify the as-built design is seismic Category II based on the review of design documents or on a basic walk-down inspection. An example where a nonsafety-related SSC is verified as seismic Category II is identified in DCD Tier 1, Section 2.7.6.4, Table 2.7.6.4-2, ITAAC 2b.

Although the response identified examples of ITAAC for nonsafety-related SSCs, the RAI response did not address an ITAAC for verification of nonsafety-related risk-significant QG D systems that may be important to safety. The requirement for an ITAAC for a seismic systems interaction walkdown is to be considered in the review of DCD Tier 2, Sections 3.7, "Seismic Design," and 14.3.2, "Chapter 1 of Tier 1, Introduction." The staff believes that an ITAAC is appropriate for this verification of important to safety SSCs, including seismic Category II SSCs, that are not considered safety-related. Until it is demonstrated that an adequate ITAAC or other verification process exists, for all applicable nonsafety-related SSCs, this concern is unresolved. As a result, **RAI 580-4584, Question 03.02.02-15,** was closed as unresolved.

In follow-up **RAI 667-5235, Question 03.02.02-18**, the staff requested the applicant to establish that an adequate ITAAC or other verification process exists for all applicable nonsafety-related mechanical systems. In its response to **RAI 667-5235, Question 03.02.02-18**, dated July 27, 2012, to the applicant indicated that SRP Section 14.3.2, SRP Acceptance Criterion 6, provides guidance that ITAAC should verify that non-seismic Category I SSCs do not interact with seismic Category I SSCs in a way that could prevent the seismic Category I SSC from performing its design-basis safety function. DCD Tier 1 Table 2.2-4, "Structural and Systems Engineering Inspections, Tests, Analyses, and Acceptance Criteria," was revised to verify appropriate seismic design of non-seismic Category I SSCs as described in the response to **RAI 571-4365, Question 09.02.02-48**, dated July 29, 2011, and the markup identified by Change ID number MIC-03-T1-00001 in MUAP-11021, "US-APWR DCD Revision 3 Tracking Report," Revision 0. DCD Tier 1, Table 2.2-4, ITAAC 23a and 23b verify that non-seismic Category I SSCs do not interact with seismic Category I SSCs in a way that could prevent the seismic Category I SSCs from performing their design-basis safety function. The staff finds the response acceptable as the applicant has identified the applicable ITAAC for nonsafety-related mechanical systems and identified where they were updated through other actions. Since the applicant has provided related DCD markups, **RAI 667-5235, Question 03.02.02-18, is being tracked as Confirmatory Item** pending revision of the DCD in response to **RAI 571-4365, Question 09.02.02-48,** referred to above.

3.2.2.4.13 Non-pressure-retaining Components and Their Supports

RG 1.26 is limited to pressure-retaining components and their supports, so the basis for assigning QG classifications and appropriate codes and standards to items that are not pressure-retaining components or their supports is not clear. For example, QG B is assigned to fuel assemblies, QG D is assigned to reactor internal structures and QG C is assigned to the new fuel storage rack and certain other non-pressure-retaining structures in DCD Tier 2, Table 3.2-2. In **RAI 914-6365, Question 03.02.02-23**, the staff requested the applicant to clarify the basis for QG classifications and appropriate codes and standards for systems and components that are not pressure-retaining or their supports and revise DCD Tier 2, Table 3.2-2 as appropriate. In its response to **RAI 914-6365, Question 03.02.02-23**, dated October 9, 2012, the applicant stated that as part of its response to **RAI 667-5235, 03.02.02-17,** DCD Tier 2, Tables 3.2-2 and 3.2-3 were revised to clarify Equipment Classifications and QA Classifications, and to provide consistency in application of RG 1.26 QG classifications. DCD Tier 2 Tables 3.2-2 and 3.2-3 are to state that RG 1.26 QGs A, B and C are applicable to Equipment Class 3 and ASME Code, Section III PSCs only. The RG 1.26 QG classifications for non-pressure-retaining SSCs outside the scope of RG 1.26, such as fuel assemblies and new fuel storage racks, are specified as Not Applicable (N/A) in DCD Tier 2, Table 3.2-2. In addition, Equipment Classes 4 and 8 have been reclassified to have only the QG D PSCs listed in RG 1.26 Table 1. Thus, the discrepancy between the applicability of RG 1.26 QG classifications and DCD Tier 2, Table 3.2-2 have been corrected as shown in the DCD markups attached to the response to **RAI 667-5235, 03.02.02-17**. However, while the response to **RAI 914-6365, Question 03.02.02-23,** clarified inconsistencies in QG classifications, it did not fully clarify the basis of QG classifications. Therefore, **RAI 914-6365, Question 03.02.02-23,** was closed as unresolved and in follow-up **RAI 978-6931, Question 03.02.02-25**, discussed in the following section, the applicant was requested to clarify the basis for QG classifications and appropriate codes and standards for systems and components that are not pressure-retaining or their supports.

3.2.2.4.14 Design-Basis for Quality Group Classifications

In response to various RAIs, DCD Tier 2, Sections 3.1, "Conformance with NRC General Design Criteria," 3.2, and Table 3.2-2 have been significantly revised from DCD Revision 3 to address the classification methodology and to reflect the design process for risk-significant SSCs. For example, the response to **RAI 667-5235, Question 03.02.02-17,** included markups to DCD Tier 2, Sections 3.1 and 3.2 and Table 3.2-2 showing numerous changes resulting from **RAI 667-5235, Question 03.02.02-17,** and additional RAIs. As a result of these changes, the staff needed further information to determine that the design-basis for QG classifications and special treatment has been finalized and verified. Therefore, in **RAI 978-6931, Question 03.02.02-25**, the staff requested the applicant to provide the basis for QG classification of each component listed in DCD Tier 2, Table 3.2-2, including the specific safety function that each component performs and any special treatment. Safety function of each component can be defined in terms of reactor coolant pressure boundary, containment boundary, specific ECCS function, minimizes radiation release, etc.; special treatment includes any codes and standards defined in the design-basis (see Note 3 Item (5) of DCD Tier 2, Table 3.2-2). In addition, the staff requested the applicant to describe the process for verifying design-basis for QG classifications with the design of the plant. If design-basis for QG classifications is not available, the staff requested the applicant to clarify when this information will be available.

In its response to **RAI 978-6931, Question 03.02.02-25**, dated February 1, 2013, the applicant provided the basis for equipment and QG classification and also described the process for verifying QG classification information with the design of the plant as follows:

1) Basis for Equipment and Quality Group Classification:

The applicant stated that Equipment Classes 1 thru 3 consist of ASME Code, Section III PSC and non-ASME Code, Section III PSC as described in DCD Tier 2, Sections 3.2.2.1, “Equipment Class 1,” through 3.2.2.3. Both types of components (ASME and non-ASME) are classified as safety-related and, therefore, controlled by Appendix B to 10 CFR Part 50. The QG classification and special treatment basis, including applicable code and standards, for ASME Code, Section III PSC conforms to 10 CFR 50.55a requirements and RG 1.26 QGs B and C as described in DCD Tier 2, Section 3.2.2 and Table 3.2-2. For example, the components with reactor coolant pressure boundary function are classified as Equipment Class 1 and meet requirements of QG A and ASME Code, Section III Class 1 as required by 10 CFR 50.55a.

The applicant stated that Table 1 of the response provides a comprehensive list of Equipment Class 2 and 3 non-ASME Code, Section III components where the design-basis code and standard is described in the applicable DCD design bases section [i.e., Note 3 Item (5) of DCD Tier 2, Table 3.2-2 as provided in the DCD revisions attached to the response to **RAI 667-5235, Question 03.02.02-17**]. Table 1 identifies the equipment classification basis (e.g., safety function) as described in DCD Tier 2, Sections 3.2.2.1 through 3.2.2.3 and special treatment of these components, such as a required quality control and DCD Tier 2 location in which the applicable design-basis codes and standards are addressed.

The applicant stated that Table 1 of the response also lists the Equipment Class 5 components from DCD Tier 2, Table 3.2-2, identifies the specific basis for the Equipment Class 5 designation and identifies the locations in the DCD where the applicable codes and standards are described. The applicant referred to the response to **RAI 914-6365, 03.02.02-22**, which states that the design-basis codes and standards are included in corresponding DCD chapters with proposed DCD changes attached to the response.

The applicant stated that for Equipment Classes 4, 6, 7 and 8, the specific basis for the Class designation and the applicable codes and standards are indicated in DCD Tier 2, Table 3.2-2 that was proposed with the **RAI 667-5235, Question 03.02.02-17** response.

The applicant stated that Equipment Classes 9 and 10 are nonsafety-related components that do not require any special treatment.

2) Process for Verifying QG Classification Information with the Design:

The applicant stated that QG classification of ASME Code, Section III PSC is generally determined in accordance with the following processes:

i) Identification of System Function and Quality Group Classification

System functions and design requirements have been identified and documented in a design document (e.g., system design package) and QG classification has been determined based on the design document and SSCs classification principle document whose contents are consistent with DCD Tier 2, Section 3.2.2. This classification provides the initial basis for the quality classification of the system or component. Note that the level of detail is similar to DCD Tier 2, Table 3.2-2.

ii) SSCs Data Sheet Development:

Based on the initial quality classification results and associated design document above, SSCs Data Sheets have been developed to provide the sufficient information to prepare the purchase specification and design specification. The data sheets typically include the description about safety functions that have to be achieved by the PSC. The data sheet is also documented as part of QG Classification basis for the PSC.

iii) Design Specification

A design specification provides more detail QG classification and basis for the PSC, such as these for individual parts or components of the PSC, using the data sheets and classification principle. The level of detail is determined by the nature and type of the PSC. The design specification will be used to develop the purchase specification. Thus, the design specification is the primary basis for the QG classification and some design documentation (e.g., data sheet and the system design package, as applicable) will support the determination basis for the QG classification.

The applicant stated that the QG classification basis for the risk-significant ASME Code, Section III PSCs will be available with the design specification and ready for the audit as discussed above.

The applicant stated that the QG classification of non-ASME Code, Section III components follows a similar design process and the design specification will be available per the engineering progress and procurement timing. The design/purchase specification for non-ASME Code, Section III components will be available prior to the procurement phase and will not be available during the DC review.

The staff finds the response acceptable since the applicant has provided the basis for the equipment and QG classifications and the applicant described the process for verifying QG classification information with the design of the plant. As discussed above, the response to **RAI 580-4584, Question 03.02.02-16**, states that the applicant agrees to make additional documents used to establish the basis for QG designations for individual SSCs available for audit by the NRC so that the classifications can be validated. Since the applicant has proposed DCD changes, **RAI 978-6931, Question 03.02.02-25, is being tracked as a Confirmatory Item** pending revision of the DCD.

3.2.2.5 Combined License Information Items

The following is a list of COL item numbers and descriptions from Table 1.8-2 of the DCD related to system quality group classification.

Table 3.2.2-1 US-APWR Combined License Information Items		
Item No.	**Description**	**Section**
3.2(5)	The equipment class and seismic category of the site-specific safety-related and nonsafety-related fluid systems, components (including pressure retaining), and equipment as well as the applicable industry codes and standards are provided in Table 3.2-201.	3.2.2
3.2(6)	The COL Applicant is to apply DCD methods of equipment classification and seismic categorization of risk-significant, nonsafety-related SSCs based on their safety role assumed in the PRA and treatment by the D-RAP.	3.2.2.5

The staff determined the above listing to be complete. Also, the list adequately describes actions necessary for the COL applicant or licensee. No additional COL information items need to be included in DCD Tier 2, Table 1.8-2 or Section 3.2.3 for QG classification consideration.

3.2.2.6 Conclusions

As a result of the open item for **RAI 1015-7054, Question 03.09.03-31**, the staff is unable to finalize its conclusions on Section 3.2.2 related to system quality group classification, in accordance with NRC regulations.

3.3 Wind and Tornado Loadings

3.3.1 Wind Loadings

3.3.1.1 Introduction

This section discusses the design of structures that must withstand the effects of the plant's design wind speed.

3.3.1.2 Summary of Application

DCD Tier 1: The Tier 1 information associated with this section is found in DCD Tier 1, Section 2.2, "Structural and Systems Engineering"; however, there are no ITAAC specifically related to wind loadings.

DCD Tier 2: The applicant has provided a DCD Tier 2 system description in Section 3.3.1, "Wind Loadings," summarized here in part, as follows: This section discusses the design of structures that must withstand the effects of the plant's design-basis wind loading.

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

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

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

Technical Reports: There are no technical reports for this area of review.

Cross-cutting Requirements (TMI, USI/ GSI, Op Ex): There are no cross-cutting requirements for this area of review.

US-APWR Interface Issues Identified in the DCD: See DCD Tier 2, Table 1.8-2, "Compilation of All Combined License Applicant Items for Chapters 1-19."

Site Interface Issues Identified in the DCD: See DCD Tier 2, Table 1.8-1, "Significant Site Specific Interfaces with the Standard US-APWR Design."

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

3.3.1.3 Regulatory Basis

The relevant requirements of the Commission's regulations for this area of review, and the associated acceptance criteria are given in Section 3.3.1, "Wind Loadings," Revision 3, issued March 2007, of NUREG-0800 and are summarized below. Review interfaces with other SRP sections can be found in Section 3.3.1 of NUREG-0800.

1. GDC 2, as it relates to the ability of SSCs without loss of capability to perform their safety function, to withstand the effects of natural phenomena, such as earthquakes, tornadoes, floods, and the appropriate combination of all loads.

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 DC has been constructed and will operate in accordance with the DC, the Atomic Energy Act of 1954, and the NRC's regulations.

Acceptance criteria adequate to meet the above requirements include:

1. The wind used in the design shall be the most severe wind that has been historically reported for the site and surrounding area with sufficient margin for the limited accuracy, quantity, and period of time in which historical data have been accumulated.

2. The acceptance criteria for the design wind speed, its recurrence interval, the speed variation with height, the applicable gust factors, and the bases for determining these site-related parameters, are stated in SRP Sections 2.3.1, "Regional Climatology," and 2.3.2, "Local Meteorology." The approved values of these parameters should serve as basic input to the review and evaluation of the structural design procedures.

3. The procedures used to transform the wind speed into an equivalent pressure to be applied to structures and parts, or portions of structures, as delineated in American Society of Civil Engineers/Structural Engineering Institute (ASCE/SEI) 7-05, "Minimum Design Loads for Buildings and Other Structures," are acceptable. In particular, the procedures used are acceptable if found in accordance with the following:

a. For a design wind speed, V, the velocity pressure, q_Z, evaluated at height, z, is given by:

$q_Z = 0.00256\ K_Z K_{dt} K_d V^2 I$ (lb/ft^2), where:
K_Z= velocity pressure exposure coefficient evaluated at height, z, as defined in ASCE/SEI 7-05, Table 6-3, but not less than 0.87
K_{dt} = topographic factor equal to 1.0
K_d = wind directionality factor equal to 1.0
V = design wind speed in miles per hour (mi/h) as stated in SRP Section 2.3.1, and
I = importance factor equal to 1.15

b. For each wind direction considered, the upwind exposure category should be based on ground surface roughness that is determined from natural topography, vegetation, and constructed facilities. Surface roughness C is defined as open terrain with scattered obstructions having heights generally less than 30 ft (9.1 m). This category includes flat open country, grasslands, and all water surfaces in hurricane prone regions. Because most nuclear power plants are located in relatively open country, Kz values in Table 6-3 should be selected from the Exposure C column. The definition of Exposure C is provided in ASCE/SEI 7-05, Section 6.5.6.3.

c. Design wind loads should be determined in accordance with the following sections in ASCE/SEI 7-05, as applicable.

   i. Section 6.5.12 - Design Wind Loads on Enclosed and Partially Enclosed Buildings.
   ii. Section 6.5.13 - Design Wind Loads on Open Buildings with Monoslope, Pitched, or Troughed Roofs.
   iii. Section 6.5.14 - Design Wind Loads on Solid Freestanding Walls and Signs.
   iv. Section 6.5.15 - Design Wind Loads on Other Structures.

3.3.1.4 Technical Evaluation

Structural design criteria for wind loading effects on seismic Category I buildings and structures described in DCD Tier 2, Revision 3 for the US-APWR were evaluated for compliance with GDC 2. Review guidance and acceptance criteria provided in SRP Section 3.3.1 were used to perform the technical evaluation. Specific areas under review included:

- validation of input parameters for the structural design criteria appropriate to account for wind loadings,

- verification that procedures for transforming the design wind speed into an equivalent pressure applied to buildings and structures and for distributing the wind speed on the buildings and structures are in accordance with wind load standards in ASCE/SEI 7-05,

- confirmation that all seismic Category I buildings and structures in the US-APWR standard design that are subject to wind loading are identified and appropriately addressed in conformance with SRP Section 14.3, “Inspections, Tests, Analyses, and Acceptance Criteria” issued March 2007.

3.3.1.4.1 Input Parameters

The 155 miles per hour (mph) (69.3 meters/second (m/s)) basic wind speed corresponding to a three-second gust at 33 ft (10 m) above ground in open terrain (ASCE/SEI 7-05, Exposure Category C) selected by the applicant as the design-basis for wind loading represents meteorological conditions that exceed the local basic wind speed at almost all locations situated in the contiguous U.S. Review of basic wind speed data presented in ASCE/SEI 7-05, Figure 6-1, which represent nominal design three-second gust speeds at 33 ft (10 m) above ground for Exposure Category C (open terrain), suggests that the maximum basic wind speed at all locations except mountainous terrain, gorges, ocean promontories, and special wind regions does not exceed 150 mph (67.1 m/s). Application of the importance factor, 1.15, adjusts the velocity pressure to an annual probability of being exceeded to 0.01 (mean reoccurrence interval of 100 years) which is consistent with guidance provided in SRP Section 2.3.1, “Regional Climatology,” Revision 3, issued March 2007.

Use of the 155 mph (69.3 m/s) basic wind speed and an importance factor of 1.15 for the seismic Category 1 buildings and structures in the US-APWR standard design will very likely envelope local basic wind speeds occurring at most, if not all, candidate sites for a US-APWR nuclear power plant located in the contiguous U.S. Per COL Information Item 3.3(1), the COL applicant is responsible for verifying the site-specific basic wind speed is enveloped by the determinations in DCD Tier 2, Section 3.3.1. In the event that the site-specific basic wind speed does exceed, a site-specific design of the seismic Category I buildings and structures must be performed. The staff assessed COL Information Item 3.3(1) and determined it to be acceptable and necessary since it assures the standard design of the seismic Category I buildings and structures can withstand the wind loads at the site.

3.3.1.4.2 Wind Speed to Wind Load Conversion

Based on technical rationale in SRP Section 3.3.1, the staff accepts the industry standard ASCE/SEI 7-05 for evaluating wind loading on structures. The methods in ASCE/SEI 7-05 include procedures for transforming wind speed into an equivalent pressure on structures, selecting pressure coefficients corresponding to the structure’s geometry and physical configuration, accounting for wind speed variation with height and direction, and applying applicable gust factors.

The staff evaluated the applicant’s application of the two methods for converting basic wind speed into design pressure loads described in ASCE/SEI 7-05. Method 1 is the simplified procedure. It can be used to determine design wind loads provided the main wind-force resisting systems (MWFRS) for the building meet all eight conditions specified in ASCE/SEI 7-05 Section 6.4.1.1. The four-step design procedure for Method 1 is presented in ASCE/SEI 7-05 Section 6.4.2. Method 2 is the analytical procedure. It can be used to determine design wind loads provided the building meets the two conditions specified in ASCE/SEI 7-05 Section 6.5.1. The ten-step design procedure for Method 2 is presented in ASCE/SEI 7-05 Section 6.5.3.

The applicant used Method 1 to analyze the main wind-force resisting systems for the power source buildings (PS/Bs). In applying this method, the applicant determined the effective wind pressure, p_s, using the following equation:

$p_s = 1.15\ \lambda\ p_{basic}$ Equation 3.3.1-1

where

1.15 = importance factor, I, for all seismic Category I buildings and structures

λ = adjustment factor for Exposure Category C from ASCE/SEI 7-05, Figure 6-2

p_{basic} = wind pressure value in pounds per square foot (psf) from ASCE/SEI 7-05, Figure 6-2 corresponding to a basic wind speed of 155 mph (69.3 m/s)

The applicant used Method 2 to design the main wind-force resisting systems for the PCCV and the reactor building (R/B). In applying this method, the applicant determined the design pressure, p, for the PCCV and the R/B using the following equation.

$p = 0.00256\ K_z\ V^2\ 1.15\ (GC_p\ +/-\ GC_{pi})$ Equation 3.3.1-2

where

p = effective wind velocity pressure, psf

K_z = velocity pressure coefficient varying with height, taken from Table 6-3 of ASCE/SEI 7-05 for Exposure Category C; however, not less than 0.87 as recommended by SRP Section 3.3.1.

V = basic wind speed of 155 mph (69.3 m/s) corresponding to a three-second gust at 33 ft (10 m) above ground in open terrain (ASCE/SEI 7-05, Exposure Category C)

G = gust effect factor = 0.85 or as determined per ASCE/SEI 7-05, Subsection 6.5.8 (where a combined gust effect and pressure coefficient factor is used from a figure(s) in ASCE/SEI 7-05, an individual gust effect factor is not applied)

C_p = external pressure coefficient from ASCE/SEI 7-05 Subsection 6.5.11

C_{pi} = internal pressure coefficient from ASCE/SEI 7-05 Subsection 6.5.11

The staff's assessment to determine the acceptability of these two methods as applicable to the PS/Bs for Method 1as well as the PCCV and R/B for Method 2 respectively is documented below:

3.3.1.4.2.1 ASCE/SEI 7-05 Method 1 – Simplified Procedure

According to requirements in ASCE/SEI 7-05, Section 6.4, Method 1 can only be used for the design of MWFRS that satisfy eight conditions. These eight conditions are evaluated by the staff in Table 3.3.1-2 of this SE. The first column of Table 3.3.1-2 lists the eight conditions. The second column of Table 3.3.1-2 describes the basis for determining whether or not the PS/B design satisfies the conditions in the first column, and the third column states whether or not the conditions are satisfied.

The staff reviewed information about the PS/Bs presented in DCD Tier 2, Revision 1 for the US-APWR standard design and determined that among all eight conditions in ASCE/SEI 7-05, Section 6.4.1.1, Conditions 1, 2, 4, and 7 are satisfied, but additional information was required to determine if Conditions 3, 5, 6, and 8 are satisfied.

Compliance with Condition 3 could not be evaluated until the applicant provided additional information about the size and construction of the large openings in the exterior walls of the PS/Bs. In **RAI 215-1906, Question 03.03.01-1**, the staff requested the applicant to provide specific additional information concerning the large openings in the PS/Bs. In its response to **RAI 215-1906, Question 03.03.01-1**, dated April 9, 2009, the applicant stated that the PS/Bs are not partially enclosed, nor are their exterior walls 80 percent open. Therefore, the PS/Bs are classified as enclosed buildings as defined in ASCE/SEI 7-05 Section 6.2. Furthermore, the PS/Bs do not utilize glazing; thus, the wind-borne provisions of ASCE/SEI 7-05 Section 6.5.9.3 do not apply. The staff finds the response acceptable since the applicant clarified the basis for classifying the PS/Bs as enclosed. The staff agrees with this classification so that Condition 3 of ASCE/SEI 7-05 Section 6.4.1.1 is satisfied. Accordingly, **RAI 215-1906, Question 03.03.01-1, is resolved**. Per COL Information item, 3.3(5), the COL applicant is to note the vented and unvented requirements in DCD Tier Section 3.3.2 to the site-specific Category I buildings. The staff finds COL Information Item, 3.3(5), to be necessary and acceptable since in assures compliance with ASCE/SEI 7-05, Section 6.4.1.1, Condition 3.

Compliance with Condition 5 could not be evaluated until the applicant provided additional information about the flexibility of the PS/Bs. In **RAI 215-1906, Question 03.03.01-2**, the staff requested the applicant to provide specific additional information concerning the natural frequencies of the PS/Bs. In its response to **RAI 215-1906, Question 03.03.01-2**, dated April 9, 2009, the applicant provided the summary of natural frequencies for PS/B on soft soil from the seismic analysis in Table 2.2-2(1) of MUAP-08002, “Enhanced Information for PS/B Design,” Revision 0, issued February 2008, which shows the lowest fundamental natural frequency is 3.03 Hz for Mode 1 in the north-south direction. The staff finds the response acceptable since the applicant provided natural frequency information on the PS/B. Based on this information, the staff agrees that the PS/Bs are classified as rigid buildings as defined in ASCE/SEI 7-05 Section 6.2 and Condition 5 of ASCE/SEI 7-05 Section 6.4.1.1 is satisfied. Accordingly, **RAI 215-1906, Question 03.03.01-2, is resolved**; and there is no impact on the DCD.

Compliance with Condition 6 has a site-specific component regarding channeling effects or buffeting in the wake of upwind obstructions that may warrant special consideration by a COL applicant. In **RAI 817-5990, Question 03.03.02-5,** the staff requested the applicant to add a COL item to require a COL applicant to verify that the PCCV, R/B, and PS/Bs do not have site locations for which channeling effects or buffeting in the wake of upwind obstructions warrant special considerations.

In its response to **RAI 817-5990, Question 03.03.02-5**, dated September 26, 2011, the applicant stated that COL Information Item 3.3(4) would be revised to add the requirement requested by the staff and included an associated DCD markup. The staff finds the response acceptable since COL Information Item 3.3(4) now includes a requirement that addresses ASCE/SEI 7-05, Section 6.4, Method 1, Condition 6. Accordingly, **RAI 817-5990, Question 03.03.02-5, is being tracked as a Confirmatory Item.**

Compliance with Condition 8 could not be evaluated until the applicant provided additional information about the torsional response characteristics of the PS/Bs. In **RAI 215-1906,**

Question 03.03.01-3, the staff requested the applicant to provide specific additional information concerning the evaluation of Condition 8.

In its response to **RAI 215-1906, Question 03.03.01-3**, dated September 9, 2009, the applicant describe calculations that shows the layout of the PS/Bs as having the exterior continuous shear walls for the MWFRS. These calculations show that the PS/B meets the requirements depicted in ASCE/SEI 7-05 Section C6.4 regarding a structure being torsionally insensitive so that Method 1, Condition 8 in ASCE/SEI 7-05 Section 6.4.1.1 is considered satisfied. The staff finds the response acceptable since the applicant clarified the PS/B torsional response characteristics. Accordingly, **RAI 215-1906, Question 03.03.01-3, is resolved.**

Because the basic formula for effective wind velocity pressure used by the applicant was not consistent with the requirements presented in ASCE/SEI 7-05, Section 6.4.2.1 for Method 1, the staff requested the applicant to provide revised text for describing Method 1. In **RAI 215-1906, Question 03.03.01-4**, the staff requested the applicant to provide specific additional information concerning inconsistencies in the basic formula for effective wind velocity pressure. In its response to **RAI 215-1906, Question 03.03.01-4**, dated April 9, 2009, the applicant agreed to correct "wind velocity" to "wind velocity pressure" in the first sentence of the second paragraph in the DCD Tier 2, Sec. 3.3.1.2. The staff finds the response acceptable because the corrected terminology makes it consistent with the requirements of ASCE 7-05. The staff confirmed that the proposed changes have been incorporated into DCD Revision 3. Accordingly, **RAI 215-1906, Question 03.03.01-4, is resolved.**

In summary, the staff reviewed those responses to **RAI 215-1906, Question 03.03.01-1** thru **RAI 215-1906, Question 03.03.01-4,** contained in the applicant's letter dated April 9, 2009, and found they were acceptable. Table 3.3.1-2 of this report summarizes the evaluation of Method 1 conditions for the PS/Bs for all eight conditions defined in ASCE/SEI 7-05, Section 6.4.1.1, including the resolutions of **RAI 215-1906, Question 03.03.01-1** thru **RAI 215-1906, Question 03.03.01-4**.

Because all eight conditions in ASCE/SEI 7-05, Section 6.4.1.1 are satisfied, the applicant is justified in using the four-step design procedure for Method 1 to analyze the PS/Bs. Details of the design procedure for Method 1 are provided in ASCE/SEI 7-05, Section 6.4. In this method, the simplified design wind pressure, p_s, is determined by the following equation:

$$p_s = \lambda\ K_{zt}\ I\ p_{s30} \qquad \text{Equation 3.3.1-3}$$

where

λ = adjustment factor for building height and exposure from ASCE/SEI 7-05, Figure 6-2

K_{zt} = topographic factor as defined in ASCE/SEI 7-05, Section 6.5.7 evaluated at a mean roof height, h

I = importance factor

p_{s30} = simplified wind pressure for Exposure Category B, at h = 30 ft (9.1 m), and for I = 1.0 from ASCE/SEI 7-05, Figure 6-2

Because Equation 3.3.1-1 and Equation 3.3.1-3 of this report are not identical, the equivalency between the two equations is not readily apparent. Therefore, the following analysis was conducted by the staff to verify that the procedure used by the applicant for transforming the basic wind speed into an equivalent pressure applied to buildings and structures yields the same results as Method 1 described in ASCE/SEI 7-05, Section 6.4. This analysis is based on the following input parameters:

- The basic wind speed is 155 mph (69.3 m/s) corresponding to a three-second gust at 33 ft (10 m) above ground in open terrain (ASCE/SEI 7-05, Exposure Category C) for both equations.
- Exposure Category C applies to both equations.
- Topographic factor, K_{zt}, equals 1.0 for flat terrain (Exposure Category C) as described in ASCE/SEI 7-05, Section 6.5.7.2.
- The importance factor, I, for both equations equals 1.15.
- The adjustment factor, λ, for both equations is identical because both input values are based on the same building height and exposure adjustment factors listed in ASCE/SEI 7-05, Figure 6-2.
- The simplified wind pressure, p_{s30}, in Equation 3.3.1-3 of this report and the wind pressure value, p_{basic}, in Equation 3.3.1-1 are equivalent because they both correspond to the same tabulated design wind pressure values listed in ASCE/SEI 7-05, Figure 6-2.

After substituting p_{s30} for p_{basic} in Equation 3.3.1-1 of this report, the two equations take the following forms.

$$p_s = 1.15\,\lambda\,p_{basic} = 1.15\,\lambda\,p_{s30} \qquad \text{Equation 3.3.1-1}$$

$$p_s = \lambda\,(1.15)\,(1.0)\,p_{s30} = 1.15\,\lambda\,p_{s30} \qquad \text{Equation 3.3.1-3}$$

Because simplified wind pressure, p_{s30}, and the wind pressure value, p_{basic}, are each based on input parameters tabulated in ASCE/SEI 7-05, Figure 6-2, Equation 1 used by the applicant is shown here to be equivalent to Equation 3 from ASCE/SEI 7-05 for Method 1. The simplified design wind pressure, p_s, determined by Method 1 represents the net pressure (sum of internal and external) to be applied to the horizontal and vertical projections of the building surfaces as shown in ASCE/SEI 7-05, Figure 6-2.

Based on the above, pending closure of the confirmatory item, the staff finds the applicant's application of Method 1 to the PS/Bs to be consistent with the guidelines of SRP Section 3.3.1 and therefore acceptable.

3.3.1.4.2.2 ASCE/SEI 7-05 Method 2 – Analytical Procedure

According to requirements in ASCE/SEI 7-05, Section 6.5, Method 2 can be used for the design of MWFRS provided two conditions are met. The two conditions are listed in the first column of Tables 3.3.1-3 and 3.3.1-4 of this report for the PCCV and the R/B, respectively. The second column of Table 3.3.1-3 of this report describes the basis for determining whether or not the PCCV design satisfies the conditions in the first column, and the third column states whether or not the conditions are satisfied. The second column of Table 3.3.1-4 of this report describes the basis for determining whether or not the R/B design satisfies the conditions in the first column, and the third column states whether or not the conditions are satisfied.

Because the basic formula for effective wind velocity pressure used by the applicant was not consistent with the requirements presented in ASCE/SEI 7-05, Section 6.4.2.1 for Method 2, the staff requested the applicant to revise the description of Method 2 and to explain the applicability of the terms in the equation to the analysis of the PCCV and the R/B. In **RAI 215-1906, Question 03.03.01-5**, the staff requested the applicant to provide specific additional information concerning uncertainties about application of Method 2 to the design of the PCCV and the R/B. In response to **RAI 215-1906, Question 03.03.01-5**, dated April 9, 2009, the applicant agreed in the mark-up of DCD Tier 2, Revision 2, Section 3.3.1.2 to (1) correct "effective wind velocity" to "effective wind velocity pressure"; (2) change "P" to "p" in the equation; and (3) change the definition of the internal pressure coefficient, C_{pi}, to include two cases that should be considered to determine the critical load requirements. The staff finds the response acceptable because the new and corrected items make them consistent with the requirements of ASCE 7-05. The staff confirmed that the proposed changes have been incorporated into DCD Revision 3. Accordingly, **RAI 215-1906, Question 03.03.01-5, is resolved.**

The staff reviewed information about the PCCV and the R/B presented in DCD Tier 2, Revision 1 for the US-APWR standard design and determined that both Condition 1 and Condition 2 in ASCE/SEI 7-05, Section 6.5.1 are partially satisfied, but additional information was required to determine if the R/B satisfies Condition 2.

Compliance with ASCE/SEI 7-05, Section 6.5.1 Condition 2 could not be evaluated until the applicant provided additional information showing that the R/B does not have response characteristics making it subject to across wind loading, vortex shedding, instability due to galloping or flutter; and does not have a site location from which channeling effects or buffeting in the wake of upwind obstructions warrant special consideration. In **RAI 215-1906, Question 03.03.01-6**, the staff requested the applicant to provide specific additional information concerning the response characteristics of the R/B. In its response to **RAI 215-1906, Question 03.03.01-6**, dated April 9, 2009, the applicant stated that as described in DCD Tier 2 Section 3.8.4.1, "Description of the Structures," the R/B is a rigid (with respect to wind loading), relatively low-rise, nearly square structure (i.e., height-to-width ratio less than unity) that does not include any unusual or irregular geometric shapes and is constructed of reinforced concrete walls, floors and roofs. Based on the configuration and the material properties of the R/B as presented in the DCD, the R/B does not fall within the limitations of Section C6.5.2 of the ASCE/SEI 7-05 Commentary. The staff agrees that the R/B is not considered to have response characteristics that make it subject to unusual wind effects such as cross wind loading, vortex shedding or instability due to galloping or flutter and Condition 2 of Section 6.51 of ASCE/SEI 7-05 is satisfied. Therefore, the response is acceptable. Accordingly, **RAI 215-1906, Question 03.03.01-6, is resolved**. The applicant also states in its response that as noted in DCD Tier 2, Section 2.3, the COL applicant is to verify that the site-specific conditions are bounded by the site parameters defined for the standard design. Should any unusual wind loading effects not bounded by the Chapter 2 site parameters, COL applicant must consider those as their wind loading parameters for the particular site. The staff finds this statement equivalent to COL Information Item 2.3(1) and therefore acceptable.

Compliance with ASCE/SEI 7-05, Section 6.5.1 Condition 2 also has a site-specific component regarding channeling effects or buffeting in the wake of upwind obstructions that may warrant special consideration by a COL applicant. In **RAI 817-5990, Question 03.03.02-5,** the staff requested the applicant to add a COL Information Item to require a COL applicant to verify that the PCCV, R/B, and PS/Bs do not have site locations for which channeling effects or buffeting in the wake of upwind obstructions warrant special considerations.

In its response to **RAI 817-5990, Question 03.03.02-5**, dated September 26, 2011, the applicant stated that COL Information Item 3.3(4) would be revised to add the requirement requested by the staff and included an associated DCD markup. The staff finds the response acceptable since COL Information Item 3.3(4) now includes a requirement that addresses ASCE/SEI 7-05, Section 6.4, Method 2, Condition 2. Accordingly, **RAI 817-5990, Question 03.03.02-5, is being tracked as a Confirmatory Item.**

In summary, the staff reviewed those responses in the letter dated April 9, 2009, to **RAI 215-1906, Question 03.03.01-5** and **RAI 215-1906, Question 03.03.01-6** and found that they were acceptable. Table 3.3.1-3 and Table 3.3.1-4 of this report summarize the evaluation of Method 2 conditions for the PCCV and R/B respectively for the two conditions defined in ASCE/SEI 7-05, Section 6.5.1, including the resolutions of **RAI 215-1906, Question 03.03.01-5,** and **RAI 215-1906, Question 03.03.01-6**.

Because both conditions in ASCE/SEI 7-05, Section 6.5.1 are satisfied, the applicant is justified in using the 10-step design procedure for Method 2 to analyze the PCCV and the R/B. Details of the design procedure for Method 2 are provided in ASCE/SEI 7-05, Section 6.5.3.

In ASCE/SEI 7-05, Method 2, the applicable design wind pressure, p, for the MWFRS for rigid buildings of all heights including the PCCV and the R/B is defined in Section 6.5.12 and determined by the following equation:

$$p = q\ GC_p - q_i\ (GC_{pi}) \qquad \text{Equation 3.3.1-4}$$

where:

$$q = q_z = 0.00256\ K_z\ K_{zt}\ K_d\ V^2\ I \qquad \text{Equation 3.3.1-5}$$

for windward walls evaluated at height z, where, z, is the height above ground level corresponding to the location where velocity pressure, q, applies (see Note 3.3.1-1)

$$q = q_h = 0.00256\ K_z\ K_{zt}\ K_d\ V^2\ I \qquad \text{Equation 3.3.1-6}$$

for leeward walls, side walls, and roofs evaluated at height h, where, h, is the mean roof height of the building, except that eave height shall be used for roof angle, θ, of less than or equal to 10° (see Note 3.3.1-1)

$$q_i = q_h = 0.00256\ K_z\ K_{zt}\ K_d\ V^2\ I \qquad \text{Equation 3.3.1-7}$$

for windward walls, side walls, leeward walls, and roofs of enclosed buildings and for negative internal pressure evaluation in partially enclosed buildings evaluated at height h, where, h, is the mean roof height of the building, except that eave height shall be used for roof angle, θ, of less than or equal to 10° (see Note 1)

$$q_i = q_z = 0.00256\ K_z\ K_{zt}\ K_d\ V^2\ I \qquad \text{Equation 3.3.1-8}$$

for positive internal pressure evaluation in partially enclosed buildings where height z is defined as the level of the highest opening in the building that could affect the positive internal pressure. For positive internal pressure evaluation, q_i may be conservatively evaluated at height h ($q_i = q_h$) (see Note 3.3.1-1)

G = gust effect factor from ASCE/SEI 7-05, Section 6.5.8

C_p = external pressure coefficient from ASCE/SEI 7-05, Figure 6-6 or 6-8

(GC_{pi}) = internal pressure coefficient from ASCE/SEI 7-05, Figure 6-5 (see Note 3.3.1-2)

K_z = velocity pressure exposure coefficient defined in ASCE/SEI 7-05, Section 6.5.6.6, but not less than 0.87 (see SRP Section 3.3.1, SRP Acceptance Criteria, Item 3A)

K_{zt} = topographic factor as defined in ASCE/SEI 7-05, Section 6.5.7 is equal to 1.0 (see SRP Section 3.3.1, SRP Acceptance Criteria, Item 3A)

K_d = wind directional factor as defined in ASCE/SEI 7-05, Section 6.5.4.4 is equal to 1.0 (see SRP Section 3.3.1, SRP Acceptance Criteria, Item 3A)

V = basic wind speed corresponding to a 3-second gust speed at 33 ft (10 m) above ground in Exposure Category C (open terrain) equal to 155 mph (69.3 m/s) for US-APWR standard design

I = importance factor is equal to 1.15 (see SRP Section 3.3.1, SRP Acceptance Criteria, Item 3A)

Note 3.3.1-1: Velocity pressure values, q and q_i, are evaluated using either Equation 3.3.1-5, 3.3.1-6, 3.3.1-7, or 3.3.1-8, as appropriate. The applicable velocity pressure exposure coefficient, K_z, is defined in ASCE/SEI 7-05, Section 6.5.6.6. Numerical values for velocity pressure exposure coefficient, K_z, are listed in ASCE/SEI 7-05, Table 6-3. Values for velocity pressure exposure coefficient, K_z, increase as height z above ground level increases. Exposure is defined in ASCE/SEI 7-05, Section 6.5.6.3. The applicant used Exposure Category C for the US-APWR standard design. Velocity pressure shall be applied simultaneously on windward and leeward walls and on roof surfaces as defined in ASCE/SEI 7-05, Figure 6-6 and Figure 6-8. Values of external and internal pressure shall be combined algebraically to determine the most critical load.

Note 3.3.1-2: Values of (GC_{pi}) used in Equation 3.3.1-4 are tabulated in ASCE/SEI 7-05, Figure 6-5. Plus and minus signs signify pressure acting towards and away from the internal surfaces, respectively. For enclosed buildings, the value of (GC_{pi}) in Equation 3.3.1-4 is either + 0.18 or − 0.18.

The first and second terms in Equation 3.3.1-4 of this report represent the external and internal pressures, respectively. Values for (GC_{pi}) listed in ASCE/SEI 7-05, Figure 6-5 can be either negative or positive depending on the enclosure classification of the particular building. The sign convention for design wind loads on enclosed and partially enclose buildings is specified in ASCE/SEI 7-05, Section 6.5.12.1.1. According to these sign convention rules, positive pressure acts toward the surface and negative pressure acts away from the surface. Determining the critical load for the appropriate conditions requires consideration of the following two cases.

1. a positive value of (GC_{pi}) applied to all internal surfaces,
2. a negative value of (GC_{pi}) applied to all internal surfaces.

Because Equation 2 used by the applicant and Equation 4 from ASCE/SEI 7-05, Section 6.5.12 are not identical, the equivalency between the two equations is not readily apparent. Therefore, the following analysis was conducted by the staff to verify that the procedure used by the applicant for transforming the basic wind speed into an equivalent pressure applied to buildings and structures yields the same results as Method 2 described in ASCE/SEI 7-05, Section 6.5. This analysis is based on the following input parameters.

- The basic wind speed, V, is 155 mph (69.3 m/s) corresponding to a three-second gust at 33 ft (10 m) above ground in open terrain (ASCE/SEI 7-05, Exposure Category C) for both equations.
- The PCCV and R/B are enclosed structures.
- Exposure Category C applies to both equations.
- Topographic factor, K_{zt}, equals 1.0 for flat terrain (Exposure Category C) as described in ASCE/SEI 7-05, Section 6.5.7.2.
- Wind directional factor, K_d, equals 1.0 based on guidance in SRP Section 3.3.1.
- Velocity pressure exposure coefficient, K_z, evaluated at height, z, for each wind direction based on tabulated values in ASCE/SEI 7-05, Table 6-3.
- The importance factor, I, for both equations equals 1.15.
- The gust-effect factor, G, equals 0.85 for rigid structures as described in ASCE/SEI 7-05, Section 6.5.8.1.

Substituting these input parameters into Equation 3.3.1-4 yields Equation 3.3.1-9, as shown below.

$$p = q\, GC_p - q_i\, (GC_{pi}) \qquad \text{Equation 3.3.1-4}$$

$$p = 0.00256\, K_z\, V^2\, 1.15\, GC_p - 0.00256\, K_z\, V^2\, 1.15\, (GC_{pi})$$

$$p = 0.00256\, K_z\, V^2\, 1.15\, (GC_p - (GC_{pi})) \qquad \text{Equation 3.3.1-9}$$

Wind pressure values, p, determined using Equation 3.3.1-4 and Equation 3.3.1-9 of this report for design wind loads on enclosed buildings as defined in ASCE/SEI 7-05, Section 6.5.12 are shown here to be equivalent. Conversion of design pressure, p, to wind loads is accomplished by simultaneously applying the design pressure on windward and leeward walls and on roof surfaces as defined in ASCE/SEI 7-05, Figures 6-6 and 6-8.

Based on the above evaluation, pending closure of the confirmatory items, the staff finds the applicant's application of Method 2 to the PCCV and to the R/B to be consistent with the guidelines of SRP Section 3.3.1 and is therefore acceptable.

DCD Tier 1, Section 2.2 includes ITAAC to verify that that seismic Category I buildings can withstand design-basis loads, which is evaluated in Section 14.3.3 of this report. Therefore, there are no ITAAC specifically related to wind loadings, since wind loadings are included in the design-basis loads. Accordingly, the staff finds that the applicant meets the requirements of 10 CFR 52.47(b)(1) with regards to wind loadings.

3.3.1.5 Combined License Information Items

The following is a list of COL item numbers and descriptions from Table 1.8-2 of the DCD related to wind loadings (Note: COL information Item 3.3(4) includes the modification made by the applicant's response to **RAI 817-5990, Question 03.03.02-5, which is being tracked as a Confirmatory Item**.):

Table 3.3.1-1 US-APWR Combined License Information Items		
Item No.	**Description**	**Section**
COL 3.3(1)	The COL applicant is responsible for verifying the site-specific basic wind speed is enveloped by the determinations in this section.	3.3.1.1
COL 3.3(2)	These requirements also apply to seismic category I structures provided by the COL Applicant. Similarly, it is the responsibility of the COL Applicant to establish the methods for qualification of tornado effects to preclude damage to safety-related SSCs.	3.3.2.2.4
COL 3.3(3)	It is the responsibility of the COL Applicant to assure that site-specific structures and components not designed for tornado loads will not impact either the function or integrity of adjacent safety-related SSCs, or generate missiles having more severe effects than those discussed in Subsection 3.5.1.4.	3.3.2.3
COL 3.3(4)	The COL applicant is to provide the wind load design method and importance factor for site-specific Category I and Category II buildings and structures. *The COL Applicant shall also verify that the site location does not have features promoting channeling effects or buffeting in the wake of upwind obstructions that invalidate the standard plant wind load design methods described above.*	3.3.1.2
COL 3.3(5)	The COL applicant is to note the vented and unvented requirements of this subsection to the site-specific Category I buildings and structures.	3.3.2.2.2

The staff finds the above listing in the table concerning wind loads to be complete, pending closure of the confirmatory items. Also, the list adequately describes actions necessary for the COL applicant to take. No additional COL information items were identified that need to be included in DCD Tier 2, Table 1.8-2, regarding wind loadings.

3.3.1.6 Conclusions

The following conclusions are based on a technical evaluation of the wind loading information submitted by the US-APWR applicant for the standard plant design as set forth above in Sections 3.3.1.4 of this report.

- The wind design parameters selected by the applicant are consistent with guidance provided in SRP 2.3.1, and thus are acceptable.

- The US-APWR applicant used either Method 1 (simplified procedure) or Method 2 (analytical procedure) in ASCE/SEI 7-05 to convert wind speed to wind loads for the following non-site-specific seismic Category I buildings and structures in the US-APWR standard plant design:

  a. PCCV
  b. R/B

c. PS/Bs

As discussed in Section 3.3.1.4.2 of this report, pending closure of the confirmatory item, the staff finds the applicant's application of Methods 1 and 2 to be consistent with the guidelines of SRP Section 3.3.1 and therefore acceptable.

- Pending closure of the confirmatory item, the DCD identifies a complete list of necessary COL information items regarding wind loadings.

Accordingly, pending closure of the confirmatory items, the staff concludes that the application meets the relevant requirements in GDC 2 and 10 CFR 52.47(b)(1) regarding wind loadings, for the non-site-specific seismic Category I buildings and structures in the US-APWR standard design.

Table 3.3.1-2: Evaluation of Method 1 Conditions for the PS/Bs

Condition*	Evaluation	Condition Satisfied
1. The building is a simple diaphragm building as defined in Section 6.2.	The buildings have reinforced concrete basemats with vertical shear walls and flat roofs that function as the MWFRS.	Yes
2. The building is a low-rise building as defined in Section 6.2.	The buildings are enclosed, have a height less than 60 ft above ground level, and a least horizontal dimension of 66 ft	Yes
3. The building is enclosed as defined in Section 6.2 and conforms to the wind-borne debris provision of Section 6.5.9.3.	The buildings do not satisfy the requirements for open or partially enclosed buildings, so they are classified as enclosed buildings in accordance with definitions in ASCE/SEI 7-05, Section 6.2. They are enclosed and have impact resistant glazing. In addition, the buildings conform to the wind-borne debris provision of Section 6.5.9.3.becasue they do not have glazing. The applicant clarified the large openings in the PS/Bs in response to **RAI 215-1906, Question 03.03.01-1**.	Yes
4. The building is a regular-shaped building as defined in Section 6.2.	The buildings have rectangular vertical walls and flat roofs with no unusual geometrical irregularities.	Yes
5. The building is not classified as a flexible building as defined in Section 6.2.	The buildings satisfy Condition 5 for Method 1 defined in ASCE/SEI 7-05 Section 6.4.1.1 because they are not considered slender buildings with fundamental natural frequencies less than 1 Hz. They are not slender and do not have natural frequencies less than 1 Hz. The applicant clarified the natural frequencies of the PS/Bs in response to **RAI**	Yes

Condition*	Evaluation	Condition Satisfied
	215-1906, Question 03.03.01-2.	
6. The building does not have response characteristics making it subject to across wind loading, vortex shedding, instability due to galloping or flutter; and does not have a site location from which channeling effects or buffeting in the wake of upwind obstructions warrant special consideration.	The buildings are located in Exposure Category C areas with minimal upwind obstructions. The buildings are rigid diaphragm reinforced concrete structures with stiff response characteristics.	Yes
7. The building has an approximately symmetrical cross section in each direction with either a flat roof or a gable or hip roof with θ ≤ 45°.	The nominal plan dimensions for each PS/B are 66 ft by 111.5 ft. The maximum height of these buildings is approximately 47.5 ft above ground level.	Yes
8. The building is exempted from torsional load cases as indicated in Note 5 of Figure 6-10, or the torsional load cases defined in Note 5 do not control the design of any of the MWFRSs of the building.	The buildings satisfy Condition 8 for Method 1 defined in ASCE/SEI 7-05, Section 6.4.1.1 because the rigid roof and floor diaphragms of the buildings distribute lateral force to the orthogonal exterior shear walls that constitute the MWFRS. For this building configuration, the torsional load cases defined in ASCE/SEI 7-05, Figure 6-10, Note 5 do not control the design of any of the MWFRS for the PS/Bs. Torsional load cases defined in Note 5 do not control the design of any of the MWFRSs of the building. The applicant clarified the torsional response characteristics of the PS/Bs in response to **RAI 215-1906, Question 03.03.01-3**.	Yes

*Conditions are defined in ASCE/SEI 7-05, Section 6.4.1.1 and reference other sections of ASCE/SEI 7-05.

Table 3.3.1-3: **Evaluation of Method 2 Conditions for PCCV**

Condition*	PCCV Evaluation	Condition Satisfied
1. The building is a regular-shaped building as defined in Section 6.2.	The PCCV is a vertical cylinder with a hemispherical dome with no unusual geometrical irregularity in spatial form.	Yes
2. The building does not have response characteristics making it subject to across	The PCCV has a relatively low profile with an approximate overall height-to-diameter ratio of 1.5, and the PCCV is surrounded by the	Yes

Condition*	PCCV Evaluation	Condition Satisfied
wind loading, vortex shedding, instability due to galloping or flutter; and does not have a site location from which channeling effects or buffeting in the wake of upwind obstructions warrant special consideration.	rectangular-shaped R/B such that approximately only the upper half of the PCCV is exposed to wind loading. The PCCV does not have response characteristics which make it subject to across wind loading, vortex shedding, or other unusual wind effects which might require investigation using Method 3 - Wind Tunnel Procedure of ASCE/SEI 7-05. Further, the site location of the PCCV is such that channeling or buffeting effects do not require special consideration.	

*Conditions are defined in ASCE/SEI 7-05, Section 6.5.1 and reference other sections of ASCE/SEI 7-05.

Table 3.3.1-4: **Evaluation of Method 2 Conditions for R/B**

Condition*	R/B Evaluation	Condition Satisfied
1. The building is a regular-shaped building as defined in Section 6.2.	The R/B has a rectangular shape with no unusual geometrical irregularity in spatial form.	Yes
2. The building does not have response characteristics making it subject to across wind loading, vortex shedding, instability due to galloping or flutter; and does not have a site location from which channeling effects or buffeting in the wake of upwind obstructions warrant special consideration.	The R/B is a low-profile building that does not have response characteristics which make it subject to across wind loading, vortex shedding, or other unusual wind effects which might require investigation using Method 3 - Wind Tunnel Procedure of ASCE/SEI 7-05. Further, the site location of the R/B is such that channeling or buffeting effects do not require special consideration. The applicant clarified the response characteristics of the R/B in response to **RAI 215-1906, Question 03.03.01-6**.	Yes

*Conditions are defined in ASCE/SEI 7-05, Section 6.5.1 and reference other sections of ASCE/SEI 7-05.

3.3.2 Tornado and Hurricane Loadings

3.3.2.1 Introduction

This section discusses the design of structures that must withstand the effects of the plant's design-basis tornado and hurricane.

3.3.2.2 Summary of Application

DCD Tier 1: The Tier 1 information associated with this section is found in DCD Tier 1, Section 2.2, "Structural and Systems Engineering"; however, there are no ITAAC specifically related to tornado loadings.

DCD Tier 2: The applicant has provided a DCD Tier 2 design description in Section 3.3.2, "Tornado Loadings," summarized here in part, as follows: This section discusses the design of structures that must withstand the effects of the plant's design-basis tornado and hurricane. Note that DCD Tier 2, Section 3.3.2, Revision 3 only addressed tornado loadings. In response to **RAI 908-6327, Question 03.03.02-6** and **RAI 908-6321, Question 02-3**, the applicant proposed revising DCD Tier 2, Section 3.3.2 to address both tornado and hurricane loadings, and provided an associated markup to be included in DCD Revision 4.

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

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 for this area of review.

Cross-cutting Requirements (TMI, USI/ GSI, Op Ex): There are no cross-cutting requirements for this area of review.

US-APWR Interface Issues Identified in the DCD: See DCD Tier 2, Table 1.8-2, "Compilation of All Combined License Applicant Items for Chapters 1-19."

Site Interface Issues Identified in the DCD: See DCD Tier 2, Table 1.8-1, "Significant Site Specific Interfaces with the Standard US-APWR Design."

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

3.3.2.3 Regulatory Basis

The relevant requirements of the Commission's regulations for this area of review, and the associated acceptance criteria are given in Section 3.3.2, "Tornado Loadings," Revision 3, issued March 2007, of NUREG-0800 and are summarized below. Review interfaces with other SRP sections can be found in Section 3.3.2 of NUREG-0800.

1. GDC 2, as it relates to the ability of SSCs without loss of capability to perform their safety function, to withstand the effects of natural phenomena, such as earthquakes, hurricanes, tornadoes, floods, and the appropriate combination of all loads.

2. GDC 4, "Environmental and dynamic effects design bases," as it relates to the protection of safety-related SSCs against dynamic effects, including the effects of tornado and hurricane-generated missiles, that may result from structural damage and from events and conditions outside the nuclear power unit.

3. 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 has been constructed and will operate in accordance with the DC, the Atomic Energy Act of 1954, and the NRC's regulations.

Acceptance criteria adequate to meet the above requirements include:

1. The tornado wind and associated missiles generated by the tornado wind used in the design shall be the most severe wind that has been historically reported for the site and surrounding area with sufficient margin for the limited accuracy, quantity, and period of time in which historical data have been accumulated.

2. The acceptance criteria for tornado parameters including maximum wind speed, translational speed, rotational speed, and atmospheric pressure change, and the bases for determining these parameters are defined in SRP Sections 2.3.1 and 2.3.2. Acceptance criteria for the spectrum of tornado-generated missiles and their characteristics, as well as the bases for determining these parameters, are defined in SRP Section 3.5.1.4. These parameters should serve as basic input to the review and evaluation for structural design.

3. The acceptance criteria for procedures used to transform tornado parameters into equivalent loads on structures are as follows:

    a. Tornado Characteristics and Effects

        Tornados are characterized, in Table 1 of RG 1.76 for the contiguous U.S. into three geographical regions and by (1) maximum wind speed, (2) translational speed, (3) maximum rotational speed, (4) radius of maximum rotational speed, (5) pressure drop, and (6) rate of pressure drop for each of the three regions. Tornado effects are subdivided into three groups:

        i. Tornado wind effects caused by the direct action of air flow on structures,

        ii. Atmospheric pressure change effects caused by the differential pressure between the interior and exterior of a structure during the passage of a tornado, and

        iii. Tornado-generated missile impact effects.

            Tornado effects considered in design should include combinations of tornado wind effects, atmospheric pressure change effects, and tornado-generated missile impact effects.

    b. Tornado Wind Effects

        Procedures delineated in American Society of Civil Engineers/ Structural Engineering Institute (ASCE/SEI) 7-05, "Minimum Design Loads for Buildings and Other Structures" are acceptable for transforming tornado wind speed into pressure-induced forces applied to structures. In particular, the following shall apply:

i. The maximum velocity pressure, q_Z, should be based on the applicable maximum tornado wind speed, V, using the following equation from ASCE/SEI 7-05, Section 6.5.10:

$$q_Z = 0.00256\ K_Z K_{dt} K_d V^2 I\ (\text{lb/ft}^2),$$

where:

K_Z= velocity pressure exposure coefficient equal to 0.87
K_{dt} = topographic factor equal to 1.0
K_d= wind directionality factor equal to 1.0
V = maximum tornado wind speed (mi/h), and
I = importance factor equal to 1.15

The maximum tornado wind speed, V, is the resultant of the maximum rotational speed and the translational speed of the tornado.

ii. Wind speed is assumed not to vary with the height above ground.

iii. Design tornado wind loads should be determined in accordance with the following sections in ASCE/SEI 7-05, as applicable.

(1) 6.5.12 Design Loads on Enclosed and Partially Enclosed Buildings.
(2) 6.5.13 Design Wind Loads on Open Buildings with Monoslope, Pitched, or Troughed Roofs.
(3) 6.5.14 Design Wind Loads on Solid Freestanding Walls and Solid Signs.
(4) 6.5.15 Design Wind Loads on Other Structures.

c. Atmospheric Pressure Change Effects

RG 1.76 provides guidance for determining the pressure drop and the rate of pressure drop caused by the passage of a tornado. “Wind Effects on Structures: Fundamentals and Applications to Design,” (Third Edition, John Wiley and Sons, Inc., New York, 1996.) by E. Simiu and R. H. Scanlan, provides methods for determining loads on structures due to atmospheric pressure changes during the passage of a tornado. For a structure that is completely open subjected to a tornado, the internal and external pressures on the structure equalize rapidly during the passage of the tornado. Therefore, the atmospheric pressure change between the interior and the exterior of that structure approaches zero. For a structure that is enclosed (unvented structure), the internal pressure remains equal to the atmospheric pressure before the passage of a tornado. The atmospheric pressure outside the structure changes during the passage of a tornado, which creates pressure differences between the interior and the exterior of that structure, and these differential pressures produce outward acting loads on the roof and walls of the enclosed structure. For a structure that is partially enclosed (vented structure), the determination of loads on the structure due to atmospheric pressure changes during the passage of a tornado is more complicated. If venting is adopted as a way to reduce the atmospheric pressure change effect on a structure, the review will be performed on a case-by-case basis.

d. Tornado-Generated Missile Impact Effects

Tornado-generated missile characteristics and the design-basis tornado missile spectrum are provided in RG 1.76. The acceptance criteria for transforming tornado-generated missile impact into equivalent static loads on structures are delineated in SRP Section 3.5.3, Subsection II.

e. Combined Tornado Effects

After tornado-generated wind effects, W_w, atmospheric pressure change effects, W_p, and missile impact effects, W_m, are determined, the combination thereof should then be established in a conservative manner for structures. An acceptable method of combining these effects and establishing the total tornado load on a structure is as follows:

$W_t = W_p$ Eq. 1

$W_t = W_w + 0.5\ W_p + W_m$ Eq. 2

where:

W_t = total tornado load
W_w = load from tornado wind effect
W_p = load from tornado atmospheric pressure change effect
W_m = load from tornado missile impact effect

The information provided to demonstrate that failure of any structure or component not designed for tornado loads will not affect the capability of other SSCs to perform necessary safety functions, is acceptable if found in accordance with either of the following:

a. The postulated failure or collapse of structures and components not designed for tornado loads, including missiles, can be shown not to result in any structural or other damage to safety-related structures, systems, or components.

b. Safety-related structures are designed to resist the effects of the postulated structural failure, collapse, or generation of missiles from structures and components not designed for tornado loads.

3.3.2.4 Technical Evaluation

Structural design criteria for tornado and hurricane loading effects on seismic Category I buildings and structures described in DCD Tier 2, Revision 3 for the US-APWR were evaluated for compliance with GDC 2 and GDC 4. Table 3.3.2-2 of this report lists the seismic classification of buildings and structure of the US-APWR standard plant. Review guidance and acceptance criteria provided in SRP Section 3.3.2 were used by the staff to perform the technical evaluation. Specific areas of review included:

- validation of design parameters for the structural design criteria appropriate to account for tornado and hurricane loadings.

- verification that procedures for transforming the tornado/hurricane wind speed into an equivalent pressure applied to buildings and structures and for distributing the tornado/hurricane wind speed on the buildings and structures are in accordance with wind load standards in ASCE/SEI 7-05 including the following procedures:

  ✓ transformation of tornado/hurricane wind into equivalent loads applied to structures taking into consideration the geometrical configuration and physical characteristics of the structures and the distribution of tornado/hurricane wind pressure on structures.
  ✓ transformation of tornado-generated atmospheric pressure changes into applied loads on structures.
  ✓ transformation of tornado and hurricane-generated missiles into equivalent loads on structures.
  ✓ combinations of individual loads in a manner that will produce the most adverse total tornado/hurricane load effect on structures.

- confirmation that all seismic Category I buildings and structures in the US-APWR standard design that are subject to tornado/hurricane loading are identified and appropriately addressed in compliance with SRP Section 14.3, "Inspections, Tests, Analyses, and Acceptance Criteria," issued March 2007.

- affirmation that COL applicant responsibilities for addressing tornado/hurricane loading effects on site-specific, non-standard US-APWR plant design seismic Category I buildings and structures are clearly defined.

3.3.2.4.1 Design Parameters

In DCD Tier 2, Section 3.3.2.1, "Applicable Design Parameters," the applicant used the following six tornado characteristic values as design parameters for determining tornado load effects on SSCs in the US-APWR standard plant.

- Maximum wind speed = 230 mph (103 m/s)
- Translational speed = 46 mph (21 m/s)
- Maximum rotational speed = 184 mph (82.3 m/s)
- Radius of maximum rotational speed = 150 ft (45.7m)
- Pressure drop = 1.2 pounds per square in. (psi) (8.3 kilopascal (kPa))
- Time rate of pressure drop = 0.5 psi/second (s) (3 kPa/second)

This set of tornado characteristic values is identical to the tabulated values listed in Table 1 of RG 1.76, "Design-Basis Tornado and Tornado Missiles for Nuclear Power Plants," Revision 1, issued March 2007 for Region I and reflects a frequency exceedance equal to or greater than 10^{-7} per year. In **RAI 218-1907 Question 03.03.02-1,** the staff requested the applicant to address a typographical error in the exceedance frequency in DCD Tier 2, Revision 1, Section 3.3.2.1. In its response to **RAI 218-1907 Question 03.03.02-1,** dated April 9, 2009, the applicant corrected the exceedance frequency. The staff finds the response acceptable since the exceedance frequency is consistent with the design-basis risk frequency stated in RG 1.76, Revision 1. The staff confirmed that the DCD change was incorporated into DCD Revision 2. Accordingly, **RAI 218-1907, Question 03.03.02-1, is resolved.** In selecting the tornado characteristic values for Region 1 as the design-basis for the US-APWR standard plant, the tornado characteristics at any site within the contiguous U.S. are enveloped by this set of tornado characteristic values.

The design-basis tornado missile spectrum in Table 2 of RG 1.76, Revision 1 is generally acceptable to the staff for the design of nuclear power plants. In addition, the staff considers the missiles listed in Table 2 to be capable of striking in all directions with horizontal velocities of V_{Mh}^{max} and vertical velocities equal to 67 percent of V_{Mh}^{max}.

According to the Rankine combined vortex model, which was used by the staff as the basis for the tornado characteristics described in RG 1.76, Revision 1, wind velocities and pressures are assumed not to vary with elevation, and the model possesses only azimuthal velocity. Because these Rankine combined vortex model features apply to all tornadoes, the maximum tornado missile velocity is considered by the staff to be independent of the missile height above ground. Consequently, any tornado-generated missile could potentially impact a SSC from any azimuthal direction and at any elevation above grade at the maximum tornado missile velocity.

Compliance with GDC 2 requires that the design bases for SSCs reflect appropriate consideration of the most severe of the natural phenomena that have been historically reported for the site and surrounding area, with sufficient margin for the limited accuracy, quantity, and period of time in which the historical data have been accumulated. DCD Tier 2, Subsection 3.3.2.2.4, “Tornado Missile Effects,” describes the applicant’s approach to addressing tornado missile effects, which references, “Impact Effect of Fragments Striking Structural Elements,” Williamson and Alvy, issued November 1973. In **RAI 218-1907, Question 03.03.02-2**, the staff requested the applicant to describe the fragment spectrum considered in the cited reference. The staff also requested the applicant to identify the various types of objects included in the fragment spectrum, if any, which are capable of producing tornado missile impact effects that are more severe than those produced by the missiles included in the missile spectrum defined in Table 2 of RG 1.76, Revision 1. The applicant was also requested to provide a description of the physical characteristics, the maximum speed, and the envelope of potential impact locations (i.e., SSC identifier, elevation above grade, and corresponding azimuthal direction) for each tornado missile included in the applicant’s design-basis tornado missile spectrum. In its response to **RAI 218-1907, Question 03.03.02-2**, dated April 9, 2009, the applicant stated that DCD Tier 2, Section 3.5.1.4, “Missiles Generated by Tornadoes and Extreme Winds,” outlines the tornado missile spectrum consistent with Table 2 of RG 1.76, Revision 1 and is listed among the key site parameters defined in DCD Tier 2, Table 2.0-1, “Key Site Parameters.” Any tornado or hurricane-generated missile fragments that could produce impact effects more severe than those defined in DCD Tier 2, Section 3.5.1.4 must be considered by COL applicants as required by COL Information Item 3.3(3) on a project-specific basis. The impact effect of fragments in “Impact Effect of Fragments Striking Structural Elements,” provides a method used to obtain an equivalent static load for use in a structural analysis. This clarification was added at the end of DCD Tier 2, Subsection 3.3.2.2.3, “Tornado Missile Effects.” The staff reviewed the response and found it acceptable based on it meeting SRP Section 3.3.2 Acceptance Criteria 3Aiii and 3D. The staff confirmed that the DCD changes were incorporated into DCD Revision 3. The response with regard to the fragment spectrum is discussed further below.

The applicant used the set of tornado characteristic values listed in DCD Tier 2, Table 2.0-1as input parameters to the single Rankine combined vortex model as described in RG 1.76, Revision 1. According to this model, wind velocities and pressures are assumed not to vary with the height above ground. This assumption was adopted by the applicant in applying the single Rankine combined vortex model to the seismic Category I structures in the US-APWR standard plant. Use of the single Rankine combined vortex model and the design-basis tornado characteristics as described in RG 1.76, Revision 1, is consistent with guidance provided in SRP Section 2.3.1, "Regional Climatology," Revision 3, issued March 2007, and SRP Section 2.3.2, "Local Meteorology," Revision 3, issued March 2007. The applicant's response also allowed the staff to conclude that the tornado-generated fragment spectrum in "Impact Effect of Fragments Striking Structural Elements," is within envelope of tornado missiles included in the missile spectrum defined in Table 2 of RG 1.76, Revision 1. In addition, the staff concludes that the physical characteristics, the maximum speed, and the envelope of potential impact locations for each tornado missile included in the applicant's design-basis tornado missile spectrum are adequately considered by the applicant in designing SSCs for tornado load effects; comply with applicable GDC 2 and GDC 4 requirements; and reflect appropriate consideration of the most severe of the natural phenomena that have been historically reported for the site and surrounding area, with sufficient margin for the limited accuracy, quantity, and period of time in which the historical data have been accumulated. Accordingly, **RAI 218-1907, Question 03.03.02-2 is resolved.**

The maximum tornado wind speed for Region I listed in Table 1 of RG 1.76, Revision 0, issued April 1974, of 360 mph (161 m/s) was reduced in the current version RG 1.76, Revision 1, to 230 mph (103 m/s). This change was based on the enhanced Fujita (EF)-scale approach that was used to replace the Fujita (F)-scale approach to project and update the design-basis tornado characteristic in RG 1.76. However, the following summary describes considerations involved in determining the maximum tornado wind speed.

(1) The EF-Scale was initially based on the need to correlate the actual damage intensity of structures related to tornados, instead of investigating the maximum tornado wind speeds. In addition, members of the F-Scale Forum insisted that the historical tornado database be preserved. These considerations resulted in the 0.6246 correlation factor (or reduction factor) between the F-scale and EF-Scale, where the EF-Scale equals 0.6246 times the F-Scale. Additional background on the F-Scale and EF Scale are provided in (1) NUREG/CR-4461, "Tornado Climatology of the Contiguous United States," Revision 2, issued February 2007; and (2) WASH-1300, "Technical Basis for Interim Regional Tornado Criteria," US-NRC, issued May 1974.

(2) The Bridge Creek tornado that occurred on May 3, 1999, in the Oklahoma City area is a good example of a recent maximum tornado event. During this tornado, researchers from the University of Oklahoma used "Doppler on Wheels" (DOW) to measure a tornado wind speed of 318 mph (142 m/s) near Bridge Creek, Oklahoma. The data obtained by the DOW team were subjected to scientific peer review, and results of this review suggest that the maximum speed actually may be less than 318 mph (142 m/s) but still in the 300 mph (134 m/s) range. Reference: DOW research from the University of Oklahoma internet website.

Because research tools such as DOW are being developed to measure tornado wind speed, the 230 mph (103 m/s) value used by the applicant for the APWR standard design may not envelope the maximum tornado wind speed that could occur during a severe tornado event. Therefore, to comply with requirements in GDC 2, per COL Information Item 3.3(1), the COL Applicant is responsible for verifying the site-specific basic wind speed is enveloped by the determinations in this DCD Tier 2, Section 3.3.2. This would include verifying that the maximum design-basis tornado characteristics listed in RG 1.76, Revision 1 are not exceeded at the site where the APWR plant will be constructed.

DCD Tier 2, Section 3.5.1.4, describes the spectrum of tornado-generated missiles. The spectrum of tornado-generated missiles selected by the applicant as the design-basis for the US-APWR plant conforms to guidelines in SRP Section 3.5.1.4, "Missiles Generated by Tornadoes and Extreme Winds" and RG 1.76, Revision 1 for Region I. Tornado-generated missiles and their associated wind speeds that are acceptable to the staff are identified in Table 2 of RG 1.76, Revision 1.

Table 3.3.2-3 of this SE presents a comparison of the missile spectrum selected by the applicant and the missile spectrum listed in Table 2 of RG 1.76 for Region I. Differences between the two spectra are noted in the Table 3.3.2-3 of this SE. The method used by the applicant to obtain an equivalent static load for use in a structural analysis to determine missile impact effects is described in "Impact Effect of Fragments Striking Structural Elements," referenced in DCD Tier 2, Subsection 3.3.2.2.3. This method is acceptable based on engineering practice in impact mechanics.

In addition to tornado, hurricane is another type of extreme wind that has potential to impose undue risk to the safety of a nuclear power plant. As discussed, the design-basis tornado wind speeds proposed in RG 1.76, Revision1 were reduced by 38 percent using the EF-Scale. It is possible that the bounding design-basis wind speeds be governed by hurricane, not tornado, in some areas of the U.S. As a result, RG 1.221, "Design-Basis Hurricane and Hurricane Missiles for Nuclear Power Plants," Revision 0, was issued in October 2011. In view of RG 1.221, in **RAI 908-6327, Question 03.03.02-6,** the staff requested the applicant to: (1) define the design-basis hurricane wind speeds and hurricane-generated missile spectrum and associated speeds; (2) provide a methodology to convert hurricane wind and missile impact effects to design load; (3) determine total wind load effects from the combination of hurricane wind effects and missile impact effects; and (4) revise COL information items to include the hurricane effects should it bound the extreme wind conditions. In its response to **RAI 908-6327, Question 03.03.02-6**, dated June 26, 2012, and the supplemental response to **RAI 907-6321, Question 02-3**, dated September 13, 2012, the applicant provided the needed information in the DCD Revision 3 Markup. Items 1, 2, and 4 are discussed below. Item 3 is discussed in Section 3.3.2.4.2 of this report. Based on these discussions, the staff finds the responses acceptable. As there are DCD changes **RAI 908-6327, Question 03.03.02-6,** and **RAI 907-6321, Question 02-3, are being tracked as Confirmatory Items**.

Regarding Item 1 of **RAI 908-6327, Question 03.03.02-6**, the applicant added the following statement to DCD Tier 2, Section 3.3.2.1:

> The design-basis hurricane wind speed is chosen from wind speed contour maps for hurricane-prone regions of the contiguous U.S. presented in RG 1.221. The hurricane wind speed selected for design of the standard plant is 160 mph (71.5 m/s), corresponding to a three-second gust at 33 ft (10 m) above ground for exposure category C applicable to nuclear power plants defined in Section 6.5.6.3 of ASCE/SEI 7-05. The design-basis hurricane wind speeds selected (160 mph) (71.5 m/s) envelop the design-basis wind speeds at most locations in the contiguous U.S. The wind speed has an exceedance frequency of 10^{-7} per year and an important factor of 1.15 in accordance with SRP Section 3.3.2

The staff reviewed the design parameter selected for hurricane wind speed, and found it indeed bounds the design-basis hurricane wind speeds at most locations given in Figure 1, "Design-Basis Hurricane Windspeeds for the Western Gulf of Mexico U.S. Coastline Representing Exceedance Probabilities of 10^{-7} per Year. Values are nominal 3-second gust windspeeds in miles per hour (meters per second) at 33 ft (10 m) above ground over open terrain (reproduced from NUREG/CR-7005 ["Technical Basis for Regulatory Guidance on Design-Basis Hurricane Wind Speeds for Nuclear Power Plants," issued November 2011])," Figure 2, "Design-Basis Hurricane Windspeeds for the Eastern Gulf of Mexico and Southeastern Atlantic U.S. Coastline Representing Exceedance Probabilities of 10^{-7} per Year. Values are nominal 3-second gust windspeeds in miles per hour (meters per second) at 33 ft (10 m) above ground over open terrain (reproduced from NUREG/CR-7005)," and Figure 3, Design-Basis Hurricane Windspeeds for the Mid- and Northern Atlantic U.S. Coastline Representing Exceedance Probabilities of 10^{-7} per year. Values are nominal 3-second gust windspeeds in miles per hour (meters per second) at 33 ft (10 m) above ground over open terrain (reproduced from NUREG/CR-7005)" of RG 1.221, Revision 0 for Gulf of Mexico U. S. Coastline, Southeastern Atlantic U.S. Coastline and Northern Atlantic U. S. Coastline respectively. Similar to the case of tornado, the design parameters defined for hurricane including wind speed, exceedance frequency, and important factor are all consistent with ASCE/SEI 7-05, SRP Section 3.3.2 and RG 1.221, Revision 0, thus are acceptable. Therefore, the staff finds that the applicant has addressed Item 1 of **RAI 908-6327, Question 03.03.02-6**.

Regarding Item 2 of **RAI 908-6327, Question 03.03.02-6**, the applicant modified the DCD description of missile impact effects to address both tornado and hurricane effects. The spectrum of hurricane-generated missiles selected by the applicant as the design-basis for the US-APWR plant conforms to guidelines in SRP Section 3.5.1.4, and RG 1.221, Table 1.

Design-Basis hurricane-generated missile velocity and their associated wind speeds that are acceptable to the staff are identified in RG 1.221, Table 2. Table 3.3.2-3 of this SE presents a comparison of the missile spectrum selected by the applicant and the missile spectrum listed in RG 1.76, Revision 1, Table 2, for Region I as well as RG 1.221, Table 1. Differences between the two spectra are noted in the Table 3.3.2-3 of this SE. The staff notes that, for a given wind speed and type of missile, hurricane produces higher missile speed than that of tornado due to distinctive high-intensity durations.

The method used by the applicant to obtain an equivalent static load for use in a structural analysis to determine missile impact effects is described in "Impact Effect of Fragments Striking Structural Elements," referenced in DCD Tier 2, Subsection 3.3.2.2.3. This is one of the accepted methods in the field of impact mechanics. In addition, the staff noted that the applicant elected to increase the design thickness of concrete barriers by 20 percent and the design thickness of steel targets by 25 percent to provide a greater safety margin for tornado/hurricane missile protection. The staff also noted that the applicant is providing a greater safety margin for tornado/hurricane missile protection by considering effects of a 4,000 lb. (1814 kg) automobile missile impact on all seismic Category I and II structures at any elevation produced by the missile striking in any direction. Therefore, the staff finds that the applicant has addressed Item 2 of **RAI 908-6327, Question 03.03.02-6**.

Regarding Item 4 of **RAI 908-6327, Question 03.03.02-6**, the applicant revised COL Information Items 3.3(2) and 3.3(3) to address hurricanes. The applicant also added a new COL Information Item 3.3(6) as follows.

The COL Applicant is responsible for verifying that the site-specific design-basis hurricane basic wind speeds, exposure category, and resulting wind forces are enveloped by the determinations in this section.

The staff finds the COL information item acceptable site it clarifies the COL applicant's responsibilities regarding hurricane loadings. Therefore, the staff finds that the applicant has addressed Item 4 of **RAI 908-6327 Question 03.03.02-6**.

3.3.2.4.2 Conversion of Extreme Wind Effects to Design Load

Regarding, Item 3 of **RAI 908-6327, Question 03.03.02-6**, the applicant modified the DCD description of conversion of extreme wind effects to design load to address both tornado and hurricane effects. As described below, the staff finds this revised description conforms to the guidelines of SRP Section 3.3.2. Therefore, the staff finds that the applicant has addressed Item 3 of **RAI 908-6327, Question 03.03.02-6**.

The applicant used ASCE/SEI 7-05, Method 2 – Analytical Procedure to transform extreme wind speed into pressure-induced forces applied to the seismic Category I buildings and structures in the US-APWR standard design. According to requirements in ASCE/SEI 7-05, Section 6.5, ASCE/SEI 7-05, Method 2 can be used for the design of MWFRS for buildings provided two conditions are met.

1. The building is a regular-shaped building as defined in ASCE/SEI 7-05 Section 6.2.

2. The building does not have response characteristics making it subject to across wind loading, vortex shedding, instability due to galloping or flutter; and does not have a site location from which channeling effects or buffeting in the wake of upwind obstructions warrant special consideration.

The staff evaluation of the PCCV, R/B, and PS/Bs for these two conditions are provided in Tables 3.3.2-4, 3.3.2-5, and 3.3.2-6 of this SE, respectively. The first column of each table lists the two conditions from ASCE/SEI 7-05, Section 6.5. The second column of each table describes the basis for determining whether or not the building design satisfies the conditions in the first column, and the third column of each table states whether or not the conditions are satisfied. The staff determined that that the PCCV, RB, and PS/Bs each meet the two conditions from ASCE/SEI 7-05, Section 6.5

In **RAI 817-5990, Question 03.03.02-5**, the staff requested the applicant to identify the responsibilities of the COL applicant for the actions that the COL applicant needs to take to verify that the two conditions from ASCE/SEI 7-05, Section 6.5. In its response to **RAI 817-5990, Question 03.03.02-5**, dated August 25, 2011, the applicant revised COL Information Item 3.3(4) to read as follows.

"The COL Applicant is to provide the wind load design method and importance factor for site-specific seismic category I and seismic category II buildings and structures. The COL Applicant shall also verify that the site location does not have features promoting channeling effects or buffeting in the wake of upwind obstructions that invalidate the standard plant wind load design methods described above."

The staff finds the response acceptable since the applicant clarified the COL applicant responsibilities with regard channeling effects or buffeting. Since the applicant has identified DCD changes, **RAI 817-5990, Question 03.03.02-5 is being tracked as a Confirmatory Item.**

Because both conditions in ASCE/SEI 7-05, Section 6.5.1 are satisfied for each building, the staff agrees that the applicant is justified in using the 10-step design procedure for ASCE/SEI 7-05, Method 2 to analyze the PCCV, R/B, and PS/Bs. Details of the design procedure for ASCE/SEI 7-05, Method 2 are provided in ASCE/SEI 7-05, Section 6.5.3.

In ASCE/SEI 7-05, Method 2, the applicable design wind pressure, *p*, for the MWFRS for rigid buildings of all heights, including the PCCV, R/B, and PS/Bs, is defined in ASCE/SEI 7-05, Section 6.5.12 and determined by the following equation:

$p = q\ GC_p - q_i\ (GC_{pi})$ Equation 3.3.2-1

where

$q = q_z = 0.00256\ K_z\ K_{zt}\ K_d\ V^2\ I$ Equation 3.3.2-2

for windward walls evaluated at height *z*, where, *z*, is the height above ground level corresponding to the location where velocity pressure, *q*, applies (see Note 1)

$q = q_h = 0.00256\ K_z\ K_{zt}\ K_d\ V^2\ I$ Equation 3.3.2-3

for leeward walls, side walls, and roofs evaluated at height *h*, where, *h*, is the mean roof height of the building, except that eave height shall be used for roof angle, θ, of less than or equal to 10° (see Note 1)

$q_i = q_h = 0.00256\ K_z\ K_{zt}\ K_d\ V^2\ I$ Equation 3.3.2-4

for windward walls, side walls, leeward walls, and roofs of enclosed buildings and for negative internal pressure evaluation in partially enclosed buildings evaluated at height *h*, where, *h*, is the mean roof height of the building, except that eave height shall be used for roof angle, θ, of less than or equal to 10° (see Note 1)

$q_i = q_z = 0.00256\ K_z\ K_{zt}\ K_d\ V^2\ I$ Equation 3.3.2-5

for positive internal pressure evaluation in partially enclosed buildings where height *z* is defined as the level of the highest opening in the building that could affect the positive internal pressure. For positive internal pressure evaluation, q_i may be conservatively evaluated at height *h* ($q_i = q_h$) (see Note 1)

G = gust effect factor from ASCE/SEI 7-05, Section 6.5.8

C_p = *external* pressure coefficient from ASCE/SEI 7-05, Figure 6-6 or 6-8

(GC_{pi}) = internal pressure coefficient from ASCE/SEI 7-05, Figure 6-5 (see Note 2)

K_z = *velocity* pressure exposure coefficient is equal to 0.87 for tornado; or to values given in Table 6-3, ASCE/SEI 7-05 for hurricane (see SRP Section 3.3.2, SRP Acceptance Criteria, Item B)

K_{zt} = *topographic* factor as defined in ASCE/SEI 7-05, Section 6.5.7.2 is equal to 1.0 (see SRP Section 3.3.2, SRP Acceptance Criteria, Item B)

K_d = *wind* directional factor as defined in ASCE/SEI 7-05, Section 6.5.4.4 is equal to 1.0 (see SRP Section 3.3.2, SRP Acceptance Criteria, Item B)

V = *basic* wind speed is equal to the maximum tornado/hurricane wind speed (see SRP Section 3.3.2, SRP Acceptance Criteria, Item B)

I = *importance* factor is equal to 1.15 (see SRP Section 3.3.2, SRP Acceptance Criteria, Item B)

Note 1: Velocity pressure values, q and q_i, are evaluated using either Equation. 3.3.2-2, 3, 4, or 5, as appropriate. The applicable velocity pressure exposure coefficient, K_z, equals 0.87 for tornado or values given in ASCE/SEI 7-05 Table 6-3 depending on the height above the ground level of the ASCE/SEI 7-05 for hurricane and the basic wind speed, V, is equal to the maximum tornado/hurricane wind speed (see SRP Section 3.3.2, SRP Acceptance Criteria, Item B). Pressure shall be applied simultaneously on windward and leeward walls and on roof surfaces as defined in ASCE/SEI 7-05, Figures. 6-6 and 6-8. Values of external and internal pressure shall be combined algebraically to determine the most critical load.

Note 2: Values of (GC_{pi}) used in Equation 3.3.2-1 are tabulated in ASCE/SEI 7-05, Figure 6-5. Plus and minus signs signify pressure acting towards and away from the internal surfaces, respectively. For enclosed buildings, the value of (GC_{pi}) in Equation 3.3.2-1 is either +0.18 or –0.18.

The first and second terms in Equation 3.3.2-1 represent the external and internal pressures, respectively. Values for (GC_{pi}) listed in ASCE/SEI 7-05, Figure 6-5 can be either negative or positive depending on the enclosure classification of the particular building. The sign convention for design wind loads on enclosed and partially enclosed buildings is specified in ASCE/SEI 7-05, Section 6.5.12.1.1. According to these sign convention rules, positive pressure acts toward the surface and negative pressure acts away from the surface. Determining the critical load for the appropriate conditions requires consideration of the following two cases.

1. A positive value of (GC_{pi}) applied to all internal surfaces.
2. A negative value of (GC_{pi}) applied to all internal surfaces.

Because Equation 3.3.2-6, which is used by the applicant and Equation 3.3.2-1, which is from ASCE/SEI 7-05, Section 6.5.12.2, are not identical, the equivalency between the two equations is not readily apparent. Therefore, the following confirmatory analysis was conducted to verify that the procedure used by the applicant for transforming the extreme wind speed into an equivalent pressure applied to buildings and structures yields the same results as ASCE/SEI 7-05, Method 2 described in ASCE/SEI 7-05, Section 6.5. This analysis is based on the following input parameters.

- The tornado/hurricane wind speed, V, is 230/160 mph (103/71.5 m/s) for both equations.
- The PCCV and R/B are enclosed structures.

- Exposure Category C applies to both equations.
- Topographic factor, K_{zt}, equals 1.0 for flat terrain (Exposure Category C) as described in ASCE/SEI 7-05, Section 6.5.7.2.
- Wind directional factor, K_d, equals 1.0 based on guidance in SRP Section 3.3.2.
- In the case of tornado, velocity pressure exposure coefficient, K_z equals 0.87 because tornado wind velocity and pressure are assumed not to vary with height above ground as described in RG 1.76, Revision 1.
- In the case of hurricane, velocity pressure exposure coefficient, K_z is given in Table 6-3 of the ASCE/SEI 7-05 because hurricane wind velocity and pressure are assumed to vary with height above ground as described in RG 1.221.
- The importance factor, *I*, for both equations equals 1.15.
- The gust-effect factor, *G*, equals 0.85 for rigid structures as described in ASCE/SEI 7-05, Section 6.5.8.1.

Substituting these input parameters into Equation3.3.2.-1 yields Equation 3.3.2-6, as shown below.

$$p = q\, GC_p - q_i\, (GC_{pi}) \qquad \text{Equation 3.3.2-1}$$

$$p = 0.00256\, K_z\, V^2\, 1.15\, GC_p - 0.00256\, K_z\, V^2\, 1.15\, (GC_{pi})$$

$$p = 0.00256\, K_z\, V^2\, 1.15\, (GC_p - (GC_{pi})) \qquad \text{Equation 3.3.2-6}$$

The resulting confirmatory analysis showed that the velocity pressure, *p*, determined using Equation 3.3.2-1 and Equation 3.3.2-6 for design wind loads on enclosed buildings as defined in ASCE/SEI 7-05, Section 6.5.12 are equivalent. Conversion of design pressure, *p*, determined by Equation 3.3.2-1 to tornado/hurricane wind loads is accomplished by simultaneously applying the design pressure on windward and leeward walls and on roof surfaces as defined in ASCE/SEI 7-05, Figures 6-6 and 6-8.

The internal pressure for the PCCV and the R/B, which are enclosed (unvented) structures, is not affected by atmospheric pressure changes resulting from passage of a tornado or hurricane. However, the pressure difference between the interior and the exterior surfaces of the PCCV and the R/B produces outward acting loads on the roof and walls of these structures. The internal pressure for the PS/Bs, which are vented buildings, vary as the atmospheric pressure changes resulting from passage of a tornado or hurricane.

According to RG 1.76, Revision 1, the magnitude of the pressure difference at a particular location varies because the maximum time rate of change of pressure is inversely proportional to the Rankine combined vortex radius and is directly proportional to the translational speed of the Rankine combined vortex. The maximum atmospheric pressure difference at a particular location occurs when the center of the tornado coincides with that location.

The maximum rotational tornado wind speed occurs at a radius of 150 ft (45.7m) from the tornado center. According to RG 1.76, Revision 1, the maximum tornado wind speed is the sum of the maximum tornado wind rotational speed and the maximum translational speed. When these two tornado characteristics are considered together, the combination of tornado wind effects, W_w, and atmospheric pressure change effects, W_p, depends on the location of the center of the tornado relative to the structural element being assessed.

Tornado characteristics, which are based on the Rankine combined vortex model referenced in RG 1.76, Revision 1, are summarized as follows.

- Rotational wind speed of a tornado, V_R, increases from zero at the tornado vortex to a maximum, V_{Rm}, at the tornado radius, R_m. Outside the tornado radius, the rotational speed decreases at a rate that is inversely proportional to the distance from the tornado vortex.
- Translational wind speed of a tornado, V_T, is the horizontal speed of the tornado.
- Rotational wind speed, V_R, and translational wind speed, V_T, of a tornado combine to produce the maximum tornado wind speed.
- Atmospheric pressure change effects, W_p, are most severe at the tornado vortex.
- Maximum time rate of change of pressure is inversely proportional to the Rankine combined vortex radius and is directly proportional to the translational speed of the Rankine combined vortex.
- Tornado wind velocities and pressures are assumed not to vary with height above ground.

Hurricane characteristics based on RG 1.221, on the other hand, are uniform and covered a broader area:

- Peak wind speed of a hurricane is uniform in horizontal as well as vertical directions.
- Hurricane wind speeds and pressures are assumed to vary with height above ground.
- No rotational wind speed.
- No atmospheric pressure change.
- No time rate of change of pressure.
- Hurricane-generated missile speeds are higher than tornado for a given type of missile and design-basis wind speed.

The applicant used guidance in SRP Section 3.5.3, “Barrier Design Procedures,” to evaluate the local structural effects of tornado or hurricane-generated missiles which are concentrated at the point of impact. Overall missile impact effects, W_m, including shear, flexural, and buckling effects are evaluated based on equivalent static loads concentrated at the impact area. These missile impact effects, W_m, are combined with tornado/hurricane wind effects, W_w, and atmospheric change effects, W_p, to establish the total wind load effect, W_t. The following two equations are used by the applicant to combine these three effects to determine the total wind load effect, W_t.

$W_t = W_p$ Equation 3.3.2-7
or
$W_t = W_w + 0.5\ W_p + W_m$ Equation 3.3.2-8
whichever is greater.

In determining maximum tornado or hurricane wind load effects on buildings, atmospheric pressure change effects, W_p, vary with time and are most severe at the point in a building where the tornado vortex is located. Because rotational wind speed is zero at the tornado vortex, missile impact effects and tornado wind effects, which are influenced by wind speed, do not occur at the tornado vortex. This phenomenon is reflected in Equation 3.3.2-7. For points in a building that are located at the tornado radius where the rotational wind speed is maximum, total wind load effect, W_t, includes a combination of tornado wind effects, W_w, atmospheric change effects, W_p, and missile impact effects, W_m. These phenomena are reflected in Equation 3.3.2-8.

The approach used by the applicant for considering the most adverse combination of total tornado/hurricane effects on the PCCV, R/B, and PS/Bs, which is based on application of these two equations, is consistent with guidance provided in SRP Section 3.3.2, Subsection II. To ensure that the combination of tornado/hurricane effects for the PCCV, R/B, and PS/Bs are established in a conservative manner, the applicant supplemented the approach described in SRP Section 3.3.2 with the design criteria and procedures provided in the Bechtel Topical Report BC-TOP-3-A, Revision 3, "Tornado and Extreme Wind Design Criteria for Nuclear Power Plants," issued August 1974. The applicant expanded the existing two combination equations to the six combination equations in Section 3.4 of the Bechtel Topical Report BC-TOP-3-A as listed below.

$W_t = W_w$
$W_t = W_p$
$W_t = W_m$
$W_t = W_w + 0.5\ W_p$
$W_t = W_w + W_m$
$W_t = W_w + 0.5\ W_p + W_m$

where

W_t = total tornado/hurricane load
W_w = load from tornado/hurricane wind effect
W_p = load from tornado atmospheric pressure change effect
W_m = load from tornado/hurricane missile impact effect

The supplementary design criteria and procedures used by the applicant are consistent with the corresponding design criteria and procedures endorsed in SRP Section 3.3.2.

Per COL Information Item 3.3(2), the COL applicant is responsible for applying this approach to the nonstandard, site-specific seismic Category I structures for US-APWR plants. These structures would include the power source fuel storage vault (PSFSV), essential service water pipe tunnel (ESWPT), and ultimate heat sink related structures (UHSRS).

The applicant provided information in DCD Tier 2, Section 3.3.2.3, "Effect of Failure of Structures or Components Not Designed for Tornado Loads," to demonstrate that limited failure of seismic Category II structures not designed for tornado/hurricane loads do not jeopardize the function and structural integrity of safety-related SSCs. This information pertains to the auxiliary building (A/B), turbine building (T/B), and access building (AC/B). Seismic Category II structures and components are committed by the applicant to be designed for the same tornado/hurricane wind loads as seismic Category I structures, in order to preclude impact on the function and integrity of safety-related SSCs. Limited failure of seismic Category II structures is acceptable provided that function and integrity of safety-related SSCs are not affected. The applicant also provided information in DCD Tier 2, Section 3.3.2.3 to demonstrate that failure of structures and components not designed for tornado/hurricane loads do not jeopardize the function and integrity of safety-related SSCs.

DCD Tier 1 Section 2.2 includes ITAAC to verify that that seismic Category I buildings can withstand design-basis loads, which is evaluated in Section 14.3.3 of this report. Therefore, there are no ITAAC specifically related to tornado and hurricane loadings, since tornado and hurricane loadings are included in the design-basis loads, Accordingly, the staff finds that the applicant meets the requirements of 10 CFR 52.47(b)(1) with regards to tornado and hurricane loadings.

3.3.2.4.3 Summary of Technical Review

The following conclusions are based on a technical evaluation of the tornado and hurricane load information submitted by the applicant for the standard plant design as set forth above in Section 3.3.2.4 of this report.

- The applicant used the single Rankine combined vortex model and the tabulated information in RG 1.76, Revision 1 to establish the design-basis tornado and tornado missile spectrum for the non-site-specific seismic Category I buildings and structures in the US-APWR standard plant design. Similarly, RG 1.221 is used for the case of hurricane. The non-site-specific seismic Category I buildings and structures in the US-APWR standard plant design include the PCCV, the R/B and the PS/Bs.

    a. The design-basis tornado parameters correspond to the tornado characteristic values for Region 1 listed in Table 1 of RG 1.76, Revision 1. These parameters are used as input to a single Rankine combined vortex model of a design-basis tornado. The tornado parameters for Region 1 envelope the design-basis tornado characteristics for the other two regions (Region 2 and Region 3) in the contiguous U.S. where tornado wind characteristics are less severe. The annual probability of exceedance of the design-basis tornado is 10^{-7} as discussed in RG 1.76 and the corresponding recurrence interval is approximately ten million years. Use of the single Rankine combined vortex model and the design-basis tornado characteristics described in RG 1.76 is consistent with guidance provided in SRP Sections 2.3.1 and 2.3.2 and acceptable to the staff. The design-basis hurricane parameters correspond to the hurricane characteristic values for coastline areas listed in Figures 1-3 of RG 1.221, Revision 0. The design-basis hurricane wind speed of 160 mph (71.5 m/s) is used as input to assess the design-basis hurricane load. The design-basis wind speed envelopes the design-basis hurricane characteristics for the most regions in the contiguous U.S. where hurricane wind characteristics are less severe. The annual probability of exceedance of the design-basis hurricane is 10^{-7}.

    b. The design-basis tornado/hurricane missile spectrum conforms to guidance provided in SRP Section 3.5.1.4 and RG 1.76, Revision 1 for Region I and RG 1.221, Revision 0. This spectrum envelope the missile spectra for the other regions in the contiguous U.S. where tornado/hurricane wind characteristics are less severe. Tornado or hurricane-generated missiles and their associated wind speeds are identified in Table 2 of RG 1.76, Revision 1 and Table 1 of RG 1.221. Use of this tabulated information for establishing the design-basis tornado/hurricane missile spectrum is acceptable to the staff.

- The applicant used the methods in ASCE/SEI 7-05 to transform extreme wind effects into design loads. These methods include procedures for transforming wind speed into an equivalent pressure on structures, selecting pressure coefficients corresponding to the structure's geometry and physical configuration, accounting for wind speed variation with height and direction, and applying applicable gust factors. Based on technical rationale in SRP Section 3.3.1, ASCE/SEI 7-05 is a well-established industry standard for evaluating wind loading on structures that is acceptable to the staff.

- The applicant used guidance provided in RG 1.76, Revision 1 to establish the pressure drop and the rate of pressure drop caused by the passage of a tornado to determine atmospheric pressure change effects. Based on technical rationale in SRP Section 3.3.2, this guidance for evaluating atmospheric pressure change effects on structures is acceptable to the staff.

- The applicant used guidance in SRP Section 3.5.3 to evaluate the local structural effects of tornado or hurricane-generated missiles which are concentrated at the point of impact. Overall missile impact effects including shear, flexural, and buckling effects are evaluated based on equivalent static loads concentrated at the impact area. Based on technical rationale in SRP Section 3.5.3, this guidance for evaluating tornado or hurricane-generated missile local structural effects is acceptable to the staff.

- The applicant used guidance in SRP Section 3.3.2 to combine missile impact effects, extreme wind effects, and atmospheric change effects to establish the total wind load effect. To ensure that the combination of tornado/hurricane effects for the PCCV, R/B, and PS/Bs are established in a conservative manner, the applicant supplemented the approach described in SRP Section 3.3.2 with design criteria and procedures that are consistent with the corresponding design criteria and procedures described in SRP Section 3.3.2, Acceptance Criteria Item C. Based on technical rationale in SRP Section 3.3.2, this guidance for combining tornado or hurricane effects is acceptable to the staff.

- The applicant provided information to demonstrate that limited failure of seismic Category II structures not designed for tornado/hurricane loads do not jeopardize the function and integrity of safety-related SSCs. Limited failure of seismic Category II structures is acceptable provided that function and integrity of safety-related SSCs are not affected. The applicant also provided information to demonstrate that failure of structures and components not designed for tornado or hurricane loads do not jeopardize the function and integrity of safety-related SSCs. This information is consistent with guidance provided in SRP Section 3.3.2, Subsection II.

- Information that defines the responsibilities of the COL applicant is provided in the following Section 3.3.2.5 "COL Information Items."

3.3.2.5 Combined License Information Items

The following is a list of COL item numbers and descriptions from Table 1.8-2 of the DCD related to tornado and hurricane loadings:

Table 3.3.2-1 US-APWR Combined License Information Items		
Item No.	**Description**	**Section**
COL 3.3(1)	The COL applicant is responsible for verifying the site-specific basic wind speed is enveloped by the determinations in this section.	3.3.1.1
COL 3.3(2)	These requirements also apply to seismic Category I structures provided by the COL applicant. Similarly, it is the responsibility of the COL applicant to establish the methods for qualification of tornado or hurricane effects to preclude damage to safety-related SSCs.	3.3.2.2.4
COL 3.3(3)	It is the responsibility of the COL applicant to assure that site-specific structures and components not designed for tornado or hurricane loads will not impact either the function or integrity of adjacent safety-related SSCs, or generate missiles having more severe effects than those discussed in Section 3.5.1.4.	3.3.2.3
COL 3.3(4)	The COL applicant is to provide the wind load design method and importance factor for site-specific Category I and Category II buildings and structures. The COL applicant shall also verify that the site location does not have features promoting channeling effects or buffeting in the wake of upwind obstructions that invalidate the standard plant wind load design methods described.	3.3.1.2
COL 3.3(5)	The COL applicant is to note the vented and unvented requirements of this section to the site-specific Category I buildings and structures.	3.3.2.2.3
COL 3.3(6)	The COL applicant is responsible for verifying that the site-specific design-basis hurricane basic wind speeds, exposure category, and resulting wind forces are enveloped by the determinations in this section.	N/A

The staff finds the above listing in the table concerning tornado and hurricane loads to be complete, pending closure of the confirmatory items. Also, the list adequately describes actions necessary for the COL applicant to take. No additional COL information items were identified that need to be included in DCD Tier 2, Table 1.8-2 regarding tornado and hurricane loadings.

Note that the second part of COL Information Item 3.3(4) is to be added in DCD Tier 2, Revision 4 as committed in the response to **RAI 817-5990, Question 03.03.02-5**. The applicant revised COL Information Items 3.3(2) and 3.3(3) to address hurricanes and added a new COL Information Item 3.3(6) in the supplemental response to **RAI 907-6321, Question 02-3**, which is linked to Item 4 of **RAI 908-6327 Question 03.03.02-6**. As discussed in Section 3.3.2.4 of this report above, **RAI 817-5990, Question 03.03.02-5, RAI 908-6327, Question 03.03.02-6, and RAI 907-6321, Question 02-3 are being tracked as Confirmatory Items**.

3.3.2.6 Conclusions

Based on the above, the staff concludes that the application meets the relevant requirements in GDC 2 and GDC 4 and 10 CFR 52.47(b)(1) for the non-site-specific seismic Category I buildings and that structures in the US-APWR standard plant design and the DCD Tier 2, Section 3.3.2, Revision 3 submittal with regards to tornado and hurricane loads are acceptable, pending resolution of the confirmatory items.

Table 3.3.2-2 - Seismic Classification of US-APWR Buildings and Structures (See Note 1)

Structure	Abbreviation	Seismic Category (See Note 2)
Reactor Building (See Note 3)	R/B	I
Prestressed Concrete Containment Vessel (See Note 3)	PCCV	I
Containment Internal Structure (See Note 3)	CIS	I
Power Source Building (East and West) (See Note 3)	PS/B	I
Power Source Fuel Storage Vault	PSFSV	I
Essential Service Water Pipe Tunnel (ESWPT) (from/to UHS)	ESWPT	I
UHS Related Structures	UHSRS	I
Auxiliary Building (See Note 3)	A/B	II
Turbine Building	T/B	II
Access Building (See Note 3)	AC/B	non-seismic
Outside Building (e.g., maintenance facility, operations office)	O/B	non-seismic
Turbine Generator Pedestal	T/G Pedestal	non-seismic

Notes:

1. Other non-standard plant building structures, such as minor non-seismic buildings and structures in the plant yard, are not listed in the above table and are not considered part of the US-APWR Nuclear Island.
2. Seismic Category I (I); Seismic Category II (II); non-seismic.
3. US-APWR nuclear island.

Table 3.3.2-3 - Comparison of Design-Basis Tornado Missile Spectra for Region I

Missile Type and Characteristic	US-APWR Applicant	Regulatory Guide 1.76, Revision 1, March 2007
Schedule 40 Pipe		
Dimensions	6.625 in. diameter by 15 ft long	6.625 in. diameter by 15 ft long
Mass, lbs.	287	287
Horizontal Velocity, ft/s	135*	56
Vertical Velocity, ft/s	90.5*	37.5
Automobile		
Dimensions	16.4 ft by 6.6 ft by 4.3 ft	16.4 ft by 6.6 ft by 4.3 ft
Mass, lbs.	4000	4000
Horizontal Velocity, ft/s	135	135
Vertical Velocity, ft/s	90.5	90.5
Solid Steel Sphere		
Dimensions	1 in. diameter	1 in. diameter
Mass, lbs.	0.147	0.147
Horizontal Velocity, ft/s	26	26
Vertical Velocity, ft/s	26*	17.4

* Exceeds design-basis tornado missile velocity in Table 2 of RG 1.76, Revision 1.

Table 3.3.2-4 - Evaluation of Method 2 Conditions for PCCV

Condition*	PCCV Evaluation	Condition Satisfied
1. The building is a regular-shaped building as defined in Section 6.2.	The PCCV is a vertical cylinder with a hemispherical dome with no unusual geometrical irregularity in spatial form.	Yes
2. The building does not have response characteristics making it subject to across wind loading, vortex shedding, instability due to galloping or flutter; and does not have a site location from which channeling effects or buffeting in the wake of upwind obstructions warrant special consideration.	The PCCV has a relatively low profile with an approximate overall height-to-diameter ratio of 1.5, and the PCCV is surrounded by the rectangular-shaped R/B such that approximately only the upper half of the PCCV is exposed to wind loading. The PCCV does not have response characteristics which make it subject to across wind loading, vortex shedding, or other unusual wind effects which might require investigation using Method 3 - Wind Tunnel Procedure of ASCE/SEI 7-05. The COL Applicant is responsible for verifying that the site location of the PCCV is such that channeling or buffeting effects do not require special consideration.(See COL Information Item 3.3(4))	Yes

*Conditions are defined in ASCE/SEI 7-05, Section 6.5.1.

Table 3.3.2-5 - Evaluation of Method 2 Conditions for R/B

Condition*	R/B Evaluation	Condition Satisfied
1. The building is a regular-shaped building as defined in Section 6.2.	The R/B has a rectangular shape with no unusual geometrical irregularity in spatial form.	Yes
2. The building does not have response characteristics making it subject to across wind loading, vortex shedding, instability due to galloping or flutter; and does not have a site location from which channeling effects or buffeting in the wake of upwind obstructions warrant special consideration.	The R/B is a low-profile building that does not have response characteristics which make it subject to across wind loading, vortex shedding, or other unusual wind effects which might require investigation using Method 3 - Wind Tunnel Procedure of ASCE/SEI 7-05. The COL Applicant is responsible for verifying that the site location of the R/B is such that channeling or buffeting effects do not require special consideration.(see COL Information Item 3.3(4))	Yes

*Conditions are defined in ASCE/SEI 7-05, Section 6.5.1.

Table 3.3.2-6 - Evaluation of Method 2 Conditions for PS/Bs

Condition*	R/B Evaluation	Condition Satisfied

1. The building is a regular-shaped building as defined in Section 6.2.	The PS/Bs have a rectangular shape with no unusual geometrical irregularity in spatial form.	Yes
2. The building does not have response characteristics making it subject to across wind loading, vortex shedding, instability due to galloping or flutter; and does not have a site location from which channeling effects or buffeting in the wake of upwind obstructions warrant special consideration.	The PS/Bs are low-profile buildings that do not have response characteristics which make it subject to across wind loading, vortex shedding, or other unusual wind effects which might require investigation using Method 3 - Wind Tunnel Procedure of ASCE/SEI 7-05. The COL Applicant is responsible for verifying that the site location of the PS/Bs is such that channeling or buffeting effects do not require special consideration.(see COL Information Item 3.3(4))	Yes

*Conditions are defined in ASCE/SEI 7-05, Section 6.5.1.

3.4.1 Internal Flood Protection

3.4.1.1 Introduction

Plant internal flooding protection includes measures to protect against or compensate for failures of all SSCs whose failures resulting in flooding could prevent safe shutdown of the plant or result in uncontrolled release of significant radioactivity. The facility design and equipment arrangements presented in the US-APWR DCD are reviewed with respect to both internal (e.g., pipe break, tank failure) and external (e.g., failure of exterior tanks) causes.

3.4.1.2 Summary of Application

DCD Tier 1: Tier 1 information associated with this section is found in DCD Tier 1, Section 2.2, "Structural and Systems Engineering."

DCD Tier 2: The applicant has provided in DCD Tier 2, Section 3.4.1, "Flood Protection," a description of features incorporated in the design of the plant to protect against internal flooding. DCD Tier 2, Section 3.4.1 also identifies the safety-related SSCs that require protection from internal flooding and describes measure used to protect those SSCs from internal flooding

DCD Tier 2, Section 3.4.1.3, "Flood Protection from Internal Sources," identifies sources of internal flooding to include the following:

- Earthquakes.
- Pipe breaks and cracks.
- Firefighting operations.
- Pump mechanical seal failures.

DCD Tier 2, Section 3.4.1.3 and DCD Tier 2, Subsection 3.4.1.5.3, "R/B Flooding Events Impacting PS/B," presents a summary of the internal flood evaluation results for the following plant areas:

- R/B (inside PCCV).

- R/B (outside PCCV).
- R/B - radiological controlled area (RCA).
- R/B - non-radiological controlled area (NRCA).
- A/B
- PS/Bs
- T/B

The US-APWR structures are designed for loads due to flooding. Design loads and load combinations consider both static and dynamic load effects for internal and/or external flooding.

ITAAC: The ITAAC associated with DCD Tier 2, Section 3.4.1 are given in DCD Tier 1, Section 2.2, Items 9 through 14 in Table 2.2-4, “Structural and Systems Engineering Inspections, Tests, Analyses, and Acceptance Criteria,” which indicates that inspections will be performed of the as-built flood prevention protection features.

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 for this area of review.

Cross-cutting Requirements (TMI, USI/ GSI, Op Ex): There are no cross-cutting requirements for this area of review.

US-APWR Interface Issues Identified in the DCD: See DCD Tier 2, Table 1.8-2, “Compilation of All Combined License Applicant Items for Chapters 1-19.”

Site Interface Issues Identified in the DCD: See DCD Tier 2, Table 1.8-1, “Significant Site Specific Interfaces with the Standard US-APWR Design.”

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

3.4.1.3 Regulatory Basis

The relevant requirements of the Commission’s regulations for this area of review, and the associated acceptance criteria, are given in Section 3.4.1, “Internal Flood Protection for Onsite Equipment Failures,” Revision 3, issued March 2007, of NUREG-0800 and are summarized below. Review interfaces with other SRP sections also can be found in Section 3.4.1 of NUREG-0800.

1. GDC 2, as it relates to SSCs important to safety being designed to withstand the effects of natural phenomena, such as earthquakes, tornados, hurricanes, floods, tsunami, and seiches without loss of capability to perform their safety functions. SSC design bases must reflect appropriate combinations of the effects of normal and accident conditions with the effects of the natural phenomena. The effects of normal and accident conditions considered could include evaluating the effects of flooding from full circumferential failures of non-seismic, moderate-energy piping, which is not considered in SRP Section 3.6.2, “Determination of Rupture Locations and Dynamic Effects Associated with the Postulated Rupture of Piping,” Revision 3, issued March 2007.

2. GDC 4, as it relates to SSCs important to safety being designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing and postulated accidents, including loss-of-coolant accidents (LOCAs).

3. 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 has been constructed 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 the following:

1. Guidance acceptable for meeting the seismic design and classification requirements of GDC 2 is found in Regulatory Guide (RG) 1.29, “Seismic Design Classification,” Revision 4, issued March 2007, Position C.1 for safety-related SSCs and Position C.2 for nonsafety-related SSCs.

2. With respect to flooding, the requirements of GDC 4 are met if SSCs important to safety are designed to accommodate the effects of discharged fluid resulting from high and moderate energy line breaks that are postulated in SRP Section 3.6.1 “Plant Design for Protection Against Postulated Piping Failures in Fluid Systems Outside Containment,” Revision 3, issued March 2007, and Section 3.6.2.

3.4.1.4 Technical Evaluation

The staff reviewed internal flood protection described in Tier 1 and Tier 2 of the DCD Revision 3, in accordance with the review procedures in SRP Section 3.4.1. The staff’s review addressed the overall design for plant internal flood protection, including safety-related SSCs whose failure as a result of flooding could prevent safe shutdown or result in an uncontrolled release of radioactivity. Conformance with the acceptance criteria of SRP Section 3.4.1 formed the basis for the evaluation of the plant’s internal flood protection with respect to the applicable regulations. The results and conclusions of the staff’s review are discussed below. The staff’s evaluation of the protection of external flood from plant site floods, precipitation, tsunami, and seiches is discussed in Sections 2.4 and 3.4.2 of this report. This evaluation addresses compliance with the SRP acceptance criteria listed in Section 3.4.1.3 above.

DCD Tier 2, Section 3.4, “Water Level (Flood) Design,” describes measures for protecting safety and nonsafety-related SSCs against the effects of flooding that can occur inside the plant. DCD Tier 2, Section 3.4.1.1, “Flood Protection for Safety and Nonsafety-Related Structures, Systems, and Components,” states that safety-related SSCs are protected from flooding by external and internal sources and list design features use to provide protection including separation of redundant trains of safety-related SSCs, protective barriers and enclosures, when necessary, placement of essential SSCs above internal flood levels. DCD Tier 2, Section 3.4.1.1 also states that protection from flooding of nonsafety-related SSCs is considered when the impact of the flooding of a nonsafety-related SSC could be a contributing factor to the flooding of safety-related SSCs or could result in an uncontrolled release of significant radioactivity.

3.4.1.4.1 System Design Considerations

3.4.1.4.1.1 SSCs That Need Protection Against Flooding

GDC 2 requires in part that “structures, systems, and components important to safety shall be designed to withstand the effects of natural phenomena such as …. floods… without loss of capability to perform their safety functions.” To verify that the standard plant is in conformance with the requirements of GDC 2, the staff reviewed the plant flood protection design, including all systems and components whose failure as a result of internal flooding could prevent safe shutdown of the plant or result in an uncontrolled release of significant radioactivity.

The design of the facility for internal flood protection was reviewed in accordance with SRP Section 3.4.1. Safety-related SSCs that should be protected against flooding were reviewed in accordance with SRP Section 3.4.1, Subsection I.1, and the location of the safety-related SSCs relative to the internal flood levels were reviewed in accordance with SRP Section 3.4.1, Subsection I.2. In this section of the SE, the plant design as specified in the DCD is reviewed to determine if all of the SSCs that need flood protection have been properly identified. The review of the protection of these SSCs is covered in Subsection 3.4.1.4.2 of this report.

SRP Section 3.4.1, Subsection I.1, states that the set of SSCs that should be protected from flooding should be reviewed. Therefore, the SSCs that need protection against flooding should be identified in the DCD. SRP Section 3.4.1, SRP Acceptance Criteria 1 states that acceptable guidance for meeting the seismic design and classification requirements of GDC 2 is provided in RG 1.29, Position C.1 for safety-related SSCs and Position C.2 for nonsafety-related SSCs.

The staff reviewed the SSCs that should be protected against flooding in accordance with Positions C.1 and C.2 of RG 1.29 and SRP Section 3.4.1, Subsection I.1. As stated in DCD Tier 1, Sections 1.2, “General Provisions,” and 3.1, “Design Description,” the standard plant design includes the following set of buildings and structures: R/B, which includes the PCCV and CIS; PS/Bs; A/B; T/B; AC/B); PSFSVs; and ESWPT. However, in DCD Tier 2, Revision 1, Section 3.8.4, “Other Seismic Category I Structures,” the applicant stated that the PSFSVs, the ESWPT, and the UHSRS were not part of the standard design. Thus, DCD Tier 1, Sections 1.2 and 3.1, and Tier 2 Section 3.8.4 provided conflicting information as to whether the PSFSVs and the ESWPT are included in the standard plant design. To support the staff’s review of the applicant’s internal flood protection, it is necessary that the applicant clearly identify the set of buildings and SSCs associated with the standard plant design. Accordingly, in **RAI 220-2058, Question 03.04.01-1**, the staff requested that the applicant clarify the set of SSCs that are part of the standard design.

In its response to **RAI 220-2058, Question 03.04.01-1,** dated April 8, 2009, the applicant stated that the PSFSVs and ESWPT are included as part of the standard plant design, though the final structural designs of the PSFSVs and ESWPT are based on the site-specific arrangement. The applicant stated that DCD Tier 1, Sections 1.2 and 3.1, and DCD Tier 2, Sections 3.8.4 and 3.8.6, “Combined License Information,” will be revised to explicitly indicate that the PSFSVs and ESWPT are included as part of the standard plant design. The staff confirmed that Revision 2 of DCD Tier 1 Sections 1.2 and 3.1, and Tier 2 Sections 3.8.4 and 3.8.6 were revised as committed in the RAI response. Accordingly, the staff finds that the applicant has adequately addressed the staff’s concern as to whether the PSFSVs and ESWPT were considered as part of the standard plant since the application now clearly identifies that the PSFSVs and ESWPT are included as part of the standard plant design.

However, the DCD did not explicitly identify SSCs within the PSFSVs and ESWPT that require protection from internal flood. Furthermore, the DCD did not describe how internal flood protection is achieved for these SSCs. According to DCD Tier 2, Table 3.2-2, "Classification of Mechanical and Fluid Systems, Components, and Equipment," and DCD Tier 2, Section 9.5.4.3, "Safety Evaluation," the PSFSVs contain seismic Category I SSCs, specifically fuel oil storage tanks for the GTGs. Also, DCD Tier 2, Table 3.2-2 indicates that the ESWPT contains some seismic Category 1 SSCs, including ESW valves. Accordingly, the staff closed as unresolved **RAI 220-2058, Question 03.04.01-1** and in follow-up **RAI 579-4481, Question 03.04.01-21,** the staff requested that the applicant (a) identify SSCs within the PSFSVs and ESWPT that require protection from internal flood, and (b) for those SSCs that require internal flood protection, describe how internal flood protection is achieved, including a description of instrumentation for flood detection

In its response to **RAI 579-4481, Question 03.04.01-21**, dated May 27, 2010, the applicant stated that the PSFSVs and ESWPT structural design are to be finalized based on site-specific arrangement, and that after finalization of the structural design, internal flooding protection for these site-specific SSCs are to be qualified by the COL applicant. The applicant also proposed, to revise DCD Tier 2, Section 3.4.3, "Combined License Information," to add COL Information Item 3.4(7), which would state, "The COL Applicant is responsible for the protection from internal flooding for those site-specific SSCs that provide nuclear safety-related functions or whose postulated failure due to internal flooding could adversely affect the ability of the plant to achieve and maintain a safe shutdown condition."

The staff has reviewed the applicant's response, including proposed COL Information Item 3.4(7), and found it to be acceptable because the addition of the new COL item to DCD Tier 2, Section 3.4.3 will ensure that the SSCs within the PSFSVs and ESWPT that need protection from internal flooding will be properly addressed by the COL applicant based on the applicable site-specific arrangements of these buildings. The staff has confirmed that DCD Tier 2 Section 3.4.3, has been revised, and COL Information Item 3.4(7) has been added to the combined license information items listed in the section. Accordingly, **RAI 579-4481, Question 03.04.01-21, is resolved**.

DCD Tier 2, Table 3.2-4, "Seismic Classification of Buildings and Structures," provides the designated seismic category of the major buildings and structures that are part of the US-APWR standard design. The PCCV, CIS, R/B, PS/B, PSFSV, ESWPT, and UHSRS are designed to the requirements of seismic Category I. The A/B and T/B are designed to meet seismic Category II requirements, while the AC/B is a non-seismic structure.

The containment structure, as discussed in DCD Tier 2, Subsections 1.2.1.2.2.1, "Defense-in-Depth," and 1.2.1.7.2.1, "Reactor Building (R/B)," consists of the PCCV and an annulus that encloses the containment penetration area. It is also indicated in the DCD that the containment annulus provides an airtight space between the PCCV and the R/B. The R/B, as indicated in DCD Tier 2, Section 3.4.1.3, is divided into a RCA and a NRCA. The RCA and NRCA are physically separated by concrete walls. The PS/B, which is described in DCD Tier 2, Subsection 1.2.1.7.2.2, "Power Source Buildings (PS/B)," and DCD Tier 2, Section 3.2.1.3, "Classification of Building Structures," consist of two buildings, PS/B (east) and PS/B (west), which are located next to the R/B. Each building contains emergency power sources and related support systems.

DCD Tier 1, Table 2.2-3, "Main Components Protected against External Floods, Internal

Floods and Internal Fires," and DCD Tier 2, Sections 3.4.1.3 and 3.4.1.5, "Evaluation of Internal Flooding," identify systems and component types that require protection from internal flood according to specific buildings or building areas. For the following buildings and SSC types requiring protection from flood are as follows:

1. **PCCV/CIS**: DCD Tier 2, Section 3.4.1.3 identifies systems to be protected within the PCCV are the RCS, the safety injection system (SIS), the residual heat removal system (RHRS), the containment spray system (CSS), and the containment boundary. Associated components of these systems that require flood protection are motor operated components, such as the emergency letdown line isolation valve, safety depressurization valves, and electric/instrumentation components, including the pressurizer backup heater. However, it did not appear that the DCD had provided a complete listing of SSCs inside the PCCV that should be protected. For example, equipment required for monitoring and actuating systems important to safety should be protected as indicated in Position C.1 of RG 1.29, Item (k). Also, Class 1E electrical systems that provide emergency power for functioning of plant features should be protected as indicated in Position C.1 of RG 1.29, Item (q). Accordingly, in **RAI 220-2058, Question 03.04.01-2**, the staff requested that the applicant provide a complete list of SSCs inside the PCCV that need flood protection.

   In its response to **RAI 220-2058, Question 03.04.01-2**, dated May 21, 2009, the applicant stated that the DCD Tier 2 will be revised to include a complete list of SSCs inside the PCCV that need flood protection. A copy of this list was included as part of the applicant's response. This list contains numerous equipment items, including equipment needed to support safety-related monitoring and actuating functions. While this list does not explicitly include entries for electrical interconnections among components (e.g., electrical cables), DCD Tier 2, Section 3.4.1.3 states that components to be protected from flooding include electric/instrumentation components.

   The staff has confirmed that Revision 2 of DCD Tier 2, Section 3.4.1 was revised as committed in the RAI response. DCD Tier 2, Appendix 3K, "Components Protected From Internal Flooding," which provides the location of components within the safety-related buildings, in comparison to maximum internal flood levels within the vicinity of the component was added to the DCD in Revision 2. Table 3K-1, "PCCV Components Protected From Internal Flooding," provides the list of components located in the PCCV. The staff has reviewed the information in Table 3K-1 and found it to be acceptable since all of the SSCs are located above flood elevation except the RV, and refueling water storage pits whose function will not be impacted by flooding, and the source, and lower power range neutron flux monitoring equipment, where the installation area of the components will not be flooded, except during a LOCA, and the functions of the components are not required during a LOCA. Therefore, the staff finds that the applicant has adequately addressed the staff concern regarding the identification of SSCs inside the PCCV and the measures taken to protect them from flooding. Accordingly, **RAI 220-2058, Question 03.04.01-2, is resolved.**

2. **RCA of the R/B**: Equipment items to be protected, based on information in DCD Tier 2, Section 3.4.1.3, are as follows: safety injection (SI) and core spray/residual hear removal (CS/RHR) pumps, elevation -8.0 meters (m) (-26 ft,

4 in.); CS/RHR HXs and safeguard component air handling units (AHU), elevation 1.1 m (3 ft 7 in.); containment isolation valves (CIVs) in piping penetration room's elevation 7.7 m (25 ft 3 in.); and annulus emergency exhaust filtration units and junction boxes and cables in the electrical penetrations rooms, elevation 15.3 m (50 ft 2 in.). DCD Tier 2 Subsection 3.4.1.5.2.1, "Radiological Controlled Area," also stated that equipment items to be protected at elevation 23.3 m (76 ft 5 in.) are "junction boxes and cables in electrical penetration room isolation valves." This statement was not clear. Accordingly, in **RAI 220-2058, Question 03.04.01-3** the staff requested that the applicant clarify this statement.

In its response to **RAI 220-2058, Question 03.04.01-3**, dated April 4, 2009, the applicant stated that the equipment items to be protected are junction boxes and cables associated with east and west PCCV electrical penetration areas. The applicant proposed to revise DCD Tier 2 Section 3.4.1.5.2.1 accordingly. The staff has confirmed that Revision 2 of DCD Tier 2, Section 3.4.1.5.2.1 was revised as committed in the RAI response. The staff finds the response acceptable since the junction boxes and cables to be protected has clearly been identified as those associated with the east and west PCCV electrical penetration areas. Accordingly, **RAI 220-2058, Question 03.04.01-3, is resolved**

The staff also found the DCD did not provide a complete listing of SSCs inside the RCA portion of the R/B that should be protected. For example, systems required for monitoring and actuating systems important to safety should be protected as indicated in Position C.1 of RG 1.29, Item (k). Accordingly, in **RAI 220-2058, Question 03.04.01-4**, the staff requested that the applicant provide a complete list of SSCs inside the RCA portion of the R/B that need flood protection.

In its response to **RAI 220-2058, Question 03.04.01-4,** dated May 21, 2009, the applicant stated that DCD Tier 2 will be revised to include a complete list of SSCs inside the RCA portion of the R/B that need flood protection. A copy of this list was included as part of the applicant's response. This list contains numerous equipment items, including equipment required to support safety-related monitoring and actuating functions. While this list does not explicitly include entries for electrical interconnections among components (e.g., electrical cables), DCD Tier 2, Section 3.4.1.3 states that components to be protected from flooding include electric/instrumentation components. It is also noted that the applicant's list of SSCs that need flood protection does not include the SFP pumps. However, as later described in the applicant's response to **RAI 220-2058, Question 03.04.01-15**, DCD Tier 2, Subsection 3.4.1.5.2.1 will be revised to identify the SFP pumps as equipment items to be protected.

The staff has confirmed that DCD Tier 2, Revision 2, Subsection 3.4.1.5.2.1 was revised as committed in the RAI responses. DCD Tier 2, Appendix 3K, which provides the location of components within the safety-related buildings, in comparison to maximum internal flood levels within the vicinity of the component was added to the DCD in Revision 2. DCD Tier 2, Table 3K-2, "R/B RCA Components Protected from Internal Flooding," provides the list of components located in the RCA of the R/B. The staff has reviewed the information in DCD Tier 2, Table 3K-2 and found it to be acceptable since all of the SSCs are protected against flooding by either being located above flood elevation, or are protected by water-tight doors and floor drain isolation valves against in-flow or flooding occurring outside its compartment. Accordingly, **RAI 220-2058, Question 03.04.01-4, is resolved**.

3. **NRCA of the R/B**: Equipment items to be protected are as follows: component cooling water system (CCWS) pump and heat exchangers, and emergency feedwater system pump trains (elevation -8.0 m (-26 ft 4 in.)); Class 1E electrical panels (elevation 1.1 m (3 ft 7 in.)); main control panel and Class 1E instrumentation and control (I&C) panels (elevation 7.7 m (25 ft 3 in.)); MCR AHUs and Class 1E electrical room AHUs (elevation 15.3 m (50 ft 2 in.)); and EFW pit instrumentation, isolation valves for main steam (MS) and main feedwater (MFW), and valves for MS depressurization (elevation 23.3 m (76 ft 5 in.)). However, the DCD did not appear to provide a complete listing of SSCs inside the NRCA portion of the R/B that should be protected. For example, all circuitry between the process and input terminals of actuator systems involved in protective actions should be protected as indicated in Position C.1 of RG 1.29, Item (j). Accordingly, in **RAI 220-2058, Question 03.04.01-5** the staff requested that the applicant provide a complete list of SSCs inside the NRCA portion of the R/B that need flood protection.

   In its response to **RAI 220-2058, Question 03.04.01-5**, dated May 21, 2009, the applicant stated that DCD Tier 2 will be revised to include a complete list of SSCs inside the NRCA portion of the R/B that need flood protection. A copy of this list was included as part of the applicant's response. This list contains numerous equipment items, including actuation system cabinets and instrumentation. While this list does not explicitly include entries for electrical interconnections among components (e.g., electrical cables), DCD Tier 2, Section 3.4.1.3, states that components to be protected from flooding include electric/instrumentation components.

   The staff has confirmed that DCD Tier 2, Revision 2 Section 3.4.1 was revised as committed in the RAI response. DCD Tier 2, Appendix 3K, which provides the location of components within the safety-related buildings, in comparison to maximum internal flood levels within the vicinity of the component was added to the DCD in Revision 2. DCD Tier 2, Table 3K-3, "R/B NRCA Components Protected From Internal Flooding," provides the list of components located in the NRCA of the R/B. The staff has reviewed the information in DCD Tier 2, Table 3K-3 and found it to be acceptable since all of the SSCs whose functions can be impacted by flooding are protected against flooding by either being located above flood elevation, or are protected by water-tight doors and floor drain isolation valves against in-flow or flooding occurring outside its compartment. The component cooling water (CCW) surge tank and EFW pits are below flood elevations but their functions will not be impacted by flooding. The MFW isolation valves are located below flood elevation but are not required for safe shutdown and the containment isolation is maintained due to installation areas of these components are not flooded during LOCA. Accordingly, **RAI 220-2058, Question 03.04.01-5, is resolved**.

4. **PS/B**: The DCD did not explicitly identify safety-related SSCs located within the PS/B that need protection from internal flood. However, as indicated in DCD Tier 2, Table 3.2-2, the PS/B contains some seismic Category I SSCs, including ESW

valves, essential chiller pumps, and miscellaneous equipment related to the emergency gas turbines. Accordingly, in **RAI 220-2058, Question 03.04.01-6,** the staff requested that the applicant provide a complete list of SSCs inside the PS/B that need flood protection. In its response to **RAI 220-2058, Question 03.04.01-6**, dated May 21, 2009, the applicant stated that the DCD Tier 2 will be revised to include a complete list of SSCs inside the PS/B that require flood protection. A copy of this list has been included as part of the applicant's response. This list contains numerous equipment items, including the Class 1E turbine generators and Class 1E batteries.

The staff has confirmed that DCD Tier 2, Revision 2, Section 3.4.1, was revised as committed in the RAI response. DCD Tier 2, Appendix 3K, which provides the location of components within the safety-related buildings, in comparison to maximum internal flood levels within the vicinity of the component was added to the DCD in Revision 2. DCD Tier 2, Table 3K-4, "PS/B Components Protected From Internal Flooding," provides the list of components located within the PS/B. The staff has reviewed the information in DCD Tier 2, Table 3K-4 and found it to be acceptable since all of the SSCs, that function can be impacted by flooding, are protected against flooding by either being located above flood elevation, or are protected by water-tight doors against in-flow or flooding occurring outside its compartment. Since the design incorporates train/component redundancy, protection against flooding occurring inside the compartment is not required.

However, the staff found that the DCD failed to identify the need to protect electrical interconnections among components (e.g., cables) within the PS/B. Accordingly, the staff closed as unresolved **RAI 220-2058, Question 03.04.01-6,** and in follow up **RAI 579-4481, Question 03.4.1-22**, the staff requested that the applicant identify electrical interconnections among components within the PS/B that require protection from internal flood**.**

Following its response to **RAI 579-4481, Question 03.04.01-22**, dated May 27, 2010, the applicant submitted an amended response, dated December 9, 2010. In its amended response, the applicant stated that the complete list of SSCs inside the PS/B that require flood protection had already been provided in DCD Tier 2, Revision 2, Table 3K-4. It also stated that the cables, including the power cable, the control cable, and instrument cable are routed in cable trays that are elevated above the floor in compartments and corridors of the PS/B, and that the electrical interconnections (cables) on level B1F (elevation -26'-4") are located above the applicable flood levels, and protection is provided for level 1F (elevation 3'7") by separation and redundancy of other trains/components.

The staff has reviewed the applicant's response and found it to be acceptable because the safety-related electrical cables in the PS/B will be routed so that they would not be subject to submergence in a flood event in the PS/B. The confirmation of equipment location relative to flood levels, the proper incorporation of divisional flood barriers, water-tight doors and water-tight seals relied on for flood protection of the SSCs in the PS/B will be verified by ITAAC, Items 9 through 12 in DCD Tier 1, Table 2.2-4 . Accordingly, **RAI 579-4481, Question 03.04.01-22, is resolved**.

5. **A/B**: DCD Tier 2, Section 3.4.1.3 states that the A/B does not contain any SSCs that need flood protection.

6. **T/B**: DCD Tier 2, Section 3.4.1.3 states that the T/B does not contain any SSCs that need flood protection.

7. **AC/B**: As indicated in DCD Tier 1, Section 2.2.1.9, "Access Building (AC/B)," DCD Tier 2, Section 1.2.1.7.2.6, "Access Building (AC/B)," and DCD Tier 2, Table 3.2-4, the AC/B is a non-seismic structure that contains the access control area, along with the chemical sampling and laboratory area. These SSCs do not need flood protection.

Containment Annulus: The containment annulus houses containment electrical and mechanical penetration areas. Mechanical penetrations include piping systems containing water. Flooding might occur following a break in these piping systems. The DCD did not explicitly identify safety-related SSCs located within the containment annulus that require protection from internal flood. Accordingly, in **RAI 220-2058, Question 03.04.01-7,** the staff requested that the applicant provide this information.

In its response to **RAI 220-2058, Question 03.04.01-7**, dated May 21, 2009, the applicant stated that there are piping and electrical penetration rooms located within the containment annulus, and that these rooms are considered part of the RCA portion of the R/B. As previously described in the applicant's response to **RAI 220-2058, Question 03.04.01-4**, DCD Tier 2 was revised to include a complete list of SSCs inside the RCA portion of the R/B that require flood protection. This list contains equipment items associated with the penetration rooms. While this list does not explicitly include entries for electrical cables, DCD Tier 2, Section 3.4.1.3 states that components to be protected from flooding include electric/instrumentation components. On the basis of its review of the applicant's response, and the review of **RAI 220-2058, Question 03.04.01-4,** which response was evaluated with the above review of the RCA of the RB, the staff finds that the concerns identified in **RAI 220-2058, Question 03.04.01-7, are resolved**.

Systems used to support SFP cooling should be protected from flooding, as indicated in Position C.1 of RG 1.29, Items (d) and (g). Important components associated with cooling the SFP are located in the R/B, for example the SFP pumps. However, the DCD did not appear to specifically address internal flood protection for SSCs associated with cooling of the SFP. Accordingly, in **RAI 220-2058, Question 03.04.01-8,** the staff requested that the applicant provide a complete list of SSCs associated with SFP cooling that need flood protection.

In its response to **RAI 220-2058, Question 03.04.01-8**, dated May 21, 2009, the applicant stated that the A and B train SFP pumps, which are located within the RCA portion of the R/B, need protection from internal flooding. As later described in the applicant's response to **RAI 220-2058, Question 03.04.01-15**, DCD Tier 2 Section 3.4.1.5.2.1 will be revised to identify the SFP pumps as equipment items to be protected from internal flooding. The staff has confirmed that, Tier 2, Revision 2, Section 3.4.1.5.2.1 was revised as committed in the RAI response. On the basis of its review of the applicant's responses to **RAI 220-2058, Question 03.04.01-8** and **RAI 220-2058, Question 03.04.01-15**, the staff finds that the concerns identified in **RAI 220-2058, Question 03.04.01-8 are resolved** since the SFP SSCs that need flood protection are now identified in the DCD.

Based on the above review, the staff concludes that the SSCs the applicant has indicated as being protected from flooding are consistent with what are specified in SRP Section 3.4.1, and the design meets the guidelines of Positions C.1 and C.2 of RG 1.29, and the requirements of GDC 2, as they relate to the identification of SSCs that require protection from internal flood.

3.4.1.4.1.2 Protection of Safety-Related SSCs from Flooding

The staff reviewed how safety-related SSCs located in buildings and structures associated with the standard plant design are protected from internal flooding. These reviews were based on compliance with GDC 2 and GDC 4. GDC 2 relates to the SSCs important to safety being designed to withstand the effects of flooding from full circumferential failures of non-seismic, moderate-energy piping. GDC 4 relates to the SSCs important to safety being designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing and postulated accidents, including LOCAs. This portion of the SE specifically focuses on how safety-related SSCs are protected from flooding.

Physical Separation and Building Layout: The US-APWR layout was reviewed to the extent of detail given in the DCD. That review determined if there is reasonable evidence that safety-related SSCs are protected against flooding from both external and internal causes. The location of the safety-related SSCs relative to the internal flood level in various buildings, rooms, and enclosures was reviewed.

DCD Tier 2, Table 3.2-2, and DCD Tier 2, Table 3D-2, “US-APWR Environmental Qualification Equipment List,” do not provide a building elevation for each SSC. However, DCD Tier 2, Appendix 3K, which was added in DCD Revision 2, provides the location of components within the safety-related buildings, in comparison to maximum internal flood levels within the vicinity of the components. Also, DCD Tier 2, Table 8.3.1-9, “Electrical Equipment Location List,” does provide building elevation information for major electrical components. Furthermore, some of the major components that need flood protection, including components associated with safe shutdown, are displayed in building layout diagrams, specifically DCD Tier 2, Figures 1.2-1, “Typical US-APWR Site Arrangement Plan,” through 1.2-51, “Access Building Sectional Views A-A and B-B,” and DCD Tier 2, Figures 9A-1, “Fire Zones and Fire Areas R/B EL -26'-4" (B1F),” through 9A-27, “Fire Zones and Fire Areas ESW Piping Tunnel and Power Source Fuel Storage Vault.” These diagrams show major compartments and divisional barriers that provide flood protection. Other related discussions are provided in DCD Tier 2 Sections 1.2, “General Plant Description,” 3.8, “Design of Category I Structures,” and 9.5, “Other Auxiliary Systems.”

The R/B utilizes internal walls and/or watertight doors as a means to achieve divisional separation to protect SSCs against flooding as described in DCD Tier 2, Section 3.4.1.5.2, “Reactor Building Flooding Events.” As previously noted, the PS/B is divided into two separate structures that are located on the east and west side of the R/B. These portions of the PS/B are referred to as PS/B (east) and PS/B (west), respectively. DCD Tier 1, Table 2.2-4 indicates that the interiors of these two PS/B structures also utilize divisional separation. The PCCV/CIS does not have divisional separation. Also, it did not appear that the containment annulus had divisional separation.

The following paragraphs describe the staff’s review of flood protection features for individual structures within the standard plant design.

PCCV/CIS: The PCCV/CIS is a structure without divisional separation. The PCCV/CIS layout was reviewed to determine if it allows water released inside the building to flow to the lowest levels of the building.

DCD Tier 2 Subsection 3.4.1.5.1, "PCCV Flood Events," describes the flood events in the PCCV. The PCCV is designed so that water discharged from a damaged component or spray water from the CSS is collected in the refueling water storage pit (RWSP), which is located at the bottom of the PCCV at elevation 1.1 m (3 ft 7 in.). The RWSP is the source of borated-water for emergency core cooling system (ECCS) and CSS. The RWSP can be visualized as roughly forming a horseshoe-shaped box around the containment perimeter.

While portions of the PCCV above the RWSP are segregated or compartmentalized, there are multiple pathways for water flow into the RWSP. To illustrate these design features, it is useful to consider major floor elevations above the RWSP. The 7.7 m (25 ft 3 in.) elevation (second floor) separates the RWSP and the balance of the PCCV. The 15.3 m (50 ft 2 in.) elevation (third floor) is a concrete floor aligned with the middle doorway into each of the steam generator (SG) compartments. The 23.3 m (76 ft 5 in.) elevation (fourth/operating floor) is a concrete floor aligned with the upper doorway into each of the SG compartments. Partitions located inside the PCCV include the four SG compartments, the pressurizer compartment, the regenerative HX room, the letdown HX room, the excess letdown HX room, the refueling cavity, the HVAC header compartment, the PCCV drain pump room, and the reactor cavity. On the 23.3 m (76 ft 5 in.) and 15.3 m (50 ft 2 in.) elevations, stairwells and the equipment hatch provide major pathways for water flow to the PCCV drain room.

Floor drains located in the regenerative heat exchanger room, the letdown heat exchanger room, and the excess letdown heat exchanger room provides a means for water to flow from upper PCCV levels into the PCCV drain pump room at elevation 1.1 m (3 ft 7 in.). In the event the PCCV sump is full or the floor drains become clogged, water will flow from the respective entrance doorways of these rooms and reach alternate pathways described above to reach the PCCV drain room. The flooding analysis does not take credit for water removal or water transport via the floor drains.

Water discharged from SG compartments flows across the floor of the HVAC header compartment to the drain line into the reactor cavity. Once the water level in the reactor cavity comes to equilibrium with water in the SG compartments, the SG compartment water level rises until it exceeds the 7.7 m (25 ft 3 in.) elevation. At this point, flood water from the SG compartment flows out through the doorway in the secondary shield wall to the floor at the 7.7 m (25 ft 3 in.) elevation.

Water that flows onto the 7.7 m (25 ft 3 in.) elevation subsequently flows into the RWSP, through a set of ten 18 in. (46 cm) transfer pipes. As discussed in DCD Tier 2, Section 6.2.2.2.5, "Refueling Water Storage Pit," the transfer piping serves to the replenishment function necessary for the ECCS to perform its safety function and are protected from large debris by vertical interceptor bars and are capped by a ceiling plate. The pipes extend through the RWSP ceiling, ending as openings into the containment floor at the 7.7 m (25 ft, 3 in.) elevation.

The PCCV flooding analysis evaluates the maximum flooding event, which is identified as a LOCA. However, the DCD did not explain how the worst case flooding source was determined. There appears to be other potential sources of flood water inside the PCCV, for example CCW and fire water as indicated in DCD Tier 2, Table 6.2.4-3, "List of Containment Penetrations and System Isolation Positions." Accordingly, in **RAI 220-2058, Question 03.04.01-9**, the staff requested that the applicant explain how the worst case flooding source was determined for the PCCV/CIS.

In its response to **RAI 220-2058, Question 03.04.01-9**, dated May 21, 2009, the applicant provided additional justification that the maximum flooding event would be a LOCA. The applicant excluded the CCWS as a potential flood source, given that CCW piping located inside the PCCV is classified as seismic Category I. The applicant also excluded the fire protection system as a potential flood source because CIVs outside the PCCV are normally closed. Furthermore, even though the reactor coolant pump (RCP) purge water head tank and containment vessel (CV) reactor coolant drain tank are non-seismic components, they were excluded as potential flood sources given that their combined water volume is significantly less than the amount of water potentially released from a LOCA. Guidance in SRP Section 3.4.1 states that for the purpose of flood analysis it is only necessary to assume, for each analyzed area, the rupture of the single worst-case pipe (or non-seismic tank/vessel). The staff reviewed the above information provided by the applicant on the justification of their selection of the LOCA for being the maximum flood event and found it to be acceptable since the CCW piping is seismic Category 1 and not assumed to fail, and the inventory associated with the non-seismic tanks are significantly less than the inventory released during a LOCA. The applicant stated that DCD Tier 2, Subsection 3.4.1.5.1 will be revised to include this additional information. The staff has confirmed that DCD Tier 2, Revision 2, Subsection 3.4.1.5.1 was revised as committed in the RAI response. Accordingly, **RAI 220-2058, Question 03.04.01-9, is resolved**.

The volume of water from the LOCA was conservatively assumed to include the RCS volume, the combined volume of the four accumulator tanks, and the volume of the RWSP. The total volume of water represented by these sources is 3,200 m^3 (113,000 ft^3). By comparison, the total water storage volume available below the 7.7 m (25 ft 3 in.) elevation is 3596 m^3 (127,000 ft^3). The DCD states that components sensitive to flooding that are required to function are located above this elevation. However, the DCD did not identify the locations of safety-related SSCs inside the PCCV relative to the internal flood level. Accordingly, in **RAI 220-2058, Question 03.04.01-10**, the staff requested that the applicant provide this information.

In its response to **RAI 220-2058, Question 03.04.01-10**, dated May 21, 2009, the applicant stated that DCD Tier 2 will be revised to include a complete list of SSCs inside the PCCV that need flood protection and the location of these SSCs relative to the flood level. A copy of this list has been included as part of the applicant's response. This list indicates that most of the SSCs that need flood protection are located above the flood level. In those instances where a SSC is not located above the flood level, this list states that the corresponding SSC function is not disabled, or that the SSC function is not required for the specific flood event (LOCA) that results in immersion. The list also reports maximum flood levels as measured above the nominal floor elevations of each SSC. For each SSC, the maximum flood level is reported to be 7.7m (25 ft 3 in.) above the nominal floor elevation, even though the SSCs are distributed among several different floor elevations. Based on a review of the PCCV interior layout and the discussion provided in DCD Tier 2, Subsection 3.4.1.5.1, it did not appear that the maximum flood level would always be 7.7 m (25 ft 3 in.) above each nominal floor elevation. Accordingly, the staff closed as unresolved **RAI 220-2058, Question 03.04.01-10,** and in follow-up **RAI 579-4481, Question 03.04.01-23**, the staff requested that the applicant explain why the maximum flood levels were 7.7 m (25 ft 3 in.) above the nominal floor elevation for each SSC.

In its response to **RAI 579-4481, Question 03.04.01-23**, dated May 27, 2010, the applicant stated that the format of DCD Tier 2, Table 3K-1 would be modified to verify whether each SSC is above the maximum flood elevation of 7.7m (25 ft 3 in.) or not by adding the elevation of each SSC, and adding a column to the table giving the SSCs level relative to the flood elevation.

The staff has reviewed the applicant's response and found it to be acceptable because the applicant's response, which includes a DCD markup of DCD Tier 2, Table 3K-1, now provides a clear indication as to whether each SSC is above or below the PCCV flood elevation. The staff has confirmed that DCD Tier 2, Revision 3, Section 3.4.1 was revised as committed in the RAI response. Accordingly, **RAI 579-4481, Question 03.04.01-23, is resolved**.

Containment Annulus: As previously noted, the containment annulus houses containment penetrations, including penetrations for piping systems. Flooding might occur following a break in these piping systems. However, the DCD did not include a flooding analysis for the containment annulus. Accordingly, in **RAI 220-2058, Question 03.04.01-11,** the staff requested that the applicant provide a flooding analysis for the containment annulus, including a description of instrumentation for flood detection.

In its response to **RAI 220-2058, Question 03.04.01-11**, dated May 21, 2009, the applicant stated that the containment annulus is considered part of the RCA portion of the R/B. (As described later in this SE, DCD Tier 2 provides a flood analysis of the RCA portion of the R/B). The applicant further stated that DCD Tier 2 will be revised to include a complete list of SSCs inside the RCA portion of the R/B (including the containment annulus) that need flood protection and the location of these SSCs relative to the flood level. A copy of this list has been included as part of the applicant's response. This list indicates that most of the SSCs that need flood protection are located above the flood level. In those instances where a SSC is not located above the flood level, this list states that the SSC is located within a water-tight compartment that is in turn protected from the in-flow of water by means of a water-tight door and floor drain isolation valves. Further, these SSCs do not need flood protection within their individual compartments, given that the design includes redundant trains and components. The staff has confirmed that DCD Tier 2, Revision 2, Section 3.4.1 was revised as committed in the RAI response. Therefore the staff's concern about the flood analysis for the containment annulus has been adequately addressed since the applicant has indicated that the R/B RCA includes the annulus, and the flood protection for the area is included in DCD Tier 2, Table 3K-2 of the Appendix 3K added in DCD Revision 2.

While the applicant's RAI response provided the requested information on the flood analysis for the annulus, the applicant did not provide a description of instrumentation for flood detection, as was requested in **RAI 220-2058, Question 03.04.01-11**. Accordingly, the staff closed **RAI 220-2058, Question 03.04.01-11,** as unresolved and in follow-up **RAI 579-4481, Question 03.04.01-24**, the staff requested that the applicant describe and demonstrate the adequacy of instrumentation for flood detection within the containment annulus.

In its response to **RAI 579-4481, Question 03.04.01-24**, dated June 21, 2010, the applicant stated that instrumentation designed to alarm when the annulus compartment is flooded will be installed in the containment annulus. The staff has reviewed the applicant response and found it to be acceptable because the design includes a means to detect flooding in the annulus and alert operators of the flooding. The staff confirmed that DCD Tier 2, Revision 3, Section 3.4.1.3 was revised as committed in the RAI response. Accordingly, **RAI 579-4481, Question 03.04.01-24, is resolved**.

As discussed earlier, the staff has also requested through **RAI 220-2058, Question 03.04.01-7,** that the applicant provide a complete list of SSCs inside the containment annulus that need flood protection. As described in the applicant's response to **RAI 220-2058, Question 03.04.01-7**, DCD Tier 2 will be revised to include a complete list of SSCs inside the containment annulus that need flood protection will be provided in DCD Revision 2. The staff has confirmed that Revision 2 of DCD Tier 2, Revision 2, Section 3.4.1, was revised as committed in the RAI response**.** Accordingly, **RAI 220-2058, Question 03.04.01-7, is resolved**.

R/B: DCD Tier 2, Section 3.4.1.5.2 and DCD Tier 2, Section 9.3.3, "Equipment and Floor Drainage Systems," describe the R/B flooding events. The R/B is designed as a divisional building. All floors within both the RCA and NRCA portions of the R/B are divided into two main areas (east and west) by means of concrete walls and/or watertight doors. Furthermore, flooding between the RCA and NRCA is precluded by concrete barrier walls. The east portions of the RCA and NRCA contain equipment associated with system trains A and B, while the west portions of the RCA and NRCA contain equipment associated with trains C and D.

A series of flooding analyses were performed by the applicant for both the RCA and NRCA. These analyses considered several types of flooding sources, specifically seismic-induced rupture of non-seismic components, high-energy and moderate energy pipe breaks, and cracks and firefighting operations. In each case, as stated in DCD Tier 2, Subsections 3.4.1.5.2.1 and 3.4.1.5.2.2, "NRCA," equipment to be protected from flooding is "located at heights above the level of flood water." However, this statement could not be confirmed. While the DCD provides elevation data for selected safety-related components located in the RCA that are to be protected from flooding, the cited equipment elevations are nominally below the maximum RCA flood levels (see DCD Tier 2 Section 3.4.1.5.2.1 and the prior discussion in this SE). Similarly, cited equipment elevations for safety-related components located in the NRCA that are to be protected from flooding are nominally below the maximum NRCA flood levels (see DCD Tier 2 Section 3.4.1.5.2.2 and prior discussion in this SE). Accordingly, in **RAI 220-2058, Question 03.04.01-12,** the staff requested that the applicant demonstrate how the safety-related equipment items in the RCA and NRCA are located above the internal flood levels.

In its response to **RAI 220-2058, Question 03.04.01-12**, dated May 21, 2009, the applicant stated that DCD Tier 2 will be revised to include complete lists of SSCs inside the RCA and NRCA portions of the R/B that require flood protection and the location of these SSCs relative to the flood level. These lists were provided as DCD Tier 2, Tables 3K-2 and 3K-3 in response to the **RAI 220-2058, Questions 03.04.01-4**, and **03.04.01-5**. These lists indicate that most of the SSCs that require flood protection are located above the flood level. In those instances where a SSC is not located above the flood level, these lists provide justification for this situation. For example, there are instances where immersion of a SSC does not disable its function. Alternatively, there are instances where a SSC function is not required for the specific flood event (LOCA) that results in immersion. Additionally, some SSCs are located within water-tight compartments that are in turn protected from the in-flow of water by means of water-tight doors and floor drain isolation valves. These SSCs do not require flood protection within their individual compartments, given that the design includes redundant trains and components.

The staff has confirmed that of DCD Tier 2, Revision 2, Section 3.4.1 was revised as committed in the RAI response. DCD Tier 2, Appendix 3K, which provides the location of components within the safety-related buildings in comparison to maximum internal flood levels within the vicinity of the component, was added to the DCD in Revision 2. DCD Tier 2, Tables 3K-2, and 3K-3 provide the list of components located inside the RCA and NRCA portion of the R/B. The staff has reviewed the information in DCD Tier 2, Tables 3K-2, and 3K-3 and found them to be acceptable since all of the SSCs whose functions can be impacted by flooding are protected against flooding by either being located above flood elevation, or are protected by water-tight doors against in-flow or flooding occurring outside its compartment as discussed above. Since the design incorporates train/component redundancy, protection against flooding occurring inside the compartment is not required. Accordingly, **RAI 220-2058, Question 03.04.01-12,** is resolved.

At elevation -8.0 m (-26 ft 4 in.) of the RCA the applicant has identified the worst case flood on the west side of the RCA is a high-energy line break (HELB) involving the chemical volume and control system (CVCS), while the worst case flood on the east side of the RCA is an earthquake followed by firefighting operations to suppress an earthquake-induced fire. In the west and east sides of the RCA at elevation -8.0 m (-26 ft 4 in.), the maximum flood levels are 0.85 m (2.79 ft) and 0.45 m (1.49 ft) above the floor, respectively. At elevation 1.1 m (3 ft 7 in.) of the RCA, the worst case floods involve an earthquake followed by firefighting operations to suppress an earthquake-induced fire. Even without credit for drainage into the basement, the maximum flood levels above floor level are 0.27 m (0.87 ft) (west side) and 0.20 m (0.67 ft) (east side). Similar flood calculations were also performed for RCA elevations 7.7 m (25 ft 3 in.), 15.3 m (50 ft 2 in.), and 23.3 m (76 ft 5 in.), where in each case, the worst case flood results from the combination of an earthquake and firefighting operations. Watertight doors are credited for protection of equipment in the RCA, including CS/RHR and SI pumps on the -8.0 m (-26 ft 4 in.) elevation, and the CS/RHR HXs and safeguard component AHUs on the 1.1 m (3 ft 7 in.) elevation. The quantity of flood water associated with fire-fighting activities (113562 liters (30,000 gallons) or 113.6 m^3 (4,010 cubic feet)) is based on operation of two hose stations for two hours, assuming a 473 L/min (125 gpm) flow rate per hose station.

A comparable set of flood calculations were performed for flooding events in the NRCA for the following elevations: -8.0 m (-26 ft 4 in.), 1.1 m (3 ft 7 in.), 7.7 m (25 ft 3 in.), 15.3 m (50 ft 2 in.), and 23.3 m (76 ft 5 in.). Except for the 23.3 m (76 ft 5 in.) elevation, the worst case flood always results from the combination of an earthquake and firefighting operations. The 23.3 m (76 ft 5 in.) elevation is divided and includes a MS/feedwater (FW) piping area that is separated from other NRCA areas by concrete walls and water tight doors. The bottom of the MS/FW piping area is at the 19.8 m (65 ft) elevation. The worst case flood for the MS/FW piping area is rupture of feedwater piping, while the worst case flood for the remainder of the NRCA is a combination of an earthquake and firefighting operations. Water released during rupture of feedwater piping in the MS/FW piping area would be confined within the MS/FW piping area, given that doorways to the MS/FW piping area are located above flood level. The fuel handling area is located at the 23.3 m (76 ft 5 in.) elevation of the RCA. The fuel handling area is separated from the other portions of the RCA through the use of water tight doors at walkway and stairwell openings. Watertight doors are also credited for protection of equipment in the NRCA, including the Class 1E electrical panels on the 1.1 m (3 ft 7 in.) elevation and the MCR and Class 1E I&C rooms on the 7.7 m (25 ft 3 in.) elevation.

There are two types of drain systems used in the R/B and outside the PCCV. More specifically, there is an equipment drain system and a room/compartment drain system for the RCA and a corresponding set of drain systems for the NRCA. Water from these drainage sources flows into sump tank systems located at elevation -8.0 m (-26 ft 4 in.) (basement first floor). Drainage from the RCA is routed into a radioactive sump tank system, while the drainage from the NRCA is routed into a non-radioactive sump tank system. Within both the RCA and NRCA, the respective floor drains in the east and west sides are connected and ultimately drain into their corresponding sump tank systems. DCD Tier 2, Section 3.4.1.5.2.1 states that there is no cross-connection between east areas drains and west area drains. Since there is no cross-connection, there will be no drainage flow between the east and west areas of the reactor building. DCD Tier 2, Revision 1, Subsection 9.3.3.1.1, “Safety Design Bases,” indicated that normally closed manual isolation valves installed in individual drainage pathways of engineered safety feature (ESF) equipment rooms preclude backflow of water into these rooms via the sump system. Based on the design information for floor drain system in DCD Tier 2, Subsection 9.3.3.1.1, and the layout of the equipment and floor drain system shown in DCD Tier 2, Figure 9.3.3-1, “Equipment and Floor Drain System Flow Schematic Radiological Controlled Area,” the staff finds that the east and west drain system design provides sufficient protection against cross-divisional flooding.

The R/B flooding analysis described above did not take credit for removal of water via the drain systems, which is conservative. The analysis also failed to provide the basis for assumptions made for evaluating flooding due to firefighting operations, and was unclear on its treatment of moderate-energy line cracks (MELCs). While the general description of how the flooding analysis was performed is helpful in understanding the approach taken in the flood analysis, the staff did not find sufficient description and detail to confirm that the applicant flood model used appropriate and conservative assumptions in all of its evaluations. Based on the information provided in the application, the staff has determined that an audit of the applicant flooding analysis will be required to confirm that appropriate methodology and assumptions were used to determine the flood levels presented in the DCD. Therefore, in **RAI 841-6055, Question 03.04.01-29,** the staff requested that the applicant provide, and include in the DCD, the basis for assumptions made for evaluating flooding due to firefighting operations (including the bases for the number of hose stations assumed, and the flow rates for those stations), and discuss how HELBs and MELCs were included in their evaluation.

In its response to **RAI 841-6055, Question 03.04.01-29**, dated December 26, 2012, the applicant provided additional information on the basis and assumptions used for the flooding analysis. The applicant indicated that for firefighting operations the water volume assumed discharged was based on the use of two hose stations with an assumed flow of 125 gpm (473 liters per minute (L/min)) in use for two hours which they stated was consistent with Section 3.2 of RG 1.189, Revision 1, issued March 2007. Based on the information provided, the staff finds the assumptions used for firefighting contributions used in the flooding analysis to be conservative and therefore acceptable.

The applicant also provided additional information on which HELBs were assumed and how the flooding resulting from breaks in high energy and moderate energy lines were evaluated. The applicant stated that flooding caused by a HELB or moderate energy line break (MELB) was evaluated in accordance with criteria in DCD Tier 2, Section 3.6 “Protection Against Dynamic Effects Associated with Postulated Rupture of Piping,” and provided details of the flood evaluation of postulated line breaks in the R/B, and the PS/B, including information on the rate at which water is released through the break, the duration of the discharge, and flooding water volume that result from the break. The applicant also provided a DCD markup to incorporate additional details of the analysis into DCD Tier 2, Section 3.4.1.3. The staff reviewed the provided information and found it to be reasonable for internal flood evaluation; however, that staff finds that an audit of the analysis is necessary to assure that appropriate modeling assumptions and break selections were used. In addition, per the RAI response and the applicant’s letter “Updated Closure Plan for US-APWR Seismic and Structural Analyses - Schedule Improvement,” dated February 15, 2013, the applicant is planning to amend the RAI response to document building layout changes that impact DCD Tier 1 and DCD Tier 2, Appendix 3K. Pending the flooding analysis audit and the submittal and review of the amended RAI response, **RAI 841-6055, Question 03.04.01-29 is being tracked as an Open Item.**

Penetrations through barrier walls are located above the maximum flood level and/or are sealed. Penetrations through divisional walls are located at least 2.5 m (8 ft 3 in.) above the floor, consistent with SRP Section 14.3.2, “Structural and Systems Engineering – Inspections, Tests, Analyses, and Acceptance Criteria," March 2007, SRP Acceptance Criteria Item 8 (2.5 m above the floor).

MCR: The MCR and Remote Shutdown Room (RSR) are located in the NRCA portion of the R/B, at elevations 7.7 m (25 ft 3 in.) and 23.3 m (76 ft 5 in.), respectively. In accordance with DCD Tier 2 Section 3.4.1.5.2.2, the MCR is isolated from the adjacent R/B corridor by concrete walls and a watertight door. However, the DCD did not discuss whether there are any internal sources of water inside the MCR, and if so, how the MCR is protected from these internal water sources. Furthermore, the DCD did not appear to address flood protection for the RSR. Accordingly, in **RAI 220-2058, Question 03.04.01-14**, the staff requested that the applicant provide additional details regarding flood protection for the MCR and RSR.

In its response to **RAI 220-2058, Question 03.04.01-14**, dated May 21, 2009, the applicant stated that the remote shutdown panel, located within the RSR, is an equipment item to be protected from internal flood. There are no flooding sources within the RSR. Additionally, the entry way into the RSR is protected from the in-flow of water by means of a water tight door. The applicant stated that DCD Tier 2, Subsection 3.4.1.5.2.2 will be revised to include this information. The staff has confirmed that DCD Tier 2, Revision 2, Subsection 3.4.1.5.2.2 was revised as committed in the RAI response. The staff finds that the concerns identified in **RAI 220-2058, Question 03.04.01-14** as related to the RSR are resolved since the RSR is protected from in-flow of water from flood sources by water-tight doors, and there are no flooding sources inside the RSR.

With regard to the MCR, the applicant stated that sanitary piping represents the only sources of water inside the MCR. The applicant further stated that internal flood countermeasures are not required because the "water lines are less than or equal to 1B." However, this statement is not clear, as the applicant did not explain or define the term "1B." Furthermore, the applicant did not propose to revise the DCD to describe how the MCR is protected from water sources inside the MCR. Accordingly, the staff closed as unresolved **RAI 220-2058, Question 03.04.01-14** and in follow-up **RAI 579-4481, Question 03.04.01-25,** that the applicant (a) clarify the response to **RAI 220-2058, Question 03.04.01-14,** with regard to how the MCR is protected from water sources inside the MCR and (b) include this discussion in the DCD

In its response to **RAI 579-4481, Question 03.04.01-25**, dated May 27, 2010, the applicant stated that the term 1B is an abbreviation for 1 in. (25 mm) diameter pipe. The applicant also stated that postulated breaks and cracks are defined in DCD Tier 2, Subsection 3.6.2.1.3, "Types of Break/Cracks Postulated" and for the moderate energy fluid systems piping, leakage cracks are not postulated in 1 in. (25 mm) nominal diameter and smaller piping, therefore the cracks were not postulated for in the sanitary piping in the MCR internal flooding evaluation.

SRP Section 3.4.1, Subsection II, Acceptance Criteria 1 provides guidance as to how the requirements of GDC 2 may be met and states in part that "meeting the requirements of GDC 2 includes evaluating the effects of flooding from full circumferential failures of non-seismic, moderate-energy piping, which is not considered in SRP Section 3.6.2." A break of small diameter piping in the MCR could result in equipment failure if the design (pipe routing, drains, etc.) does not take adequate measure to prevent wetting of equipment due to spray or flooding.

The staff closed **RAI 579-4481, Question 03.04.01-25.** as unresolved and in follow up **RAI 842-5863, Question 03.04.01-31**, the staff requested that the applicant provide additional justification for not considering the sanitary piping in the MCR internal flood evaluation, and to provide information on flood protection features incorporated in the design of the control room.

In its response to **RAI 842-5863, Question 03.04.01-31**, dated December 19, 2011, the applicant stated that only source of water in the MCR compartment is the potable and sanitary water system (PSWS) and that the piping within the MCR is designed to be excluded from the postulating of leaks and cracks during normal plant operation in accordance with the criteria described in DCD Tier 2, Section 3.6. In addition, the applicant states that PSWS piping will be routed to avoid the MCR area, and as a result only be installed in the kitchen and the restroom. Since there are walls between the MCR and other areas of the MCR compartment, including the kitchen and the restroom, the equipment in the MCR would not be affected by spray from the piping.

Since the design incorporates features to protect against breaks, and uses piping routing to prevent spray onto safety-related SSCs, the staff finds that the PSWS does not present a significant source of flood water to MCR, and that the PSWS design provides adequate protection to prevent wetting of equipment due to spray by routing of system piping outside of areas of the MCR containing equipment. Since the applicant has identified DCD changes, **RAI 842-5863, Question 03.04.01-31, is being tracked as a Confirmatory Item**.

In addition to protecting safety-related SSCs in the MCR from flooding, the flood protection features associated with protection of the MCR (i.e., doors, barriers, etc.) should also allow for access to the MCR in cases where flooding occurs in the corridors outside of the MCR. In DCD Tier 2, Section 3.4.1.5.2.2, it is stated that “the main control room subject to regular access is protected by use of barriers,” and that the Class 1 I&C panels are installed in the room, which prevents flow-in by the use of barriers and water-tight doors. The review of control room habitability, including a review of control room access, is addressed in Section 6.4 of this SE. However, as part of the internal flood review the staff reviewed the DCD Tier 2, Appendix 3K to verify the location of the barriers and water-tight doors described in DCD Tier 2, Section 3.4.1.5.2.2. The staff reviewed the DCD Figures showing the flood barriers and water-tight doors for the control room (DCD Tier 1. Figure 2.2-18, “Flood barriers and water-tight doors R/B EL 25’ -3” (2F)” and DCD Tier 2, Figure 3K-5, “Location of Watertight Doors and Flood Barrier Walls R/B Plan View Elevation 25'-3") and was not able to find the barriers described in Section 3.4.1.5.2.2, of the DCD. Therefore, in **RAI 841-6055, Question 03.04.01-30**, the staff requested that the applicant identify the flood barrier that is credited in the flood evaluation in Section 3.4 of the DCD.

In its response to **RAI 841-6055, Question 03.04.01-30**, dated October 19, 2012, the applicant indicated that an updated DCD Tier 2, Figure 3K-5 that clearly indicates the barriers was provided in the revised response to **RAI 559-4387, Question 06.04-11**, dated September 20, 2012. The staff reviewed the revised DCD Tier 2, Figure 3K-5 in the **RAI 559-4387, Question 06.04-11,** response and found that the revised figure does show the flood barriers and water-tight doors used as flood protection measures for the MCR. Based on the revised DCD Tier 2, Figure 3K-5, the staff finds this portion of the response to **RAI 841-6055, Question 03.04.01-30** acceptable. However, per the RAI response and the applicant’s letter “Updated Closure Plan for US-APWR Seismic and Structural Analyses - Schedule Improvement,” dated February 15, 2013, the applicant is planning to amend the RAI response to document building layout changes that impact DCD Tier 1 and DCD Tier 2, Appendix 3K. Pending the submittal and review of the amended RAI response, **RAI 841-6055, Question 03.04.01-30, is being tracked as an Open Item**.

SFP Cooling: The DCD did not describe how internal flood protection is achieved for SSCs used to provide SFP cooling. The SFP is located in the RCA portion of the R/B. Accordingly, the staff requested in **RAI 220-2058, Question 03.04.01-15** that the applicant demonstrate how SSCs used to provide SFP cooling are protected from flooding.

In its response to **RAI 220-2058, Question 03.04.01-15**, dated May 21, 2009, the applicant stated that the top of the foundations for the two SFP pumps are 0.30 m (1.0 ft) above the corresponding floor elevation, whereas the maximum flood water heights are as follows: 0.20 m (0.67 ft) at the "A train" SFP pump, and 0.26 m (0.87 ft) at the "B train" SFP pump. As a result, the SFP pumps would not be flooded. The applicant stated that DCD Tier 2, Section 3.4.1.5.2.1 will be revised to provide this information. The staff has confirmed that Revision 2 of DCD Tier 2, Section 3.4.1.5.2.1 was revised as committed in the RAI response. The staff finds the response acceptable since the SFP pumps are located above the maximum flood water height. Accordingly, **RAI 220-2058, Question 03.04.01-15, is resolved.**

As discussed earlier, in **RAI 220-2058, Question 03.04.01-8,** the staff requested that the applicant provide a complete list of SSCs associated with SFP cooling that need flood protection. In its response to **RAI 220-2058, Question 03.04.01-8**, the applicant identified the SFP pumps as being the SSCs associated with SFP cooling that need protection from internal flooding. Accordingly, **RAI 220-2058, Question 03.04.01-8, is resolved**.

PS/B: As previously noted, the PS/B is divided into two separate structures, namely PS/B (east) and PS/B (west). It seems plausible that the PS/B design provides flood protection for safety-related SSCs based on the physical separation of PS/B (east) and PS/B (west), given that these two areas contain redundant trains of equipment. However, the DCD did not describe how internal flood protection is achieved for safety-related SSCs located inside the PS/B. Accordingly, in **RAI 220-2058, Question 03.04.01-16**, the staff requested that the applicant demonstrate how safety-related SSCs located in the PS/B are protected from flooding, including a description of instrumentation for flood protection.

In its response to **RAI 220-2058, Question 03.04.01-16**, dated May 21, 2009, the applicant stated that DCD Tier 2 will be revised to include a complete list of SSCs inside the PS/B areas that need flood protection and the location of these SSCs relative to the flood level. A copy of this list was included as part of the applicant's response. This list indicates that many of the SSCs that need flood protection are located above the flood level. In those instances where a SSC is not located above the flood level, this list states that the SSC is located within a water-tight compartment that is in turn protected from the in-flow of water by means of a water-tight door and floor drain isolation valves. However, the DCD did not describe in further detail how internal flood protection is achieved for safety-related SSCs located inside the PS/B, for example by identifying potential flood sources and flood pathways. Accordingly, the staff closed **RAI 220-2058, Question 03.04.01-16,** as unresolved and in follow-up **RAI 579-4481, Question 03.04.01-26**, the staff requested that the applicant demonstrate how safety-related SSCs located in the PS/B are protected from flooding**.**

In its response to **RAI 579-4481, Question 03.04.01-26**, dated June 21, 2010, the applicant stated that it would include a description in the DCD of the worst case flood sources and how safety-related SSCs in the PS/B are protected from internal flooding. In the proposed new DCD Tier 2, Subsection 3.4.1.5.3, the applicant stated that doorways provide potential flow paths from the NRCA of the R/B to the PS/B. They also stated that all floors in the NRCA of the R/B are divided into two areas, east and west, by concrete walls and/or water-tight doors, and that there is no cross-connection between east and west area drains. Accordingly the applicants flood analysis considers flow into the PS/B due to flooding in the NRCA of the R/B building, but does not consider flow between the east and west side of the NRCA of the R/B.

The applicant performed a flood analysis for elevation -8.0 m (-26 ft 4 in.) and determined that the worst case flooding resulted from a combination of an earthquake and firefighting operations. For that event the applicant calculated the maximum water level due to flooding to be 0.137 m (0.45 ft) above elevation - 8.0 m (-26 ft 4 in.) for the east side, and 0.183m (0.60 ft) above elevation -8.0 m (-26 ft 4 in.) for the west side.

The equipment in these areas that the applicant identified as needing protection from flooding at this elevation is the essential chiller units. The pump foundations for these units are 0.305m (1.0 ft) above floor elevation -8.0 m (-26 ft 4 in.), which is above flood elevation. Also, the applicant stated in the RAI response that the instrumentation for each pump is designed to be located above the level of flood water.

The applicant performed a flood analysis for elevation 1.1m (3 ft 7 in.) and determined that the worst case flooding resulted from a combination of an earthquake and firefighting operations. For that event the applicant calculated the maximum water level due to flooding to be 0.957m (3.14 ft) above elevation 1.1m (3 ft 7 in.) for both the east and west areas.

The equipment in these areas that the applicant identified as needing protection from flooding at this elevation is the Class 1E GTGs. The Class 1E GTGs are located in the Class 1E GTG rooms, which are isolated from the corridor of the R/B NRCA by concrete walls and water-tight doors. Therefore, the GTG rooms will not be flooded by this worst case flooding event.

The staff has reviewed the applicant's response to **RAI 579-4481, Question 03.04.01-26,** and found it to be acceptable because it provided information from the flood analysis that demonstrate how safety-related SSCs located in the PS/B are protected from flooding. The staff confirmed that DCD Revision 3 incorporated the proposed DCD changes. Accordingly, **RAI 579-4481, Question 03.04.01-26, is resolved.**

Interconnected Paths: A review was performed for possible flow paths from interconnected nonsafety-related areas to buildings, rooms, and enclosures that house safety-related SSCs (e.g., leakage through interconnecting doorways). The review was based on information provided in DCD Tier 2, Sections 3.4.1.3 and 3.4.1.5, "Evaluation of Internal Flooding"; DCD Tier 2, Figures 1.2-1 through 1.2-51; and DCD Tier 2 Figures 9A-1 through 9A-27. The candidate nonsafety areas are the AC/B, the A/B, and the T/B. The AC/B does not directly connect with any adjacent safety-related buildings, but instead connects only with the A/B. However, the A/B and T/B do adjoin the R/B. Furthermore, the A/B adjoins the west portion of the PS/B. The A/B consists of a RCA and NRCA that are physically separated.

There are doorways between the RCA portions of the R/B and A/B, and between the NRCA portions of the R/B and A/B. These interconnecting pathways between the R/B and A/B are secured by means of watertight doors. There are also ground level doorways between the NRCA portion of the R/B and the T/B. These interconnecting pathways between the R/B and the T/B are also secured with watertight doors. Furthermore, the T/B exterior walls include a flood relief panel system. This flood relief panel system would drain excess flood water into the yard once flood water has filled the lower level of the T/B so as to prevent water from affecting equipment in the R/B.

DCD Tier 2 Subsection 3.4.1.5.2.2 states that there are doorways between the NRCA of the R/B and east and west PS/B at the -8.0 m (-26 ft 4 in.) elevation. However, these doorways are not watertight. Therefore, flood waters that collect at this elevation of the R/B are assumed to flow into the entire area of the PS/B and were considered in the flood analysis.

Interconnections between the RCA portion of the R/B and PCCV are a normal personnel airlock located at the floor level on the 7.7 m (25 ft 3 in.) elevation, and an equipment hatch and emergency airlock, located at the 23.3 m (76 ft 5 in.) elevation. The annulus is designed to maintain a leak-tight barrier under all postulated conditions, thereby precluding the flow of water from the annulus into the R/B.

Based on the above, the staff finds that the design provides adequate protection from interconnected paths.

In-Leakage Sources: The staff performed a review to assess the protection against possible external in-leakage sources, such as non-mechanistic cracks in structures and exterior openings and penetrations in structures located at a lower elevation than the internal flood level. The following information was extracted from DCD Tier 2, Section 3.4.1.2, "Flood Protection from External Sources." The US-APWR was found to withstand these effects by means of the following design features of seismic Category I and II structures:

- Plant site grade is sloped away from structures;
- Structures below grade have water stops and waterproofing;
- Structures can withstand hydrostatic loads from groundwater pressure;
- Electrical conduit and piping penetrations below grade have watertight seals;
- Number of below-grade exterior wall penetrations is minimized; and
- All above grade exterior doors and equipment access openings that represent a potential pathway for floodwater to reach safety-related SSCs are located above the design-basis flood level (DBFL).

The US-APWR design does not include a permanent dewatering system to protect safety-related SSCs from below-grade groundwater seepage. The need for a permanent dewatering system is addressed by the COL applicant per COL Information Item 3.4(5), which is evaluated in Section 3.4.2.4.2 of this report.

The staff finds that the design provides adequate protection from external in-leakage sources.

Watertight Doors: As previously discussed, the US-APWR design utilizes watertight doors as an important element of the overall flood protection strategy. However, there was no mention of methods used for assuring the functionality of the watertight doors, for example position indicators, door seals, aging degradation, testing, and maintenance procedure requirements for the door seals. In accordance with SRP Section 3.4.1, Item III.2, the adequacy of techniques used to prevent flooding, including watertight doors, should be assessed. Accordingly, in **RAI 220-2058, Question 03.04.01-17,** the staff requested that the applicant describe how the functionality of the watertight doors is assured. In its response to **RAI 220-2058, Question 03.04.01-17**, dated May 21, 2009, the applicant stated that several means are used to assure the functionality of watertight doors. These methods include remote indication of door positions provided to operators, and periodic visual inspections and functional tests. The periodic inspections will ensure that aging-related degradation of the seals is identified. The applicant proposed to revise DCD Tier 2, Section 3.4.1.3 to include a brief summary discussion that outlines this approach. However, in the proposed revision to the DCD, the applicant made reference to door "position indication" instead of door "remote position indication." Furthermore, the applicant's proposed revision to DCD Tier 2 did not indicate that remote position indication is available to operators. The ability of operators to remotely monitor the position of watertight doors is important in assuring door functionality. Accordingly, the staff closed as unresolved **RAI 220-2058, Question 03.04.01-17** and in follow-up **RAI 579-4481, Question 03.04.01-27,** the staff requested that the applicant modify the proposed revision to DCD Tier 2, Section 3.4.1.3 to explicitly state that the design of watertight doors includes remote position indication for operators.

In its response to **RAI 579-4481, Question 03.04.01-27,** dated June 21, 2010, the applicant proposed to revise DCD Tier 2, Section 3.4.1.3 to indicate that "water-tight doors have remote position indication for closure verification and are periodically inspected and tested to ensure proper functionality."

The staff has reviewed the applicant response regarding the use of remote position indication for water-tight doors and found it to be acceptable because the applicant's response clearly states that water-tight door will have remote position indication. The staff also reviewed the applicant's response in regards to requiring programs for testing and inspection of the doors. While the applicant has revised the DCD to indicated periodic testing and inspection of the doors will be performed to ensure proper functionality, the applicant has not provide information on the what is considered acceptable in terms of maintenance requirements and procedures, and the basis for what it considers acceptable, also the applicant has not proposed to add a COL information item to the DCD to instruct the COL applicant to implement procedures to ensure that the proper maintenance and inspection of the doors will be performed to ensure proper functionality. As a result, the staff closed as unresolved **RAI 579-4481, Question 03.04.01-27** and in **RAI 842-5863, Question 03.04.01-32**, the staff requested the applicant to provide additional information on the maintenance and inspection of the water-tight doors, and to include in the DCD a COL information item to ensure that a maintenance and inspection program is implemented to ensure proper functionality of the water-tight doors. In its response to **RAI 842-5863, Question 03.04.01-32**, dated October 25, 2012, the applicant indicated that the COL applicant should address periodic inspections and testing when developing operating and maintenance procedures, to ensure the integrity of water-tight doors in terms of sealing function. To ensure that the COL addresses this issue, the applicant added a new COL Information Item 3.4(8) to DCD Tier 2, Table 1.8.2 and Section 3.4.3. The COL information item states the following:

> The COL applicant is responsible for developing inspection and testing procedures in accordance with manufacturer recommendations so that each water-tight door remains capable of performing it intended function.

The staff finds that applicant's response to **RAI 842-5863, Question 03.04.01-32,** to be acceptable. Since the applicant has identified DCD changes, **RAI 842-5863, Question 03.04.01-32, is being tracked as a Confirmatory Item**. The staff has also confirmed that Revision 3 of DCD Tier 2, Section 3.4.1 was revised as committed in the responses to **RAI 220-2058, Question 03.04.01-17** and **RAI 579-4481, Question 03.04.01-27**.

Submerged SSCs: DCD Tier 2, Section 3.4.1.1 states the following: "SSCs are mounted above the flood level. While safety-related SSCs that are environmentally protected in accordance with Section 3.11 are permitted below potential flood level, no components requiring active operation to achieve their intended safety function are located below the potential flood level."

In accordance with SRP Section 3.4.1, Item III.5, safety-related SSCs being located below the flood level should be reviewed by the staff, and therefore, should be identified in the DCD. Further, the qualification program should be described in the DCD for the staff review. Exceptions, if any, should be justified in the DCD. DCD Tier 2, Table 3D-2, lists components subject to graded environmental conditions. However, this table does not indicate which components, if any, are credited for operation while being submerged. The staff could not determine whether the applicant intends to include the option of submerged SSCs in the DC stage or in the COLA stage. Accordingly, the staff requested in **RAI 220-2058, Question 03.04.01-18,** that the applicant provide further details regarding submerged SSCs.

In its response to **RAI 220-2058, Question 03.04.01-18**, dated April 8, 2009, the applicant stated that the design will preclude submergence of components that require active operation to achieve safety functions. This design constraint applies to both the DC and COLA stages. The applicant further stated that there are other safety-related SSCs that are permitted to be submerged and environmentally protected as described in DCD Tier 2, Section 3.11, "Environmental Qualification of Mechanical and Electrical Equipment." However, DCD Tier 2, Section 3.11 and its associated tables (e.g., Table 3D-2) did not explicitly identify those components that are credited for operation while being submerged. It is noted that the applicant's responses to **RAI 220-2058, Questions 03.04.01-2, and 03.04.01-5,** identified instances in which credit was taken for submergence of safety-related SSCs in the PCCV and the NRCA portion of the R/B. However, it was not clear that the applicant had identified all instances in which credit was taken for a submerged SSC (e.g., there does not appear to be any instance of credit taken for a submerged SSC located outside of the PCCV and the NRCA portion of the R/B). Furthermore, the applicant did not explain how the DCD will ensure that the COL applicant will address the operability of submerged SSCs that do not require active operation. Accordingly, the staff closed as unresolved **RAI 220-2058, Question 03.04.01-18.** In follow-up **RAI 579-4481, Question 03.04.01-28,** the staff requested that the applicant (a) explicitly identify all of the safety-related SSCs that are credited for operation while being submerged, (b) demonstrate how these SSCs will retain their normal function while submerged, and (c) if submerged, non-active SSCs are to be credited in the COL stage, explain how the DCD will ensure that the COL applicant will address the operability of these non-active SSCs.

In its response to **RAI 579-4481, Question 03.04.01-28**, dated June 21, 2010, the applicant clarified that DCD Tier 2, Table 3K-3 lists all of the safety-related SSCs that are credited for operation while being submerged. The SSCs submerged in a flooding event are the RV, source range neutron flux detector, power range neutron flux detector, MFW isolation valves, CCW surge tank and level control valves, and associated piping. The applicant also indicated that the notes on DCD Tier 2, Table 3K-3 include information on how these components retain their function when submerged, and that notes 3, and 7 of DCD Tier 2, Table 3K-3 were being revised to provide more information. A DCD markup with the proposed changes to DCD Tier 2, Table 3K-3 were included as part of the RAI response.

The staff has reviewed the applicant response and found it to be acceptable because it confirms that all of the safety-related SSCs that are credited for operation while being submerged are listed in DCD Tier 2, Table 3K-3, and it provides additional information on why submergence of these SSCs do not adversely impact their ability to perform their safety function. The staff confirmed that DCD Tier 2, Revision 3, Section 3.4.1 was revised as committed in the RAI response. Accordingly, **RAI 579-4481, Question 03.04.01-28, is resolved.**

External Equipment: A review was done regarding SSCs that are external to the buildings within the standard plant design, such as outside storage tanks and yard piping. In accordance with DCD Tier 2, Section 3.4.1.2, flood protection from failure of these types of external SSCs will be accomplished by the COL applicant by various means, specifically dikes, levees, retention basins, component location, and/or the site grading and drainage. Flood protection measures from site-specific SSCs (for example, circulating water piping) are the responsibility of the COL applicant, as indicated in COL Information Item 3.4(3). Section 3.4.1.5 of this report provides additional details regarding COL information items.

Instrumentation: The staff evaluated the instrumentation needed for flood protection, including the adequacy of detectors and alarms necessary to detect rising water levels within structures, and the consequences of flooding on other safety-related instrumentation and electrical equipment. As stated in DCD Tier 2, Section 9.3.3.3, "Safety Evaluation," flood detection is provided in R/B rooms housing ESF equipment and where a flooding potential exists. The flood detection equipment associated with these areas includes a wall-mounted level switch and a leak-detecting floor drain box. An alarm system for ESF room leakage is provided in the MCR. A number of other systems are in place to alert operators to potential flood conditions, including in the PCCV. Examples of these other instrumentation systems include the SG water level system, the containment sump level monitoring system, and the pressurizer level monitoring system. In accordance with DCD Tier 2, Section 3.4.1.3, safety-related instrumentation and electrical equipment sensitive to flooding in the PCCV and R/B are located above potential flood elevations. The DCD did not appear to describe any instrumentation related to detection of flood conditions inside the containment annulus or PS/B. As discussed earlier in this SE Section, the staff had requested in **RAI 220-2058, Question 03.04.01-11** and **RAI 220-2058, Question 03.04.01-16,** that the applicant demonstrate how safety-related SSCs located in the containment annulus and PS/B are protected from flooding. Within these RAIs, the applicant was requested to describe means to assess flood-detection instrumentation within the PS/B and containment annulus. The applicant did not provide any information regarding flood instrumentation in the responses to **RAI 220-2058, Question 03.04.01-11** and **RAI 220-2058, Question 03.04.01-16**. Accordingly, the staff closed, as unresolved, **RAI 220-2058, Question 03.04.01-11** and issued follow-up **RAI 579-4481, Question 03.04.01-24**. The staff also closed, as unresolved, **RAI 220-2058, Question 03.04.01-16** and issued follow-up **RAI 579-4481, Question 03.04.01-26,** to request information regarding flood instrumentation. Similarly, in **RAI 579-4481, Question 03.04.01-21,** as a follow-up to **RAI 220-2058, Question 03.04.01-1**, the staff requested the applicant to demonstrate how safety-related SSCs located within the PSFSVs and ESWPT are protected from internal flood, including a description of instrumentation for flood detection. **RAI 579-4481, Questions 03.04.01-21, 03.04.01-24, and 03.04.01-26**, as discussed earlier in this SE Section, have been resolved. Based on the staff's review and the resolution of the above mentioned RAIs, the staff finds the instrumentation adequate for flood protection.

Based on the above review, and pending resolution of **RAI 841-6055, Questions 03.04.01-29 and 03.04.01-30, which are being tracked as Open Items**, the staff concludes that the design meets the requirements of GDC 2 and GDC 4 as they relate to protecting safety-related SSCs from internal flood.

3.4.1.4.2 Technical Specifications

SRP Section 16.0, "Technical Specifications," does not specifically reference flood protection. There are no US-APWR TS sections for internal flood protection. The staff finds this aspect of the DCD acceptable since it is consistent with SRP Section 16, and the standard TS in NUREG-1431, "Standard Technical Specifications - Westinghouse Plants," Revision 4, dated April 2012.

3.4.1.4.3 Inspections, Tests, Analyses, and Acceptance Criteria

Proposed ITAAC related to flood protection are given in DCD Tier 1, Table 2.2-4, and DCD Tier 1, Table 2.5.1-5, “RT System and ESF System Inspections, Tests, Analyses, and Acceptance Criteria.”

Inspections listed in DCD Tier 1, Table 2.2-4 confirm: (1) existence and proper placement of divisional flood barriers within the R/B and PS/B [Item 9], (2) existence and proper placement of watertight doors within the R/B [Item 10] (3) proper placement and sealing of penetrations in divisional walls of the R/B and PS/B (except watertight doors) [Item 11], (4) installation of safety-related electrical and I&C equipment items at appropriate elevations above flood level [Item 12], (5) adequate thicknesses of external walls of the R/B and PS/B to protect against water seepage [Item 13], (6) installation of flood barriers for the R/B and PS/B up to finished plant grade to protect against water seepage [Item 14].

The staff noted that ITAAC Acceptance Criteria No. 11 in DCD Tier 1, Table 2.2-4, included the following wording: “…equipment are located at sufficient height the floor surface against the design flood level.” For clarity, this phase needed to be revised, for example by inserting the word “above” or similar wording between “height” and “the floor.” Accordingly, the staff requested in **RAI 220-2058, Question 03.04.01-19,** that the applicant revise the wording in this ITAAC item.

In its response to **RAI 220-2058, Question 03.04.01-19,** dated April 8, 2009, the applicant stated that this ITAAC item will be reworded to explicitly indicate that safety-related electrical, instrumentation, and control equipment will be located above design flood levels. The staff has confirmed that DCD Tier 1, Revision 2, Table 2.2-4 was revised as committed in the RAI response. Accordingly, **RAI 220-2058, Question 03.04.01-19, is resolved**.

Inspections listed in DCD Tier 1, Table 2.5.1-6, “RT System and ESF System Inspections, Tests, Analyses, and Acceptance Criteria,” confirm that certain Class 1E equipment associated with the protection and safety monitoring system (PSMS), the reactor trip (RT) system, and the ESF system are located in plant areas that provide protection from various potential hazards, including flooding [Item 8]. In accordance with DCD Tier 1, Table 2.5.1-1, “Equipment Names and Classifications of PSMS and Field Equipment for RT System and ESF System,” most of this equipment is not otherwise qualified for harsh environments.

As previously discussed in Section 3.4.1.4.1 of this report, floor drains in the east and west areas of the RCA portion of the R/B are isolated by means of normally closed valves or check valves in individual drainage pathways prior to connecting into a common sump tank system. This design is used to prevent flood waters from the east (or west) from passing into the west (or east) side of the building via the floor drain system. A similar arrangement is used within the NRCA portion of the R/B to preclude cross-flow of floor drain water. Also, normally closed manual isolation valves installed in individual drainage pathways of ESF equipment rooms preclude backflow of water into these rooms via the sump system. However, the staff could not find an ITAAC entry or DCD Tier 1 discussion that specifically addresses the check valves and manual valves that are used to prevent cross-divisional flooding via floor drain and sump systems. Therefore, in **RAI 220-2058, Question 03.04.01-20,** the staff requested that the applicant include these valves as part of the ITAAC process.

In its response to **RAI 220-2058, Question 03.04.01-20,** dated April 8, 2009, the applicant noted that DCD Tier 1, Section 2.7.6.8, “Equipment and Floor Drainage Systems,” describes isolation valves associated with ESF equipment rooms that are used to protect these rooms from flooding caused by backflow. As specified in DCD Tier 1, Section 2.7.6.8, the drain systems from ESF equipment rooms are designed to detect a flooded condition and to prevent flooding due to backflow by the virtue of a difference in elevation of the ESF equipment rooms and the collection sump. A common alarm in the MCR is provided for indication of a flooded condition. Figure 2.7.6.8-1 is included to show the safety-related portions of the EFDS in order to complete ITAAC to verify functional arrangement. Furthermore, ITAAC Item 1 in DCD Tier 1, Table 2.7.6.8-1, “Equipment and Floor Drainage Systems Inspections, Tests, Analyses and Acceptance Criteria,” represents a design commitment related to the functional arrangement of floor drainage equipment. On the basis of its review of the applicant’s response, the staff finds that the concerns identified in **RAI 220-2058, Question 03.04.01-20** are resolved, and the staff finds the ITAAC adequate for flood protection. Accordingly, **RAI 220-2058, Question 03.04.01-20 is resolved**.

The review of the remaining items in DCD Tier 1, Table 2.7.6.8-1, is addressed in Section 14.3.2 of this report.

3.4.1.4.4 Initial Test Program

The initial test program for US-APWR is evaluated in Section 14.2 of this report, and evaluation of the initial test program in this section is an extension of the evaluation provided in Section 14.2 of this report.

Applicants for standard plant design approval must provide plans for preoperational testing and initial operations in accordance with 10 CFR 50.34(b)(6)(iii) requirements. SRP Section 14.2, “Initial Plant Test Program - Design Certification and New License Applicants,” Revision 3, issued March 2007, Section II, SRP Acceptance Criteria 2.A states that the applicant should commit to the revision of RG 1.68, “Initial Test Programs for Water-Cooled Nuclear Power Plants,” Revision 3, issued March 2007 (the applicable version for the US-APWR DC) in effect six months prior to submittal.

RG 1.68, Appendix A, Subsection 1(h), “Engineered Safety Features,” states the following: “Appropriate tests should also be conducted to verify the functioning of protective devices … provided to protect engineered safety features from flooding….” However, the staff could not find in the DCD any pre-operational tests that specifically address the functionality of check valves and manual valves that are used to prevent cross-divisional flooding via floor drain and sump systems. Therefore, in **RAI 243-2044, Question 14.02-110,** the staff requested that the applicant include these valves as part of the initial test program. In its response to **RAI 243-2044, Question 14.02-110**, dated March 27, 2009, the applicant stated that DCD Tier 2, Section 14.2 will be expanded to include a preoperational test for the floor drainage system. The staff finds the response acceptable since the piping and valves that prevent cross-divisional flooding will be adequately tested. The staff has confirmed that DCD Tier 2, Revision 2, Subsection 14.2.12.1.116 “Equipment and Floor Drainage System Test,” has been added to DCD Tier 2, and it includes acceptance criteria requiring that it be demonstrated that the system piping and valves will prevent backflow to prevent cross-divisional flooding. Accordingly, **RAI 243-2044, Question 14.02-110, is resolved**.

In accordance with RG 1.68, Appendix A, Subsection 1(j), “Instrumentation and Control Systems,” Item (20), “instrumentation used to detect external and internal flooding conditions that could result from such sources as fluid system piping failures” should be included in the test program. Preoperational tests for leakage detection system components installed in each ESF equipment room are provided in DCD Tier 2, Subsection 14.2.12.1.77, “Miscellaneous Leakage Detection System Preoperational Test.”

Preoperational tests for the liquid waste management system (LWMS) are listed in DCD Tier 2, Subsection 14.2.12.1.80, “Liquid Waste Management System Preoperational Test.” This test demonstrates the operation of building sump drain systems. The LWMS is a nonsafety-related system.

The staff finds that the initial test program adequately addresses internal flood protection since it verifies the operability of plant features that are used to detect internal flooding conditions, and verifies the operability of drainage systems, and other plant features (valves that prevent cross-divisional flooding) used to mitigate the effects of internal flooding that could result from such sources as fluid system piping failures. Therefore, the staff finds that the initial test program conforms to RG 1.68 regarding addresses internal flood protection.

3.4.1.5 Combined License Information Items

The following is a list of COL item numbers and descriptions from Table 1.8-2 of the DCD related to internal flood protection:

Table 3.4.1-1
US-APWR Combined License Information Items

Item No.	Description	Section
3.4(3)	Site-specific flooding hazards from engineered features, such as from cooling water system piping, is to be addressed by the COL Applicant.	3.4.1.2
3.4(7)	The COL Applicant is responsible for the protection from internal flooding for those site-specific SSCs that provide nuclear safety-related functions or whose postulated failure due to internal flooding could adversely affect the ability of the plant to achieve and maintain a safe shutdown condition.	3.4.1.3
3.4(8)	The COL Applicant is responsible for developing inspection and testing procedures in accordance with manufacturer recommendations so that each water-tight door remains capable of performing its intended function.	3.4.1.3

The staff finds the above listing to be complete regarding internal flood protection. Also, the list adequately describes actions necessary for the COL applicant or licensee. No additional COL information items were identified that need to be included in DCD Tier 2 Table 1.8-2 regarding internal flooding. As discussed above, COL Information Item 3.4(8) was added in response to **RAI 842-5863, Question 03.04.01-32.** Since the applicant has identified DCD changes, **RAI 842-5863, Question 03.04.01-32 is being tracked as a Confirmatory Item**.

3.4.1.6 Conclusions

As a result of the open items for **RAI 841-6055, Questions 03.04.01-29 and 03.04.01-30**, the staff is unable to finalize its conclusions on Section 3.4.2 related to internal flood protection, in accordance with NRC regulations.

3.4.2 Analysis Procedures

3.4.2.1 Introduction

This section discusses the design of seismic Category I structures to withstand the effects of the highest flood and groundwater levels specified for the plant.

3.4.2.2 Summary of Application

DCD Tier 1: The Tier 1 information associated with this section is found in DCD, Revision 3, Tier 1, Section 2.2, “Structural and Systems Engineering.”

DCD Tier 2: The applicant has provided a DCD Tier 2, Revision 3 system description in Section 3.4.2, “Analysis Procedures,” summarized here in part, as follows: This section discusses the design of seismic Category I structures to withstand the effects of the highest flood and groundwater levels specified for the plant.

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

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

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

Technical Reports: The technical reports associated with DCD Tier 2, Section 3.4.2 are:

1. MUAP-10001, “Seismic Design Bases of the US-APWR Standard Plant,” Revision 1, issued May 2010.

2. MUAP-10006, “Soil-Structure Interaction Analyses and Results for the US-APWR Standard Plant,’ Revision 3, issued November 2012.

Cross-cutting Requirements (TMI, USI/ GSI, Op Ex): There are no cross-cutting requirements for this area of review.

US-APWR Interface Issues Identified in the DCD: See DCD Tier 2, Table 1.8-2, “Compilation of All Combined License Applicant Items for Chapters 1-19.”

Site Interface Issues Identified in the DCD: See DCD Tier 2, Table 1.8-1, “Significant Site Specific Interfaces with the Standard US-APWR Design.”

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

3.4.2.3 Regulatory Basis

The relevant requirements of the Commission’s regulations for this area of review, and the associated acceptance criteria are given in Section 3.4.2, “Analysis Procedures,” Revision 3, issued March 2007, of NUREG-0800, the SRP, and are summarized below. Review interfaces with other SRP sections can be found in Section 3.4.2 of NUREG-0800.

1. GDC 2, as it relates to the ability of SSCs without loss of capability to perform their safety function, to withstand the effects of natural phenomena, such as earthquakes, tornadoes, floods, and the appropriate combination of all loads.

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 DC has been constructed 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 1.102, "Flood Protection for Nuclear Power Plants," Revision 1, issued September 1976.

2. RG 1.59, "Design-Basis Floods for Nuclear Power Plants," Revision 2, issued August 1977.

3. SRP Section 14.3.2, "Structural and Systems Engineering - Inspections, Tests, Analyses, and Acceptance Criteria," issued March 2007.

3.4.2.4 Technical Evaluation

The staff reviewed the analysis procedures used by the applicant to transform static and dynamic effects of the highest flood and groundwater levels into effective loads on seismic Category I structures described in DCD Tier 2, Revision 3. In addition, the staff reviewed the appropriateness of loads used by the applicant to account for flood and groundwater on these structures. The staff conducted its review based on guidance and acceptance criteria in SRP Section 3.4.2 to ensure conformance with GDC 2. The scope of the review included consideration of guidance in SRP Section 2.0, "Site Characteristics and Site Parameters," issued March 2007; SRP Section 2.4.3, "Probable Maximum Flood (PMF) on Streams and Rivers," Revision 4, issued March 2007; SRP Section 2.4.12, "Groundwater," Revision 3, issued March 2007; and the associated regulations; information in DCD Tier 2, Sections 2.4, "Hydrologic Engineering," 3.4, "Water Level (Flood) Design," and 3.8, "Design of Category I Structures"; and procedures in recently published reference documents and standards for determining effects of flood and groundwater loads on structures. The review focused on the following activities.

- Validation of input parameters for the structural design criteria appropriate to account for flood and ground water loadings.

- Verification of flood protection for safety-related SSCs from external and internal sources.

- Verification that the analysis procedures used to transform static and dynamic effects of the highest flood and groundwater level into effective loads applied to seismic Category I structures are consistent with SRP Section 3.4.2 and industry standards including ASCE/SEI-7-05, "Minimum Design Loads for Building and Other Structures."

3.4.2.4.1 Design-Basis Parameters

The design-basis parameters for the US-APWR standard design include the following flood, precipitation, and groundwater parameters in DCD Tier 2, Revision 3, Section 2.4 and Table 2.0-1:

Parameter Description	Parameter Value
Maximum flood (or tsunami) level	1 ft below plant grade
Maximum rainfall rate (hourly)	19.4 in./hr. for seismic Category I/II structures
Maximum rainfall rate (short-term)	6.3 in./5 min for seismic Category I/II structures
Maximum groundwater level	1 ft below plant grade

As discussed in Section 3.4.2.5 of this report, the COL applicant is responsible for providing sufficient site-specific information to verify that hydrological events will not affect the safety-basis for the US-APWR and that the contents of the COLA satisfy applicable regulatory requirements.

To ensure that the applicant considered the probable maximum flood in the design of seismic Category I structures, in **RAI 219-1908, Question 03.04.02-1,** the staff requested the applicant to describe the basis for the PMF. In its response to **RAI 219-1908, Question 03.04.02-1**, dated April 9, 2009, the applicant stated that a COLA is required to determine the maximum ground water elevation that may occur from external flood and tsunami sources, including the probable maximum surge and probable maximum hurricane estimates. DCD Tier 2, Section 2.4.5, "Probable Maximum Surge and Seiche Flooding," indicates that, if applicable, the site-specific data relating to surges and seiches includes the effects of seismic and non-seismic information on the postulated design bases, and how the data relates to surge and seiche in the vicinity of the site and the site region. The applicant further stated that the US-APWR is designed for a maximum groundwater elevation of one ft (0.3 m) below plant grade as well as a maximum level for flood or tsunami, including storm surge, of one ft (0.3 m) below plant grade. If a hydrologic-related event exceeds the design-basis parameters for the US-APWR, the COL applicant is responsible for providing a site-specific design to prevent a threat to the safety-basis of the US-APWR. This commitment is addressed by COL Information Item 3.4(2), which states that the COL applicant is to demonstrate the DBFL bounds their specific site, or is to identify and address applicable site conditions where static flood level exceed the DBFL and/or generate dynamic flooding forces. The staff finds COL Information Item 3.4(2) acceptable as it meets the SRP Section 3.4.2 Acceptance Criterion II(1).

Based on an evaluation of the applicant's response to **RAI 219-1908, Question 03.04.02-1**, the staff concludes that the PMF is properly considered in the design of the US-APWR seismic Category I structures and that the COL applicant, consistent with COL Information Item 3.4(2), is responsible for providing a site-specific design to prevent a threat to the safety-basis of the US-APWR, if site-specific hydrologic conditions exceed the design-basis parameters for the standard design. The staff also concludes that the input parameter associated with the PMF is consistent with the acceptance criteria in SRP Section 3.4.2, Section II and that the applicant's method for establishing extreme site-specific hydrologic-related events is acceptable. Therefore, the staff finds the response acceptable. Accordingly, **RAI 219-1908, Question 03.04.02-1, is resolved.**

3.4.2.4.2 Flood Protection from Internal and External Sources

In DCD Tier 2, Section 3.4.1, the applicant indicated that the US-APWR seismic Category I structures are designed for maximum water levels caused by external flooding. The applicant also stated that the design-basis for external flooding complies with GDC 2. This compliance is accomplished by designing SSCs to withstand the effects of natural phenomena such as floods, tsunami, and seiches without the loss of capability to perform their safety functions. Based on the DBFL and the plant elevation with regard to the DBFL, the applicant stated that the safety-related SSCs are protected from flooding along with the static and dynamic forces associated with a design-basis flood in accordance with the guidelines of RG 1.102.

Per COL Information Item 3.4(1), the COL applicant is to address the site-specific design of plant grading and drainage. The staff finds COL Information Item 3.4(1) acceptable as this meets the SRP Section 3.4.2 Acceptance Criteria II(2). Per COL Information Item 3.4(2), the COL applicant is to demonstrate the DBFL bounds their specific site, or is to identify and address applicable site conditions where static flood level exceed the DBFL or generate dynamic flooding forces. Per COL Information Item 3.4(5), the COL applicant is to identify and design, if necessary, any site-specific flood protection measures such as levees, seawalls, floodwalls, site bulkheads, revetments, or breakwaters per the guidelines of RG 1.102, or dewatering system if the plant is not built above the DBFL. The staff finds COL Information Item 3.4(5) acceptable as this meets the SRP Section 3.4.2 Acceptance Criteria II(2). In addition, per COL Information Item 3.4(4), the COL applicant is to address any additional measures below grade to protect against exterior flooding and the intrusion of groundwater into seismic Category I buildings and structures. Per COL Information Item 3.4(3), the COL Applicant is to address site-specific flooding hazards from engineered features, such as from service water or circulating water piping. The staff finds these COL Information Items acceptable since they appropriately identify site-specific topics a COL applicant needs to address as they meet the SRP Section 3.4.2 Acceptance Criteria II(2).

The applicant also indicated in DCD Tier 2, Section 3.4.1 that the US-APWR seismic Category I structures are designed to accommodate the effects of, and to be compatible with, the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including LOCAs. These design criteria for flood protection from internal sources are applicable to earthquake pipe breaks and cracks, firefighting operations, and pump mechanical seal failure events.

Based on a review of information in the DCD Tier 2, Sections 3.4.1.2, “Flood Protection from External Sources,” and 3.4.1.3, “Flood Protection from Internal Sources,” the staff concludes that flood protection from internal and external sources are consistent with acceptance criteria in SRP Section 3.4.2, Subsection II and guidance in RG 1.102 and are therefore acceptable.

3.4.2.4.3 Evaluation of Analysis Procedures for External Flooding

The applicant outlined its external flood evaluation process in DCD Tier 2, Section 3.4.1.4, “Evaluation of External Flooding.” However, in the design of seismic Category I structures for flood loads, the applicant did not identify specific codes or standards used to conduct hydrostatic and hydrodynamic load evaluations nor the use of the “Shore Protection Manual,” U.S. Army Coastal Engineering Research Center, issued June 2002; or the “Coastal Engineering Manual,” U.S. Army Corps of Engineers, issued April 2002; which are identify as acceptable procedures for determining dynamic loads in SRP Section 3.4.2, Section I, SRP Acceptance Criteria 3. In order to evaluate the analysis procedures used by the applicant for external flooding, in **RAI 219-1908, Question 03.04.02-2,** the staff requested the applicant to describe the basis for determining hydrostatic and hydrodynamic loads on seismic Category I structures.

In its response to **RAI 219-1908, Question 03.04.02-2**, dated April 9, 2009, the applicant stated that the US-APWR is designed for a maximum groundwater elevation of one ft (0.3 m) below plant grade and a maximum level for flood or tsunami of one ft (0.3 m) below plant grade. As a result, the subject of above-grade hydrostatic and hydrodynamic loadings described in the "Shore Protection Manual" or the "Coastal Engineering Manual" are not applicable for the design of the US-APWR standard plant. The applicant further stated that should site-specific conditions exist which require the evaluation of above-grade hydrostatic and hydrodynamic loading, the COL applicant is responsible for providing sufficient information to verify that hydrologic-related events will not affect the safety-basis for the US-APWR. The COL applicant may therefore utilize the "Shore Protection Manual" and the "Coastal Engineering Manual" as applicable for the analysis of the site-specific conditions.

Based on an evaluation of the applicant's response to **RAI 219-1908, Question 03.04.02-2**, the staff concludes that the flood load design criterion is properly addressed by the applicant because the US-APWR is designed for a maximum ground water elevation of one ft (0.3 m) below plant grade and a maximum level for flood or tsunami of one ft (0.3 m) below plant grade. The staff also concurs that the COL applicant, consistent with COL Information Item 3.4(2), is to provide sufficient information to verify that hydrologic-related events will not affect the safety-basis for the US-APWR. Therefore, the staff finds the response acceptable. Accordingly, **RAI 219-1908, Question 03.04.02-2, is resolved.**

The factors of safety against overturning, sliding, and flotation are determined by the applicant using procedures defined in DCD Tier 2, Section 3.8.5.5, "Structural Acceptance Criteria." To evaluate the acceptability of these procedures, in **RAI 219-1908, Question 03.04.02-3** and **RAI 489-3516, Question 03.04.02-5**, the staff requested the applicant to describe the basis for these procedures. In its responses to **RAI 219-1908, Question 03.04.02-3**, dated April 9, 2009, and **RAI 489-3516, Question 03.04.02-5**, dated December 23, 2009, the applicant stated that foundation sliding for the deeply embedded mat foundations was analyzed assuming that the resistance to sliding is provided by shear resistance along the base of the mat, and if necessary, from passive soil resistance in front of the mat in the direction of sliding. As a result of this request, the applicant modified DCD Tier 2, Subsection 3.8.5.5.2, "Sliding Acceptance Criteria," to clarify that passive soil pressure is not used in the design of the standard plant to resist sliding loads and stated that no credit is taken for passive soil pressure in calculating the factor of safety against sliding in standard plant structures. The applicant also stated that the coefficient of friction at the base-to-soil interface used in the standard plant analyses is 0.7. This relatively high coefficient of friction can be obtained by special treatment of the interface. The coefficient of friction at the concrete-to-concrete interface (i.e., between the fill concrete and foundation concrete), can also be taken as 0.7, which can be achieved by minor roughening of the top of the fill concrete where necessary at certain sites. The applicant further stated that the groundwater level is an important parameter for estimating the buoyant forces on the structure, which are a part of the stability analysis. However, the effect of groundwater level on the coefficient of friction at the soil-foundation interface is negligible.

Based on its review of the response to **RAI 219-1908, Question 03.04.02-3**, the staff concurs with the applicant that calculating the factor of safety against sliding without taking credit for passive soil pressure is appropriate. Therefore, the staff finds the response acceptable. The staff confirmed that the DCD changes were incorporated into DCD Revision 3. Accordingly, **RAI 219-1908, Question 03.04.02-3, is resolved.**

However, the staff identified the following concerns regarding the shear friction coefficient in the response to **RAI 489-3516, Question 03.04.02-5**.

- Use of a concrete-to-concrete interface is not conservative if the soil-structure interface is weaker.

- The soil friction coefficient may be influenced by effects of groundwater.

To address these concerns, the staff closed as unresolved **RAI 489-3516, Question 03.04.02-5** and in follow-up **RAI 546-4345, Question 03.04.02-6,** the staff requested requesting the applicant to further clarify the use of 0.7 as the coefficient of friction at the soil-concrete interface. The applicant's response to **RAI 546-4345, Question 03.04.02-6**, dated April 16, 2010, stated that the soil angle of internal friction of 0.7 is selected as representative of soil properties anticipated for US-APWR sites and that the technical basis for the coefficient of friction at the concrete-soil interface will be further addressed in revised MUAP-10001, "Seismic Design Bases of the US-APWR Standard Plant," Revision 1. The applicant also stated that the revised report will present data and analyses results that provide the basis for the selection of the value for friction coefficient at the soil-structure interface and will consider the effect of groundwater level on the interface friction coefficient.

Subsequently, in a letter, "Updated Closure Plan for US-APWR Seismic and Structural Analyses - Schedule Improvement," dated October 12, 2012, the applicant stated that MUAP-10001 would be incorporated into MUAP-10006, "Soil-Structure Interaction Analyses and Results for the US-APWR Standard Plant," Revision 3. In addition, the letter stated that the responses to **RAI 489-3516, Question 03.04.02-5** and **RAI 546-4345, Question 03.04.02-6,** would be revised**.** Pending receipt and review of these documents, **RAI 546-4345, Question 03.04.02-6, is being tracked as an Open Item**.

3.4.2.4.4 Evaluation of Analysis Procedures for Internal Flooding

The applicant described procedures for determining liquid loads in DCD Tier 2, Subsection 3.8.3.3.2, "Liquid Loads (F)." According to these procedures, structures supporting fluid loads during normal operation and accident conditions are designed for hydrostatic and hydrodynamic loads including seismic sloshing (convective) pressures and seismic inertia (impulsive) pressures. Based on a review of liquid load effects on the load and load combinations described in DCD Tier 2, Section 3.8.4.3, "Loads and Load Combinations," the staff identified an inconsistency in earthquake load terminology. To clarify this inconsistency, in **RAI 219-1908, Question 03.04.02-4,** the staff requested the applicant to provide a consistent definition for earthquake load. In its response to **RAI 219-1908, Question 03.04.02-4**, dated December 23, 2009, the applicant stated that the term, E_s, in DCD Tier 2, Subsection 3.8.4.3.2, "Liquid Loads (F)," is intended to refer in general to earthquake-induced sloshing. To prevent a misinterpretation and enhance clarity, the applicant changed the relevant text in DCD Tier 2, Subsection 3.8.4.3.2.

The staff concludes that the change is acceptable and adequately addresses the inconsistency in earthquake load terminology. Therefore the response is acceptable. The staff confirmed that the DCD change was incorporated into DCD Revision 3. Accordingly, **RAI 219-1908, Question 03.04.02-4, is resolved**,

3.4.2.4.5 Flood Protection for Safety-Related and Nonsafety-Related SSCs

The applicant indicated in DCD Tier 2, Section 3.4.1, "Flood protection," that seismic Category I and II structures are designed to protect SSCs such that plant nuclear safety functions are not jeopardized by flooding due to the potential failure(s) of the plant SSCs or the operation of the plant fire protection system. The plant nuclear safety functions are defined as any function that is necessary to assure the following:

- The integrity of the reactor coolant pressure boundary.
- The capability to shut down the reactor and maintain it in a safe-shutdown condition.
- The capability to prevent or mitigate the consequences of plant conditions that could result in potential offsite exposures that are comparable to the guideline exposures of 10 CFR 100, "Reactor Site Criteria."

In addition, the applicant stated that the US-APWR plant is designed to assure control room habitability and operator access to areas requiring local actuation of equipment required to achieve or maintain the conditions described above.

The applicant further stated that safety-related SSCs are protected from flooding by external and internal sources and are designed with the following features:

- The separation of redundant trains of safety-related SSCs as addressed in DCD Tier 2, Chapter 1, "Introduction and General Description of the Plant."
- Protective barriers and enclosures, where necessary, as addressed in this DCD Tier 2, Section 3.4.1.
- The placement of essential SSCs above internal flood levels.
- SSCs are mounted above the flood level. While safety-related SSCs that are environmentally protected in accordance with DCD Tier 2, Section 3.11, "Environmental Qualification of Mechanical and Electrical Equipment," are permitted below the potential flood level, no components requiring active operation to achieve their intended safety function are located below the potential flood level.

The applicant also stated that protection from flooding of nonsafety-related SSCs will be considered when the impact of the flooding on a nonsafety-related SSC could be a contributing factor to the flooding of safety-related SSCs or could result in an uncontrolled release of significant radioactivity.

Based on a review of flood design information in DCD Tier 2, Section 3.4.1, the staff concludes that protection of safety and nonsafety-related SSCs against flooding due to potential failures of the plant SSCs or the operation of the plant fire protection systems is acceptable and that the design of these SSCs is consistent with applicable regulatory guidance in RG 1.59, "Design-Basis Floods for Nuclear Power Plants," Revision 2, issued August 1977, RG 1.102, and SRP

Section 14.3.2, “Structural and Systems Engineering - Inspections, Tests, Analyses, and Acceptance Criteria,” issued March 2007.

3.4.2.5 Combined License Information Items

The following is a list of COL item numbers and descriptions from DCD Tier 2, Table 1.8-2, “Compilation of All Combined License Applicant Items for Chapters 1-19,” related to analysis procedures:

Table 3.4.2-1
US-APWR Combined License Information Items

Item No.	Description	Section
COL 3.4(1)	The COL applicant is to address the site-specific design of plant grading and drainage.	3.4.1.2
COL 3.4(2)	The COL applicant is to demonstrate the design-basis flooding level (DBFL) bounds their specific site, or is to identify and address applicable site conditions where static flood level exceed the DBFL and/or generate dynamic flooding forces.	3.4.1.4
COL 3.4(3)	Site-specific flooding hazards from engineered features, such as from cooling water system piping, is to be addressed by the COL applicant.	3.4.1.2
COL 3.4(4)	The COL applicant is to address any additional measures below grade to protect against exterior flooding and the intrusion of ground water into seismic Category I buildings and structures.	3.4.1.2
COL 3.4(5)	The COL applicant is to identify and design, if necessary, any site-specific flood protection measures such as levees, seawalls, floodwalls, site bulkheads, revetments, or breakwaters pursuant to the guidelines of RG 1.102 (Reference 3.4-3), or dewatering system if the plant is not built above the DBFL.	3.4.1.2
COL 3.4(6)	The COL applicant is to identify any site-specific physical models used to predict prototype performance of hydraulic structures and systems.	3.4.2
COL 3.4(7)	The COL Applicant is responsible for the protection from internal flooding for those site-specific SSCs that provide nuclear safety-related functions or whose postulated failure due to internal flooding could adversely affect the ability of the plant to achieve and maintain a safe shutdown condition.	3.4.1.5

The staff’s evaluation of COL Information Items 3.4(1) to 3.4(5) is in Section 3.4.2.4 of this report. The staff finds COL Information Item 3.4(6) acceptable as this meets the SRP Section 3.4.2 Acceptance Criteria II.

Based on the above, the staff finds the above listing to be complete. Also, the list adequately describes actions necessary for the COL applicant or licensee. No additional COL information items were identified that need to be included in DCD Tier 2, Table 1.8-2 regarding flood protection.

3.4.2.6 Conclusions

As a result of the open item for **RAI 546-4345, Question 03.04.02-6**, the staff is unable to finalize its conclusions on Section 3.4.2 related to analysis procedures, in accordance with NRC regulations.

3.5 Missile Protection

3.5.1 Missile Selection and Description

3.5.1.1 Internally Generated Missiles (Outside Containment)

3.5.1.1.1 Introduction

This section discusses operations and performance requirements for SSCs outside containment, identification of SSCs outside containment necessary for the safe shutdown of the reactor, and the failure of SSCs outside containment that could cause a significant release of radioactivity. Also discussed is the adequacy of methods of protection from internally-generated missiles for SSCs outside containment necessary to perform functions required to attain and maintain a safe shutdown or to mitigate the consequences of an accident.

3.5.1.1.2 Summary of Application

DCD Tier 1: The Tier 1 information associated with this section is found in DCD Tier 1 Section 2.2, “Structural and Systems Engineering.”

DCD Tier 2: The applicant has provided a DCD Tier 2 system description in Section 3.5.1.1, “Internally Generated Missiles (Outside Containment),” summarized here in part, as follows:

> This section discusses operations and performance requirements for SSCs outside containment, identification of SSCs necessary for the safe shutdown of the reactor facility and the failure of SSCs that could cause a significant release of radioactivity. Also discussed is the adequacy of methods of protection from internally-generated missiles for SSCs necessary to perform functions required to attain and maintain a safe shutdown or to mitigate the consequences of an accident.

ITAAC: The ITAAC associated with DCD Tier 2, Section 3.5.1.1 are given in DCD Tier 1, Section 2.2.4, “Inspection, Tests, Analyses, and Acceptance Criteria.”

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 for this area of review.

Cross-cutting Requirements (TMI, USI/ GSI, Op Ex): There are no cross-cutting requirements for this area of review.

US-APWR Interface Issues Identified in the DCD: See DCD Tier 2, Table 1.8-2, “Compilation of All Combined License Applicant Items for Chapters 1-19.”

Site Interface Issues Identified in the DCD: See DCD Tier 2, Table 1.8-1, "Significant Site Specific Interfaces with the Standard US-APWR Design."

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

3.5.1.1.3 Regulatory Basis

The relevant requirements of the Commission's regulations for this area of review, and the associated acceptance criteria are given in Section 3.5.1.1, "Internally Generated Missiles (Outside Containment)," Revision 3, issued March 2007, of NUREG-0800, the SRP, and are summarized below. Review interfaces with other SRP sections can be found in Section 3.5.1.1 of NUREG-0800.

1. GDC 4, as it relates to the protection of SSCs against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that may result from equipment failures and from events and conditions inside the nuclear power unit.

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 DC has been constructed and will operate in accordance with the DC.

Acceptance criteria adequate to meet the above requirements include:

1. RG 1.115, "Protection against Low-Trajectory Turbine Missiles," Revision 1, issued July 1977, Regulatory Positions C.1 and C.3, as they relate to the protection of the SSCs important to safety from the effects of turbine missiles.

2. RG 1.117, "Tornado Design Classification," Revision 1, issued April 1978.

3.5.1.1.4 Technical Evaluation

The staff reviewed the US-APWR design for protecting SSCs important to safety against internally-generated missiles (outside containment) in accordance with the guidance of the SRP Section 3.5.1.1, Revision 3. The staff reviewed DCD Tier 2, Revision 3, Section 3.5.1.1. The staff also reviewed DCD Tier 1, Revision 3, Section 2.0, "Design Descriptions and ITAAC," and other DCD Tier 2 sections noted below.

3.5.1.1.4.1 Compliance with GDC 4

Compliance with GDC 4 may be based on meeting the guidance RG:

- RG 1.115, Regulatory Positions C.1 and C.3, as they relate to the protection of the SSCs important to safety from the effects of turbine missiles.

  - Regulatory Position C.1 specifies that essential systems of a nuclear power plant should be protected against low-trajectory turbine missiles due to failure of main turbine-generator (T/G) sets. Consideration may be limited to the SSCs listed in the Appendix to RG 1.117, "Tornado Design

Classification." The effect of physical separation of redundant or alternative systems may also be considered. Each essential system and its location should be identified on dimensioned plan and elevation layout drawings.

- Regulatory Position C.3 specifies that when protection of essential systems is provided by barriers, dimensioned plan and elevation layout drawings should include information on wall or slab thicknesses and materials of pertinent structures.

- RG 1.117, Appendix A, as to which SSCs should be protected from missile impacts.

DCD Tier 2, Section 3.5, "Missile Protection," addresses SSCs important to safety to be protected from internally-generated missiles inside and outside containment. DCD Tier 2, Table 3.2-2, "Classification of Mechanical and Fluid Systems, Components, and Equipment," lists all the SSCs (safety-related and nonsafety-related) in various locations of the plant (inside and outside the containment) and identifies for each SSC the associated seismic category, quality group, equipment classification and location. DCD Tier 2, Table 7.4-1, "Component Controls for Shutdown," lists the systems and components required for safe shutdown. General arrangement drawings further defining the building locations are provided in DCD Tier 2, Section 1.2, "General Plant Description."

In DCD Tier 2, Section 3.5.1.1, the applicant evaluated the potential internally-generated missiles that could result from failure of the plant equipment located outside the containment. The applicant stated that:

1. These potential missiles internally-generated outside containment are:

   a. Missiles resulting from in-plant rotating equipment over speed failures.
   b. Missiles resulting from in-plant high-pressure system ruptures such as valves, piping, fittings, tank man ways and hand holes, bolts in high energy systems, valve bonnets valve stems, pressure vessel, thermo wells, and retaining bolts.
   c. Missiles generated by onsite explosions of stored gases. Equipment that uses or generates hydrogen gas is also potential sources of internally-generated missiles outside containment.
   d. Gravitational missiles resulting from falling objects from non-seismically designed SSCs during a seismic event and maintenance equipment that is required during maintenance and is not either removed during operation to a location where it is not a potential hazard to safety-related equipment or seismically restrained to prevent it from becoming a missile.
   e. Missiles resulting from turbine over speed failures.

2. Once a potential missile is identified, its statistical significance of the identified missile is evaluated by a probability analysis. Its statistical significance is determined in the following manner:

   a. The probability of missile occurrence (P_1).
   b. The probability of impact on a significant target (P_2).

c. The probability of significant damage (P_3).
d. The combined probability ($P_4 = P_1 \times P_2 \times P_3$).

If the combined probability of the potential missile is greater than 1×10^{-7} per year, the missile is considered as credible and protection of safety-related SSCs against the credible missile will be provided. If the combined probability of the potential missile is less than 1×10^{-7} per year, the missile is not considered statistically significant, the missile is considered as not credible, and protection of safety-related SSCs against the not credible missile would not be provided.

3. Protection of safety-related SSCs against the credible missile will be provided by one or more of the following methods:

   a. Locating the system or component in a missile-proof structure.
   b. Separating redundant systems or components of the system from the missile path or range.
   c. Providing local shields or barriers for systems and components.
   d. Designing the equipment to withstand the impact of the most damaging missile.
   e. Providing design features to prevent the generation of missile.
   f. Orientating a missile source to prevent missiles from striking equipment important to safety.
   g. Missile barriers are designed if the ability to achieve and maintain safe shutdown is not determined.

Section 3.5.1.3, "Turbine Missiles," of this report addresses the staff's evaluation of protection of safety-related equipment and stored fuel from the effects of turbine missiles including compliance with the guidance of RG 1.115.

Section 3.5.3, "Barrier Design Procedures," of this report addresses the staff's evaluation of the design of structures, shields, and barriers required for missile protection inside and outside the containment.

Section 3.6.1 of this report addresses the staff's evaluation of the design of structures, shields, and barriers used for missile protection against dynamic effects of high energy line break outside the containment.

Section 3.7.3, "Seismic Subsystem Analysis," of this report addresses the staff's evaluation of the impact of the fall or overturning of non-seismic components on safety-related SSCs resulting from a seismic event.

In reviewing DCD Tier 2, Revision 1, Section 3.5.1.1, regarding protection for safety-related SSCs against internally-generated missiles outside containment, the staff identified areas in which additional information was necessary to complete its review.

In DCD Tier 2, Revision 1, Subsection 3.5.1.1.2.1, "Missiles Not Considered Credible," the applicant provided the rationale to exclude certain types of equipment from consideration as credible missile sources outside the containment. For example, missiles originating from valves, threaded connections and piping in high energy systems would not be credible due to ASME code criteria that control quality from production through operation, material characteristics, and ISIs. Qualitative discussions were also used to exclude other types of equipment (e.g. piping and valves of non-high energy fluid systems, gas explosions, gravitation missiles such as crane drops and falling objects resulting from non-seismic SSCs during a seismic event, secondary missiles, and unsecured maintenance equipment) from consideration as credible missile sources. However, the applicant had not provided the analysis to demonstrate that these missiles were of insufficient energy to cause unacceptable impact or to cause unacceptable damage. Also, it was not clear to the staff whether the applicant had followed the guidance described in SRP Section 3.5.1.1 for probabilistic analyses to determine which missiles might be non-credible by demonstrating that the product of the probability of missile occurrence, probability of impact on a significant target, and probability of significant damage was less than 1×10^{-7} per year.

Therefore, in **RAI 127-1641, Question 03.05.01.01-1, Subquestion 3.5.1.1-01**, the staff requested the applicant to demonstrate how DCD Tier 2 had excluded equipment items from consideration as credible missile sources based on design features and other qualitative considerations, and how these design features and qualitative considerations would ensure a level of protection from missiles that is equivalent to the probability criteria described in SRP Section 3.5.1.1 SRP Acceptance Criterion 1.

In its response to **RAI 127-1641, Question 03.05.01.01-1, Subquestion 3.5.1.1-01**, dated January 28, 2009, the applicant proposed to revise DCD Tier 2, Section 3.5.1.1, as marked in the response, to incorporate the probabilistic design and evaluation requested by the RAI in DCD Revision 2.

In the response, the applicant stated that potential missile sources will be statistically evaluated in accordance with the probability criteria described in SRP Section 3.5.1.1. In situations where the probability of occurrence of a missile (P_1) is greater than 10^{-7} per year, additional evaluations are made to assess the probability of impact (P_2) on a significant target. In situations where the product of P_1 and P_2 exceeds 10^{-7} per year, additional assessments are made to assure the ability to achieve and maintain safe shutdown. In these additional assessments, missiles are evaluated for the probability of significant damage (P_3) to an impacted SSC important to safety, or alternatively, evaluations are made to determine whether sufficient redundancy would remain after missile impact to achieve and maintain safe shutdown. Additional missile protection measures are not required if the product of P_1, P_2, and P_3 is less than 10^{-7} or if sufficient SSC redundancy remains following missile impact.

The applicant provided probabilistic arguments for component reliability that are based on component design features and administrative controls. For example, the probability of missile generation from high pressure piping is reduced below 10^{-7} because of design compliance with ASME Code, Section III and an inspection program that is in compliance with ASME Code Section XI. Given that piping is secured in place by means of pipe supports, the ruptured section would remain attached to the remainder of the piping system, and the overall probability of missile generation (P_1) remains below 10^{-7}.

The applicant presented other examples of design features that help to limit the probability of missile generation below 10^{-7}. For example, valves with bolted bonnets (covers) are designed such that they can withstand a bolt failure and still maintain bonnet integrity. Additionally, there are several features of high-speed rotating equipment that limit the probability of missile generation below 10^{-7}. For example, the equipment casing is postulated to prevent missile penetration. High pressure gas cylinders have design fabrication features that reduce the probability of missile occurrence (P_1) while the storage orientation and restraints reduce the probability of missile occurrence and probability of impacting a significant target.

The applicant indicated that the probability (P_2) of missile impact to safety-related SSCs will be minimized mainly by: (1) locating the potential missile sources or potential target outside the zone of postulated missile strike, (2) robust walls and/or slabs designed to withstand missile strikes, or (3) separation of the missile sources from the potentially impacted SSCs.

Additionally, in its response to **RAI 127-1641, Question 03.05.01.01-1, Subquestion 3.5.1.1-02**, dated January 28, 2009, the applicant proposed to add an ITAAC item that requires the COL applicant to verify, through inspection, that safety-related SSCs are adequately protected from missiles outside containment and inside containment.

Section 19.0 of this report addresses the staff's evaluation of the applicant's PRA methodology used to assess the internally-generated missiles.

Based on its review and evaluation discussed in the above, the staff finds the applicant's response to **RAI 127-1641, Question 03.05.01.01-1, Subquestion 3.5.1.1-01** acceptable, because the applicant: demonstrated how the DCD Tier 2 had excluded equipment items from consideration as credible missile sources based on design features and other qualitative considerations, and how these design features and qualitative considerations would ensure a level of protection from missiles that is equivalent to the probability criteria described in SRP Section 3.5.1.1; and proposed to add an ITAAC item that requires the COL applicant to verify that SSCs are adequately protected from missiles outside containment. The staff has confirmed that DCD Tier 2, Revision 2, was revised as committed in the RAI response. Accordingly, the **RAI 127-1641, Question 03.05.01.01-1, Subquestion 3.5.1.1-01, is resolved**.

In DCD Tier 2, Revision 1, Section 3.5.1.1, the applicant stated that the COL applicant needed to address the procedures, analysis, and design to ensure that pressurized gas bottles will not become missiles capable of damaging SSCs important to safety to the extent that safety-related functions are compromised. In DCD Tier 2, Revision 1, Section 3.5.1.1.2.1, the applicant described the installation and storage of the hydrogen supply system and gas bottles to prevent the buildup of hydrogen concentrations to explosive levels, thereby preventing a gas explosion that could result in missile generation. However, the applicant did not provide any design features or procedures, analysis, or design details to ensure that pressurized gas cylinders would not become/generate missiles that might adversely impact safety-related SSCs during seismic events.

Therefore, in **RAI 127-1641, Question 03.05.01.01-1, Subquestion 3.5.1.1-03**, the staff requested the applicant to revise DCD Tier 2, Section 3.5.1.1 to describe in detail any design features for missile protection from pressurized gas cylinders and to revise Tier 2, Table 1.8-2, "Compilation of all Combined License Applicant Items for Chapters 1-19," to include a COL information item which requires the COL applicant to establish/provide procedures to ensure that portable pressurized gas cylinders located/stored outside containment will not become/generate missiles that may adversely impact safety-related SSCs during seismic events.

In its response to **RAI 127-1641, Question 03.05.01.01-1, Subquestion 3.5.1.1-03**, dated January 28, 2009, the applicant proposed to revise DCD Tier 2 Section 3.5.1.1 to include pressurized gas bottles/cylinders as potential sources of missiles. Per SRP Section 3.5.1.1, the applicant identified protective measures to be taken as recommended by NUREG/CR-3551, "Safety Implications Associated with In-Plant Pressurized Gas Storage and Distribution Systems in Nuclear Power Plants," issued May 1985. These measures include design features of the bottles to prevent the generation of a missile, bottle restraints and orientation, and preparation of procedures to maintain these protective measures. The staff finds the applicant's proposal to provide protective measures to prevent pressurized gas bottles/cylinders to become potential sources of missiles as recommended by NUREG/CR-3551, acceptable.

Also, in its response to **RAI 127-1641, Question 03.05.01.01-1, Subquestion 3.5.1.1-03**, the applicant stated that the features of a pressurized gas cylinder/bottle to mitigate the cylinder from becoming a missile are provided through standard plant criteria, thereby eliminating the necessity to specify on a site-specific basis. Therefore, it is not necessary for a COL information item to be added to require the COL applicant to establish/provide procedures to ensure that portable pressurized gas cylinders located/stored outside containment will not become/generate missiles that may adversely impact safety-related SSCs during seismic events.

The staff did not agree with the applicant that it was not necessary for a COL information item to be added to require the COL applicant to establish/provide procedures to ensure that portable pressurized gas cylinders located/stored outside containment would not become/generate missiles that might adversely impact safety-related SSCs during seismic events. The staff found that some features associated with pressurized gas bottles, such as the requirement to restrain gas cylinders as well as to store in an orientation to minimize the potential for missile generation, are operational issues that require the use of procedures to ensure consistent implementation. Accordingly, in follow-up **RAI 359-2590, Question 03.05.01.01-2,** the staff requested the applicant to include a COL information item which requires the COL applicant to establish/provide procedures to ensure that portable pressurized gas cylinders located/stored outside containment are properly restrained, oriented, and utilized in a manner that minimizes the likelihood that pressurized gas cylinders will not become/generate missiles that may adversely impact safety-related SSCs. This COL item should also address implementation of procedures to remove unsecured maintenance equipment located outside the containment, prior to operations, to a location where it is not potential hazard to SSCs important to safety or seismically restrained to prevent it from becoming a missile.

In its response to **RAI 359-2590, Question 03.05.01.01-2,** dated June 5, 2009, the applicant agreed that procedural controls relating to unsecured maintenance equipment are necessary to prevent a potential hazard to important to safety SSCs whether it is located inside or outside containment. The applicant stated that DCD Tier 2, Subsection 3.5.1.1.4, "Gravitational Missiles," would be revised to require the COL applicant to have plant procedures in place prior to fuel load that specify unsecured equipment, including portable pressurized gas cylinders, located inside or outside containment and required for maintenance or undergoing maintenance is to be removed from containment prior to operation, moved to a location where it is not a potential hazard to SSCs important to safety, or seismically restrained to prevent it from becoming a missile.

The staff finds the applicant's proposed revision of COL Information Item 3.5-1 listed in DCD Tier 2, Table 1.8-2 to implement these procedural controls acceptable because potential missiles generated outside containment resulting from unsecured and non-seismically restrained compressed gas cylinders during a seismic event will be minimized. The staff has confirmed that DCD Tier 2, Revision 2, was revised as committed in the RAI response. Accordingly, **RAI 127-1641, Question 03.05.01.01-1, Subquestion 3.5.1.1-03,** and **RAI 359-2590, Question 03.05.01.01-2, are resolved**.

With regard to unsecured maintenance equipment, in DCD Tier 2, Revision 1, Section 3.5.1.1.2.1 the applicant specified, per COL Information Item 3.5(1), that the COL Applicant is to address implementation of procedures to remove unsecured maintenance equipment from containment prior to operations, to a location where it is not potential hazard to SSCs important to safety or seismically restrained to prevent it from becoming a missile. COL Information Item 3.5(1) is also specified in DCD Tier 2, Table 1.8-2. However, the applicant did not specify a COL information item to address implementation of procedures to remove unsecured maintenance equipment located/stored outside containment prior to operations.

Therefore, in **RAI 127-164,1 Question 03.05.01.01-1, Subquestion 3.5.1.1-04**, the staff requested the applicant to provide an assessment of potential gravitational missiles generated outside containment from unsecured maintenance equipment and discuss in the DCD the measures provided to prevent the impact of a falling object on safety-related equipment necessary to achieve a safe shutdown. Additionally the staff requested the applicant to revise DCD Tier 2 Table 1.8.2 to include a COL information item which requires the COL applicant to establish/provide procedures to ensure that unsecured maintenance equipment (outside containment) must be removed prior to operations to prevent those items from becoming missiles during seismic events.

In its response to **RAI 127-1641, Question 03.05.01.01-1, Subquestion 3.5.1.1-04**, dated January 28, 2009, the applicant indicated that SRP Section 3.5.1.2, "Internally Generated Missiles (Inside Containment)," Revision 3, March 2007, requires that prior to operations unsecured maintenance equipment will be removed from inside containment to a location where it is not a potential hazard to SSCs important to safety, or to seismically restrain the equipment. This requirement is not specified in SRP Section 3.5.1.1. The applicant further acknowledged that DCD Tier 2 had listed the requirement for removing unsecured maintenance equipment inside containment in DCD Tier 2, Section 3.5.1.1, which also addresses procedural controls for unsecured maintenance equipment outside containment.

The staff finds the applicant's response acceptable because, as stated in its response to **RAI 359-2590, Question 03.05.01.01-2**, the applicant stated that DCD Tier 2, Subsection 3.5.1.1.4, would be revised to require the COL applicant to have plant procedures in place prior to fuel load that specify unsecured equipment, including portable pressurized gas cylinders, located inside or outside containment and required for maintenance or undergoing maintenance is to be removed from containment prior to operation, moved to a location where it is not a potential hazard to SSCs important to safety, or seismically restrained to prevent it from becoming a missile. The staff has confirmed that the COL Information Item 3.5(1) listed in Table 1.8-2 of the DCD Tier 2, Revision 2 was revised to implement these procedural controls for potential missiles generated outside containment resulting from unsecured and non-seismically restrained compressed gas cylinders during a seismic event. Accordingly, **RAI 127-1641, Question 03.05.01.01-1, Subquestion 3.5.1.1-04, is resolved**.

Section 13.5, of this report addresses the staff's evaluation of plant procedures, including procedures to remove or seismically restrain equipment, such as a hoist that is used during maintenance, when not in use to prevent it from becoming a missile.

In DCD Tier 2, Revision 1, Section 3.5.1.1 the applicant stated that the following components also had the potential to produce missiles:

- RV.

- Control rod drive mechanism.
- Fittings of RV.

It was not clear to the staff how the above cited components located inside the containment would become potential sources of missiles outside the containment. Therefore, in an **RAI 127-1641, Question 03.05.01.01-1, Subquestion 3.5.1.1-05**, the staff requested the applicant to provide detailed discussion in the DCD to demonstrate that the above cited components would become potential sources of missiles outside the containment.

In its response to **RAI 127-1641, Question 03.05.01.01-1, Subquestion 3.5.1.1-05**, dated January 28, 2009, in order to clarify that the components identified in the staff's request for information are potential missiles inside containment and not applicable to DCD Tier 2, Section 3.5.1.1, the applicant proposed to re-format the affected sections in DCD Revision 2 to move any reference to items unique to inside containment from Section 3.5.1.1 to Section 3.5.1.2, "Internally Generated Missiles (Inside Containment)." The staff finds the applicant's response acceptable since it clarifies that the above cited components are potential missiles inside containment and meant to be discussed under DCD Tier 2 Section 3.5.1.2. The staff has confirmed that DCD Tier 2, Revision 2, was revised as committed in the RAI response. Accordingly, **RAI 127-1641, Question 03.05.01.01-1, Subquestion 3.5.1.1-05, is resolved**.

Based on the review described above, the staff finds the applicant's approach to identify potential missiles, determine the statistical significance of potential missiles, and provide measures for SSCs needing protection against the effects of missiles to be acceptable. Also, in DCD Tier 2, Table 3.2-2, the applicant lists SSCs that need missile protection and in DCD Tier 2, Section 7.4.1, "System Description," lists the safe-shutdown systems. The staff finds that the scope of the SSCs afforded missile protection conforms to the guidance of RG 1.115, Positions C.1 and C.3, and RG 1.117, Appendix A. Therefore, the staff concludes that the applicant's evaluation of potential internally-generated missiles outside the US-APWR containment resulting from equipment and component failures satisfies GDC 4.

3.5.1.1.4.2 Inspections, Tests, Analyses, and Acceptance Criteria

In DCD Tier 2, Revision 1, Section 3.5.1.1, the applicant described its approach to identify potential missiles, determine the potential credible and not credible missiles, and provide measures for safety-related SSCs requiring protection against the effects of missiles outside containment. However, DCD Tier 1, Chapter 2.0, did not contain an ITAAC to verify that safety-related SSCs outside containment are designed and constructed in accordance with the requirements as described in DCD Tier 2, Section 3.5.1.1, to prevent or mitigate the effects of internally-generated missiles outside containment.

Therefore, in **RAI 127-1641, Question 03.05.01.01-1, Subquestion RAI 3.5.1.1-02,** the staff requested the applicant to provide an ITAAC that requires COL applicant to perform a walk-down of the SSCs to ensure that safety-related SSCs described in the above cited sections are protected from internally-generated missiles (outside containment) in accordance with the requirements as described in DCD Tier 2 Section 3.5.1.1. The staff also requested that the applicant identify in the DCD which of the SSCs are outside and which of the SSCs are inside the containment.

In its response to **RAI 127-1641, Question 03.05.01.01-1, Subquestion 3.5.1.1-02**, dated January 28, 2009, the applicant stated that DCD Tier 2, Section 3.2, "Classification of

Structures, Systems, and Components," and Section 3.11, "Environmental Qualification of Mechanical and Electrical Equipment," list the SSCs requiring missile protection and their locations (Inside and Outside Containment). In addition, the applicant proposed to add: a new subsection (Subsection 2.2.2.5, "Internally Generated Missiles (Inside and Outside Containment)") to DCD Tier 1, Section 2.2.2, "Protection against Hazards," to discuss the protection of safety-related SSCs against credible missiles from internal sources inside and outside the containment; and an ITAAC (Item 19) to DCD Tier 1, Table 2.2-4, "Structural and Systems Engineering Inspections, Tests, Analyses, and Acceptance Criteria," to verify that safety-related SSCs inside and outside the containment are protected from credible missiles.

The staff finds the applicant's response to **RAI 127-1641, Question 03.05.01.01-1, Subquestion 3.5.1.1-02,** acceptable because DCD Tier 2, Section 3.2 and Section 3.11 list the SSCs requiring missile protection and their locations (both inside and outside containment), and the proposed ITAAC will verify these SSCs inside and outside the containment are protected from credible missiles.. The staff notes that the ITAAC regarding protection of safety-related SSCs against credible missiles from internal sources inside and outside the containment was moved from Item 19 to Item 21 in DCD Tier 1, Revision 2, Table 2.2-4. Otherwise, the staff has confirmed that DCD Tier 2, Revision 2, was revised as committed in the RAI response. Accordingly, **RAI 127-1641, Question 03.05.01.01-1, Subquestion 3.5.1.1-02, is resolved**.

In DCD Revision 3, the applicant reorganized DCD Tier 1, Section 2.2.2 to move the design description of internally-generated missiles (inside and outside containment) from DCD Tier 1, Section 2.2.2.5 to DCD Tier 1, Section 2.2.2.1. In addition, the description of the inspections, tests and analyses and the description of the acceptance criteria were revised to reflect the need for both analysis and testing, and to provide a reference to acceptable missile protection methods in the relocated DCD Tier 1, Section 2.2.2.1. The staff finds the ITAAC acceptable since the methods in DCD Tier 1, Section 2.2.2.1 are consistent with those the staff has reviewed and accepted, as described above in DCD Tier 2, Section 3.5.1.1. The ITAAC also confirm that the as built safety-related SSCs are protected from credible missile sources. Therefore, the staff concludes that the ITAAC for missile protection provided for US-APWR plant safety-related SSCs complies with the requirements of 10 CFR 52.47(b)(1).

3.5.1.1.4.3 Initial Testing

DCD Tier 2, Revision 2, Sections 14.2, "Initial Plant Test Program," does not have any initial testing requirements associated with this review item. The staff reviewed DCD Tier 2, Revision 2, Section 3.5.1.1, against the guidance in SRP Section 14.2, "Initial Plant Test Program," Revision 3, issued March 2007, and found that no additional initial testing is needed in connection with this section.

3.5.1.1.4.4 Technical Specifications

DCD Tier 2, Revision 2, Chapter 16, "Technical Specifications," does not have any TS requirements associated with this review item. The staff reviewed DCD Tier 2, Revision 2, Section 3.5.1.1 against 10 CFR 50.36, "Technical Specifications," and agrees that no TS are needed in connection with this section.

3.5.1.1.5 Combined License Information Items

The following is a list of COL item numbers and descriptions from Table 1.8-2 of the DCD related to internally-generated missiles outside containment:

Table 3.5.1.1-1
US-APWR Combined License Information Items

Item No.	Description	Section
COL 3.5(1)	The COL Applicant is to have plant procedures in place prior to fuel load that specify unsecured equipment, including portable pressurized gas cylinders, located inside or outside containment and required for maintenance or undergoing maintenance is to be removed from containment prior to operation, moved to a location where it is not a potential hazard to SSCs important to safety, or seismically restrained to prevent it from becoming a missile.	3.5.1.1.4

The staff finds the above listing to be complete. Also, the list adequately describes actions necessary for the COL applicant. No additional COL information items need to be included in DCD Tier 2, Table 1.8-2 for internally-generated missile considerations.

3.5.1.1.6 Conclusions

Based on the foregoing, the staff concludes that the US-APWR design satisfies the guidelines of SRP Section 3.5.1.1, Revision 3. Therefore, the staff concludes that adequate protection has been provided for the US-APWR plant to protect safety-related SSCs against internally-generated missiles outside the containment and the US-APWR design meets the requirements of GDC 4, and 10 CFR 52.47(b)(1) regarding internally-generated missiles outside containment.

3.5.1.2 Internally Generated Missiles (Inside Containment)

3.5.1.2.1 Introduction

This section discusses operations and performance requirements for SSCs inside containment, identification of SSCs inside containment necessary for the safe shutdown of the reactor and the failure of SSCs inside containment that could cause a significant release of radioactivity. Also discussed is the adequacy of methods of protection from internally-generated missiles for SSCs inside containment necessary to perform functions required to attain and maintain a safe shutdown or to mitigate the consequences of an accident.

3.5.1.2.2 Summary of Application

DCD Tier 1: The Tier 1 information associated with this section is found in DCD Tier 1, Section 2.2, “Structural and Systems Engineering.”

DCD Tier 2: The applicant has provided a DCD Tier 2 system description in Section 3.5.1.2, “Internally Generated Missiles (Inside Containment),” summarized here in part, as follows:

> This section discusses operations and performance requirements for SSCs inside containment, identification of SSCs necessary for the safe shutdown of the reactor and the failure of SSCs that could cause a significant release of radioactivity. Also discussed is the adequacy of methods of protection from internally-generated missiles for SSCs necessary to perform functions required to attain and maintain a safe shutdown or to

mitigate the consequences of an accident.

ITAAC: The ITAAC associated with DCD Tier 2, Section 3.5.1.2 is given in DCD Tier 1, Section 2.2.4, "Inspection, Tests, Analyses, and Acceptance Criteria."

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 for this area of review.

Cross-cutting Requirements (TMI, USI/ GSI, Op Ex): There are no cross-cutting requirements for this area of review.

US-APWR Interface Issues Identified in the DCD: See DCD Tier 2, Table 1.8-2, "Compilation of All Combined License Applicant Items for Chapters 1-19."

Site Interface Issues Identified in the DCD: See DCD Tier 2, Table 1.8-1, "Significant Site Specific Interfaces with the Standard US-APWR Design."

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

3.5.1.2.3 Regulatory Basis

The relevant requirements of the Commission's regulations for this area of review, and the associated acceptance criteria are given in Section 3.5.1.2, "Internally Generated Missiles (Inside Containment)," Revision 3, issued March 2007, of NUREG-0800, the SRP, and are summarized below. Review interfaces with other SRP sections can be found in Section 3.5.1.2 of NUREG-0800.

1. GDC 4, as it relates to the protection of SSCs against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that may result from equipment failures and from events and conditions inside the nuclear power unit.

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 DC has been constructed and will operate in accordance with the DC.

Acceptance criteria adequate to meet the above requirements are given in SRP Section 3.5.1.2.

The design of the SSCs important to safety is acceptable if the integrated design affords protection from the internally-generated missiles (inside containment) that may result from equipment failure, in order to maintain their safety functions in accordance with GDC 4.

1. The applicant's statistical significance of an identified missile can be evaluated by a probability analysis. The statistical significance for a potential missile is determined by calculating the probability of missile occurrence. If this probability is less than 10^{-7} per year, the missile is not considered significant. If the probability of occurrence is greater than 10^{-7} per year, the probability that it will

impact a significant target is determined. If the product of these two probabilities is less than 10^{-7} per year, the missile is not considered significant. If the product is greater than 10^{-7} per year, the probability of significant damage is determined. If the combined probability (product of all three) is less than 10^{-7} per year, the missile is not considered significant. If the combined probability is greater than 10^{-7} per year, missile protection of SSCs important-to-safety, and of nonsafety-related SSCs whose failure could affect an intended safety function of the safety-related SSCs, should be provided by one or more of the six methods listed below.

2. The missile protection for SSCs important to safety is adequate if provided by one or more of the following methods: (1) locating the system or component in a missile-proof structure, (2) separating redundant systems or components for the missile path or range, (3) providing shields and barriers for systems and components, (4) designing the equipment to withstand the impact of the most damaging missile, (5) providing design features to prevent the generation of missiles, or (6) orienting missile sources to prevent missiles from striking equipment important to safety.

In summary, a Safety Analyses Report (SAR) statement that SSCs important to safety will be afforded protection by locating them in individual missile-proof structures, physically separating redundant systems or system components, or providing special protective shields or barriers is an acceptable method to meet this criterion.

3.5.1.2.4 Technical Evaluation

The staff reviewed the US-APWR design for protecting SSCs important to safety against internally-generated missiles (inside containment) in accordance with the guidance of SRP Section 3.5.1.2, Revision 3, issued March 2007. The staff reviewed DCD Tier 2, Revision 3, Section 3.5.1.2. The staff also reviewed DCD Tier 1, Revision 3, Section 2.0, “Design Descriptions and ITAAC,” and other DCD Tier 2 sections noted below.

DCD Tier 2, Section 3.5, “Missile Protection,” addresses SSCs to be protected from internally-generated missiles inside and outside containment. DCD Tier 2 Table 3.2-2, “Classification of Mechanical and Fluid Systems, Components, and Equipment,” lists all the SSCs (safety-related and nonsafety-related) in various locations of the plant (inside and outside the containment) and identifies for each SSC the associated seismic category, quality group and equipment classifications. DCD Tier 2, Table 7.4-1, “Component Controls for Shutdown,” lists the systems and components required for safe shutdown. General arrangement drawings further defining the building locations are provided in DCD Tier 2, Section 1.2, “General Plant Description.”

In DCD Tier 2, Section 3.5.1.2, the applicant evaluated the potential internally-generated missiles that could result from failure of the plant equipment located inside the containment. These potential missiles internally-generated inside containment are:

- Missiles resulting from in-plant rotating equipment overspeed failures.
- Missiles resulting from in-plant high-pressure system ruptures such as valves, piping, fittings, control rod drive mechanism (CRDM), valve bonnets, valve stems, and pressure vessel.

- Components (e.g. missiles originating from the RV, SG, RCP pressurizer, etc.) and piping, within the reactor coolant pressure boundary.
- RCP flywheel.
- Gravitational missiles including falling objects generated by non-seismic SSCs during a seismic event, and unsecured maintenance equipment that is required during maintenance and is not either removed during operation to a location where it is not a potential hazard to safety-related equipment or seismically restrained to prevent it from becoming a missile.

Once a potential missile is identified, its statistical significance is determined by the combined probability of an event that is defined as the product of:

- The probability of missile occurrence.
- The probability of impact on a significant target.
- The probability of significant damage.

If the combined probability associated with a potential missile is greater than $1x10^{-7}$ per year, the missile is considered as credible and protection of safety-related SSCs against the credible missile will be provided. If the combined probability associated with a potential missile is less than $1x10^{-7}$ per year, the event is considered not statistically significant, the missile is considered as not credible, and protection of safety-related SSCs against the not credible missile would not be provided.

The discussion in DCD Tier 2, Section 3.5.1.1, “Internally Generated Missiles (Outside Containment),” regarding missiles not considered credible is also applicable to missiles generated inside the containment. In addition, the applicant identified the non-credible missiles generated from the following internally sources inside containment and provided rationale to justify why they are not credible missiles:

- Components (e.g. missiles originating from the RV, SG, RCP pressurizer, etc.) and piping, within the reactor coolant pressure boundary.
- RCP flywheel.
- CRDM.

Once a potential credible missile is identified, the applicant stated that protection of safety-related SSCs against the credible missile will be provided by one or more of the following methods:

- Locating the system or component in a missile-proof structure.
- Separating redundant systems or components of the system from the missile path or range.
- Providing local shields or barriers for systems and components.
- Designing the equipment to withstand the impact of the most damaging missile.
- Providing design features to prevent the generation of missile.
- Orientating a missile source to prevent missiles from striking equipment important to safety.

Section 3.5.3 of this report addresses the staff’s evaluation of the design of structures, shields, and barriers required for missile protection.

Section 3.6.2 of this report addresses the staff's evaluation of the dynamic effects associated with the postulated rupture of piping inside the containment.

Section 3.7.3 of this report addresses the staff's evaluation of the impact of the fall or overturn of non-seismic components on safety-related SSC as a result of a seismic event.

In reviewing DCD Tier 2, Section 3.5.1.2, regarding protection for safety-related SSCs against internally-generated missiles inside containment, the staff identified areas in which additional information was necessary to complete its review.

For the postulated missiles inside containment, in DCD Tier 2, Revision 1,Section 3.5.1.2.2.1, "Missiles Not Considered Credible," the applicant referred to Section 3.5.1.1, for discussion of its rationale to exclude certain types of equipment (e.g. missiles originating from valves, threaded connections and piping in high energy systems, gravitational missiles resulting from non-seismic SSCs during a seismic event, unsecured maintenance equipment, etc.) from consideration as credible missile sources. However, in DCD Tier 2, Revision 1, Section 3.5.1.1, the applicant did not provide the analysis to demonstrate that these missiles were of insufficient energy to cause unacceptable impact or to cause unacceptable damage.

Similar to the request described in **RAI 127-1641, Question 03.05.01.01-1, Subquestion 3.5.1.1-01,** for DCD Tier 2, Revision 1, Section 3.5.1.1, the staff in **RAI 152-1642, Question 03.05.01.02-1, Subquestion 3.5.1.2-01,** requested the applicant to demonstrate in the DCD how these design features and qualitative considerations ensured a level of protection from missiles that was equivalent to the probability criteria specified in the SRP.

In its response to **RAI 152-1642, Question 03.05.01.02-1, Subquestion 3.5.1.2-01**, dated February 4, 2009, the applicant proposed to revise DCD, Tier 2, Section 3.5.1.2 in Revision 2 to reflect that for certain SSCs postulated as capable of generating missiles, the probability of missile occurrence (P_1), the product of the probability of missile occurrence and probability of missile impact (P_1 x P_2), or the combined product of the probability of missile occurrence, probability of missile impact, and probability of significant damage (P_1 x P_2 x P_3) demonstrate through probabilistic analyses that the events are not statistically significant.

Based on its review and evaluation of the applicant's response to **RAI 127-1641, Question 03.05.01.01-1, Subquestion 3.5.1.1-01,** regarding the use of PRA methodology to determine the credibility of internally-generated missiles, the staff finds the applicant's response to **RAI 152-1642, Question 03.05.01.02-1, Subquestion 3.5.1.2-01,** acceptable because it follows the guidance described in SRP Section 3.5.1.1 for probabilistic analyses to determine which missiles might be non-credible. The staff has confirmed that DCD Tier 2, Revision 2, was revised as committed in the RAI response. Accordingly, **RAI 152-1642, Question 03.05.01.02-1, Subquestion 3.5.1.2-01, is resolved**.

In DCD Tier 2, Revision 1 Section 3.5.1.2.2.3, "Credible Sources of Internally Generated Missiles (Inside Containment)," the applicant described two credible sources (items containing high-energy fluids and high-speed rotating equipment) of internally-generated missiles inside containment. Items containing high-energy fluids were dismissed as a credible source of an internally-generated missile, given that all high-energy systems within the PCCV comply with ASME Code, Section III. Reference was made to a few nonsafety-related, high-speed rotating equipment items that remained as credible sources of internally-generated missiles inside containment. However, information provided in the DCD Tier 2, Revision 1 did not provide a specific listing of the nonsafety-related, high-speed rotating equipment items located inside containment that might be the source of credible missiles, or potential damage to or failure of SSCs important to safety as a result of missile impingement.

Therefore, in **RAI 152-1642, Question 03.05.01.02-1, Subquestion 3.5.1.2-02**, the staff requested the applicant to demonstrate in the DCD how SSCs inside containment were afforded protection from the nonsafety-related, high-speed rotating equipment items that remain as credible sources of internally-generated missiles inside containment.

In its response to **RAI 152-1642, Question 03.05.01.02-1, Subquestion 3.5.1.2-02**, dated February 4, 2009, the applicant stated that various design features and administrative controls are used to prevent nonsafety-related, high-speed rotating equipment from being a source of credible missiles.

In general, the probability of occurrence, P_1, of rotating equipment is maintained less than 10^{-7} by virtue of the equipment design and manufacturing criteria. Justification for a low P_1 includes the fact that rotating equipment energized by alternating current power are governed by the frequency of the power supply. The narrow range of frequency variation for the alternating current power supply makes it highly unlikely that an overspeed condition of rotating equipment can occur. While it is postulated that missiles such as a fan blade may occur at rated speeds, the design of the casing prevents missile penetration. However, in the unlikely case of a high-speed rotating component penetrating the casing and P_1, is greater than 10^{-7} the probability of impact, P_2, is also considered. P_2 is minimized by locating a potential missile source or potential target outside the zone of postulated missile strike, by the robust building walls and slabs that are designed for applicable missile strikes, or the separation of missile sources from potentially impacted SSCs important to safety. When considering both the probability of occurrence and the probability of impact, the product of P_1 *x* P_2 is less than 10^{-7}, and therefore high-speed rotating equipment is not a credible missile source.

The applicant committed to revising information in DCD Tier 2, Subsections 3.5.1.1.2, “Missile Selection,” and 3.5.1.2.2, “Missile Selection,” to reflect the missile protection measures described above.

Based on its review of the applicant’s response to **RAI 152-1642, Question 03.05.01.02-1, Subquestion 3.5.1.2-02,** regarding missile protection from nonsafety-related high-speed rotating equipment, the staff finds the applicant’s response acceptable because it follows the guidance described in SRP Section 3.5.1.2 for statistical significance in determining the credibility of a missile. The staff has confirmed that DCD Tier 2, Revision 2, was revised as committed in the RAI response. Accordingly, **RAI 152-1642, Question 03.05.01.02-1, Subquestion 3.5.1.2-02, is resolved**.

Section 13.5, of this report addresses the staff’s evaluation of plant procedures, including procedures to remove or seismically restrain equipment, such as a hoist that is used during maintenance, when not in use to prevent it from becoming a missile.

Based on the review described above, the staff finds the applicant’s approach to identify potential missiles, determine the statistical significance of potential missiles, and provide measures for SSCs needing protection against the effects of missiles to be acceptable. Also, in DCD Tier 2, Table 3.2-2, the applicant lists SSCs that need missile protection and in DCD Tier 2, Section 7.4.1, “System Description,” lists the safe-shutdown systems. Therefore, the staff concludes that the applicant’s evaluation of potential internally-generated missiles inside the

US-APWR containment resulting from equipment and component failures satisfies GDC 4 and is consistent with the recommendations described in SRP Section 3.5.1.2.

3.5.1.2.4.1 Inspections, Tests, Analyses, and Acceptance Criteria

In DCD Tier 2, Revision 1, Section 3.5.1.2, the applicant referred to Section 3.5.1.1 for discussion of its approach to identify potential missiles, determined the statistical significance of potential missiles, and provided measures for SSCs requiring protection against the effects of missiles inside containment. However, DCD Tier 1, Chapter 2.0, did not contain an ITAAC to verify that SSCs inside containment are designed and constructed in accordance with the requirements as described in DCD Tier 2, Section 3.5.1.2 to prevent or mitigate the effects of internally-generated missiles inside containment.

Therefore, in **RAI 152-1642, Question 03.05.01.02-1, Subquestion 3.5.1.2-03**, the staff requested the applicant to provide a commitment in the ITAAC that requires COL applicant to verify that SSCs and missile sources inside containment are designed and constructed in accordance with the requirements as described in DCD Tier 2, Section 3.5.1.2.

In its response to **RAI 152-1642, Question 03.05.01.02-1, Subquestion 3.5.1.2-03,** the applicant referred to its response to **RAI 127-1641, Question 03.05.01.01-1, Subquestion 3.5.1.1-02**. The response to **RAI 127-1641, Question 03.05.01.01-1, Subquestion 3.5.1.1-02** commits to add discussion during Revision 2 of DCD Tier 1, Subsection 2.2.2.5, “Internally Generated Missiles (Inside and Outside Containment),” regarding protection of safety-related SSCs against credible missiles from internal sources inside and outside the containment, and to provide an ITAAC (Item 19) in DCD Tier 1, Table 2.2-4, to verify that SSCs inside and outside the containment are protected from credible missiles.

As stated in staff’s evaluation of the applicant’s response to **RAI 127-1641, Question 03.05.01.01-1, Subquestion 3.5.1.1-02**, the staff finds the applicant’s response to **RAI 127-1641, Question 03.05.01.01-1, Subquestion 3.5.1.1-02,** acceptable. The staff notes that the ITAAC regarding protection of safety-related SSCs against credible missiles from internal sources inside and outside the containment was moved from Item 19 to Item 21 in DCD Tier 1, Revision 2, Table 2.2-4. Otherwise, the staff has confirmed that DCD Tier 1 and Tier 2, Revision 2, were revised as committed in **RAI 127-1641, Question 03.05.01.01-1, Subquestion 3.5.1.1-02,** response. Accordingly, **RAI 152-1642, Question 03.05.01.02-1, Subquestion 3.5.1.2-03 is resolved**.

In DCD Revision 3, the applicant reorganized DCD Tier 1, Section 2.2.2 to move the design description of internally-generated missiles (inside and outside containment) from DCD Tier 1, Section 2.2.2.5 to DCD Tier 1, Section 2.2.2.1. In addition, the description of the inspections, tests and analyses and the description of the acceptance criteria were revised to reflect the need for both analysis and testing, and to provide a reference to acceptable missile protection methods in the relocated DCD Tier 1, Section 2.2.2.1. The staff finds the ITAAC acceptable since the methods in DCD Tier 1, Section 2.2.2.1 are consistent with those the staff has reviewed and accepted, as described above in DCD Tier 2, Section 3.5.1.2. The ITAAC also confirm that the as built safety-related SSCs are protected from credible missile sources. Therefore, the staff concludes that the ITAAC for missile protection provided for US-APWR plant safety-related SSCs complies with the requirements of 10 CFR 52.47(b)(1).

3.5.1.2.4.2 Initial Testing

DCD Tier 2, Section 14.2, “Initial Plant Test Program,” does not have any initial testing requirements associated with this review item. The staff reviewed DCD Tier 2, Section 3.5.1.2 against the guidance in SRP Section 14.2, “Initial Plant Test Program,” Revision 3, issued March 2007, and found that no additional initial testing is needed in connection with this section.

3.5.1.2.4.3 Technical Specifications

DCD Tier 2 Chapter 16, “Technical Specifications,” does not have any TS requirements associated with this review item. The staff reviewed DCD Tier 2, Section 3.5.1.2 against 10 CFR 50.36, “Technical Specifications,” and agrees that no TS are needed in connection with this section.

3.5.1.2.5 Combined License Information Items

The following is a list of COL item numbers and descriptions from Table 1.8-2, “Compilation of All Combined License Applicant Items for Chapters 1-19,” of the DCD related to internally-generated missiles inside containment:

Table 3.5.1.2-1
US-APWR Combined License Information Items

Item No.	Description	Section
COL 3.5(1)	The COL Applicant is to have plant procedures in place prior to fuel load that specify unsecured equipment, including portable pressurized gas cylinders, located inside or outside containment and required for maintenance or undergoing maintenance is to be removed from containment prior to operation, moved to a location where it is not a potential hazard to SSCs important to safety, or seismically restrained to prevent it from becoming a missile.	3.5.1.1.4

The staff finds the above listing to be complete. Also, the list adequately describes actions necessary for the COL applicant. No additional COL information items need to be included in DCD Tier 2, Table 1.8-2 for internally-generated missile considerations.

3.5.1.2.6 Conclusions

Based on the foregoing, the staff concludes that the US-APWR design satisfies the guidelines of SRP Section 3.5.1.2, Revision 3. Therefore, the staff concludes that adequate protection has been provided for the US-APWR plant to protect safety-related SSCs against internally-generated missiles inside containment and the US-APWR design meets the requirements of GDC 4, and 10 CFR 52.47(b)(1) regarding internally-generated missiles inside containment.

3.5.1.3 Turbine Missiles

3.5.1.3.1 Introduction

GDC 4 requires that SSC’s important to safety shall be designed and protected against the effects of missiles that might result from equipment failures. The failure of a rotor in a large steam turbine may result in the generation of high-energy missiles that could affect safety-related SSCs. The probability of a strike by a turbine missile should be sufficiently low such that the risk from turbine missiles on safety-related SSCs is acceptably small.

3.5.1.3.2 Summary of Application

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

DCD Tier 2: The applicant has provided a DCD Tier 2 description in Section 3.5.1.3, "Turbine Missiles," of potential turbine missiles and turbine missile generation evaluation, summarized here in part, as follows:

> Protection against damage from turbine missiles to safety-related SSCs is provided by the turbine-generator (T/G), by the robust turbine rotors, and by the redundant and failure safe turbine design control system.
>
> The T/G is located south of the nuclear island with the shaft oriented along the north-south axis. In this orientation, the potential for low trajectory turbine missiles to impact safety-related SSCs within the same unit is minimized since safety-related SSCs are located outside the high-velocity, low-trajectory missile strike zone. The T/G and associated equipment, with respect to essential safety-related SSCs, are shown in DCD Tier 2, Section 1.2, "General Plant Description." The COL applicant is responsible for assessing the orientation of the T/G at a multi-unit site.
>
> The rotor design, material selection, pre-service and in-service programs and redundant T/G control system support the low probability of turbine missile generation. The turbine rotor design, material selection, fracture toughness/fracture analysis, and the inspection and testing program that is used to maintain an acceptably low probability of missile generation are evaluated in Section 10.2 of this report. The COL applicant is to commit to actions to maintain the probability of turbine failure resulting in the ejection of turbine rotor (or internal structure) fragments through the turbine casing (P_1) to less than 10^{-5}, based on the applicant's Technical Report MUAP-07028-P, "Probability of Missile Generation From Low Pressure Turbines," Revision 1, issued January 2011.

ITAAC: The ITAAC associated with this area of review are specified in DCD Tier 1, Section 2.7.1.1, "Turbine Generator (T/G)." The specific ITAAC are given in DCD Tier 1, Table 2.7.1.1-1, "Turbine Generator Inspections, Tests, Analyses, and Acceptance Criteria," Design Commitment 2.

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

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

Technical Reports: The technical reports associated with DCD Tier 2, Section 3.5.1.3 are as follows:

1. MUAP 07028-P, "Probability of Missile Generation From Low Pressure Turbines," Revision 0, issued December 2007.

2. MUAP 07028-P, "Probability of Missile Generation From Low Pressure Turbines," Revision 1, issued January 2011.

3. MUAP-07029-P, "Probabilistic Evaluation of Turbine Valve Test Frequency," Revision 0, issued December 2007.

4. MUAP-07029-P, “Probabilistic Evaluation of Turbine Valve Test Frequency,” Revision 1, issued January 2011.

Cross-cutting Requirements (TMI, USI/ GSI, Op Ex): There are no cross-cutting requirements for this area of review.

US-APWR Interface Issues Identified in the DCD: See DCD Tier 2, Table 1.8-2, “Compilation of All Combined License Applicant Items for Chapters 1-19.”

Site Interface Issues Identified in the DCD: See DCD Tier 2, Table 1.8-1, “Significant Site Specific Interfaces with the Standard US-APWR Design.”

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

3.5.1.3.3 Regulatory Basis

The relevant requirements of the Commission’s regulations for this area of review, and the associated acceptance criteria, are given in Section 3.5.1.3, “Turbine Missiles,” Revision 3, issued March 2007, of NUREG-0800, the SRP, and are summarized below. Review interfaces with other SRP sections can be found in Section 3.5.1.3 of NUREG-0800.

1. GDC 4 requires SSCs important to safety to be appropriately protected against environmental and dynamic effects, including the effects of missiles that may result from equipment failure. Failure of the large steam turbine rotor at a high rotating speed could generate high-energy missiles that have the potential to damage SSCs important to safety.

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 DC has been constructed 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. Per SRP Section 3.5.1.3 Item II.1, criteria adequate to meet the requirements of GDC 4 to reduce the probability of turbine missile generation are: (a) the integrity of the reactor coolant pressure boundary; (b) the capability to shut down and maintain the reactor in a safe condition; and (c) the capability to prevent accidents that could result in potential offsite exposure, which represent a significant fraction of the guideline exposures specified in 10 CFR Part 100, “Reactor Site Criteria.” Examples of safety-related systems that should be protected are described in the appendix to RG 1.117, “Tornado Design Classification.”

2. RG 1.115, “Protection Against Low-Trajectory Turbine Missiles,” Revision 1, issued July 1977, as it relates to the identification of low-trajectory missiles resulting from turbine failure.

3. The minimum reliability values (i.e., P_1) for loading the turbine and bringing the system on line are that the minimum reliability values of P_1 should be less than $1x10^{-4}$ per reactor year for favorably oriented turbines, and be less than $1x10^{-5}$ per reactor year for unfavorably oriented turbines. The P_1 calculation is related to maintenance and inspection of turbine rotors and control valves, operating experience of similar equipment, and inspection results.

3.5.1.3.4 Technical Evaluation

The failure of a rotor in a large steam turbine may result in the generation of high energy missiles that could affect safety-related SSCs. The probability of a strike by a turbine missile should be sufficiently low so that the risk from turbine missiles on safety-related SSCs are acceptably small. DCD Tier 2, Section 3.5.1.3 provides information that the probability of the favorably oriented T/G generating a turbine missile is less than 1×10^{-4}. The staff reviewed this information using the guidelines in SRP Section 3.5.1.3.

SRP Section 3.5.1.3 states that with the use of proper turbine rotor design, materials that satisfy the acceptance criteria in SRP Section 10.2.3, "Turbine Rotor Integrity," Revision 2, issued March 2007, and acceptable preservice and inservice nondestructive examination methods, the probability of turbine missile generation, P_1, is expected to be no greater than $1x10^{-5}$ per reactor year for an unfavorably oriented turbine and no greater than $1x10^{-4}$ for a favorably oriented turbine. This probability represents the general minimum reliability requirement for loading the turbine and bringing the system on line.

DCD Tier 2, Subsection 3.5.1.3.1, "Geometry," states that its T/G is located south of the nuclear island and oriented along the north-south axis, and safety-related equipment is located outside of the high-velocity, low-trajectory missile strike zone. Based on this information and the corresponding figures in DCD Tier 2, Section 1.2, "General Plant Description," the US-APWR T/G is favorably oriented with respect to RG 1.115, Revision 1, since safety-related equipment described in DCD Tier 2, Section 1.2.1.7, "Plant Arrangement," is located outside of the high velocity, low-trajectory turbine missile strike zone.

DCD Tier 1, Revision 1, Table 2.7.1.1-1, provided an ITAAC (commitment 1) for the arrangement of the T/G. The acceptance criteria for this ITAAC stated, "the as-built T/G conforms to the functional arrangement as described in Subsection 2.7.1.1.1." However, the NRC staff noted that DCD Tier 1, Subsection 2.7.1.1.1, "Design Description," did not provide arrangement criteria, except that the orientation of T/G was such that high-energy missiles would be directed at approximately 90 degrees away from safety-related equipment. This was not completely consistent with RG 1.115, Revision 1, and DCD Tier 2, Revision 1, Section 3.5.1.3.1, which stated the high energy, low-trajectory missile strike zone was the area bounded by lines inclined at 25 degrees to the turbine wheel planes and passing through the end wheels of the low pressure stages. Therefore, in **RAI 323-2071, Question 03.05.01.03-3**, the staff requested the applicant to clarify the ITAAC or DCD Tier 2, Section 2.7.1.1.1 to provide accurate acceptance criteria, or reference the criteria of the applicable DCD Tier 2, Section or RG 1.115, Revision 1. In its response to **RAI 323-2071, Question 03.05.01.03-3**, dated May 20, 2009, the applicant modified DCD Tier 1, Section 2.7.1.1 to be consistent with RG 1.115, Revision 1. The staff finds the response acceptable since it is consistent with staff guidance in RG 1.115, Revision 1. The staff confirmed that the DCD changes were incorporated into DCD Revision 2. Accordingly, **RAI 323-2071, Question 03.05.01.03-3, is resolved.**

In **RAI 782-5910, Question 14.03.07-58**, the staff requested the applicant include acceptance criteria for the T/G arrangement and for the turbine missile probability in the DCD Tier 1, Section 2.7.1.1 ITAAC. The staff noted that acceptable ITAAC were included in DCD Revision 2 but were modified in DCD Revision 3 to be unacceptable. As the resolution of this issue will impact DCD Tier 2, Section 3.5.1.3, **RAI 782-5910, Question 14.03.07-58, is being tracked as an Open Item.**

DCD Tier 2, Section 3.5.1.3.2, "Evaluation," indicates, due to the favorably oriented T/G, that the probability of a turbine failure resulting in the ejection of a turbine missile (P_1) is acceptable if it is less than $1x10^{-4}$. The applicant relies on the turbine missile methodology and the analytical results documented in the applicant's technical reports MUAP-07028-P, Revision 0, and MUAP-07029-P, Revision 0.

MUAP-07028-P, Revision 0, provides the turbine missile analysis for the probability of generating missiles by assessing the potential of the turbine rotor failure due to fracture from high cycle fatigue (HCF) cracking, low cycle fatigue (LCF) cracking and fracture due to stress corrosion cracking. This report also provides a deterministic analysis on turbine rotor failure due to ductile burst from destructive overspeed condition. A detailed evaluation on ductile burst is provided in MUAP-07029-P, Revision 0, which uses T/G failure data to determine the probability of a turbine missile generation at destructive overspeed due to the failure of the overspeed protection system.

Concerning MUAP-07028-P, Revision 0, in **RAI 324-1997, Question 03.05.01.03-4, Subquestion 03.05.01.03-01,** the staff requested the applicant to clarify whether this report bounded all of the material used in the turbine rotor specified in DCD Tier 2, Section 10.2.3, "Turbine Rotor Integrity." The material specification to be used is necessary to assure that turbine rotor will have adequate material properties including fracture toughness. In its response to **RAI 324-1997, Question 03.05.01.03-4**, **Subquestion 03.05.01.03-01**, dated May 20, 2009, the applicant stated that the material of the low pressure (LP) turbine rotor is made from Ni-Cr-Mo-V alloy steel that is equivalent to ASTM A470 Classes 5, 6 and 7. The purchase specification of this material requires a minimum yield strength, and this minimum yield strength is used in the MUAP-07028-P, Revision 0. Based on this, the applicant stated that the class of material that is not bounded by the report will be deleted from the US-APWR DCD. Therefore, the staff found the response acceptable with respect to the applicant clarifying the rotor materials properties. However, DCD Tier 2, Revision 2, Section 10.2.3.1, "Materials Selection," was modified by deleting the reference to all of the class material. The staff notes that DCD Tier 2, Revision 2, Section 10.2.3.1 allows all classes of material in ASTM A470, which are not bounded by the MUAP-07028-P, Revision 0. Therefore, the staff notes that the material class (i.e., ASTM A470, Grade C, Class 7) that bounds the report should be included in the DCD. This issue is addressed by **RAI 574-4633, Question 10.02.03-8** and **RAI 574-4633, Question 10.02.03-10** and further evaluation of this issue is discussed in Section 10.2.3 of this report. Accordingly, as the staff found the remaining aspects of the response acceptable, **RAI 324-1997, Question 03.05.01.03-4**, **Subquestion 03.05.01.03-01, is resolved.**

In addition, to validate the material property homogeneity of the turbine rotor in **RAI 324-1997, Question 03.05.01.03-5, Subquestion 03.05.01.03-02**, the staff requested the applicant to identify where the material testing will be performed, especially at the bore region. In its response to **RAI 324-1997, Question 03.05.01.03-5, Subquestion 03.05.01.03-02**, dated May 20, 2009, the applicant stated that the material testing is performed on the outside periphery of the large diameter portion of the rotor and not at the center. In addition, the applicant stated that the chemical composition and mechanical properties of the rotor core can be evaluated by the test results of the rotor periphery based on a comparison of material test results of a similar sized rotor taken at the rotor periphery and rotor center. The response clarified that issues in the RAI, including the use of a bored or non-bored rotor and the location of the testing to be performed on the rotor, are addressed by **RAI 199-2073, Question 10.02.03-9**, which is discussed further in Section 10.2.3 of this report. Accordingly as the issues in **RAI 324-1997, Question 03.05.01.03-4, Subquestion 03.05.01.03-02,** are being addressed in Chapter 10 of this report, the staff considers **RAI 324-1997, Question 03.05.01.03-5, Subquestion 03.05.01.03-02, to be resolved.**

In **RAI 324-1997, Question 03.05.01.03-6, Subquestion 03.05.01.03-3**, the staff requested the applicant to discuss what methodology was used in the ductile burst analysis in MUAP-07028-P, Revision 0. In its response to **RAI 324-1997, Question 03.05.01.03-6, Subquestion 03.05.01.03-3**, dated May 20, 2009, the applicant provided a proprietary methodology used in performing the ductile burst analysis. In **RAI 324-1997, Question 03.05.01.03-7, Subquestion 03.05.01.03-4**, the staff requested the applicant to discuss influence of temperature variation on material properties. In its response to **RAI 324-1997, Question 03.05.01.03-7, Subquestion 03.05.01.03-04,** the applicant provided the burst analysis which was revised to use material properties that included the influence of temperature variation. After using these material properties based on temperature variations, the minimum destructive overspeed was slightly reduced, but was well beyond the design overspeed. Therefore, the analysis concludes a ductile burst due to destructive overspeed would only generate missile if the over-speed protection system malfunctions and the turbine speed increases to a point at which the low-pressure rotor undergoes ductile fracture. With this information, MUAP-07029-P, Revision 0, then evaluated the probability of the overspeed protection system malfunctioning, and determines the appropriate valve testing interval to ensure the probability of reaching a destructive overspeed caused by the protection system malfunctioning is low. The applicant identified corresponding changes to MUAP-07028-P and MUAP-07029-P. The staff finds the responses acceptable since they clarified the methodology and material properties used in the ductile burst analysis and provided updated ductile burst analysis results. The staff confirmed that MUAP-07028-P, Revision 1 and MUAP-07029-P, Revision 1 were revised to incorporate the changes to the destructive overspeed result. Accordingly, **RAI 324-1997, Question 03.05.01.03-6, Subquestion 03.05.01.03-3 and RAI 324-1997, Question 03.05.01.03-7, Subquestion 03.05.01.03-4, are resolved.**

MUAP-07028-P, Revision 0, Section 3.3, provides the analysis for LCF, which is based on the probability of fatigue crack growth due to the stresses imposed during startup and shutdown. The use of the Paris fatigue crack growth rate equation is consistent with other approved methodologies. This equation uses variables that were either determined by actual test data, or conservatively assumed based on operational experience and test data. The maximum undetectable crack size was deemed appropriate by the staff since it is well within the range of being detected by current inspection methods. In addition, the parameters used in the equation were derived experimentally from applicable test material data. In addition, the staff finds the fracture toughness value applied in the analysis is acceptable because it is conservative compared to the actual material to be used for the turbine.

MUAP-07028-P, Revision 0, Section 3.3, "Fracture Due to Low Cycle Fatigue (LCF) – Startup/Shutdown Cycles," states that for low-cycle fatigue, it takes a proprietary number of start and stops that are assumed to occur weekly, for the initial cracks to grow up to the critical size, and concludes that this cannot happen under the actual plant operation. However, with these assumptions, this scenario can take place in the life cycle of the turbine rotor. Therefore, the conclusion in MUAP-07028-P, Revision 0, Section 4.0, "Discussion and Conclusions," that stated low-cycle fatigue can be excluded as one of the mechanisms for determining the inspection interval, is not accurate. In **RAI 324-1997, Question 03.05.01.03-8, Subquestion 03.05.01.03-5,** the staff requested the applicant to provide a basis for its conclusion regarding LCF. In its response to **RAI No. 324-1997, Question 03.05.01.03-8, Subquestion 03.05.01.03-5**, dated May 20, 2009, the applicant stated that the assumed weekly start and stops is not typical, and therefore this assumption is very conservative. However, even though this is a very conservative assumption, the applicant agrees that the unit run time (the number of start and stop cycles) should be a limiting factor for determining the inspection interval. Therefore, the applicant proposed to revise MUAP-07028-P, Section 4.0 to delete the statement which excludes LCF (with the assumed start and stop cycles) from determining the inspection interval. The staff finds the response acceptable since the applicant will consider LCF in determining the inspection interval. The staff confirmed MUAP-07028-P, Revision 1, Section 4.0, incorporate the changes identified in the RAI response. Accordingly, **RAI 324-1997, Question 03.05.01.03-8, Subquestion 03.05.01.03-5, is resolved.** The staff notes that the proposed inspection interval of 10 years stated in DCD Tier 2, Section 10.2.3, still bounds the time necessary for a crack to grow to a critical size based on the LCF analysis using the very conservative assumption of weekly starts and stops. Since, the inservice inspections (ISI) planned every ten years would detect any crack before it reaches critical size, the staff finds that the LCF analysis is acceptable as described in MUAP-07028-P, Revision 1, Section 4.0.

MUAP-07028-P, Revision 0, Section 3.4, “Fracture Due to Stress Corrosion Cracking,” provides the analysis to determine the probability of the rotor rupturing due to stress corrosion cracking. A surface stress corrosion crack is assumed to initiate at the rim where the stresses are the highest, and propagate inward until it reaches the critical crack size. MUAP-07028-P, Revision 0, Section 3.4 also states that experience with built-up rotors has also shown that the probability of cracking and crack growth rates of the some of the discs are so low that it is not necessary to consider them in determining the probability of a rotor burst. In **RAI 324-1997, Question 03.05.01.03-9, Subquestion 03.05.01.03-6,** the staff requested the applicant to discuss why these discs have a low probability of cracking. In its response to **RAI 324-1997, Question 03.05.01.03-9, Subquestion 03.05.01.03-6**, dated May 20, 2009, in lieu of providing additional justification of why these discs were not considered, the applicant provided the actual probability for each of the discs that were not initially considered. The staff finds the response acceptable since the staff agrees that the probabilities of these discs are so low that they have a negligible effect on the total probability of missile generation. Accordingly, **RAI 324-1997, Question 03.05.01.03-9, Subquestion 03.05.01.03-6, is resolved.**

In addition, MUAP-07028-P, Revision 0, Table 3.4-1, “3.5% Ni-Cr-Mo-V Rotor Steel, Crack Growth Rate Deviation 3.4-3 from Calculation,” uses a statistical analysis of a proprietary number of data sources for predicting the stress corrosion crack growth rate. In **RAI 324-1997, Question 03.05.01.03-10, Subquestion 03.05.01.03-7,** the staff requested the applicant to discuss why the data sources for predicting the stress corrosion cracking growth rate are statistically sufficient. In its response to **RAI 324-1997, Question 03.05.01.03-10, Subquestion 03.05.01.03-7**, dated May 20, 2009, the applicant clarified that each data source contains numerous data points, so that the actual number of data points representing the material is large and is statistically significant. The applicant also provided a comparison of the calculated values and the measured values from the data sources. The staff finds that the use of measured values from actual material samples was large enough to be statistically significant and provides a good representation of how actual materials will behave. Therefore, the staff finds the response acceptable and **RAI 324-1997, Question 03.05.01.03-10, Subquestion 03.05.01.03-7, is resolved.**

Using the information evaluated in the previous two paragraphs, MUAP-07028-P, Section 3.4, determined critical crack depths at normal operating speed and design overspeed, and determined the probability of rupturing the rotor based on the assumed cracks growing to critical size within a time interval. Based on these calculations, an inspection interval was determined so that the missile generation probability is less than 10^{-5} per year. The inspection interval of 10 years stated in DCD Tier 2, Section 10.2.3, is within the inspection interval determined in the technical report with some margin. The staff notes that this margin provides additional assurance that a potential flaw that was initially missed during initial inspection will be detected at the next inspection before it reaches the critical crack size. Therefore, the staff considers the MUAP-07028-P, Revision 0 uses an acceptable methodology to provide a bounding turbine missile analysis to support the inspection interval of 10 years proposed by DCD Tier 2, Section 10.2.3. The staff notes that the 10 year inspection interval provides margin to assure that a potential flaw that was initially missed during inspection will be detected before it reaches the critical crack size.

As stated in DCD Tier 2, Section 10.2.3.5, "Inservice Inspection," MUAP-07029-P, Revision 0, provides the evaluation on the probability of a turbine missile generation at destructive overspeed due to the failure of the overspeed protection system (valves, etc.). This probability varies based on the valve inspection and test interval used. The method used for calculating the probability of destructive overspeed in this report is based on historical failure data pertinent to the operating experiences of Japanese T/Gs. This report also outlines the use of this failure data to calculate failure rates for various components. This failure rate calculational methodology is acceptable to the staff since it is consistent with industry practice that has resulted in satisfactory operational performance, and is a bounding methodology. This report demonstrates that the probability of turbine missiles generation with quarterly valve test intervals is less than $1x10^{-5}$, which exceeds the guideline of 1×10^{-4} in RG 1.115, Revision 1 and SRP Sections 3.5.1.3 and 10.2.3 for a favorably oriented T/G.

Based on the above, the staff finds that the T/G is favorably oriented with respect to safety-related systems. In addition, the staff finds that the turbine missile probability analysis provided MUAP-07028-P, Revision 1 and MUAP-07029-P, Revision 1, are acceptable bounding turbine missile analysis that demonstrate the probability of the favorably oriented T/G generating a turbine missile is less than 1×10^{-5}, which is more conservative than the RG 1.115, Revision 1, criteria of 1×10^{-4} for a favorably oriented T/G.

Section 10.2.3 of this report provides additional discussion of the staff's evaluation of the turbine ISI program and turbine rotor integrity related to the turbine rotor design and materials used. Section 10.2.2 of this report discusses the staff's detailed evaluation of the turbine overspeed protection system of the US-APWR design. On the basis of the above evaluation, the staff concludes that the probability of turbine missile generation and turbine orientation as required in DCD Tier 2, Section 3.5.1.3 are consistent with the acceptance criteria in SRP Section 3.5.1.3 and RG 1.115, Revision 1.

3.5.1.3.5 Combined License Information Items

The following is a list of COL item numbers and descriptions from DCD Tier 2, Table 1.8-2, "Compilation of All Combined License Applicant Items for Chapters 1-19," related to turbine missiles:

Table 3.5.1.3-1 US-APWR Combined License Information Items		
Item No.	**Description**	**Section**
COL 3.5(2)	The COL Applicant is to commit to actions to maintain P_1 within this acceptable limit as outlined in RG 1.115, "Protection Against Low-Trajectory Turbine Missiles" (Reference 3.5-6) and SRP Section 3.5.1.3, "Turbine Missiles" (Reference 3.5-7).	3.5.1.3.2
COL 3.5(6)	The COL applicant is responsible to assess the orientation of the T/G of this and other unit(s) at multi-unit sites for the probability of missile generation using the evaluation of [DCD Tier 2,] Subsection 3.5.1.3.2.	3.5.1.3.1

The staff evaluated whether sufficient COL information items were identified in Table 1.8-2 of the US-APWR DCD.

DCD Tier 2, Revision 1, Section 3.5.4, "Combined License Information," and Table 1.8-2 provided COL Information Item COL 3.5(2), which stated that the COL applicant will commit to actions to maintain P_1 within acceptable limits as provided by turbine and rotor design features, material specifications and recommended inspections during preservice and inservice periods based on MUAP-07028-P." However, the staff noted that inservice testing of the overspeed protection system was also needed to prevent destructive overspeed conditions and maintain the turbine missile probability within acceptable limits as discussed in MUAP-07029-P. As noted in Section 3.5.1.3.4 of this report, MUAP-07029-P, identifies an acceptable, and provides the basis for, the turbine valve testing frequency. Therefore, in **RAI 323-2071, Question 03.05.01.03-1**, the staff requested that the COL item should also include inservice testing based on MUAP-07029-P. In its response to **RAI 323-2071, Question 03.05.01.03-1**, dated May 20, 2009, the applicant proposed that DCD Tier 2, Section 3.5.4 and Table 1.8.2 be modified to include inservice testing based on MUAP-07029-P. The staff finds the response acceptable since the applicant agreed to include in-service testing based on MUAP-07029-P in COL Information Item 3.5(2).

In addition, DCD Tier 2, Revision 2 modified Sections 3.5.1.3.2 and 3.5.4 and Table 1.8.2 to clarify that the COL applicant will commit to actions to maintain the turbine missile generation probability, P_1, with acceptable limits as outlined in RG 1.115 and SRP Section 3.5.1.3. In addition DCD Tier 2, Revision 2 modified Section 3.5.1.3.2 to state that Technical Reports MUAP-07028-P and MUAP-07029-P are to be used in establishing programs and criteria for preservice inspection, ISI interval and turbine valve testing frequency in order to maintain the turbine missile generation probability, P_1, less than the acceptable limit of 1×10^{-5} and that ISI programs be maintained as outlined in SRP Section 3.5.1.3, Paragraph II.4. The staff finds that the Revision 2 clarification follows the guidance in SRP Section 3.5.1.3 in that applicants with turbine missile generation probability analysis are to meet the criteria appropriate to the T/G orientation in accordance with Table 3.5.1.3-1 of SRP Section 3.5.1.3 for loading and bringing the T/G online. Based on the above, **RAI 323-2071, Question 03.05.01.03-1, is resolved.**

The staff noted that DCD Tier 2, Revision 1, Sections 3.5.4, 3.5.5, "References," and Table 1.8-2 referenced MUAP-070028-P, Revision 0. However, the applicant submitted MUAP-07028-P, Revision 0. In **RAI 323-2071, Question 03.05.01.03-2**, the staff requested the applicant to correct the report number in the applicable section. In its response to **RAI 323-2071, Question 03.05.01.03-2**, dated May 20, 2009, the applicant identified that MUAP-07028-P is the correct report number and identified applicable changes to the DCD. The staff finds the response acceptable since MUAP-07028-P, Revision is consistent with the submitted technical report. The staff confirmed that in DCD Tier 2, Revision 2, the applicable sections in the DCD were revised to reference the correct report. Accordingly, **RAI 323-2071, Question 03.05.01.03-2, is resolved.**

COL Information Item 3.5(6) states that the COL applicant is responsible to assess the orientation of the T/G of this and other units at multi-unit site for the probability of missile generation using the evaluation of DCD Tier 2, Subsection 3.5.1.3.2. The staff finds COL Information Item 3.5(6) acceptable as the evaluation of multi-unit site is by its nature site-specific.

3.5.1.3.6 Conclusions

As a result of the open item for **RAI 782-5910, Question 14.03.07-58**, the staff is unable to finalize its conclusions on Section 3.5.1.3 related to turbine missiles, in accordance with NRC regulations.

3.5.1.4 Missiles Generated By Tornado and Hurricane Winds

3.5.1.4.1 Introduction

This section discusses possible hazards attributable to missiles generated by high-speed winds, such as tornados, hurricanes, and any other extreme winds. Because of the higher wind speed and the resulting higher kinetic energy, the design for wind-generated missiles is governed by tornado and hurricane missiles.

3.5.1.4.2 Summary of Application

DCD Tier 1: The Tier 1 information associated with this section is found in DCD Tier 1, Section 2.2, “Structural and Systems Engineering.”

DCD Tier 2: The applicant has provided a DCD Tier 2 system description in Section 3.5.1.4, “Missiles Generated by Tornadoes and Extreme Winds,” summarized here in part, as follows:

> This section discusses possible hazards attributable to missiles generated by tornados and hurricanes. The spectrum of tornado and hurricane missiles discussed include the associated velocities for a 4,000 pound (1814 kg) automobile, a 15 ft (4.6 m) length of schedule 40 pipe, and a 1 in. (3 cm) diameter solid steel sphere. Note that DCD Tier 2, Section 3.5.1.4, Revision 3 only addressed tornado missiles. In the supplemental response **RAI 908-6321, Question 02-3**, dated September 13, 2012, the applicant proposed revising DCD Tier 2, Section 3.5.1.4 to address both tornado and hurricane missiles, and provided an associated markup to be included in DCD Revision 4.

ITAAC: The ITAAC associated with DCD Tier 2, Section 3.5.1.4 are given in DCD Tier 1, Section 2.2.4, “Inspection, Tests, Analyses, and Acceptance Criteria.”

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 for this area of review.

Cross-cutting Requirements (TMI, USI/ GSI, Op Ex): There are no cross-cutting requirements for this area of review.

US-APWR Interface Issues Identified in the DCD: See DCD Tier 2, Table 1.8-2, "Compilation of All Combined License Applicant Items for Chapters 1-19."

Site Interface Issues Identified in the DCD: See DCD Tier 2, Table 1.8-1, "Significant Site Specific Interfaces with the Standard US-APWR Design."

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

3.5.1.4.3 Regulatory Basis

The relevant requirements of the Commission's regulations for this area of review, and the associated acceptance criteria are given in Section 3.5.1.4, "Missiles Generated by Tornadoes and Extreme Winds," Revision 3, issued March 2007, of NUREG-0800, the SRP, and are summarized below. Review interfaces with other SRP sections can be found in Section 3.5.1.4 of NUREG-0800.

1. GDC 2, as it relates to the ability of SSCs without loss of capability to perform their safety function, to withstand the effects of natural phenomena, such as earthquakes, tornadoes, floods, and the appropriate combination of all loads.

2. GDC 4, as it relates to the protection of SSCs against the effects of missiles that may result from events and conditions outside the nuclear power unit.

3. 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 has been constructed and will operate in accordance with the DC and NRC regulations.

Acceptance criteria adequate to meet the above requirements include:

1. RG 1.76, "Design-Basis Tornado and Tornado Missiles for Nuclear Power Plants," Revision 1, issued March 2007.

2. RG 1.221, "Design-Basis Hurricane and Hurricane Missiles for Nuclear Power Plants," Revision 0, October 2011.

3.5.1.4.4 Technical Evaluation

The staff reviewed the US-APWR design for protecting SSCs important to safety against missiles generated by tornados and hurricane winds in accordance with SRP Section 3.5.1.4, Revision 3. The staff reviewed DCD Tier 2, Revision 3, Section 3.5.1.4, "Missiles Generated by Tornadoes and Extreme Winds," and proposed changes included in the supplemental response to **RAI 907-6321, Question 02-3**, dated September 13, 2012. The staff also reviewed DCD Tier 1, Revision 3, Section 2.0, "Design Descriptions and ITAAC," and other DCD Tier 2 sections noted below.

3.5.1.4.4.1 Compliance with GDC 2 and GDC 4

Compliance with GDC 2 and GDC 4 with respect to missiles generated by high winds, tornados, and hurricanes may be based on meeting the following guidance:

- RG 1.76, Positions C.1, “Design-Basis Tornado Parameters,” and C.2, “Design-Basis Tornado-Generated Missile Spectrum,” and

- RG 1.221, Position C.1, “Design-Basis Hurricane Wind Speeds,” and C.2 “Design-Basis Hurricane-Generated Missiles.”

The guidance of RG 1.76 only applies to the continental U.S., which is divided into three regions: Region I − the central portion of the U.S.; Region II − a large region of the U.S. along the east coast, the northern border, and western Great Plains; and Region III − the western U.S. The tornado parameter values specified in RG 1.76, Table 1 for Region I are most severe and bound all the tornado parameter values specified for Regions II and III. However, RG 1.76 only applies to the continental U.S. Therefore, COL Information Item COL 2.3(1) requires the COL applicant regardless of whether the plant is to be sited inside or outside the continental U.S., to verify that site-specific regional climatology and local metrology are bounded by the site parameters for the standard US-APWR design.

In DCD Tier 2, Section 3.3.2.1, “Applicable Design Parameters,” the applicant provided the following parameters as design-basis tornado parameters for the US-APWR design:

- Maximum wind speed of 103 m/s (230 mph).
- Maximum rotational speed of 82.3 m/s (184 mph).
- Maximum translational speed of 21 m/s (46 mph).
- Radius of maximum rotational speed of 45.7 m (150 ft).
- Atmospheric pressure drop of 8.3 kPa (1.2 psi).
- Rate of pressure drop of 3.4 kPa/s (0.5 psi/s).

In addition, the applicant stated that the above parameters are those of a Region 1 tornado characteristics and the annual probability of exceedance of the design-basis tornado described above is $1x10^{-7}$ as discussed in RG 1.76.

In DCD Tier 2, Section 3.5.1.4, the applicant provided tornado-generated missile spectra for the US-APWR design as follows:

- A 1814.4 kg (4,000 lb.) automobile, 5 m (16.4 ft) by 2 m (6.6 ft) by 1.3 m (4.3 ft), impacting the structure at normal incidence with a horizontal velocity of 41.1 m/s (135 ft/s) or a vertical velocity of 27.6 m/s (90.5 ft/s).

- A 16.83 cm (6.625 in.) diameter by 4.6 m (15 ft) long schedule 40 pipe, weighing 130 kg (287 lb.), impacting the with a horizontal velocity of 41.1 m/s (135 ft/s) or a vertical velocity of 27.6 m/s (90.5 ft/s).

- A 2.5 cm (1 in.) diameter solid steel sphere assumed to impinge upon barrier openings in the most damaging direction with a horizontal velocity of 7.9 m/s (26 ft/s) or a vertical velocity of 5.3 m/s (17.4 ft/s).

In reviewing DCD Tier 2, Revision 1, Section 3.5.1.4 regarding protection provided for safety-related SSCs against missiles generated by tornados and extreme winds, the staff identified areas in which additional information was necessary to complete its review.

The staff found that the above design-basis tornado parameters and tornado-generated missile spectra were consistent with the guidance as described in RG 1.76 for Region 1. However, the staff also found that not all of the above design-basis tornado parameters were included in DCD Tier 1, Revision 1, Table 2.1-1, “Key Site Parameters.” Therefore, the staff in **RAI 154-1643 Question 03.05.01.04-1, Subquestion 3.5.1.4-01,** requested the applicant to revise DCD Tier 1, Revision 1, Table 2.1-1 to include maximum rotational speed of 184 mph (82.3 m/s), maximum translational speed of 46 mph (21 m/s), radius of maximum rotational speed of 150 ft (45.7 m), rate of pressure drop of 0.5 psi/s (3.4 kPa/s), and exceedance frequency of 1×10^{-7} per year.

In its response to **RAI 154-1643, Question 03.05.01.04-1, Subquestion 3.5.1.4-01**, dated April 4, 2009, the applicant stated that DCD, Tier 1 Table 2.1-1 (along with DCD Tier 2 Table 2.0-1, “Key Site Parameters,”) will be revised to include the following additional entries: maximum rotational speed (184 mph) (82.3 m/s), maximum translational speed (46 mph) (21 m/s), radius of maximum rotational speed (150 ft) (45.7 m), and rate of pressure drop (0.5 psi/s) (3.4 kPa/s). The applicant further stated that the exceedance frequency is an acceptance value for frequency of occurrence, which is not a key site parameter, therefore, the exceedance frequency is not applicable for inclusion in these tables.

Based on its review, the staff finds the applicant’s response to **RAI 154-1643, Question 03.05.01.04-1, Subquestion 3.5.1.4-01,** acceptable because the staff finds that the above design-basis tornado parameters and tornado-generated missile spectra are in accordance with the guidance described in RG 1.76, Table 1, for Region I and because the applicant commits to include all of the above design-basis tornado parameters in DCD Tier 1, Table 2.1-1 and DCD, Tier 2, Table 2.0-1. The staff also finds the applicant’s clarification that the exceedance frequency is an acceptance criterion for frequency of occurrence and that this acceptance criterion is not a key site parameter, acceptable. The staff has confirmed that DCD Revision 2, was revised as committed in the RAI response. Accordingly, **RAI 154-1643, Question 03.05.01.04-1, Subquestion 3.5.1.4-01, is resolved**.

In the DCD markup in the supplemental response to **RAI 907-6321, Question 02-3**, DCD Tier 2, Section 3.3.2.1 states that 160 mph (71.5 m/s) is the plant standard design-basis hurricane wind speed. This value was chosen from the contour maps provided in RG 1.221, and though this wind speed is conservative for most locations in the U.S., it is not bounding for all locations. Therefore, the applicant proposed to add the new COL Information Item COL 3.3(6), which requires the COL applicant to verify site-specific design-basis hurricane wind speeds are enveloped by the assumptions in the DCD. Since the applicant has proposed DCD changes, **RAI 907-6321, Question 02-3, is being tracked as a Confirmatory Item.**

In the DCD markup in the supplemental response to **RAI 907-6321, Question 02-3**, the applicant proposed to revise DCD Tier 2, Section 3.5.1.4 and provided a hurricane-generated missile spectra for the US-APWR design as follows:

- A 1814.4 kg (4,000 lb.) automobile, 5 m (16.4 ft) by 2 m (6.6 ft) by 1.3 m (4.3 ft), impacting the structure at normal incidence with a horizontal velocity of 41.1 m/s (135 ft/s) or a vertical velocity of 26 m/s (85 ft/s).

- A 16.83 cm (6.625 in.) diameter by 4.6 m (15 ft) long schedule 40 pipe, weighing 130 kg (287 lb.), impacting the with a horizontal velocity of 31.1 m/s (102 ft/s) or a vertical velocity of 26 m/s (85 ft/s).

- A 2.5 cm (1 in.) diameter solid steel sphere assumed to impinge upon barrier openings in the most damaging direction with a velocity of 27 m/s (89 ft/s) and a vertical velocity of 26 m/s (85 ft/s).

The staff finds the proposed hurricane missile spectrum acceptable since it conforms to the guidelines of RG 1.221. Since the applicant has proposed DCD changes, **RAI 907-6321, Question 02-3, is being tracked as a Confirmatory Item.**

Section 2.3 of this report addresses the staff's evaluation of meteorological site parameters.

RG 1.76 assumes automobile missiles are considered to impact at an altitude of less than 9.1 m (30 ft) above plant grade. Therefore, for sites with surrounding ground elevations higher than plant grade, the staff in **RAI 154-1643, Question 03.05.01.04-1, Subquestion 3.5.1.4-05,** requested the applicant to revise the DCD Tier 2, Revision 1, Table 1.8.2, to include a COL information item to require the COL applicant that references the US-APWR DC to confirm that automobile missiles could not be generated within a 0.8 km (0.5 mile) radius of safety-related SSCs that would lead to impact higher than 9.1 m (30 ft) above plant grade.

In its response to **RAI 154-1643, Question 03.05.01.04-1, Subquestion 3.5.1.4-05**, the applicant stated that due to the robustness of the exterior wall design, all seismic Category I structures are capable of withstanding the impact of each identified tornado missile at any elevation, including the potential impact of a 4,000 lb. (1814 kg) automobile greater than 30 ft (9.1 m) above grade.

Also, the applicant stated that DCD Tier 2, Section 3.5.1.4 will be revised regarding the ability of seismic Category I buildings to withstand tornado-generated missiles. More specifically, this revision will state that seismic Category I buildings have sufficient thicknesses throughout their entire height to withstand impact from the identified tornado-generated missiles at any elevation, including an 1814.4 kg (4,000 lb.) automobile. Therefore, an additional COL item is not necessary to evaluate plant elevation in excess of building grade level.

Based on its review, the staff finds the applicant's response to **RAI 154-1643, Question 03.05.01.04-1, Subquestion 3.5.1.4-05,** acceptable because all seismic Category I structures are capable of withstanding the impact of each identified tornado or hurricane missile at any elevation. Also, the staff agrees with the applicant that no additional COL information item is necessary to evaluate plant elevation in excess of building grade level. The staff has confirmed that DCD Tier 2, Revision 2, was revised as committed in the RAI response. Accordingly, **RAI 154-1643, Question 03.05.01.04-1, Subquestion 3.5.1.4-05, is resolved**.

Section 3.5.2 of this report addresses the staff's evaluation of the adequacy protection provided for US-APWR plant structures and SSCs important to safety against the effects of externally generated missiles.

Section 3.5.3, of this report addresses the staff's evaluation of the adequacy of the barriers and structures designed to withstand the effects of the identified tornado or hurricane missiles, including the potential impact of a 4,000 lb. (1814 kg) automobile greater than 30 ft (9.1 m) above grade.

Section 3.8.4, of this report addresses the staff's evaluation of the US-APWR structural design.

Based on its review, the staff finds that the applicant conforms with the guidance in RG 1.76 for design-basis tornado and tornado missiles for nuclear power plants, and RG 1.221 for design-basis hurricane and hurricane missiles for nuclear power plants. Therefore, the staff concludes that the US-APWR design meets the requirements of GDC 2 and GDC 4 with respect to protection for safety-related SSCs against the effects of natural phenomena such as tornadoes and hurricanes.

3.5.1.4.4.2 Inspections, Tests, Analyses, and Acceptance Criteria

In DCD Tier 1, Section 2.0 , the applicant provides the design descriptions and ITAAC which commit to verify that the US-APWR plant safety-related SSCs and standard plant buildings are designed and constructed to be protected from externally and internally-generated missiles, and performed as described in DCD Tier 2. The ITAAC associated with specific external and internal environmental hazards are addressed specifically in DCD Tier 1, Table 2.2-4, "Structural and Systems Engineering Inspections, Tests, Analyses, and Acceptance Criteria."

Based on its review, the staff finds the above cited ITAAC Items acceptable as they require a licensee to verify that the safety-related SSCs are protected from missiles generated by tornadoes and hurricane winds, and are designed and constructed as described in DCD Tier 2. Therefore, the staff concludes that the missile protection provided for USAPWR plant safety-related SSCs and structures complies with the requirements of 10 CFR 52.47(b)(1).

3.5.1.4.4.3 Initial Testing

DCD Tier 2, Revision 2, Section 14.2, "Initial Plant Test Program," does not have any initial testing requirements associated with this review item. The staff reviewed DCD Tier 2, Revision 2, Section 3.5.1.4 against the guidance in SRP Section 14.2, "Initial Plant Test Program," Revision 3, issued March 2007, and found that no additional initial testing is needed in connection with this section.

3.5.1.4.4.4 Technical Specifications

DCD Tier 2, Revision 2, Chapter 16, "Technical Specifications," does not have any TS requirements associated with this review item. The staff reviewed DCD Tier 2, Revision 2, Section 3.5.1.4 against 10 CFR 50.36, "Technical Specifications," and agrees that no TSs are needed in connection with this section.

3.5.1.4.5 Combined License Information Items

The following is a list of COL item numbers and descriptions from Table 1.8-2, "Compilation of All Combined License Applicant Items for Chapters 1-19," of the DCD related to missiles generated by tornadoes and hurricanes:

Table 3.5.1.4-1 US-APWR Combined License Information Items

Item No.	Description	Section
COL 2.3(1)	The COL Applicant, whether the plant is to be sited inside or outside the continental U.S., is to provide site-specific pre-operational and operational programs for meteorological measurements, and is to verify the site-specific regional climatology and local meteorology are bounded by the site parameters for the standard US-APWR design or demonstrate by some other means that the proposed facility and associated site-specific characteristics are acceptable at the proposed site.	2.3
COL 3.5(5)	The COL applicant is responsible to evaluate site-specific hazards for external events that may produce missiles more energetic than tornado missiles and hurricane missiles for the standard plant presented in Subsection 3.5.1.4, and assure that the design of seismic Category I and II structures meet these loads.	3.5.1.4

The staff finds the above listing to be complete. Also, the list adequately describes actions necessary for the COL applicant. No additional COL information items need to be included in DCD Tier 2, Table 1.8-2 related to missiles generated from tornadoes and hurricanes.

Note that the applicant revised COL Information Item 3.5(5) in the supplemental response to **RAI 907-6321, Question 02-3.** As discussed in Section 3.5.1.4.4 of this report above, **RAI 907-6321, Question 02-3, is being tracked as a Confirmatory Item.**

3.5.1.4.6 Conclusions

Based on the foregoing, pending resolution of **RAI 907-6321, Question 02-3, which is being tracked as a Confirmatory Item**, the staff concludes that the US-APWR design satisfies the guidelines of SRP Section 3.5.1.4, Revision 3, and conforms with RG 1.76 for design-basis tornado and tornado missiles and RG 1.221 for design-basis hurricane and hurricane missiles for nuclear power plants. Therefore, the staff concludes that the US-APWR design-basis tornado and hurricane parameters and tornado-generated and hurricane-generated missile spectra for the US-APWR design meet the requirements of GDC 2 and GDC 4, and 10 CFR 52.47(b)(1) regarding SSCs to be protected from missiles generated by the design-basis tornado and hurricane winds.

3.5.1.5 Site Proximity Missiles (Except Aircraft)

3.5.1.5.1 Introduction

This section explains that the design is based on tornado missiles as being the most severe general case, although hurricane missiles are considered, and that the COL applicant will establish site-specific missile spectra. The potential threat to the plant from site proximity missiles is site-specific and cannot be assessed at the DC stage.

3.5.1.5.2 Summary of Application

DCD Tier 1: The Tier 1 information associated with this section is found in DCD Tier 1, Section 2.2, “Structural and Systems Engineering.”

DCD Tier 2: The applicant has provided a DCD Tier 2 system description in Section 3.5.1.5, "Site Proximity Missiles (Except Aircraft)," summarized here in part, as follows:

DCD Tier 2, Section 3.5.1.5 addresses site proximity missiles (except aircraft) with COL Information Item 3.5(3), which states that as described in DCD Tier 2, Section 2.2, "Nearby Industrial, Transportation, and Military Facilities," the COL Applicant is to establish the presence of potential hazards, except aircraft, which are reviewed in DCD Tier 2, Section 3.5.1.6, "Aircraft Hazards," and the effects of potential accidents in the vicinity of the site.

Note that DCD Tier 2, Section 3.5.1.5, Revision 3 states that externally initiated missiles considered for the US-APWR standard design are based on tornado missiles as described in Subsection 3.5.1.4. As described in Section 3.5.1.4 of this report, the applicant added in DCD Tier 2, Section 3.5.1.4 and 3.5.1.5 that hurricane missiles are also considered for the US-APWR standard design in response to **RAI 908-6327, Question 03.03.02-6** and **RAI 908-6321, Question 02-3**.

ITAAC: The ITAAC associated with DCD Tier 2, Section 3.5.1.5 are given in DCD Tier 1, Section 2.2.4.

TS: No TS for this area of review.

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

Technical Reports: The technical report associated with DCD Tier 2, Section 3.5.1.5 is as follows:

1. MUAP 07028 P, "Probability of Missile Generation From Low Pressure Turbines," Revision 0, issued December 2007.

Cross-cutting Requirements (TMI, USI/ GSI, Op Ex): There are no cross-cutting requirements for this area of review.

US-APWR Interface Issues Identified in the DCD: See DCD Tier 2, Table 1.8-2, "Compilation of All Combined License Applicant Items for Chapters 1-19."

Site Interface Issues Identified in the DCD: See DCD Tier 2, Table 1.8-1, "Significant Site Specific Interfaces with the Standard US-APWR Design."

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

3.5.1.5.3 Regulatory Basis

The relevant requirements of the Commission's regulations for this area of review, and the associated acceptance criteria are given in Section 3.5.1.5, "Site Proximity Missiles (Except Aircraft)," Revision 4, issued March 2007, of NUREG-0800 and are summarized below. Review interfaces with other SRP sections can be found in Section 3.5.1.5 of NUREG-0800.

1. GDC 4, as it relates to the protection of SSCs against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that may result from equipment failures and from events and conditions outside the nuclear power unit.

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 DC has been constructed and will operate in accordance with the DC, the provisions of the Atomic Energy Act of 1954, and the NRC's regulations.

3. 10 CFR Part 100, 10 CFR 100.10, 10 CFR 100.20, 10 CFR 100.21, and 10 CFR Part 52, as they relate to the factors to be considered in the evaluation of sites, which indicate that reactors should reflect through their design, construction, and operation an extremely low probability for accidents that could result in the release of significant quantities of radioactive fission products.

4. 10 CFR 100.10 and 10 CFR 100.20, as they relate to the site assuring a low risk of public exposure.

Acceptance criteria adequate to meet the above requirements include:

1. The SRP Acceptance Criterion in SRP Section 3.5.1.5, Section II, Item 1 states that to meet the requirements of 10 CFR Part 100, the probability that site proximity missiles will impact the plant and cause radiological consequences greater than the 10 CFR Part 100 exposure guidelines must be less than an order of magnitude of 10^{-7} per year (see also guidance in SRP Section 2.2.3, "Evaluation of Potential Accidents"). If the review indicates that the above criterion is not met, then the acceptance criterion described in Item 2 below applies.

2. The SRP Acceptance Criterion in SRP Section 3.5.1.5, Section II, Item 2 states that, if Item 1 is not met, the plant will meet the relevant requirements of GDC 4 and will be considered appropriately protected against site proximity missiles' design if the SSCs important to safety are capable of withstanding the effects of the postulated missiles without loss of safe-shutdown capability and without causing a release of radioactivity in excess of the 10 CFR Part 100 dose guidelines.

3. RG 1.91, "Evaluations of Explosions Postulated To Occur on Transportation Routes Near Nuclear Power Plants," Revision 1, issued February 1978. This RG describes methods acceptable to the NRC for determining whether a plant meets the previous two criteria.

3.5.1.5.4 Technical Evaluation

The potential threat to the plant from site proximity missiles is site-specific and cannot be assessed at the DC stage. Missiles generated from nearby facilities are identified as a COL information item in the US-APWR DCD Tier 2, Table 1.8-2, "Compilation of All Combined License Applicant Items for Chapters 1-19," (Item 3.5(3)). A COL applicant that references the US-APWR DCD will evaluate the potential for site proximity explosions and missiles generated by these explosions for their potential impact on missile protection design features.

DCD Tier 2, Section 3.5.1.5, states that externally initiated missiles considered for the US-APWR standard design are based on tornado and hurricane missiles as described in DCD Tier 2, Section 3.5.1.4, which is evaluated in Section 3.5.1.4 of this report.

DCD Tier 2, Section 3.5.1.5, addresses site proximity missiles (except aircraft) with COL Information Item 3.5(3), which states that as described in DCD Tier 2, Section 2.2, "Nearby Industrial, Transportation, and Military Facilities," the COL applicant is to establish the presence of potential hazards, except aircraft, which are reviewed in DCD Tier 2, Section 3.5.1.6, and the effects of potential accidents in the vicinity of the site.

The staff agrees that the potential threat to the plant from site proximity missiles is site-specific and cannot be assessed at the DC stage. Therefore the staff finds COL Information Item 3.5(3) acceptable. Therefore, Section 3.5.1.5 is reviewed at the COL stage.

3.5.1.5.5 Combined License Information Items

The following is a list of COL item numbers and descriptions from Table 1.8-2 of the DCD related to site proximity missiles (except aircraft):

Table 3.5.1.5-1
US-APWR Combined License Information Items

Item No.	Description	Section
COL 3.5(3)	As described in DCD, Section 2.2, the COL applicant is to establish the presence of potential hazards, except aircraft, which is reviewed in Section 3.5.1.6, and the effects of potential accidents in the vicinity of the site.	3.5.1.5

As discussed above, the staff finds the above listing in the table concerning site proximity missiles (except aircraft) to be complete. Also, the list adequately describes actions necessary for the COL applicant to take. No additional COL information items were identified that need to be included in DCD Tier 2, Table 1.8-2 regarding site proximity missiles (except aircraft).

3.5.1.5.6 Conclusions

As set forth above, per COL Information Item 3.5(3), the applicant has stated that the COL applicant will provide the site-specific information. Since this information is site-specific, the applicant's statement that the COL applicant is to supply this site-specific information is acceptable. Therefore, Section 3.5.1.5 is reviewed at the COL stage.

3.5.1.6 Aircraft Hazards

3.5.1.6.1 Introduction

This section assures that the risks from aircraft hazards are sufficiently low. The COL applicant verifies the site parameters with respect to aircraft hazards. Additional analyses may be required as appropriate.

3.5.1.6.2 Summary of Application

DCD Tier 1: The Tier 1 information associated with this section is found in DCD Tier 1, Section 2.2.

DCD Tier 2: The applicant has provided a DCD Tier 2 system description in Section 3.5.1.6, "Structural and Systems Engineering," summarized here in part, as follows:

> DCD Tier 2, Section 3.5.1.6 addresses aircraft hazards with COL Information Item 3.5(4), which states that it is the responsibility of the COL Applicant to verify the site interface parameters with respect to aircraft crashes and air transportation accidents as described in Section 2.2.

ITAAC: The ITAAC associated with DCD Tier 2, Section 3.5.1.6 are given in DCD Tier 1, Section 2.2.4.

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

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

Technical Reports: There are no technical reports for this area of review.

Cross-cutting Requirements (TMI, USI/ GSI, Op Ex): There are no cross-cutting requirements for this area of review.

US-APWR Interface Issues Identified in the DCD: See DCD Tier 2, Table 1.8-2, "Compilation of All Combined License Applicant Items for Chapters 1-19."

Site Interface Issues Identified in the DCD: See DCD Tier 2, Table 1.8-1, "Significant Site Specific Interfaces with the Standard US-APWR Design."

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

3.5.1.6.3 Regulatory Basis

The relevant requirements of the Commission's regulations for this area of review, and the associated acceptance criteria are given in Section 3.5.1.6, "Aircraft Hazards," Revision 3, issued March 2007, of NUREG-0800, the SRP, and are summarized below. Review interfaces with other SRP sections can be found in Section 3.5.1.6 of NUREG-0800.

1. GDC 3 requires that SSCs important to safety be appropriately protected against the effects of fires.

2. GDC 4, as it relates to the protection of SSCs against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that may result from equipment failures and from events and conditions outside the nuclear power unit.

3. 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 has been constructed and will operate in accordance with the DC, the provisions of the Atomic Energy Act of 1954, and

the NRC's regulations.

4. 10 CFR Part 100, 10 CFR 100.10, 10 CFR 100.20, 10 CFR 100.21, and 10 CFR 52.79, as they relate to the reactors reflecting through their design, construction, and operation an extremely low probability for accidents that could result in the release of significant quantities of radioactive fission products.

5. 10 CFR Part 100, 10 CFR 100.10, and 10 CFR 100.20, as they relate to the site assuring a low risk of public exposure.

Acceptance criteria adequate to meet the above requirements include:

1. SRP Acceptance Criteria in SRP Section 3.5.1.6, Section II, Item 1 state that the requirements of 10 CFR 100.10, 10 CFR 100.20, 10 CFR 100.21, 10 CFR 52.17, and 10 CFR 52.79 are met if the probability of aircraft accidents resulting in radiological consequences greater than the 10 CFR Part 100 exposure guidelines is less than an order of magnitude of 10^{-7} per year (see also SRP Section 2.2.3, "Evaluation of Potential Accidents"). The probability is considered to be less than an order of magnitude of 10^{-7} per year by inspection if the distances from the plant meet all of the criteria listed below:

   A. The plant-to-airport distance D is between 5 and 10 statute miles (8 and 16 km), and the projected annual number of operations is less than 500 D^2, or the plant-to-airport distance D is greater than 10 statute miles (16 km), and the projected annual number of operations is less than 1000 D^2.

   B. The plant is at least five statute miles (8 km) from the nearest edge of military training routes, including low-level training routes, except for those associated with usage greater than 1000 flights per year, or where activities (such as practice bombing) may create an unusual stress situation.

   C. The plant is at least two statute miles (3 km) beyond the nearest edge of a Federal airway, holding pattern, or approach pattern

   The projected number of operations in item A above, as well as the 1000 flights per year in item B above, should represent the maximum aircraft activity expected during the permit term in construction permit (CP) and early site permit (ESP) applications or for the license duration in COL and COLAs.

2. SRP Acceptance Criteria in SRP Section 3.5.1.6, Section II, Item 2, state that if the above proximity criteria (in SRP Acceptance Criteria in SRP Section 3.5.1.6, Section II, Item 1) are not met, or if sufficiently hazardous military activities are identified (see item B above), a detailed review of aircraft hazards must be performed. Aircraft accidents that could lead to radiological consequences in excess of the exposure guidelines of 10 CFR Part 100 with a probability of occurrence greater than an order of magnitude of 10^{-7} per year should be considered in the design of the plant. If the results of the review do not support a finding that the risk from aircraft activities is acceptably low, then the design-basis acceptance criteria outlined in GDC 4 apply.

The plant meets the relevant requirements of GDC 3 and GDC 4, and is considered appropriately protected against design-basis aircraft impacts and fires if the SSCs important to safety are capable of withstanding the effects of the postulated aircraft impacts and fires without loss of safe-shutdown capability and without causing a release of radioactivity that could exceed the 10 CFR Part 100 dose guidelines.

3. RG 1.117, “Tornado Design Classification,” Revision 1, issued April 1978, provides acceptable methods for determining those SSCs that should be protected. The selection of SSCs to be protected is based upon not allowing offsite exposures to exceed an appropriate fraction of the offsite dose guidelines of 10 CFR Part 100. Basing the limits upon an appropriate "fraction" ensures protection for those events that are not as severe as the design-basis event, but have a higher probability of occurrence. Protecting those SSCs important to safety from the effects of externally generated missiles due to aircraft hazards prevents failure of those systems required for safe shutdown and prevents the release of radioactivity with the potential for causing exposures in excess of the 10 CFR Part 100 guidelines.

4. The expected rate of exposure identified in 10 CFR 50.34(a)(1) dose guideline as it relates to the requirements identified in 10 CFR 100.20(b) should be about an order of magnitude of 10^{-6} per year. If it can be shown with rigorous analysis, using realistic assumptions and reasonable arguments that the estimated probability could be lower, then, in accordance with the SRP Section 2.2.3, it is acceptable.

3.5.1.6.4 Technical Evaluation

DCD Tier 2, Section 3.5.1.6, states that the US-APWR standard plant design-basis is that the plant is located such that an aircraft crash and air transportation accidents are not required to be considered as part of the design-basis. Per COL Information Item 3.5(4), it is the responsibility of the COL applicant to verify the site interface parameters with respect to aircraft crashes and air transportation accidents as described in DCD Tier 2, Section 2.2. Additional analyses may be required to evaluate potential aircraft missiles.

The staff agrees that aircraft hazards are site-specific and cannot be assessed at the DC stage. Therefore the staff finds COL Information Item 3.5(4) acceptable. Therefore, Section 3.5.1.6 is reviewed at the COL stage.

3.5.1.6.5 Combined License Information Items

The following is a list of COL item numbers and descriptions from Table 1.8-2, “Compilation of All Combined License Applicant Items for Chapters 1-19,” of the DCD, related to aircraft hazards:

Table 3.5.1.6-1
US-APWR Combined License Information Items

Item No.	Description	Section

Table 3.5.1.6-1 US-APWR Combined License Information Items		
Item No.	**Description**	**Section**
COL 3.5(4)	It is the responsibility of the COL applicant to verify the site interface parameters with respect to aircraft crashes and air transportation accidents as described in Section 2.2.	2.2 3.5.1.6

As discussed above, the staff finds the above listing in the table concerning aircraft hazards to be complete. Also, the list adequately describes actions necessary for the COL applicant to take. No additional COL information items were identified that need to be included in DCD Tier 2, Table 1.8-2 regarding aircraft hazards.

3.5.1.6.6 Conclusions

As set forth above, per COL Information Item 3.5(4), the applicant has stated that the COL applicant will provide the site-specific information. Since this information is site-specific, the applicant’s statement that the COL applicant is to supply this site-specific information is acceptable. Therefore, Section 3.5.1.6 is reviewed at the COL stage.

3.5.2 Structures, Systems, and Components to be Protected from Externally Generated Missiles

3.5.2.1 Introduction

This section discusses the SSCs to be protected from externally-generated missiles, which include all plant site safety-related SSCs supporting the reactor facility, and such elements as ESW intakes, buried components, and structure access openings and penetrations.

3.5.2.2 Summary of Application

DCD Tier 1: The Tier 1 information associated with this section is found in DCD Tier 1, Section 2.2, “Structural and Systems Engineering.”

DCD Tier 2: The applicant has provided a DCD Tier 2 system description in Section 3.5.2, “Structures, Systems, and Components to be Protected from Externally Generated Missiles,” summarized here in part, as follows:

> This section discusses the SSCs to be protected from externally-generated missiles, which includes all plant site safety-related SSCs supporting the reactor facility, and such elements as ESW intakes, buried components, and structure access openings and penetrations. Note that DCD Tier 2, Section 3.5.2, Revision 3 only addressed tornado missiles. In the supplemental response to **RAI 908-6321, Question 02-3**, dated September 24, 2012, the applicant proposed revising DCD Tier 2, Section 3.5.1.4 to address both tornado and hurricane missiles, and provided an associated markup to be included in DCD Revision 4.

ITAAC: The ITAAC associated with DCD Tier 2, Section 3.5.2 are given in DCD Tier 1, Section 2.2.4, “Inspection, Tests, Analyses, and Acceptance Criteria.”

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 for this area of review.

Cross-cutting Requirements (TMI, USI/ GSI, Op Ex): There are no cross-cutting requirements for this area of review.

US-APWR Interface Issues Identified in the DCD: See DCD Tier 2, Table 1.8-2, "Compilation of All Combined License Applicant Items for Chapters 1-19."

Site Interface Issues Identified in the DCD: See DCD Tier 2, Table 1.8-1, "Significant Site Specific Interfaces with the Standard US-APWR Design."

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

3.5.2.3 Regulatory Basis

The relevant requirements of the Commission's regulations for this area of review, and the associated acceptance criteria are given in Section 3.5.2, "Structures, Systems, and Components to be Protected from Externally-Generated Missiles," Revision 3, issued March 2007, of NUREG-0800, the SRP, and are summarized below. Review interfaces with other SRP sections can be found in Section 3.5.2 of NUREG-0800.

1. GDC 2, as it relates to the ability of SSCs without loss of capability to perform their safety function, to withstand the effects of natural phenomena, such as earthquakes, tornadoes, floods, and the appropriate combination of all loads.

2. GDC 4, as it relates to the protection of SSCs against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that may result from equipment failures and from events and conditions outside the nuclear power unit.

3. 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 has been constructed and will operate in accordance with the DC and NRC regulations.

Acceptance criteria adequate to meet the above requirements include:

1. RG 1.13, "Spent Fuel Storage Facility Design-Basis," Revision 2, issued March 2007,

2. RG 1.27, "Ultimate Heat Sink for Nuclear Plants," Revision 2, issued January 1976,

3. RG 1.115, "Protection Against Low-Trajectory Turbine Missiles," Revision 1, issued July 1977,

4. RG 1.117, "Tornado Design Classification," Revision 1, issued April 1978.

3.5.2.4 Technical Evaluation

The staff reviewed the US-APWR design for protecting SSCs important to safety against externally generated missiles in accordance with the guidance of SRP Section 3.5.2, Revision 3. The staff reviewed DCD Tier 2, Revision 3, Section 3.5.2. The staff also reviewed DCD Tier 1, Revision 3, Section 2.0, "Design Descriptions and ITAAC," and other DCD Tier 2 sections noted below.

3.5.2.4.1 Compliance with GDC 2 and GDC 4

Per the SRP Acceptance Criteria, in SRP Section 3.5.2, compliance with GDC 2 and GDC 4 is based on meeting the guidance of the following RGs:

- RG 1.13, as it relates to the capacity of the SFP cooling systems and structures to withstand the effects of externally generated missiles and to prevent missiles from contacting the stored fuel assemblies,

- RG 1.27, as it relates to the capability of the UHS and connecting conduits to withstand the effects of externally generated missiles,

- RG 1.115, as it relates to the protection of the SSCs important to safety from the effects of turbine missiles,

- RG 1.117, as it relates to the protection of the SSCs important to safety from the effects of tornado missiles.

SRP Section 3.5.2, states that the SSCs required for safe shutdown of the reactor should be identified. RG 1.115 Position C.1 and RG 1.117, Appendix A, provide guidance as to which SSCs should be protected from missile impacts. In DCD Tier 2, Table 3.2-2, "Classification of Mechanical and Fluid Systems, Components, and Equipment," the applicant identifies the SSCs that require missile protection, and in DCD Tier 2, Table 7.4-1, "Component Controls for Shutdown," the applicant identifies the SSCs that are needed for safe-shutdown.

US-APWR structures that provide protection for SSCs important to safety against externally generated missiles include:

- R/B.
- PS/B.
- PCCV.
- PSFSV.
- UHSRS.

Section 3.5.3, of this report addresses the staff's evaluation of the design of structures, shields, and barriers required for missile protection.

Section 3.8, of this report addresses the staff's evaluation of the US-APWR structural design including evaluation of US-APWR structure conformance to the guidance of RG 1.13 with respect to protection of spent fuel from externally generated missiles.

In reviewing DCD Tier 2, Revision 1, Section 3.5.2, the staff identified the areas in which additional information was necessary to complete the evaluation of the US-APWR plant design for protection against external missiles generated by natural phenomena.

In DCD Tier 2, Revision 1, Section 3.5.2, the applicant stated that protection of SSCs from external missiles is provided by the external walls and roof of the structures identified above. However, the applicant did not include a COL information item in DCD Tier 2, Revision 1, Table 1.8-2, "Compilation of All Combined License Applicant Items for Chapters 1-19," to require a COL applicant to conduct evaluations of the effects of missile impact on openings in exterior walls. Therefore, in **RAI 153-1646, Question 03.05.02-1**, the staff requested the applicant to revise DCD Tier 2, Revision 1, Table 1.8-2 to include a COL information item that requires the COL applicant to evaluate the effects of externally generated missile impact through openings in exterior walls.

In its response to **RAI 153-1646, Question 03.05.02-1**, dated February 4, 2009, the applicant proposed to revise DCD Tier 2, Section 3.5.2 and Section 3.5.1.4, "Missiles Generated by Tornadoes and Extreme Winds," to clarify that openings through the exterior walls of the seismic Category I structures within the US-APWR standard plant and the location of equipment in the vicinity of such openings are arranged so that a missile passing through the opening would not prevent the safe shutdown of the plant and would not result in an offsite release exceeding the limits defined in 10 CFR 100. In the response, the applicant explained that the COL applicant is not responsible for evaluating the effects of externally generated missile impact on openings in exterior walls within standard plant structures. For the additional site-specific design (applicant-designed) seismic Category I structures other than the standard plant structures, the COL applicant is responsible only for the design in accordance with the requirements of DCD Tier 2 Chapter 3, "Design of Structures, Systems, Components, and Equipment," which includes consideration of the effects of externally generated missile impact on openings in applicant-designed seismic Category I structures. Therefore, no COL information item needs to be included in DCD Tier 2, Table 1.8-2 for any additional applicant-designed activity.

On the basis of its review, the staff finds the applicant's response acceptable and agrees with the applicant that no COL information item is needed for any additional applicant-designed activity as the need to address openings in the exterior walls of seismic Category I structures is included in DCD Tier 2, Sections 3.5.1.4 and 3.5.2. The staff has confirmed that DCD Tier 2, Revision 2, was revised as committed in the RAI response. Accordingly, **RAI 153-1646, Question 03.05.02-1, is resolved**.

Section 9.2.5, "Ultimate Heat Sink," of this report addresses the staff's evaluation the US-APWR design of the UHS including the evaluation of US-APWR structure conformance to the guidance of RG 1.27, Revision 2.

Section 3.5.1.3, "Turbine Missiles," of this report addresses the staff's evaluation of the protection from low-trajectory turbine missiles including the evaluation of US-APWR structure conformance to the guidance of RG 1.115, Revision 1.

In the DCD markup in the supplemental response to **RAI 907-6321, Question 02-3**, dated September 13, 2012, the applicant proposed to revise DCD Tier 2, Section 3.5.1.4 and Section 3.5.2 to include hurricane-generated missiles. The applicant also proposed to revise COL Information Items 3.3(3) and 3.5(5) to include hurricane loads. The proposed revised COL Information Item 3.3(3) states:

> It is the responsibility of the COL Applicant to assure that site-specific structures and components not designed for tornado and hurricane loads will not impact either the function or integrity of adjacent safety-related SSCs, or generate missiles having more severe effects than those discussed in Subsection 3.5.1.4.

The proposed revised COL Information Item 3.5(5) states:

> The COL Applicant is responsible to evaluate site-specific hazards for external events that may produce missiles more energetic than tornado missiles, and hurricane missiles for the standard plant presented in Subsection 3.5.1.4 and assure that the design of seismic Category I and II structures meet these loads.

Therefore, the applicant considers tornado and hurricane-generated missiles as the limiting externally generated missiles on a plant site. As stated above, protection of SSCs important to safety against external missiles, including the effects of tornado and hurricane missiles, is provided by the R/B, PS/B, PCCV, PSFSV and UHSRS structures. Therefore, the staff concludes that the US-APWR design conforms with RG 1.117, Revision 1, Appendix A. Since the applicant has proposed DCD changes, **RAI 907-6321, Question 02-3, is being tracked as a Confirmatory Item.**

Section 3.5.1.4, “Missiles Generated by Tornados and Extreme Winds,” of this report addresses the staff’s evaluation the tornado missile spectrum used for the US-APWR design.

Section 3.3.2 of this report addresses the staff’s evaluation of the design of Category I structures regarding the protection of these structures from the effects resulting from the failure of adjacent non-seismic Category I structures during a tornado or hurricane.

On the basis of its review and for the reasons set forth above, the staff concludes that the US-APWR missile protection for SSCs important to safety, including stored spent fuel and the UHS, against externally generated missiles is in accordance with the guidelines of RG 1.13, RG 1.27, RG 1.115 and RG 1.117. Therefore, the staff concludes that the US-APWR design meets the guidelines described in SRP Section 3.5.2 and the requirements of GDC 2 and GDC 4 for providing protection for US-APWR plant SSCs important to safety against externally generated missiles.

3.5.2.4.2 Inspections, Tests, Analyses, and Acceptance Criteria

In DCD Tier 1, Section 2.0, the applicant provides the design descriptions and ITAAC which commit to verify the US-APWR structures to be protected from externally-generated missiles are designed and performed as described in DCD Tier 2. Specifically, ITAAC Item 6 listed in DCD Tier 1, Table 2.2-4, “Structural and Systems Engineering Inspections, Tests, Analyses, and Acceptance Criteria,” requires analyses to be performed to verify that the as-built safety-related standard plant structures, other than the PCCV structural design-basis loads are reconciled.

The staff finds that the above cited ITAAC Item which commits to verify that the safety-related SSCs are protected from externally-generated missiles and are designed and performed as described in DCD Tier 2, acceptable. Therefore, the staff concludes that the missile protection provided for US-APWR SSCs important to safety comply with the requirements of 10 CFR 52.47(b)(1).

3.5.2.4.3 Initial Testing

DCD Tier 2, Revision 3, Section 14.2, "Initial Plant Test Program," does not have any initial testing requirements associated with this review item. The staff reviewed DCD Tier 2, Revision 3, Section 3.5.2 against the guidance in SRP Section 14.2, "Initial Plant Test Program," Revision 3, and found that no additional initial testing is needed in connection with this section.

3.5.2.4.4 Technical Specifications

DCD Tier 2, Revision 3, Chapter 16, "Technical Specifications," does not have any TS requirements associated with this review item. The staff reviewed DCD Tier 2, Revision 3, Section 3.5.2 against 10 CFR 50.36, "Technical Specifications," and agrees that no TSs are needed in connection with this section.

3.5.2.5 Combined License Information Items

The following is a list of COL item numbers and descriptions from Table 1.8-2 of the DCD related to the protection of SSCs from externally generated missiles:

Table 3.5.2-1
US-APWR Combined License Information Items

Item No.	Description	Section
COL 3.3(3)	It is the responsibility of the COL applicant to assure that site-specific structures and components not designed for tornado and hurricane loads will not impact either the function or integrity of adjacent safety-related SSCs, or generate missiles having more severe effects than those discussed in Subsection 3.5.1.4.	3.3.2.3
COL 3.5(5)	The COL Applicant is responsible to evaluate site-specific hazards for external events that may produce missiles more energetic than tornado missiles and hurricane missiles for the standard plant presented in Subsection 3.5.1.4, and assure that the design of seismic Category I and II structures meet these loads.	3.5.2

The staff finds the above listing to be complete. Also, the list adequately describes actions necessary for the COL applicant. No additional COL information items need to be included in DCD Tier 2, Table 1.8-2 for externally generated missile considerations.

Note that the applicant revised COL Information Items 3.3(3) and 3.5(5) in the supplemental response to **RAI 907-6321, Question 02-3.** As discussed in Section 3.5.2.4 of this report above, **RAI 907-6321, Question 02-3, is being tracked as a Confirmatory Item.**

3.5.2.6 Conclusions

Based on the forgoing, pending resolution of **RAI 907-6321, Question 02-3, which is being tracked as a Confirmatory Item**, the staff concludes that the that the US-APWR design satisfies the guidelines of SRP Section 3.5.2, Revision 3, and the SSCs to be protected from externally-generated missiles are in conformance with the guidance described in RG 1.13, RG 1.27, RG 1.115, and RG 1.117 and, therefore, meet the requirements of GDC 2 and GDC 4.

The staff further concludes that adequate protection features have been provided for the US-APWR design to protect safety-related SSCs against externally generated missiles.

3.5.3 Barrier Design Procedures

3.5.3.1 Introduction

The barrier design procedures provide the following information concerning the design of structures or barriers to resist missile hazards:

- Local damage prediction methods.
- Methodology for estimating penetration depth.
- How barrier thickness is estimated to prevent perforation.
- Methods used to estimate the potential of concrete barriers generating secondary missiles (i.e., spalling and scabbing effects).
- Prediction methods used to model the response of the barrier to the impact force of the missile.

3.5.3.2 Summary of Application

DCD Tier 1: The Tier 1 information associated with this section is found in DCD, Tier 1, Section 2.2.2, "Protection Against Hazards."

DCD Tier 2: The applicant has provided a DCD Tier 2 system description in Section 3.5.3, "Barrier Design Procedures," summarized here in part, as follows:

> As necessary, equipment and shields/barriers are provided to prevent damage to safety-related SSCs. This section provides the design methodology used to provide that protection.

ITAAC: The ITAAC associated with DCD Tier 2, Section 3.5.3 are given in DCD Tier 1, Section 2.2.4, "Inspection, Tests, Analyses, and Acceptance Criteria."

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

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

Technical Reports: The technical report associated with DCD Tier 2 Section 3.5.3 is:

> MUAP-07028-P, "Probability of Missile Generation from Low Pressure Turbines," Revision 1, issued January 2011.

Cross-cutting Requirements (TMI, USI/ GSI, Op Ex): There are no cross-cutting requirements for this area of review.

US-APWR Interface Issues Identified in the DCD: See DCD Tier 2, Table 1.8-2, "Compilation of All Combined License Applicant Items for Chapters 1-19."

Site Interface Issues Identified in the DCD: See DCD Tier 2, Table 1.8-1, "Significant Site Specific Interfaces with the Standard US-APWR Design."

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

3.5.3.3 Regulatory Basis

The relevant requirements of the Commission's regulations for this area of review, and the associated acceptance criteria are given in Section 3.5.3, "Barrier Design Procedures," Revision 3, issued March 2007, of NUREG-0800 and are summarized below. Review interfaces with other SRP sections can be found in Section 3.5.3 of NUREG-0800.

1. GDC 2, as it relates to the ability of SSCs without loss of capability to perform their safety function, to withstand the effects of natural phenomena, such as earthquakes, tornadoes, hurricanes, floods, and the appropriate combination of all loads.

2. GDC 4, as it relates to the protection of SSCs against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that may result from equipment failures and from events and conditions outside the nuclear power unit.

3. 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 has been constructed and will operate in accordance with the DC and NRC regulations.

Specific criteria sufficient to meet the relevant requirements of GDC 2 and 4 are as follows:

1. For Local Damage Prediction

   a. Concrete

      Sufficient thickness of concrete should be provided to prevent perforation, spalling, or scabbing of the barriers in the event of missile impact. Several empirical equations, such as the modified National Defense Research Council (NDRC) formula; proposed in "A Review of Procedures for the Analysis and Design of Concrete Structures to Resist Missile Impact Effects," by R.P. Kennedy, Nuclear Engineering and Design 1976 Pages 183-203 are available to estimate missile penetration into concrete. These equations should be used to determine the required barrier thicknesses. Thicknesses resulting from such calculations should not be less than those listed in SRP Section 3.5.3, Table 1, "Minimum Acceptable Barrier Thickness Requirements," which specifies the minimum thicknesses necessary to protect against tornado missiles.

      SRP Section 3.5.3, Table 1, provides minimum concrete barrier thickness requirements for preventing local damage against tornado generated missiles for tornado spectrum shown in Table 2 of RG 1.76.

      Barrier thicknesses less than those listed in SRP Section 3.5.3, Table 1 may be used, provided that sufficient justification (including test data) is presented to support them. This justification will be reviewed on a case-by-case basis.

Other types of missiles are specified in SRP Sections 3.5.1.1 through 3.5.1.6.

For turbine missile barriers, penetration and scabbing predictions should be based on empirical equations such as the modified NDRC formula or the results of a valid test program.

b. Steel

The results of tests conducted by the Stanford Research Institute (SRI) on the penetration of missiles into steel plates are summarized in "U.S. Reactor Containment Technology" (Oak Ridge National Laboratory/Nuclear Safety Information Center (ORNL/NSIC)-5, Vol.1, Chapter 6, Oak Ridge National Laboratory, 1965) by W.B. Cottrell and A.W. Savolainen. The equations presented in aforementioned document are acceptable. Other equations such as the Ballistic Research Laboratory formula described in, "Reactor Safeguards," by C. R. Russell, published by MacMillan, New York, 1962, may be used, provided the results are either comparable to those obtained by using the aforementioned "U.S. Reactor Containment Technology" method or are validated by penetration tests.

c. Composite Sections

For composite or multi-element barriers, procedures for prediction of local damage are acceptable if the residual velocity of the missile perforating the first element is considered as the striking velocity for the next element. For determining this residual velocity, the equations presented in "Ballistic Perforation Dynamics," Journal of Applied Mechanics, Transactions of the ASME, Vol. 30, Series E, No. 3, September 1963 by R. F. Recht and T. W. Ipson, are acceptable when the first barrier of a multi-element missile barrier is steel. When the first barrier is concrete, procedures used are reviewed on a case-by-case basis.

2. For Overall Damage Prediction

The response of a structure or barrier to missile impact depends largely on the location of impact (e.g., midspan of a slab or near a support), on the dynamic properties of the target and missile, and on the kinetic energy of the missile. In general, the assumption of plastic collisions is acceptable, where all of the missile's initial momentum is transferred to the target and only a portion of its kinetic energy is absorbed as strain energy within the target. However, where elastic impacts are expected, the additional momentum transferred to the target by missile rebound should be considered in the analyses.

After it has been demonstrated that the missile will not penetrate the barrier, an equivalent static load concentrated at the impact area should then be determined, from which the structural response, in conjunction with other design loads, can be evaluated using conventional design methods. An acceptable procedure for such an analysis, where the impact is assumed to be plastic, is presented in “Impact Effect of Fragments Striking Structural Elements,” Holmes and Narver, Inc., revised November 1973 by R. A. Williamson and R. R. Alvy. Other procedures may be used, with adequate justification provided the results obtained are comparable to that of the above reference.

Maximum allowable ductility ratios for steel and reinforced concrete barriers, in the above analysis, are given in American National Standard Institute/ American Institute of Steel Construction (ANSI/AISC) N690-1994 including supplement 2(2004), American National Standard Specification for the Design, Fabrication, and Erection of Steel Safety-Related Structures for Nuclear Facilities (1994) and in RG 1.142, respectively.

3.5.3.4 Technical Evaluation

The staff reviewed the procedures used to design seismic Category I structures, shields, and barriers described in DCD Tier 2, Revision 3 to withstand the effects of missile impacts. These areas were evaluated to ensure compliance with GDC 2 and 4.

The review focused on procedures used to predict local damage in the impacted area and procedures used to predict the overall response of the barrier or portion thereof due to missile impact. This included (1) estimation of the depth of penetration and, in case of concrete barriers, the potential for generation of secondary missiles by spalling or scabbing effects; and (2) assumptions on acceptable ductility ratios where elasto-plastic behavior is relied upon, and procedures for estimation of forces, moments, and shears induced in the barrier by the impact force of the missile. The staff also reviewed the adequacy of missile parameters cited in support of the applicant’s conclusions concerning their suitability for the plant and the effects of missiles on SSCs. Review activities also addressed COL information items.

The staff also evaluated the adequacy of missile parameters cited in support of the applicant's conclusions concerning their suitability for the plant in accordance with SRP Sections 3.5.1.1, “Internally Generated Missiles (Outside Containment),” Revision 3, issued March 2007; 3.5.1.2, “Internally-Generated Missiles (Inside Containment),” Revision 3, issued March 2007; 3.5.1.4, “Missiles Generated by Tornadoes and Extreme Winds,” Revision 3, issued March 2007; 3.5.1.5, “Site Proximity Missiles (Except Aircraft),” Revision 4, March 2007; and 3.5.1.6, “Aircraft Hazards,” Revision 4, issued March 2010. In addition, the staff reviewed turbine missile parameters cited in support of the applicant's conclusions concerning their suitability for the plant based on guidance in SRP Section 3.5.1.3 “Turbine Missiles,” Revision 3, issued March 2007. The staff also reviewed the SSCs that require protection from externally generated missiles including all plant site safety-related SSCs that support the reactor facility based on guidance in SRP Section 3.5.2, “Structures, Systems, and Components to be Protected from Externally-Generated Missiles,” Revision 3, issued March 2007.

The staff performed the technical evaluation of barrier design procedures based on specific acceptance criteria and review procedures in the referenced SRP sections.

3.5.3.4.1 Barrier design parameters

The staff reviewed the US-APWR standard design-basis to verify that the barrier design parameters satisfy GDC 2 and GDC 4.

3.5.3.4.1.1 Tornado and hurricane-generated missiles

The applicant determined tornado forces on structures including equivalent static loads for tornado missile impact based on guidance in RG 1.76, "Design-Basis Tornado and Tornado Missiles for Nuclear Power Plants," Revision 1, issued March 2007, for Region 1. To evaluate barrier design procedures, in **RAI 221-1909, Question 03.05.03-1**, the staff requested the applicant to clarify that historical tornado wind speed data are enveloped by the 230 mph (103 m/s) design-basis tornado wind speed for Region 1. In its response to **RAI 221-1909, Question 03.05.03-1**, dated April 8, 2009, the applicant stated that the key site parameter of 230 mph (103 m/s) for maximum tornado wind speed considers data presented in Draft RG (DG)-1143, "Design-Basis Tornado and Tornado Missiles for Nuclear Power Plants," issued January 2006, and RG 1.76, Revision 1. The applicant also provided the following statement.

> DCD Section 2.0 states the site-specific parameters for the US-APWR bound an estimated 75% to 80% of the United States landmass. COL Applicant Item COL 2.3(1) as stated in DCD Section 2.3 is to verify the site-specific regional climatology and local meteorology are bounded by the site parameters for the standard US-APWR design, or to demonstrate by some other means that the proposed facility and associated site-specific characteristics are acceptable at the proposed site. Therefore, the COL Applicant is to address any site-specific tornado wind speeds that are determined to exceed the key site parameters for the standard US-APWR.

Based on an evaluation of the applicant's response to **RAI 221-1909, Question 03.05.03-1**, the staff concludes that using RG 1.76 for maximum tornado wind speed evaluation is appropriate. The staff also concurs with the applicant's statement, consistent with COL Information Item 2.3(1), that the "COL Applicant is to address any site-specific tornado wind speeds that are determined to exceed the key site parameters for the standard US-APWR." Accordingly, the staff finds the response acceptable and **RAI 221-1909, Question 03.05.03-1, is resolved.**

The applicant determined hurricane forces on structures including equivalent static loads for hurricane missile impact based on guidance in RG 1.221, "Design-Basis Hurricane and Hurricane Missiles for Nuclear Power Plants," Revision 0, issued October 2011. To evaluate barrier design procedures, in **RAI 907-6321, Question 2.0-3**, the staff requested the applicant to add hurricane wind speed and hurricane missile spectra to the list of site parameters in Tier 1 and Tier 2 of the DCD. In its response to **RAI 907-6321, Question 02-3**, dated June 26, 2012, as supplemented on September 24, 2012, the applicant stated that the key site parameter of 160 mph (71.5 m/s) for maximum hurricane wind speed considers data presented in RG 1.221, Revision 0, Figures 1-3. The applicant also proposed to add the following to DCD Tier 2, Section 3.3.2.1.

> The design-basis hurricane wind speed is chosen from wind speed contour maps for hurricane-prone regions of the contiguous U.S. presented in RG 1.221. The wind speed due to hurricanes that is selected for design of the standard plant is 160 mph, (71.5 m/s) corresponding to a three-second gust at 33 ft (10 m) above ground for exposure Category C. Exposure Category C, defined in ASCE/SEI 7-05 Section 6.5.6.3, is a typical exposure category for nuclear power plants and includes flat open country, grasslands, and all water surfaces in hurricane-prone regions.

The applicant stated that this designed hurricane wind speed envelops the design-basis hurricane wind speeds at most locations in the contiguous U.S. The applicant also proposed to add COL Information Item 3.3(6) which states that, "The COL Applicant is responsible for verifying that the site-specific design-basis hurricane basic wind speeds exposure category, and resulting wind forces are enveloped by the determinations in this section."

Based on an evaluation of the applicant's response to **RAI 907-6321, Question 2.0-3**, the staff concludes that using RG 1.221 for maximum hurricane wind speed evaluation is appropriate. The staff finds the commitment that the COL applicant will verify that the site-specific design-basis hurricane basic wind speeds are enveloped by this selected value to be acceptable. Therefore, the staff considers the relevant part of the response to **RAI 907-6321, Question 2.0-3, to be** acceptable. Since the applicant has identified DCD changes**, RAI 907-6321, Question 2.0-3, is being tracked as a Confirmatory Item.**

The US-APWR design-basis spectra of tornado and hurricane-generated missiles are discussed in DCD Tier 2, Section 3.5.1.4, "Missiles Generated by Tornadoes and Extreme Winds." The US-APWR design-basis spectra conforms to the spectra of missiles defined in RG 1.76, Revision 1, Table 2 for Region I and RG 1.221, Rev. 0, Table 1. The spectra of missiles are chosen to represent: (1) a massive high-kinetic-energy missile that deforms on impact, (2) a rigid missile that tests penetration resistance, and (3) a small rigid missile of a size sufficient to pass through any opening in protective barriers. Although only three types of missiles are included in the design-basis tornado and hurricane missile spectrum, the applicant is required to consider including other types of tornado or hurricane-generated missiles in the design bases to ensure that the design has sufficient margin for the limited accuracy, quantity, and period of time in which the historical data have been accumulated.

To verify compliance with GDC 2, in **RAI 221-1909, Question 03.05.03-6,** the staff requested the applicant to characterize the massive missiles included in the US-APWR standard plant design-basis. In its response to **RAI 221-1909, Question 03.05.03-6**, dated April 9, 2009, the applicant stated the following.

> The discussion in the first paragraph of DCD Subsection 3.5.1.4 recognizes the spectrum of missiles includes a massive high-kinetic-energy missile that deforms on impact. By the position provided by RG 1.76, the NRC staff considers an automobile to be acceptable as the massive missile for use in the design of the nuclear power plant.
>
> DCD Subsection 3.5.3.1 provides the formulas to predict the penetration depth (x), scabbing thickness (t_s) and the perforation thickness (t_p) potential created by the missile impact. As noted by the NRC staff in **RAI 221-1909, Question 03.05.03-2**, the Modified NDRC formula that is applicable for the design of concrete surfaces includes a missile shape factor N that is dependent on the projectile nose shape. The impact of very soft missiles subjected to large deformations upon the impact may be considered as caused by a blunt or flat nose object. Therefore, a reduced value for the missile shape factor N is applicable; however massive high-kinetic-energy missiles may be conservatively analyzed using N equal to 1.0 to limit the justification of the basis for selection of any lower value of N.

The staff finds the applicant's response acceptable since the staff concludes that an automobile is an acceptable massive missile for barrier design and use of shape factor, N, equal to 1.0 is conservative and therefore acceptable. Accordingly, **RAI 221-1909, Question 03.05.03-6, is resolved.**

To verify compliance with GDC 2, in **RAI 686-4557, Question 03.05.03-9,** the staff requested the applicant to provide a safety analysis that assesses potential damage to seismic Category I structures resulting from an extreme wind-generated missile impact at any elevation above grade and any azimuthal direction. Specifically, the applicant was requested to address the following two types of damage in the analysis and to provide technical evidence that potential impacts from each type of missile within the applicant's design-basis tornado and hurricane missile spectrum does not compromise the structural integrity of any seismic Category I structure or adversely affect its ability to perform its intended safety functions.

(1) Local damage:

   (a) full penetration of missile "punch-thru" due to shear failure

   (b) crack initiation and propagation due to partial penetration at a building location under highest stress

(2) Global damage:

   (a) building "tip-over" or "sliding" due to foundation failure

   (b) failure of critical section due to severe impact/dynamic loads

The applicant was also requested to include as part of the analysis a description of the physical characteristics, the maximum speed, and the envelope of potential impact locations (i.e., SSC identifier, elevation above grade, and corresponding azimuthal direction) for each type of missile included in the applicant's design-basis tornado and hurricane missile spectrum.

In its response to **RAI 686-4557, Question 03.05.03-9**, dated April 9, 2009, the applicant stated the following.

> With respect to item (2) global damage in the RAI question: Design for building "tip over" and foundation sliding failure are dominated by the seismic design load combinations, not by load combinations involving tornado missiles. For example, the heaviest missile, which is the 4000 lb. automobile missile, is about 0.013% of the weight of a Power Source Building, which weighs roughly 30,000,000 lb. (Reference MHI Technical Report MUAP-10001, "Seismic Design Bases of the US-APWR Standard Plant", Revision 2, Table 5.4.2-1). Due to this small ratio, lateral building load due to transfer of the automobile missile kinetic energy will have negligible impact on "tip-over" and sliding. This ratio is even less for the PCCV, which weighs roughly 70,000,000 lb. (shell cylinder and dome portions only)

Based on the above information, the staff finds that the building "tip-over" and foundation sliding failure are indeed bounded by the design-basis seismic loadings. However, the impact loading by the automobile missile (> 500 Hz, high pulse) imposes a different structural response from seismic loading (<33 Hz, inertia force induced, quasi-static). Thus, the seismic analysis methodology may not apply directly in this case. One of the major differences is the applied load distribution wherein the impact force induced by the auto missile at the impact site is much bigger than the inertia force induced at that location locally by the SSE, even though the overall seismic loads are higher. Accordingly, the local effects induced by the localized dynamic impact load can still affect the structural integrity at or near the impact location of the building.

DCD Tier 2, Section 3.5.1.4, states that "all seismic Category I structures are capable of withstanding the impact of each identified tornado missile at any elevation, including the potential impact of a 4,000 lb. (1814 kg) automobile greater than 30 ft (9.1 m) above grade." Thus, the staff requested investigation of local impact effects to assure structural integrity of all seismic Category I structures under an automobile missile striking at any elevation. The analyses should include local shear response of the building at the critical elevation level near the auto missile impact site as well as the possibility of auto missile penetrations at the weakest locations wherever the missile can strike. It should be noted that the case of auto missile impact at elevations higher than 30 ft (9.1 m) above grade is not covered by RG 1.76, thus not addressed in Item (1) of the response to **RAI 686-4557, Question 03.05.03-9**. Therefore, the staff closed as unresolved **RAI 686-4557, Question 03.05.03-9,** and in follow-up **RAI 758-5680, Question 03.05.03-10,** the staff requested the applicant to provide an analysis assessing the local effects of an automobile missile on all seismic Category I structures. **RAI 758-5680, Question 03.05.03-10 is being tracked as an Open Item**.

3.5.3.4.1.2 Aircraft missiles

DCD Tier 2, Section 3.5.1.6 states the following concerning aircraft hazards.

> The US-APWR standard plant design-basis is that the plant is located such that an aircraft crash and air transportation accidents are not required to be considered as part of the design-basis. It is the responsibility of the COL Applicant to verify the site interface parameters with respect to aircraft crashes and air transportation accidents as described in Section 2.2. Additional analyses may be required to evaluate potential aircraft missiles.

To further clarify this statement, in **RAI 221-1909, Question 03.05.03-5,** the staff requested the applicant to provide additional justification for excluding aircraft crashes and air transportation accidents from the US-APWR standard plant design-basis. The applicant included the following statement in its response to **RAI 221-1909, Question 03.05.03-5**.

> It is the COL Applicant's responsibility to verify the site interface parameters with respect to aircraft crashes and air transportation accidents. Evaluations of malevolent threats, such as terrorist attack using airplane missiles, are of beyond DBEs. Aircraft impact assessment has been conducted to meet NRC rule 10 CFR 50.150 using NEI 07-13 methodology and NRC specified aircraft threat. The technical report for the beyond-design-basis aircraft impact is being prepared now, and is scheduled to be submitted to the NRC this year. Content of the assessment and the results will be safeguards information. After the release of the technical report, DCD Chapter 19 will be updated to provide non-safeguards information applicable to the DCD, or to reference the technical report.

Based on an evaluation of the applicant's response to **RAI 221-1909, Question 03.05.03-5**, the staff concurs with the applicant that per COL Information Item 3.5(4), the COL applicant is responsible to verify the site interface parameters with respect to aircraft crashes and air transportation accidents. In addition, the applicant has conducted beyond design-basis aircraft impact assessment in accordance with 10 CFR 50.150 and provided the results in DCD Tier 2, Chapter 19, Appendix 19A. Therefore, the staff finds the response acceptable. Accordingly, **RAI 221-1909, Question 03.05.03-5, is resolved.**

3.5.3.4.1.3 Turbine missiles

DCD Tier 2, Section 3.5.1.3, “Turbine Missiles,” discusses the turbine missiles considered in the US-APWR standard design. According to the applicant, current evidence suggests that low trajectory turbine missile strikes are concentrated within an area bounded by lines inclined at 25 degrees to the turbine wheel planes and passing through the end wheels of the low pressure stages. The T/G is located south of the nuclear island with its shaft oriented along the north-south axis. In this orientation, the R/B, PCCV, and PS/B, and SSCs, which are defined by the guidance and examples in RG 1.117, “Tornado Design Classification,” Revision 1, issued April 1978 within the same unit, are located outside the high velocity and low trajectory missile strike zone as defined by RG 1.115, “Protection Against Low-Trajectory Turbine Missiles,” Revision 1, issued July 1977. Per COL Information Item 3.5(6), the COL applicant is responsible to assess the orientation of the T/G of this and other unit(s) at multi-unit site for the probability of missile generation using the evaluation of DCD Tier 2, Subsection 3.5.1.3.2. The review of this COL Information Item is in Section 3.5.1.3 of this report.

The applicant does not rely on turbine missile barriers to meet the guidelines of RG 1.115 and RG 1.117. The protection against turbine missiles is evaluated in Section 3.5.1.3 of this report.

3.5.3.4.2 Procedures for local damage prediction

The staff reviewed procedures used by the applicant for local damage prediction including identified differences between the design features, analytical techniques, and procedural measures proposed for its facility and the SRP acceptance criteria. The staff also evaluated how the proposed alternatives to the SRP acceptance criteria provide acceptable methods of compliance with NRC regulations.

3.5.3.4.2.1 Concrete barriers

The applicant used the NDRC procedure, “A Review of Procedures for the Analysis and Design of Concrete Structures to Resist Missile Impact Effects,” issued 1976, for the analysis and design of concrete structures to resist missile impact effects. The staff concludes that use of the NDRC procedure for the analysis and design of concrete structures to resist missile impact effects is consistent with acceptance criteria provided in SRP Section 3.5.3, Subsection II.

<u>Missile shape factor</u>

In DCD Tier 2, Revision 1, Subsection 3.5.3.1.1, “Concrete,” the applicant assigned a value of unity to missile shape factor, N, in the Modified NDRC formula. However, the missile shape factor is dependent on the projectile nose shape as described below.

N = 0.72 for flat nosed body
N = 0.84 for blunt nose body
N = 1.0 for average bullet nose (spherical end)
N = 1.14 for very sharp nose

Use of N=1.0 in DCD Tier 2, Subsection 3.5.3.1.1 is acceptable as this value is for the average bullet nose commonly encountered in real case and is very close to the upper bound value of 1.14 associated with a very sharp nose which is rarely encountered in reality. Based on these findings, the staff concluded that use of missile shape factors based on projectile nose shape is appropriate.

In addition, during its review of the Modified NDRC formula, the staff identified a typographical error in the equation for t_p/d. The correct term for t_p/d is stated below.

$$t_p/d = 3.19\ (\ x/d\ ) - 0.718\ (x/d\ )^2 \text{ for } x/d \leq 1.35$$

Furthermore, the applicant used variable, *W,* in the Modified NDRC formula for concrete in DCD Tier 2, Subsection 3.5.3.1.1 and in the SRI formula for steel in DCD Tier 2, Subsection 3.5.3.1.2, "Steel," to represent different parameters. In order to avoid possible confusion, the staff identified that different variables should be used in these two equations.

In **RAI 221-1909, Question 03.05.03-2**, the staff requested the applicant to respond to these issues. In its response to **RAI 221-1909, Question 03.05.03-2**, dated April 8, 2009, the applicant proposed to revise the DCD Tier 2, Subsection 3.5.3.1.1 to adjust the missile shape factor, N, based on the projectile nose shape, replace variable, *W*, with, W_m, in the Modified NDRC formula, and correct the typographical error. The staff finds the response acceptable since the applicant corrected the issues identified by the staff. The staff confirmed that the DCD changes were incorporated into DCD Revision 3. Accordingly, **RAI 221-1909, Question 03.05.03-2, is resolved.** With these corrections, the staff finds the missile shape factor acceptable.

<u>Concrete barrier thickness</u>

In order to provide a sufficient margin of safety, the applicant stated in DCD Tier 2, Section 3.5.3.1.1 that the design thickness of concrete barriers is 20 percent greater than the threshold value for the phenomenon being prevented. The applicant also selected concrete wall thicknesses to satisfy the minimum barrier thicknesses to prevent local damage against tornado or hurricane generated missiles to be consistent with thickness values in SRP Section 3.5.3, Table 1. Therefore, the staff finds the approach to design thickness of concrete barriers acceptable.

3.5.3.4.2.2 Steel barriers

The applicant stated in DCD Tier 2, Subsection 3.5.3.1.2 that the perforation threshold for steel plate thickness is determined by satisfying both the Ballistic Research Laboratory (BRL) formula available in "Reactor Safeguards," and the SRI formula summarized in NSIC-5, "US Reactor Containment Technology," issued 1965. The use of these two formulas to determine the penetration of missiles into steel plates is consistent with acceptance criteria provided in SRP Section 3.5.3, Subsection II.1.B.

After reviewing the BRL and the SRI formulas in DCD Tier 2, Subsection 3.5.3.1.2, in **RAI 221-1909, Question 03.05.03-3,** the staff requested the applicant to address the designation for the SRI formula and to clarify and restructure the SRI formula for calculating penetration depth. In its response to **RAI 221-1909, Question 03.05.03-3**, dated April 8, 2009, the applicant stated that the parenthetical abbreviation of "Stanford Formula" in DCD Tier 2, Revision 1, Subsection 3.5.3.1.2, is not intended to be a reference to any institution other than Stanford Research

Institute as referenced within the DCD and to add “SRI” as the acronym for “Stanford Research Institute.”

The applicant clarified that the use of the SRI formula is defined within the DCD Tier 2, Subsection 3.5.3.1.2 as valid within specified ranges. However, the introductory paragraph in DCD Tier 2, Subsection 3.5.3.1.2 indicates that the BRL formula is an alternate method that provides comparable results to the SRI formula. Since the SRI formula is based on empirical data derived from parameter-dependent tests, the applicant proposed to restructure DCD Tier 2. Subsection 3.5.3.1.2 to use the BRL Formula in determining the required plate thickness; however, the plate thickness is not to be less than that calculated using the SRI Formula.

The staff finds the applicant’s response acceptable since the applicant proposed to modify the DCD regarding the BRL and SRI formula to be consistent with industry practice. The staff confirmed that the proposed changes to DCD Tier 2, Subsection 3.5.3.1.2 were incorporated into DCD Revision 3. Accordingly, **RAI 221-1909, Question 03.05.03-3, is resolved.** With these corrections, the staff finds the equations used for steel barriers acceptable.

3.5.3.4.2.3 Composite barriers

DCD Tier 2, Revision 1, Subsection 3.5.3.1.3, “Composite (Modular) Sections,” states concerning composite (modular) sections, “Composite or multi-element barriers consider the residual velocity of the missile perforating the first element as the striking velocity for the next element.” In order to evaluate consistency with SRP Section 3.5.3, in **RAI 221-1909, Question 03.05.03-4,** the staff requested the applicant to define the formula used to determine residual velocity. In its response to **RAI 221-1909, Question 03.05.03-4**, dated April 8, 2009, the applicant proposed including an equation in the DCD Tier 2, Subsection 3.5.3.1.3 for calculating residual velocity after missile penetration of the first layer in a composite section. The revised statement in DCD Tier 2, Subsection 3.5.3.1.3 follows.

> The residual velocity after missile penetration of the first layer (or outer shield) is determined by the formula:
>
> $$V_r = \sqrt{V^2 - {V_B}^2}$$
>
> Where
>
> V_r = residual velocity after missile penetration of the first layer (or outer shield).
> V = impact (or striking) velocity of the missile object.
> V_B = perforation velocity associated with the energy absorbed up to the threshold of perforation.

The staff finds that the proposed equation is consistent with the equations presented in “Ballistic Perforation Dynamics,” Journal of Applied Mechanics, Transactions of the ASME, Vol. 30, Series E, No. 3, issued September 1963 by R. F. Recht and T. W. Ipson, which are recommended for use in the acceptance criteria provided in SRP Section 3.5.3, Subsection II.1.C. Therefore, the staff finds the response acceptable. The staff confirmed that the proposed changes to DCD Tier 2, Subsection 3.5.3.1.3 were incorporated into DCD Revision 3. Accordingly, **RAI 221-1909, Question 03.05.03-4 is resolved.** With these corrections, the staff finds the equations used for composite barriers acceptable.

3.5.3.4.3 Evaluation of Overall Structural Effects

The staff reviewed the procedures in DCD Tier 2, Section 3.5.3.2, “Evaluation of Overall Structural Effects,” for evaluating overall structural effects from missile impacts on concrete and steel barriers for consistency with acceptance criteria in SRP Section 3.5.3, Subsection II. According to the applicant,

> Elements required to remain elastic are evaluated to assure that the usable strength capacity exceeds the demand. For structures allowed to displace beyond yield (elasto-plastic response), an evaluation confirms that acceptable deformation limits to demonstrate ductile behavior are not exceeded by comparing computed demand ductility ratios with capacity values.

For concrete barrier design, the applicant qualified overall structural response including flexural, shear, and buckling effects on structural members using the equivalent static load obtained from the evaluation of missile impact on structural response. Stress and strain limits for the equivalent static load are consistent with RG 1.142, “Safety-Related Concrete Structures for Nuclear Power Plants (Other than Reactor Vessels and Containments),” Revision 2, issued November 2001, and ANSI/ AISC N690-1994 “Specification for the Design, Fabrication and Erection of Steel Safety-Related Structures for Nuclear Facilities”, including Supplement 2 (2004), 1994 and 2004. However, in the DCD Tier 2, Subsection 1.2.1.2.10, “Design Criteria for Natural Phenomena”, it is stated that static analysis is employed to investigate missile damage for both cases of tornado and hurricane. Thus, in **RAI 981-6968, Question 03.05.03-11**, the staff requested the applicant to change the “static analysis” to “equivalent static analysis” DCD Tier 2, Subsection 1.2.1.2.10, so as to be consistent with the methodology described in DCD Tier 2, Section 3.5.3.2. In its response to **RAI 981-6968, Question 03.05.03-11**, dated February 5, 2013, the applicant agreed to delete the misleading description of the analytical method. The staff finds the response acceptable since the applicant modified DCD Chapter 1 to be consistent with DCD Tier 2, Section 3.5.3.2. Since the applicant proposed DCD changes, **RAI 981-6968, Question 03.05.03-11, is being tracked as a Confirmatory Item.**

For steel barrier design, the applicant qualified overall structural response using maximum allowable ductility ratios provided by AISC N690 including Supplement 2.

After evaluating the concrete and steel barrier design procedures used by the applicant for evaluating overall structural effects from missile impacts, the staff finds that the acceptance criteria for overall damage prediction in SRP Section 3.5.3, Subsection II are satisfied.

3.5.3.5 Combined License Information Items

The following is a list of COL item numbers and descriptions from Table 1.8-2, “Compilation of All Combined License Applicant Items for Chapters 1-19,” of the DCD related to barrier design procedures:

Table 3.5.3-1
US-APWR Combined License Information Items

Item No.	Description	Section

Table 3.5.3-1 US-APWR Combined License Information Items		
Item No.	**Description**	**Section**
COL 3.5(4)	It is the responsibility of the COL applicant to verify the site interface parameters with respect to aircraft crashes and air transportation accidents as described in Section 2.2.	2.2 3.5.1.6
COL 3.5(5)	The COL applicant is responsible to evaluate site-specific hazards for external events that may produce missiles more energetic than tornado and hurricane missiles contained in Sec. 3.5.1.4, and assure that the design of seismic Category I and II structures meet these loads.	3.5.2
COL 3.5(6)	The COL applicant is responsible to assess the orientation of the turbine generator (T/G) of this and other unit(s) at multi-unit site for the probability of missile generation using the evaluation of Section 3.5.1.3.2.	3.5.1.3.2

The staff finds COL Information Item 3.5(4) acceptable since it identifies necessary site interface requirements and it is discussed further in Section 3.5.3.4.1.2 of this report. The staff finds COL Information Item 3.5(5) acceptable since it is necessary in order to assure the design is adequate for protection against site-specific missile attacks in compliance with GDC 4. The staff finds COL Information Item 3.5(6) acceptable since it supports compliance with GDC 4 and it is discussed further in Section 3.5.3.4.1.3 of this report.

Based on the above, the staff finds the above listing to be complete. Also, the list adequately describes actions necessary for the COL applicant or licensee. No additional COL information items were identified that need to be included in DCD Tier 2 Table 1.8-2 regarding barrier design procedures.

3.5.3.6 Conclusions

As a result of the open item for **RAI 758-5680, Question 03.05.03-10**, the staff is unable to finalize its conclusions on Section 3.5.3 related to barrier design procedures, in accordance with NRC regulations.

3.6 Protection against the Dynamic Effects Associated with the Postulated Rupture of Piping

3.6.1 Plant Design for Protection Against Postulated Piping Failures in Fluid Systems Outside Containment

3.6.1.1 Introduction

This section addresses the US-APWR design bases and criteria required to demonstrate essential systems and components are protected against postulated piping failures outside of containment. High and moderate energy systems representing potential sources of dynamic effects associated with pipe rupture are identified, and the criteria for separation and the evaluation of adverse consequences are defined. The design objective of these piping systems is to ensure that the plant can be safely shut down and the functions of safety-related systems are not affected in the event of piping failures outside containment.

3.6.1.2 Summary of Application

DCD Tier 1: Tier 1 information associated with this section is found in DCD Tier 1, Section 2.3, "Piping Systems and Components," Revision 3. This section contains 2 items that are directly related to protection against postulated piping failures in fluid systems outside of containment. DCD Tier 1, Section 2.3, Table 2.3-2, "Piping Systems and Components ITAAC," the applicant provided the piping break hazard analyses ITAAC pertaining to as-designed pipe break hazard analysis and the as-built pipe break hazards analysis reconciliation.

DCD Tier 2: The applicant has provided a DCD Tier 2 system description in DCD Tier 2, Section 3.6.1, "Plant Design for Protection against Postulated Piping Failure in Fluid Systems Inside and Outside Containment." This section describes the methodology used in designing the protection of essential systems and components from the consequences of postulated piping failures outside containment.

ITAAC: The ITAAC associated with DCD Tier 2, Section 3.6.1 are delineated in DCD Tier 1, Table 2.3-2, "Piping Systems and Components ITAAC," which provide the piping break hazard analyses ITAAC pertaining to as-designed pipe break hazard analysis and the as-built pipe break hazards analysis reconciliation.

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 for this area of review.

Cross-cutting Requirements (TMI, USI/ GSI, Op Ex): The cross-cutting requirements associated with DCD Tier 2, Section 3.6.1 are TMI III.D.1.1, Item A-18 of NUREG-0933, and New Generic Issue #156.6.1 and New Generic Issue #163 discussed in DCD Tier 2, Section 1.9.3.

US-APWR Interface Issues Identified in the DCD: See DCD Tier 2, Table 1.8-2, "Compilation of All Combined License Applicant Items for Chapters 1-19."

Site Interface Issues Identified in the DCD: See DCD Tier 2, Table 1.8-1, "Significant Site Specific Interfaces with the Standard US-APWR Design."

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

3.6.1.3 Regulatory Basis

The relevant requirements of the Commission regulations for this area of review, and the associated acceptance criteria, are given in Section 3.6.1, "Plant Design for Protection Against Postulated Piping Failures in Fluid Systems Outside Containment," Revision 3, issued March 2007, of NUREG-0800, Standard Review Plan (SRP), and are summarized below. Review interfaces with other SRP sections also can be found in SRP Section 3.6.1.

1. GDC 2, as it relates to protection against natural phenomena, such as seismically-induced failures of non-seismic piping. The application of GDC 2 to this section is to incorporate environmental effects of full-circumferential ruptures

of non-seismic moderate-energy piping in areas where effects are not already bounded by failures of high-energy piping. Acceptance is based on conformance to the Branch Technical Position (BTP) 3-3, "Protection Against Postulated Piping Failures in Fluid Systems Outside Containment."

2. GDC 4, as it relates to SSCs important to safety being designed to accommodate the effects of and to be compatible with the environmental conditions associated with postulated pipe rupture. Acceptance is based on conformance to BTP 3-3.

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 DC has been constructed 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. BTP 3-3, "Protection Against Postulated Piping Failures in Fluid Systems Outside Containment," Revision 3, issued March 2007.

2. BTP 3-4, "Postulated Rupture Locations in Fluid System Piping Inside and Outside Containment," Revision 2, issued March 2007.

3.6.1.4 Technical Evaluation

In DCD Tier 2, Section 3.6.1, Revision 3, the applicant presents the general protection criteria to mitigate the postulated piping failure in fluid systems.

The staff reviewed the US-APWR design regarding the protection of essential SSCs against postulated piping failures in fluid systems outside the containment in accordance with GDC 2, GDC 4 and the guidance provided by SRP Section 3.6.1. DCD Tier 2, Section 3.6.1 discusses the measures provided to protect the essential SSCs from the consequences of a postulated piping failure. BTP 3-4, “Postulated Rupture Locations in Fluid System Piping Inside and Outside Containment,” Revision 2, issued March 2007, defines essential SSCs as systems and components necessary to shut down the reactor and mitigate the consequences of a postulated pipe rupture without offsite power. The staff could not find in the DCD the definition of an essential SSC, therefore, the staff could not confirm that the applicant’s definition of an essential SSC is consistent with the definition provided in BTP 3-4. In **RAI 884-6223, Question 03.06.01-10**, the staff requested the applicant to update the DCD in order to include the definition of an essential SSC.

In its response to **RAI 884-6223, Question 03.06.01-10,** dated March 9, 2012, the applicant stated that the term "essential SSCs" used in the DCD is defined in BTP 3-4 as "systems and components necessary to shut down the reactor and mitigate the consequences of a postulated pipe rupture without offsite power." The applicant’s response also proposed to revise DCD Tier 2, Section 3.6.1 to add the statement that the essential SSCs addressed by this section and BTP 3-4 are the safety-related SSCs defined in DCD Tier 2, Section 3.2, “Classification of Structures, Systems, and Components.” The staff reviewed the applicant’s response and determined that the definition of “essential SSCs” in the DCD is consistent with the staff’s definition of “essential SSCs” provided in BTP 3-4. The staff also determined that the proposed changes to DCD Tier 2, Section 3.6.1 are bound by the definition of essential SSCs. Therefore, the staff finds the RAI response and the proposed DCD changes to be acceptable. Since the applicant has proposed DCD changes, **RAI 884-6223, Question 03.06.01-10, is being tracked as a Confirmatory Item.**

3.6.1.4.1 GDC 2 Design Bases for Protection Against Natural Phenomena

The plant design for protection against postulated piping failure in fluid systems outside containment must meet the requirements of GDC 2, as it relates to protection against natural phenomena, such as seismically-induced failures of non-seismic piping. The application of GDC 2 to this section incorporates environmental effects of full-circumferential ruptures of non-seismic moderate energy piping in areas where effects are not already bounded by failures of high energy piping. Acceptance is based on conformance to BTP 3-3, "Protection Against Postulated Piping Failures in Fluid Systems Outside Containment," Revision 3, issued March 2007.

BTP 3-3, Section B.2, "Design Features," states that protective structures and compartments should be designed to seismic Category I requirements. The staff reviewed the DCD Tier 2, Section 3.6.1 and did not find a confirmatory statement that the protective structures and equipment used to protect essential SSCs from pipe rupture would be designed to seismic standards. In **RAI 180-1594, Question 03.06.01-1**, the staff requested the applicant to include in the DCD the seismic standards that the protective structures and equipment use to protect essential SSCs from pipe rupture would be designed to.

In its response to **RAI 180-1594, Question 03.06.01-1,** dated March 3, 2009, the applicant stated that the protective structures relied to protect essential SSCs from pipe ruptures are designed as Category I structures, the applicant's RAI response also proposed to update the DCD to specify the seismic standard of the protective structures use to protect essential SSCs from pipe rupture.

The staff reviewed the applicant's response and determined that the seismic design of the protecting structure is in accordance with the criteria specified in BTP 3-3. The staff confirmed that the proposed DCD changes have been incorporated into Revision 3 of the DCD. Therefore, the staff finds the RAI response to be acceptable. Accordingly, **RAI 180-1594, Question 03.06.01-1, is resolved**.

In DCD Tier 2, Section 3.6.1.1, "Design-Basis," Part E, the applicant states that for systems not analyzed for seismic considerations, it is assumed that a SSE event will cause pressure boundary failure at any location. The DCD also states that non-essential systems for the US-APWR plant are not required to be protected from the dynamic and environmental effects associated with pipe rupture. However, if a failure of a non-essential system, in turn, could affect the functioning of safety-related equipment to an unacceptable level, then this piping is also evaluated for protection. This is consistent with the recommendations of SRP Section 3.6.1 and BTP 3-3.

Based on the discussion above, the staff finds that the applicant has provided sufficient design information to demonstrate conformance with the portions of BTP 3-3 that relate to GDC 2. Therefore, the staff finds the applicant's design meets the requirements of GDC 2 as they relate to the protection of essential systems from natural phenomena.

3.6.1.4.2 GDC 4 Environmental and Dynamic Effects Design Bases

The plant design for protection against postulated piping failure in fluid systems outside containment must meet the requirements of GDC 4, as it relates to accommodating the dynamic effects of postulated pipe ruptures, including the effects of pipe whipping and discharging fluids. The design is considered to comply with GDC 4 if it conforms to BTP 3-3, with regard to high and moderate energy fluid systems outside the containment.

The DCD defines the postulated pipe failure type based on whether the piping system is a high or moderate energy system. DCD Tier 2, Section 3.6.1.1 states that a system is considered to be high energy when the maximum operating temperature exceeds 95°C (200°F) and/or the maximum operating pressure exceeds 1900 kPa (275 psig) during normal conditions. Moderate-energy fluid systems are defined to be those systems or portion of systems that, during normal plant conditions are either in operation or maintained pressurized (above atmospheric pressure) under conditions where the maximum operating temperature is 95°C (200°F) or less and where the maximum operating pressure is 1900 kPa (275 psig) or less. The staff found this to be consistent with criteria provided in Appendix A to BTP 3-3.

DCD Tier 2, Appendix 3E, "High Energy and Moderate Energy Piping in the Prestressed Concrete Containment Vessel and Reactor Building," identifies high- and moderate-energy piping (greater than 2.2 cm (1 in.) diameter) within the containment vessel and the reactor building. DCD Tier 1, Table 2.3-1, "High and Moderate Energy Piping System Considered for Protection of Essential Systems," (and DCD Tier 2, Table 3.6-1, "High and Moderate Energy Fluid Systems") identifies all high- and moderate-energy piping systems within the US-APWR design. The staff was unable to confirm that all high- and moderate-energy piping systems were properly identified since the maximum normal operating pressures and temperatures were not specified. The staff also noted that some systems that typically are considered high or moderate energy system were not included in these lists. In **RAI 180-1594, Question 03.06.01-2**, the staff requested the applicant to update the DCD to include the maximum normal operating pressures and temperatures for all the fluid containing systems.

In its response to **RAI 180-1594, Question 03.06.01-2,** dated March 3, 2009, the applicant stated that the US-APWR design documents contain the maximum normal operating pressures and temperatures for all fluid containing systems, which were used to identify high- and moderate-energy piping systems, described in DCD Tier 1, Table 2.3-1 and DCD Tier 2, Table 3.6-1. The applicant further stated that the maximum normal operating pressures and temperatures for all fluid containing systems is a level of detail not typically included in the DCD. The response also confirms that the applicant used the criteria and assumptions presented in BTP 3-3 to identify high- and moderate-energy piping systems. The staff reviewed the applicant's response and determined that it is not typical to include in the DCD the maximum normal operating pressures and temperatures for all fluid containing systems. In addition, the staff determined that the applicant used an acceptable method; i.e., the criteria and assumptions presented in BTP 3-3, to identify all high- and moderate-energy piping systems. Therefore, the staff finds the RAI response to be acceptable. Accordingly, **RAI 180-1594, Question 03.06.01-2, is resolved**.

In DCD Tier 2, Appendix 3D, "US-APWR Equipment Qualification List Safety and Important to Safety Electrical and Mechanical Equipment," the applicant identifies the systems and components important to plant safety or shutdown. However, the applicant did not identify which of the safety systems are located near to high- or moderate-energy piping systems. The applicant also did not provide the layout of the site piping systems (the drawing should present the location of all the safety-related or important to safety SSCs, the pipe layout, and the barriers), in order to allow the staff to verify that all the essential SSCs that need to be protected have been identified. In **RAI 180-1594, Question 03.06.01-3**, the staff requested the applicant to provide detailed layout drawings of the site piping systems (the drawing should present the location of all the safety-related/important to safety SSCs, the pipe layout, and the barriers).

In its response to **RAI 180-1594, Question 03.06.01-3**, dated March 3, 2009, the applicant stated that the detailed layout drawings of the site piping systems, which include the location of all the safety-related or important to safety SSCs, the pipe layout, and the barriers, are available to the NRC staff to audit. The staff finds that since all the drawings exist and are available for staff review, that the applicant has addressed the staff concerns described in **RAI 180-1594, Question 03.06.01-3**. Accordingly, **RAI 180-1594, Question 03.06.01-3,** is resolved.

DCD Tier 2, Section 3.6.1 states that piping systems meeting the leak-before-break (LBB) criteria are not subject to postulated pipe breaks. High-energy fluid system piping that meets the LBB criteria will be evaluated for the effects of leakage cracks. A leakage crack is the crack size that results in leakages that will be determined in the LBB analysis. The applicant's evaluation of LBB is provided in DCD Tier 2, Section 3.6.3, "LBB Evaluation Procedures." The staff's evaluation of LBB is discussed in Section 3.6.3 of this report. High-energy fluid system piping that does not meet the LBB criteria should be evaluated for the dynamic effects of postulated pipe failures which include pipe whip and jet impingement. The staff finds that these design criteria are in accordance with the recommendations of SRP Section 3.6.1 and therefore are acceptable.

In DCD Tier 2, Section 3.6.1, the applicant stated that pipe failure evaluations will be based on circumferential or longitudinal pipe breaks, through-wall cracks, or leakage cracks. Through-wall cracks will be postulated in both high- and moderate-energy piping, and will be assumed to be a circular opening with an area equal to that of a rectangle one-half pipe diameter in length and one-half pipe wall thickness in width, as specified in BTP 3-3. Subcompartment pressurization, jet impingement, jet reaction thrust, internal fluid decompression loads, spray wetting, flooding, and pipe whip will be considered for postulated pipe breaks. Through-wall cracks will not be postulated in the break exclusion zone. Pressurization, spray wetting, and flooding effects for pipe failures in the break exclusion zone for high-energy piping (including MS and MFW piping) near containment penetrations will assume a 0.093 m^2 (1 ft^2) break. Postulated break, through-wall crack, and leakage crack locations will be determined in accordance with DCD Tier 2, Sections 3.6.2, "Determination of Rupture Locations and Dynamic Effects Associated with the Postulated Rupture of Piping," and 3.6.3. The staff's evaluation of pressurization loading on structures is described in Section 3.8 of this report. The staff finds that these criteria are all consistent with BTP 3-3 and therefore are acceptable.

DCD Tier 2, Subsection 3.6.1.2.3.1, "Subcompartment Pressurization," provides the basis of the calculation of subcompartment pressurization levels. All areas with high energy lines will be evaluated. The RV annulus will be evaluated for asymmetric compartment pressurization. Safety systems are to be designed to withstand the effects of pressurization. The staff finds that these criteria are consistent with BTP 3-3 and therefore are acceptable.

The applicant identified that there are no high-energy fluid lines near the MCR. The high-energy fluid lines are separated from the MCR by two floors composed of thick, reinforced concrete. This assures that the habitability of the MCR will be assured in case of a high-energy fluid line break. The staff finds that this design criterion is in accordance with the guidance of BTP 3-3 and therefore is acceptable.

DCD Tier 2, Section 3.6.1 discusses the design-basis assumptions used in the dynamic effects analysis for pipe failures. These assumptions include the following:

- Offsite power is assumed to be unavailable if a trip of the T/G system or reactor protection system is a direct consequence of the postulated piping failure, or during and following seismic events.

- A single active component failure occurs in systems needed to mitigate the consequences of the piping failure or to safely shut down the reactor. The single active component failure occurs in addition to the pipe failure (including any direct consequences of the pipe failure, such as a unit trip or loss of offsite power (LOOP)).

- All available systems, including those actuated by an operator, are credited with mitigating the consequences of a postulated piping failure (given a single active component failure). For breaks in non-seismic piping systems only seismically qualified systems are assumed available.

- If the direction of the initial pipe movement caused by the thrust force is such that the pipe impacts a flat surface normal to its direction of travel, it is assumed that the pipe comes to rest against the surface with no pipe whip in other directions. Pipe whip restraints are used wherever pipe breaks could impair the functioning of essential SSCs.

- Regarding components impacted by jets from breaks in high-pressure steam piping (greater than 5,998 kPa (870 psia)), or subcooled water that would flash, components within 10 diameters of the broken pipe are assumed to fail, unless the components are required for safe shutdown and accident mitigation capability in which case the jet loads are computed and evaluated based on the criteria given in DCD Tier 2, Section 3.6.2.

The staff evaluated these assumptions and finds that they are consistent with the recommendations of BTP 3-3 and SRP 3.6.1. Therefore, the staff finds them acceptable.

In DCD Tier 2, Subsection 3.6.1.2.2, “Basic Protection Measures,” the applicant defines the three basic protection measures used in the protection of essential SSCs from postulated pipe failures. Separation is the preferred protection strategy. Redundant systems are located in separate compartments. For pipe whip, adequate separation is based on the distance between the equipment and the pipe, as well as the length of the whipping pipe. For jet impingement, equipment located more than 10 pipe diameters from the source of the jet is considered to be adequately protected from the jet.

Barriers and shields will be used when separation is not practicable. Floors and walls provide barriers; additional barriers will be installed where needed. Piping restraints will be added as a last resort. The priority of these choices is consistent with the guidance provided in BTP 3-3.

In DCD Tier 2, Section 3.6.4, “Combined License Information,” COL Information Item 3.6(1) states that the COL applicant is to identify the site-specific systems or components that are safety-related or required for safe shutdown that are located near high-energy or moderate-energy piping systems, and are susceptible to the consequences of these piping failures. The

COL applicant is to provide a list of site-specific high-energy and moderate energy piping systems, which includes a description of the layout of all piping systems where physical arrangement of the piping systems provides the required protection, the design-basis of structures and compartments used to protect nearby essential systems or components, or the arrangements to assure the operability of safety-related features where neither separation nor protective enclosures are practical. Additionally, the COL applicant is to provide the failure modes and effect analyses that verifies the consequences of failures in site-specific high-energy and moderate-energy piping does not affect the ability to safely shut down the plant.

In DCD Revision 3 and in its responses to **RAI 71-986, Question 03.06.02-18**, dated October 7, 2008, **RAI 459-3331, Question 03.06.02-39**, dated December 1, 2009, and **RAI 636-4732, Question 03.06.02-48** dated November 24, 2010, and the revised response dated December 15, 2010; the applicant added the new Section 3.6.2.6, "Outline of Pipe Break Hazard Analysis Report(s)," to DCD Tier 2. The staff identified that DCD Tier 1, Table 2.3-2, Items 4 and 5 include ITAAC that requires the COL applicant to perform a pipe hazards analysis that demonstrate that all essential SSCs are protected from pipe failures from the piping systems identified in DCD Tier 1, Table 2.3-1. Since the completion of the ITAAC and the COL information item are the responsibility of the COL applicant, it is expected that these reports will contain similar information. In **RAI 795-5884, Question 03.06.01-9**, the staff requested the applicant to update COL Information Item 3.6(1) to instruct the COL applicant to modify DCD Tier 1, Table 2.3-1 to include all site-specific high and moderate piping systems, or to update the as-design pipe hazards analysis report to include the impact of all site-specific high and moderate piping systems.

In its response to **RAI 795-5884, Question 03.06.01-9,** dated October 26, 2011, the applicant stated that the COL Information Item COL 3.6(1) will be revised in the next DCD revision to indicate that the COL applicant must update the as-designed pipe hazards analysis report to include the impact of all site-specific high and moderate piping systems. The response also proposed conforming changes to DCD Tier 2, Table 1.8-2, “Compilation of All Combined License Applicant Items for Chapters 1-19,” Section 3.6.1, and Section 3.6.4.

The staff evaluated the applicant’s response and the proposed DCD changes and determined that the changes would reduce redundancy between the ITAAC and the COL information item, and minimize confusion. Therefore, the staff finds the RAI response and the proposed changes to be acceptable. Since the applicant has proposed DCD changes, **RAI 795-5884, Question 03.06.01-9 is being tracked as a Confirmatory Item.**

Based on the above, pending resolution of **RAI 795-5884, Question 03.06.01-9, which is being tracked as a Confirmatory Item,** the staff concludes that the applicant has provided sufficient design information to ensure that the guidelines of BTP 3-3 and SRP Section 3.6.1 will be met. Specifically, the US-APWR design provides protection of essential SSCs from the effects of high- and moderate-energy pipe failures by separation of redundant divisions of these systems, by separating (distance) essential SSCs from piping failures, or by locating essential SSCs such that necessary protection is provided. Furthermore, piping failures are postulated in accordance with BTP 3-3, pipe restraints will be provided in accordance with the guidelines in SRP Section 3.6.2 (addressed in Section 3.6.2 of this report), and the effects of the postulated piping failures, including those from non-seismic systems, are considered. Based on this information, the staff concludes that the guidelines of both BTP 3-3 and SRP Section 3.6.1 will be met. Therefore, the staff finds that the requirements of GDC 4 are met.

3.6.1.4.3 Inspections, Tests, Analyses, and Acceptance Criteria

A number of ITAAC items involve piping systems within the US-APWR design. DCD Tier 1, Revision 1 only contained one ITAAC item that directly involved SRP Section 3.6.1.

The staff reviewed DCD Revision 1, Tier 1 Section 2.2, “Structural and Systems Engineering,” and Section 2.3, “Piping Systems and Components,” and generated three RAIs related to these ITAAC (**RAI 180-1594, Questions 03.06.01-4, 03.06.01-5, and 03.06.01-6**). In DCD Revision 3 and in response to **RAI 71-986, Question 03.06.02-18; RAI 459-3331, Question 03.06.02-39; and RAI 636-4732, Question 03.06.02-48**; the applicant revised DCD Tier 1, Sections 2.2 and 2.3 and added the DCD Tier 2, Section 3.6.2.6. These changes superseded the previously issued RAI responses. Accordingly, **RAI 180-1594, Questions 03.06.01-4, 03.06.01-5, and 03.06.01-6 are considered resolved**.

DCD Tier 1 Table 2.3-2, ITAAC Item 4 requires the completion of the as-design pipe break hazards analysis report, and ITAAC Item 5, requires the reconciliation of the pipe break hazards analysis report with the actual as-built site. DCD Tier 2, Section 3.6.2.6 defines the content of the pipe break hazards analysis report. The staff evaluated this section and determined that SRP 3.6.2 allows for a break exclusion zone, where no pipe breaks needs to be postulated. However, SRP 3.6.1 states that the effects of flooding, spray wetting, and subcompartment pressurization should be evaluated for a postulated 0.093 m^2 (1 ft^2) break for the MS and feedwater lines within the break exclusion zone, at a location that has the greatest effect on essential equipment. The staff determined that DCD Tier 2, Section 3.6.2.6 was not clear enough to ensure that the pipe break hazards analysis report will include the evaluation of the consequences of this break within the break exclusion zone. In **RAI 795-5884, Question 03.06.01-7,** the staff requested the applicant to update DCD Tier 2, Section 3.6.2.6 (fifth bullet or the section notes) to explicitly state that the pipe break hazards analysis will evaluate the consequences of a postulated 0.093 m^2 (1 ft^2) break for the MS and feedwater lines within the break exclusion zone.

In its response to **RAI 795-5884, Question 03.06.01-7,** dated October 26, 2011, the applicant stated that the DCD Tier 2 Section 3.6.2.6 (fifth bullet) will be modified to explicitly state that pipe break hazards analysis will evaluate the consequences of a postulated 0.093 m^2 (1 ft^2) break for the MS and feedwater lines within the break exclusion zone. The staff evaluated the applicant’s response and concluded that the proposed DCD modification will ensure that the pipe break hazards analysis will evaluate the consequences of a postulated 0.093 m^2 (1 ft^2) break for the MS and feedwater lines within the break exclusion zone, as recommended by SRP 3.6.1 and BTP 3-3. Therefore, the staff finds the RAI response acceptable. Since the applicant has proposed DCD changes, **RAI 795-5884, Question 03.06.01-7 is being tracked as a Confirmatory Item.**

DCD Tier 1, Table 2.3-2, ITAAC Items 4 and 5 make reference to DCD Tier 1, Table 2.3-1, which contains a list of the high and moderate energy piping system. DCD Tier 2 Section 9.2.5, "Ultimate Heat Sink" describes a water system that the staff considers should be included in the list. In **RAI 795-5884, Question 03.06.01-8**, the staff requested the applicant to include the UHS in DCD Tier 1, Table 2.3-1 or to justify why this system was excluded from the table.

In its response to **RAI 795-5884, Question 03.06.01-8,** dated October 26, 2011, the applicant stated that the UHS is CDI and will not be certified as part of the DCD. The response also stated that the COL applicant has the responsibility to complete the design and the evaluation of CDI and site-specific systems.

The staff evaluated the applicant's response and confirmed that the UHS is a CDI system; therefore, it is not included in the certified design. The COL applicant is responsible for the completion of the design and safety analysis of the system. Additionally, the staff identified that COL Information Item 3.6(1) instructs the COL applicant to update the as-design pipe hazards analysis report to include the impact of all site-specific high and moderate piping systems, COL Information Item 3.6(4) instructs the COL applicant to implement the criteria for defining break and crack locations and configurations for site-specific high-energy and moderate-energy piping systems. These two COL Information Items ensure that all site-specific essential SSCs are protected from pipe failures and failures from site-specific pipe failures have been properly identified and evaluated. Based on this, the staff finds the RAI response acceptable. Accordingly, **RAI 795-5884, Question 03.06.01-8, is resolved**.

Based on the discussion above, pending resolution of **RAI 795-5884, Question 03.06.01-7, which is being tracked as a Confirmatory Item**, the staff concludes that the applicant has proposed adequate ITAAC in DCD Tier 1, Table 2.3-2, ITAAC Items 4 and 5 to confirm that the pipe break hazards analysis report will be completed as described in DCD Tier 2, Sections 3.6.1 and 3.6.2, and this report will confirm that all essential SSCs are protected from postulated pipe failures in fluid systems outside containment. The staff notes that the evaluation of these ITAAC with regard to protection from postulated pipe failure in fluid systems inside containment is discussed in Section 14.3.3 of this report.

3.6.1.4.4 Technical Specifications and Initial Plant Test Program

There are no TS or initial plant testing program requirements in the DCD related to the plant design for protection against postulated piping failures in fluid systems outside containment. Based on the staff review of DCD Tier 2, Section 3.6.1, the staff finds this acceptable.

3.6.1.5 Combined License Information

The following is a list of COL item numbers and descriptions from Table 1.8-2 of the DCD related to the plant design for protection against postulated piping failure in fluid systems inside and outside containment.

Table 3.6.1-1
U.S. APWR Combined License Information Items

Item No.	Description	Section
COL 3.6(1)	The COL Applicant is to identify the site-specific systems or components that are safety-related or required for safe shutdown that are located near high-energy or moderate-energy piping systems, and are susceptible to the consequences of these piping failures. The COL Applicant is to provide a list of site-specific high-energy and moderate-energy piping systems, which includes a description of the layout of all piping systems where physical arrangement of the piping systems provides the required protection, the design-basis of structures and compartments used to protect nearby essential systems or components, or the arrangements to assure the operability of safety-related features where neither separation nor protective enclosures are practical. Additionally, the COL Applicant is to provide the failure modes and effect analyses that verifies the consequences of failures in	3.6.1

	site-specific high-energy and moderate-energy piping does not affect the ability to safely shut down the plant. The COL Applicant is to update the as-design pipe hazards analysis report to include the impact of all site-specific high and moderate piping systems.	
COL 3.6(4)	The COL Applicant is to implement the criteria for defining break and crack locations and configurations for site-specific high-energy and moderate-energy piping systems. The COL Applicant is to identify the postulated rupture orientation of each postulated break location for site-specific high-energy and moderate-energy piping systems. The COL Applicant is to implement the appropriate methods to assure that as-built configuration of site-specific high-energy and moderate-energy piping systems is consistent with the design intent and provide as-built drawings showing component locations and support locations and types that confirms this consistency.	3.6.1

ITAAC items have been identified to complete the pipe break hazards analyses report. COL Information Item 3.6(1) instructs the COL applicant to update the as-design pipe hazards analysis report to include the impact of all site-specific high and moderate piping system. As discussed above in Section 3.6.1.4 of this report, the applicant modified COL Information Item 3.6(1) in response to **RAI 795-5884, Question 03.06.01-9, which is being tracked as a Confirmatory Item.** COL Information Item 3.6(4) instructs the COL applicant to implement the criteria for defining break and crack locations and configurations for site-specific high-energy and moderate-energy piping systems. These COL information items were evaluated above in Section 3.6.1.4 of this report and found acceptable.

The staff finds the above listing to be complete. Also, the list adequately describes actions necessary for the COL applicant. No additional COL information items were identified that need to be included.

3.6.1.6 Conclusions

Based on its review as described above, and pending satisfactory resolution of **RAI 795-5884, Question 03.06.01-9 and RAI 884-6223, Question 03.06.01-10, which are being tracked as Confirmatory Items**, the staff concludes that the US-APWR design, as it relates to the protection of safety-related SSCs from the effects of piping failures outside containment, meets the requirements of GDC 2 and GDC 4 with respect to accommodating the effects of postulated pipe failures. The staff also finds that, pending satisfactory resolution of **RAI 795-5884, Question 03.06.01-7, which is being tracked as a Confirmatory Item**, the US-APWR design has proposed acceptable ITAAC that meet the requirements of 10 CFR 52.47(b)(1). The staff determined that the applicant has not proposed TS, or initial test program considerations that are related to this area of review. The staff finds this acceptable. Therefore, the DCD is considered to be acceptable with respect to protection against piping failures outside of containment considerations.

3.6.2 Determination of Rupture Locations and Dynamic Effects Associated with the Postulated Rupture of Piping

3.6.2.1 Introduction

This section of the DCD addresses the criteria for defining break and crack location and methods of analysis for evaluating the dynamic effects associated with postulated breaks and cracks in high-energy and moderate-energy piping systems inside and outside containment. The criteria and methods of analysis are to ensure that the plant can be safely shut down or the consequences of a postulated pipe rupture can be mitigated.

3.6.2.2 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," Revision 3. Specifically, in DCD Tier 1, Section 2.3, Table 2.3-2, "Piping Systems and Components ITAAC," the applicant provided the piping break hazard analyses ITAAC pertaining to as-designed pipe break hazard analysis and the as-built pipe break hazards analysis reconciliation.

DCD Tier 2: The applicant has provided a DCD Tier 2 system description in Section 3.6.2, "Determination of Rupture Locations and Dynamic Effects Associated with the Postulated Rupture of Piping," Revision 3. This section provides the following information to address GDC 4:

- DCD Tier 2, Section 3.6.2.1, "Criteria used to Define Break and Crack Location and Configuration."
- DCD Tier 2, Section 3.6.2.2, "Guard Pipe Assembly Design Criteria."
- DCD Tier 2, Section 3.6.2.3, "Analytical Methods to Define Forcing Functions and Response Models."
- DCD Tier 2, Section 3.6.2.4, "Dynamic Analysis Methods to Verify Integrity and Operability."
- DCD Tier 2, Section 3.6.2.5, "Implementation of Criteria Dealing with Special Features."
- DCD Tier 2, Section 3.6.2.6, "Outline of Pipe Break Hazard Analysis Report(s)."

DCD Tier 2, Table 3.6-1, "High and Moderate Energy Fluid Systems," lists the safety-related high- and moderate-energy fluid systems for fluid systems both inside and outside the containment. DCD Tier 2, Table 3.6-2, "List of High Energy Lines for Pipe Break Hazards Analysis, Including Properties of Internal and External Fluids," provides the list of high-energy lines for pipe break hazard analysis, including properties of internal and external fluid conditions.

DCD Tier 2, Section 3.6.2.1, provides the criteria for the locations and configuration of the postulated breaks and cracks, except for piping that satisfies the requirements for LBB. DCD Tier 2, Section 3.6.2.2, discusses the design criteria for guard pipe assembly as part of the containment penetration design for high-energy piping. DCD Tier 2, Section 3.6.2.3 provides analytical methods to define forcing functions and pipe response models to determine the forcing functions and reaction forces that can dynamically excite the piping systems as a result of pipe breaks. DCD Tier 2, Section 3.6.2.4 discusses the dynamic analysis methods used to evaluate the effects of the postulated pipe breaks or cracks on the surrounding safety-related SSCs. The applicant's Technical Reports, MUAP-10017-P, "US-APWR Methodology of Pipe Break Hazard Analysis," Revision 3, issued May 2012; and MUAP-10022-P, "Evaluation on Jet Impingement Issues Associated with Postulated Pipe Rupture," Revision 2, issued May 2012, provide a description of the applicant's approaches for assessing the effects of all types of loads induced on surrounding safety-related SSCs from the postulated pipe failures. (Discussion of these reports will refer to these revisions unless otherwise noted.) These effects include: blast waves, static jet impingement loads, and dynamic jet impingement loads, including the effects of resonance within jets impinging upon nearby targets. DCD Tier 2, Section 3.6.2.5 refers to DCD Tier 2, Subsection 3.6.2.4.4, "Pipe Whip Restraints, Barriers and Shields," which describes the design criteria for special features, such as pipe whip restraints, barriers, shields, that will be used in the pipe rupture evaluation. Finally, DCD Tier 2, Section 3.6.2.6 provides an outline of methodology and evaluation for the pipe break hazard analysis report that will address high- and moderate-energy piping systems.

ITAAC: The ITAAC associated with DCD Tier 2, Section 3.6.2 are delineated in DCD Tier 1, Table 2.3-2, "Piping Systems and Components ITAAC," which provide the piping break hazard analyses ITAAC pertaining to as-designed pipe break hazard analysis and the as-built pipe break hazards analysis reconciliation.
TS for this area of review.

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

Technical Reports: Technical reports associated with DCD Tier 2, Section 3.6.2 are as follows:

1. MUAP-10017-P, "US-APWR Methodology of Pipe Break Hazard Analysis," Revision 3, issued May 2012.

2. MUAP-10022-P, "Evaluation on Jet Impingement Issues Associated with Postulated Pipe Rupture," Revision 2, issued May 2012.

Cross-cutting Requirements (TMI, USI/ GSI, Op Ex): There are no cross-cutting requirements for this area of review.

US-APWR Interface Issues Identified in the DCD: See DCD Tier 2, Table 1.8-2, "Compilation of All Combined License Applicant Items for Chapters 1-19."

Site Interface Issues Identified in the DCD: See DCD Tier 2, Table 1.8-1, "Significant Site Specific Interfaces with the Standard US-APWR Design."

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

3.6.2.3 Regulatory Basis

The relevant requirements of the Commission's regulations for this area of review, and the associated acceptance criteria are given in Section 3.6.2, "Determination of Rupture Locations and Dynamic Effects Associated with the Postulated Rupture of Piping," Revision 2, issued March 2007, of NUREG-0800 and are summarized below. Review interfaces with other SRP sections can be found in Section 3.6.2 of NUREG-0800.

1. GDC 4, as it relates to SSCs important to safety being designed to accommodate the dynamic effects associated with postulated pipe rupture.

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 DC has been constructed 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. BTP 3-3, "Protection Against Postulated Piping Failures in Fluid Systems Outside Containment," Revision 3, issued March 2007, which contains staff guidelines for protection against postulated piping failures in fluid systems outside the containment.

2. BTP 3-4, "Postulated Rupture Locations in Fluid System Piping Inside and Outside Containment," Revision 2, issued March 2007, which contains the staff guidelines for defining postulated rupture locations in fluid system piping inside and outside the containment.

3.6.2.4 Technical Evaluation

GDC 4 requires, in part, that SSCs important to safety be designed to be compatible with, and to accommodate, the effects of the environmental conditions resulting from postulated pipe rupture accidents, including LOCAs. GDC 4 also requires that they are adequately protected against dynamic effects (including the effects of pipe whipping and discharging fluids) that may result from postulated pipe rupture events.

In accordance with the staff guidelines as described in SRP Section 3.6.2 and the BTP 3-4, the staff reviewed the proposed criteria and methodology presented by the applicant in the DCD 3.6.2 and other associated sections. The staff also reviewed two technical reports described in Section 3.6.2.2 of this report.

3.6.2.4.1 Criteria Used to Define Pipe Break and Crack Locations and Configurations

In DCD Tier 2, Section 3.6.2.1, the applicant provides the criteria for defining the location and configuration of postulated pipe breaks and leakage cracks. DCD Tier 2, Subsection 3.6.2.1.1, "High-Energy Fluid Systems Piping," provides criteria for the postulated pipe break locations for high-energy piping in containment penetration areas (known as break exclusion area), and both pipe break and pipe leakage cracks in areas other than the containment penetrations, including non-ASME class piping. DCD Tier 2, Subsection 3.6.2.1.2, "Moderate-Energy Fluid System Piping Break Locations," provides criteria of postulated pipe crack locations for moderate-energy piping in both containment penetrations and other areas. Finally, DCD Tier 2, Subsection 3.6.2.1.3, "Types of Break/Cracks Postulated," provides criteria for defining the types of pipe breaks (i.e., break configuration) to be postulated in high-energy fluid system piping, and through-wall leakage crack configurations to be postulated in either high- and moderate-energy fluid systems or portions of systems. Specifically, DCD Tier 2, Subsection 3.6.2.1.1.1, "High-Energy Fluid System Piping in PCCV Penetration Area," provides the break exclusion criteria for high-energy piping in the PCCV penetration area. The applicant states that breaks and cracks are not postulated in those portions of piping from the PCCV penetration to an anchor or five-way restraint, the segment known as break exclusion area. The applicant also provides its design criteria that must be met for Class 2 piping in this break exclusion area.

Based on its review of the information included in DCD Tier 2, Section 3.6.2.1, the staff found that the applicant initially did not adequately address some of the staff guidelines included in SRP 3.6.2 and BTP 3-4 as described below.

BTP 3-4, Part B, Item A(i) for high-energy systems and Item B(i) for moderate-energy systems refers to BTP 3-3 Part B, Item 1.a for the separation provisions of plant arrangement. It further states that a review of the piping layout and plant arrangement drawings should clearly show that the effects of postulated piping breaks at any location in high-energy fluid systems, or through-wall leakage cracks at any location in piping designed to seismic and non-seismic standards, are isolated or physically remote from essential systems and components. At the designer's option, break locations in high-energy piping as determined from BTP 3-4, Part B, Item A(iii) of this regulatory position on postulation of pipe breaks in areas other than containment penetration exclusion areas may be assumed for this purpose.

The staff noted that DCD Tier 2, Section 3.6.2 does not clearly address the above staff guidelines for the separation provisions of plant arrangement. The staff also noted that DCD Tier 2, Subsection 3.6.1.2.2.1, "Separation," provides separation criteria as one of the basic protection measures for safety-related SSCs located in both inside and outside the containment. Therefore, in **RAI 71-986, Question 03.06.02-1**, the staff requested the applicant to clarify if the separation provisions of plant arrangement as described in DCD Tier 2, Subsection 3.6.1.2.2.1 are also applicable for the assessments of postulated high- and moderate- energy piping systems pipe ruptures inside and outside the containment for US-APWR standard plant design; otherwise, to describe the criteria that address the pertinent BTP 3-4 staff guidelines as described above.

In its response to **RAI 71-986, Question 03.06.02-1**, dated October 7, 2008, the applicant clarified that the design methodology of separation as included in DCD Tier 2, Subsections 3.6.1.2.2.1 and 3.6.1.2.3.2, "MCR Habitability," are applicable for postulating pipe break locations inside and outside the containment for both high- and moderate-energy fluid systems of US-APWR standard plant design. The applicant also stated that DCD Tier 2, Subsection 3.6.1.2.3.2 is consistent with the BTP 3-3, Part B, Item 1.a, which states that high-energy fluid system piping is not to be located close to the MCR. Furthermore, DCD Tier 2, Subsection 3.6.2.1.1.1 is consistent with the BTP 3-3, Part B, Item 1.a guideline for defining the environmental condition due to a postulated break of one square foot for the MS supply system and feedwater system in the MS pipe room.

Based on the above discussion, the staff concluded that DCD Tier 2, Subsections 3.6.1.2.2.1, 3.6.1.2.3.2, and 3.6.2.1.1.1 adequately address the separation criteria applicable to both high- and moderate-energy fluid systems inside and outside the containment for US-APWR standard plant design and are consistent with the BTP 3-4, Part B, Items A(i) and B(i), guidelines as clarified in the applicant's response. Therefore, **RAI 71-986, Question 03.06.02-1, is resolved** because it adequately addressed the staff's concerns regarding the BTP 3-4 guidelines.

BTP 3-4, Part B, Item A(ii) states that breaks and cracks need not be postulated in those portions of piping from the containment wall up to and including the inboard or outboard isolation valves (containment penetration area/break exclusion area), provided they meet the requirements of ASME Code, Section III, Subarticle NE-1120 with the additional seven design conditions described in BTP 3-4, Part B, Item A(ii). Based on its review of the information described in DCD Tier 2, Subsection 3.6.2.1.1.1, the staff determined that there were discrepancies between the criteria stated in the DCD Tier 2, Subsection 3.6.2.1.1.1 and the staff guidelines in BTP 3-4, Part B, Item A(ii) for the break exclusion area. Therefore, in **RAI 71-986,**

Question 03.06.02-2 and two follow-up RAIs, **RAI 459-3331, Questions 03.06.02-20 and 03.06.02-21**, the staff requested the applicant to clarify the US-APWR design criteria for those portions of piping in the break exclusion area.

The applicant provided its response to **RAI 71-986, Question 03.06.02-2,** by a letter dated October 7, 2008, and later amended that response by a letter dated December 27, 2011. The applicant also provided its response to **RAI 459-3331, Questions 03.06.02-20 and 03.06.02-21** by a letter dated October 19, 2009**.** The applicant's responses to address the above concerns are described below.

In its October 7, 2008, and October 19, 2009, responses to **RAI 71-986, Question 03.06.02-2** and **RAI 459-3331, Questions 03.06.02-20 and 03.06.02-21**, the applicant redefined the break exclusion area for the US-APWR design. Specifically, the applicant stated that with the exception of the MS supply system and feedwater system piping in the MS piping room, breaks and cracks need not be postulated in the portions of piping from the containment wall up to and including the inboard and outboard isolation valves. In addition, the applicant incorporated its responses into DCD Tier 2 Section 3.6.2.1.1.1 and provided the following design criteria for the piping systems within the break exclusion area:

1. PCCV penetrations meet the design criteria of the ASME Code, Section III, Subarticle NE-1120.

2. The maximum stress ranges as calculated by the sum of Equations 9 and 10 in Paragraph NC-3653 of ASME Code, Section III, considering those loads and conditions thereof, for which Level A and Level B stress limits have been specified in the system's design specification, does not exceed $0.8(1.8S_h + S_a)$.

3. The maximum stress, as calculated by ASME Code, Section III, NC-3653, Equation 9 under the loadings resulting from a postulated pipe failure beyond this portion of piping does not exceed the smaller of 2.25 S_h or 1.8 S_y, except that following a pipe failure outside containment, the pipe between the outboard isolation valve and the first restraint may be permitted higher stresses provided a plastic hinge is not formed and the operability of the valves with such stresses is ensured in accordance with the pertinent guidance included in SRP Section 3.9.3. When the piping between the outboard isolation valve and the restraint is constructed in accordance with the Power Piping Code ANSI B31.1, the piping should either be of seamless construction with full radiography of all circumferential welds or all longitudinal and circumferential welds should be fully radiographed.

4. The number of circumferential and longitudinal piping welds and branch connections are minimized.

5. Welded attachments, for pipe supports or other purposes, to this portion of piping are avoided. Where welded attachments are necessary, the welds are 100 percent volumetrically examinable and detailed stress analyses are performed to demonstrate compliance with the limits of DCD Tier 2, Subsection 3.6.2.1.1.2.

6. 100 percent volumetric examination of all pipe welds is performed in accordance with IWA-2400 of ASME Code, Section XI.

7. Anchors or five-way restraints do not prevent the access required to conduct the ASME Code, Section XI inservice examination for the circumferential and the longitudinal welds within the boundary of this portion of piping.

8. The length of these portions of piping is to be the minimum length practical.

In its amended response dated December 27, 2011, for **RAI 71-986, Question 03.06.02-2**, the applicant also stated that no breaks are postulated from the PCCV penetration outboard weld to the wall of MS pipe room for the MS supply system and feedwater system piping. The applicant included in the RAI response a revised DCD Tier 2, Figure 3.6-1, "Break Exclusion Region-Main Steam Pipe Room." In addition, the applicant indicated that a five-way restraint (free only in axial direction) is installed at the nearby outboard isolation valve in the MS pipe room wall penetration as shown in Appendix A of the RAI response. The five-way restraint is designed to protect against any postulated pipe break between the isolation valve and the five-way restraint. Therefore, the break exclusion area is extended to include this portion of piping. Furthermore, the applicant stated that in addition to meeting the eight design criteria described above, the pipe is routed straight to lower the stresses and the length between the outboard isolation valve and the MS pipe room wall is to be reduced to the minimum length practical. Moreover, the applicant stated that in accordance with the staff guideline included in BTP 3-3, Part B, Item 1(a)(1), essential equipment is protected from the environmental effects of an assumed non-mechanistic longitudinal break for this portion of the MS supply system and feedwater system piping. Each assumed non-mechanistic longitudinal break has a cross sectional area of one square foot and is to be postulated at a location that has the greatest effect on essential equipment. Moreover, the applicant clarified that breaks are postulated in the branch pipes connected to these portions of MS supply system and feedwater system piping in accordance with the criteria for postulating break as described in DCD Tier 2, Subsection 3.6.2.1.1.2, "Postulation of Pipe Breaks in Areas Other than PCCV Penetrations." Furthermore, the applicant stated that DCD Tier 2, Subsection 3.6.2.1.1.1 and DCD Tier 2, Figure 3.6-1, are to be revised accordingly. In addition, the applicant stated that it will correct an US-APWR section numbering error. Specifically, it will replace "Subsection 3.6.2.4.5" in the first paragraph of DCD Tier 2, Subsection 3.6.2.4.2.2, "Piping Other Than RCL Piping," with "Subsection 3.6.2.4.4.3."

The staff reviewed the applicant's design criteria for the break exclusion region for high-energy fluid system piping in PCCV penetration area. The staff determined that the applicant's design criteria as described in its responses are consistent with the pertinent staff guidelines described in BTP 3-4, Part B, Item A(ii) for the fluid system piping in containment penetration areas and are acceptable. Therefore, **RAI 459-3331, Questions 03.06.02-20 and 03.06.02-21 are resolved. RAI 71-986, Question 03.06.02-2 is being tracked as a Confirmatory Item** until the applicant updates the DCD as described above**.**

In DCD Tier 2, Subsection 3.6.2.1.1.2, for high-energy systems in areas other than PCCV penetrations (i.e., outside the break exclusion area), the applicant provides criteria for pipe breaks in Class 1, 2, and 3, and non-ASME Class piping. However, the applicant did not address the staff guideline described in BTP 3-4, Part B, Item A(iii)(4) concerning the design of the separating structure between a high-energy line and an essential component. Also, the staff guideline in BTP 3-4, Part B, Item A(iii)(5) states that safety-related equipment must be environmentally qualified in accordance with SRP Section 3.11. Appropriate pipe breaks and leakage cracks must be considered in the design bases for defining the qualifying environment for these components both inside and outside the containment. Therefore, in **RAI 71-986, Question 03.06.02-3**, the staff requested the applicant to explain why these two specific guidelines were not addressed in the DCD.

In its response to **RAI 71-986, Question 03.06.02-3**, dated October 7, 2008, the applicant clarified that the staff guideline of BTP 3-4, Part B, Item A(iii)(4) associated with separating structures will be added to DCD Tier 2, Subsection 3.6.2.1.1.2, and the staff guideline of BTP 3-4, Part B, Item A(iii)(5) concerning qualifying electrical and mechanical equipment for the environmental effects of postulated pipe failure in accordance with SRP Section 3.11 are addressed in the DCD Tier 2, Section 3.6.1, “Plant Design for Protection against Postulated Piping Failure in Fluid Systems Inside and Outside Containment.” The staff verified that Revision 3 of DCD Tier 2, Subsections 3.6.2.1.1.2 and 3.6.1 adequately address the staff’s concerns stated in the original RAI and are consistent with the staff guidelines included in BTP 3-4, Part B, Items A(iii)(4) and A(iii)(5). Therefore, the staff found this acceptable and **RAI 71-986, Question 03.06.02-3, is resolved.**

The staff guideline included in BTP 3-4, Part B, Item A(iv) states that in complex systems such as those containing arrangements of headers and parallel piping running between headers, the designer should identify and include all such piping within the designated run in order to postulate the number of breaks required by the criteria in BTP 3-4, Part B, Item A(iii). However, the DCD did not provide information related to postulating pipe breaks in complex systems. Therefore, in **RAI 71-986, Question 03.06.02-4** and its follow-up **RAI 459-3331, Question 03.06.02-22**, the staff requested the applicant to clarify whether this staff guideline for postulating pipe break in complex systems is applicable to the US-APWR plant design.

The applicant provided its response to **RAI 71-986, Question 03.06.02-4** by a letter dated October 7, 2008 and its response to **RAI 459-3331, Question 03.06.02-22** by a letter dated October 19, 2009.

In these responses, the applicant proposed to add to DCD Tier 2, Subsection 3.6.2.1.1, a paragraph consistent with the staff guideline for postulating pipe break locations in complex systems such as those containing arrangements of headers and parallel piping running between headers, as included in BTP 3-4, Part B, Item A(iv). The staff verified that these changes are appropriately incorporated in DCD Revision 3 as stated in the MHI’s response and are acceptable. Since a follow-up RAI was necessary, **RAI 71-986, Question 03.06.02-4 is closed and unresolved.** Its follow-up **RAI 459-3331, Question 03.06.02-22, is resolved.**

DCD Tier 2, Subsection 3.6.2.1.2, provides the criteria for defining break and crack location in moderate-energy fluid systems piping. DCD Tier 2, Subsection 3.6.2.1.2.1, “Moderate-Energy Fluid System Piping in PCCV Penetration Areas,” provides leakage crack postulation for moderate-energy fluid system piping in the PCCV penetration area. In DCD Tier 2, Subsection 3.6.2.1.2.2, “Moderate-Energy Fluid System Piping in Areas Other than PCCV Penetrations,” for moderate-energy fluid system piping outside the break exclusion area, the applicant states that leakage cracks are postulated in the piping systems located adjacent to SSCs required for safe shutdown. However, the staff guideline included in BTP 3-4, Part B, Item B(iii)(1) states that leakage cracks should be postulated in piping located adjacent to SSCs important to safety. The applicant’s criterion is applicable to the SSCs required for safe shutdown and may not include those SSCs required for accident mitigation and other functions that are important to safety. Therefore, in **RAI 71-986, Question 03.06.02-5**, the staff requested the applicant to clarify this difference between SSCs important to safety and SSCs required for safe shutdown for the US-APWR plant design.

In its response to **RAI 71-986, Question 03.06.02-5**, dated October 7, 2008, the applicant stated that the terminology “SSCs required for safe shutdown” is synonymous with “SSCs important to safety." However, for consistency, the applicant proposed to revise DCD Tier 2, Subsection 3.6.2.1.2.2 from "SSCs required to safe shutdown" to "SSCs important to safety" for postulating leakage cracks in moderate-energy fluid system piping. The staff verified the changes were incorporated into Revision 3 of the DCD Tier 2, Subsection 3.6.2.1.2.2 and finds this acceptable. Therefore, **RAI 71-986, Question 03.06.02-5, is resolved.**

Based on its review of the applicant's criteria for moderate-energy fluid system pipe break postulations as described in DCD Tier 2, Subsection 3.6.2.1.2.2, the staff found that the applicant did not address several staff guidelines included in BTP 3-4, Part B, Item B. Therefore, in **RAI 71-986, Question 03.06.02-6,** the staff requested the applicant to provide technical justification for not addressing these staff guidelines.

By a letter dated October 7, 2008, the applicant provided its response to **RAI 71-986, Question 03.06.02-6.** However, the staff found that the applicant did not adequately address the staff's concerns regarding BTP 3-4, Part B, Item B guidelines in the response. Therefore, the staff closed as unresolved **RAI 71-986, Question 03.06.02-6** and issued three follow-up RAIs **(RAI 459-3331, Question 03.06.02-23; RAI 459-3331, Question 03.06.02-24; and RAI 636-4732, Question 03.06.02-47)** to request the applicant to provide further technical justification for not adequately addressing BTP 3-4, Part B, Item B guidelines. The applicant provided its responses to **RAI 459-3331, Question 03.06.02-23** and **RAI 459-3331, Question 03.06.02-24** by a letter dated October 19, 2009. The applicant also provided its response to **RAI 636-4732, Question 03.06.02-47** by a letter dated November 22, 2011. The applicant's responses to the staff's concerns are discussed below.

In its response to **RAI 459-3331, Question 03.06.02-23**, dated, the applicant stated that DCD Tier 2, Subsection 3.6.2.1.2 did not address the staff guideline included in BTP 3-4, Part B, Item B(iv) for moderate-energy piping in proximity to high-energy piping. The applicant proposed to revise DCD Tier 2, Subsection 3.6.2.1.2. Specifically, the applicant stated that leakage cracks are not postulated in moderate-energy fluid system piping where a break in the high-energy fluid system is postulated, provided that such a crack does not result in environmental conditions more severe than the high-energy break. The applicant further stated that if the effects of breaks of moderate-energy fluid system piping is more severe than those of high-energy fluid system piping, then the provisions of DCD Tier 2 Subsection 3.6.2.1.2.2 for postulating leakage cracks is applied. The staff reviewed the applicant's response and found it acceptable because the applicant's proposed change is consistent with the pertinent staff guideline in BTP 3-4 Part B Items B(iii) and B(iv) for postulating leakage cracks of moderate-energy piping in proximity to high-energy piping. With the proposed change, BTP 3-4 Part B Items B(iii) and B(iv) are addressed by DCD Tier 2 Subsections 3.6.2.1.2.2 and 3.6.2.1.2, respectively. In addition, the staff verified that Revision 3 of the DCD Tier 2, Subsection 3.6.2.1.2 is appropriately revised. Therefore, **RAI 459-3331, Question 03.06.02-23, is resolved.**

In its response to **RAI 71-986, Question 03.06.02-6**, the applicant clarified that its criterion related to through-wall leakage cracks in moderate-energy fluid system piping is consistent with the staff guideline as included in BTP 3-4, Part B, Item B(v). That staff guideline is to qualify a piping system as a moderate-energy piping when that piping system only operated as a high energy system for a short operational period. The operational period is considered "short" if the fraction of time that piping system operates as a high-energy fluid system is about two percent of the time that piping system operates as a moderate-energy fluid system. The staff determined that the applicant's response is acceptable because its criterion is consistent with the staff guideline included in BTP 3-4, Part B, Item B(v) guideline. However, the staff determined that the staff guideline should be added to the DCD.

In **RAI 459-3331, Question 03.06.02-24**, the staff requested the applicant to add this criterion as described in its RAI response to DCD Tier 2, Subsection 3.6.2.1.2. The applicant's response to **RAI 459-3331, Question 03.06.02-24,** agreed with the staff to revise the DCD accordingly. The staff verified that the applicant has appropriately incorporated this criterion into Revision 3 of DCD Tier 2, Subsection 3.6.2.1.2 and found it acceptable. Therefore, **RAI 459-3331, Question 03.06.02-24 is resolved.**

In **RAI 636-4732, Question 03.06.02-47**, the staff identified that DCD Tier 2, Subsection 3.6.2.1.2.2 did not adequately address the staff's guidelines included in BTP 3-4, Part B, Item B(iii)(1)(b) related to postulating leakage cracks for moderate-energy piping in areas other than PCCV penetration. The applicant's response proposed to revise DCD Tier 2, Subsection 3.6.2.1.2.2. Specifically, the first bullet of the first paragraph of DCD Tier 2, Subsection 3.6.2.1.2.2 is to state that for the ASME Code, Section III, Class 1 piping, leakage cracks are postulated where the stress range calculated by Equation 10 in NB-3653 is more than or equal to 1.2 S(m). The staff reviewed the applicant's response and found it acceptable because the applicant's proposed change is consistent with the staff guideline described in BTP 3-4 for postulating leakage cracks for moderate-energy piping in areas other than PCCV penetration. Therefore, **RAI 636-4732, Question 03.06.02-47 is being tracked as a Confirmatory Item** pending the applicant updating the DCD accordingly.

In DCD Tier 2, Subsection 3.6.2.1.3, the applicant provides the criteria for defining break and crack types in both high- and moderate-energy fluid systems piping. Specifically, DCD Tier 2, Subsections 3.6.2.1.3.1, "Circumferential Pipe Breaks," and 3.6.2.1.3.2, "Longitudinal Pipe Breaks," provide criteria for postulating circumferential pipe breaks and longitudinal pipe breaks respectively. In addition, DCD Tier 2, Subsection 3.6.2.1.3.3, "Leakage Cracks," provides criteria for postulating leakage cracks. Based on its review of the DCD information, the staff determined that the applicant did not address some of staff guidelines included in BTP 3-4, Part B, Items C(i), C(ii), and C(iii). **Therefore, in RAI 71-986, Question 03.06.02-7 and its follow-up RAI 459-3331, Question 03.06.02-25**, the staff requested the applicant to address the remaining items from BTP 3-4, Part B, Items C(i), C(ii), and C(iii).

DCD Tier 2, Subsection 3.6.2.1.3.1 did not address the BTP 3-4, Part B, Item C(i)(2) guideline for postulating break locations without the benefit of stress calculations. In its response to **RAI 71-986, Question 03.06.02-7,** dated October 7, 2008, the applicant clarified that the break is postulated at the piping welds and the description of the fourth paragraph of DCD Tier 2, Subsection 3.6.2.1.3.1 will be modified as described in BTP 3-4, Part B, Item C(i)(2). The staff verified that the changes in Revision 3 of the DCD are consistent with the BTP 3-4 guideline and, therefore, found the response acceptable.

DCD Tier 2, Subsection 3.6.2.1.3.1 did not address the BTP 3-4, Part B, Item C(i)(3) guideline related to pipe severance and separation of a ruptured piping section resulting from a postulated circumferential break. Specifically, the staff requested the applicant to clarify how the pipe stiffness will be used. In its response to **RAI 71-986, Question 03.06.02-7**, the applicant also stated that piping stiffness is used only when a plastic hinge is not developed in the piping. This applicant's design provision is consistent with the BTP 3-4 guideline. However, the applicant did not propose to add this design provision to the DCD Tier 2, Subsection 3.6.2.1.3.1. Therefore the staff closed as unresolved **RAI 71-986, Question 03.06.02-7**, and in follow-up **RAI 459-3331, Question 03.06.02-25**, the staff requested the applicant to add this design provision to the DCD Tier 2, Subsection 3.6.2.1.3.1. In its response to **RAI 459-3331, Question 03.06.02-25**, dated October 19, 2009, the applicant agreed to add this design provision into

DCD Tier 2, Subsection 3.6.2.1.3.1. The staff verified this modification in Revision 3 of the DCD and, therefore, found the response acceptable.

DCD Tier 2, Subsection 3.6.2.1.3 did not clearly address the BTP 3-4, Part B, Item C(i)(4) guideline associated with the determination of the effective cross-sectional flow area of the pipe in the jet discharge evaluation. In its response to **RAI 71-986, Question 03.06.02-7**, the applicant also stated that as stated in DCD Tier 2, Subsection 3.6.2.1.3.2 for longitudinal pipe breaks, the line restrictions, flow limiters, positive pump-controlled flow, and the absence of energy reservoirs may be taken into account, as applicable, in the reduction of jet discharge. The applicant further stated that DCD Tier 2, Subsection 3.6.2.1.3.1 will be revised to include these same design provisions for circumferential pipe break in the jet evaluation. The staff verified that the changes in Revision 3 of DCD Tier 2, Subsection 3.6.2.1.3.1 are consistent with the BTP 3-4, Part B, item C(i)(4) guideline and, therefore, found the response acceptable.

DCD Tier 2, Subsection 3.6.2.1.3.2 did not address the staff guideline included in BTP 3-4, Part B, Item C(ii)(2) related to not postulating longitudinal breaks at terminal ends. In its response to **RAI 71-986, Question 03.06.02-7**, the applicant also stated that DCD Tier 2, Subsection 3.6.2.1.3.2 will be revised to state that the longitudinal breaks are not postulated at terminal ends. The staff verified that the changes in Revision 3 of DCD Tier 2, Subsection 3.6.2.1.3.2 are consistent with the BTP 3-4, Part B, Item C(ii)(2) guideline and, therefore, found the response acceptable.

Based on the staff's evaluation as described above, the staff determined that the applicant has adequately addressed all of the staff's concerns included in **RAI 71-986, Question 03.06.02-7** and the follow-up **RAI 459-3331, Question 03.06.02-25**. Since a follow-up RAI was necessary, **RAI 71-986, Question 03.06.02-7 is closed and unresolved.** Its follow-up **RAI 459-3331, Question 03.06.02-25, is resolved.**

3.6.2.4.2 Guard Pipe Assembly Design

DCD Tier 2, Section 3.6.2.2 describes the design for PCCV piping penetrations. The applicant states that piping penetrations are an integral part of the PCCV pressure boundary. The annular space of the PCCV consists of multiple compartments that segregate the PCCV electrical and mechanical penetrations into their own isolated compartments (specifically, electrical penetration rooms and mechanical penetration rooms). The applicant also states that these compartments are designed to address postulated piping failures and their effects; as such, guard pipe assemblies are not required for the US-APWR plant design.

The staff noted that NRC RG 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants (LWR Edition)," Revision 3, issued November 1978, Section 3.6.2.4 defines that a guard pipe is a device to limit pressurization of the space between dual barriers of certain containments to acceptable levels. (Note that an equivalent definition for guard pipe is included in RG 1.206, "Combined License Applications for Nuclear Power Plants (LWR Edition)," issued June 2007, Subsection C.I.3.6.2.2.) The RG includes a guard pipe assembly as part of the piping penetrations. BTP 3-4, Part B, items A(ii)(3) and (6) also provide guidelines for the design of guard pipes. However, based on a review of the information provided in DCD Tier 2, Section 3.6.2.2, it was not clear whether these staff guidelines are applicable to the specific PCCV piping penetration design. Therefore, in **RAI 71-986, Question 03.06.02-8 and its follow-up RAI 459-3331, Question 03.06.02-26**, the staff requested the applicant to clarify whether these staff guidelines are applicable to the PCCV piping penetrations as shown in DCD Tier 2, Figure 3.8.1-8, "Containment Penetrations."

The applicant provided its response to **RAI 71-986, Question 03.06.02-8**, by a letter dated October 7, 2008. The applicant provided its response to **RAI 459-3331, Question 03.06.02-26** by a letter dated October 19, 2009. The applicant's responses to the staff's concerns are discussed below.

The applicant stated that the PCCV penetrations are isolated in compartments made of concrete. The annulus area is composed of several closed compartments surrounding the PCCV. The applicant also stated that the containment penetrations for these high-energy lines are designed to ASME Code, Section III, Subarticle NE-1120 and meet all the pertinent code design requirements as described in DCD Tier 2, Subsection 3.6.2.1.1.1. In addition, each piping penetration is designed to the break exclusion zone design provision described in DCD Tier 2, Subsection 3.6.2.1.1.1. The applicant further concluded that guard pipes and their associated design guidelines are not considered necessary around the PCCV piping penetrations.

Based on the review of the applicant's clarification as described above, the staff concluded that the applicant has appropriately justified that the PCCV piping penetrations are designed to withstand all the pertinent loads for the high-energy lines described above and no guard pipe assembly is required for the US-APWR plant configuration. Accordingly, the staff finds that the applicant's responses have adequately addressed the staff's concerns. Since a follow-up RAI was necessary, **RAI 71-986, Question 03.06.02-8, is closed and unresolved.** Its follow-up **RAI 459-3331, Question 03.06.02-26, is resolved.**

3.6.2.4.3 Analysis Methods to Define Blowdown Forcing Functions and Response Models

In DCD Tier 2, Section 3.6.2.3, the applicant discusses criteria for the analytical methods to be used to calculate the blow-down forcing functions as a result of a pipe rupture. The blow-down force is characterized as a function of time and space, and depends upon fluid state within the pipe prior to rupture, break flow area, frictional losses, plant system characteristics, piping system, and other factors. The applicant states that these forcing functions are determined for the US-APWR design by the SRP Section 3.6.2 guidelines and the applicant's in-house developed methodologies. DCD Tier 2, Section 3.6.2.3 describes a steady-state method and a time-dependent method to calculate the forcing functions. The applicant also stated that the rupture of a pressurized pipe causes the flow characteristics of the system to change, creating reaction forces that can dynamically excite the piping system. Furthermore, the fluid conditions at the upstream source and at the break exit determine the analytical approach for determining the forcing functions for postulated pipe breaks. For the postulated pipe breaks, one of the following situations exists: superheated or saturated steam, saturated or sub-cooled water, and cold water (non-flashing). The applicant also stated that the time dependent forcing function is affected by the thrust pulse resulting from the sudden pressure drop at the initial moment of pipe rupture, the thrust transient resulting from wave propagation and reflection, and the blowdown thrust resulting from the buildup of the discharge flow rate, which may reach a steady state if there is a fluid energy reservoir having sufficient capacity to develop a steady jet for a significant interval. Moreover, the applicant stated that, alternatively, a steady state jet thrust function may be used as outlined in DCD Tier 2, Subsection 3.6.2.1.

Based on its review of the information included in DCD Tier 2, Subsection 3.6.2.3, the staff determined that the applicant did not address some of the staff guidelines included in SRP 3.6.2. Therefore, in **RAI 71-986, Question 03.06.02-9** and two follow-up RAIs**, RAI 459-3331,**

Question 03.06.02-27 and **RAI 459-3331, Question 03.06.02-28,** the staff requested the applicant to provided additional information on the analysis methods to be used in the US-APWR piping design under a pipe rupture loading condition.

The applicant provided its response to **RAI 71-986, Question 03.06.02-9** by a letter dated October 7, 2008. The applicant provided its responses to **RAI 459-3331, Question 03.06.02-27** and **RAI 459-3331, Question 03.06.02-28** by letters dated October 19, 2009, and December 1, 2009, respectively. In addition, in a letter dated June 20, 2012, the applicant amended its responses to **RAI 71-986, Question 03.06.02-9** and **RAI 459-3331, Question 03.06.02-28**. The applicant's responses to the staff's concerns are discussed below.

DCD Tier 2, Section 3.6.2.3 did not address the SRP Section 3.6.2, Item II.2.A guideline related to the initial condition for piping pressurized during normal operation at power. In its response to **RAI 71-986, Question 03.06.02-9**, the applicant stated that the initial condition for piping pressurized during normal operation at power is the greater of the contained energy at hot standby or at 102 percent power. The staff found that this criterion is consistent with the pertinent SRP Section 3.6.2 guideline but needed to be added to the DCD. Therefore, **RAI 71-986, Question 03.06.02-9** was unresolved and in follow-up **RAI 459-3331, Question 03.06.02-27**, the staff requested that the applicant add this criterion to the DCD. In its response to **RAI 459-3331, Question 03.06.02-27**, the applicant stated that DCD Tier 2, Section 3.6.2.3 will be revised to include this criterion. The staff verified that the Revision 2 of DCD Tier 2, Subsection 3.6.2.3 includes this statement and found the response acceptable. Accordingly, **RAI 459-3331, Question 03.06.02-27, is resolved.**

DCD Tier 2, Section 3.6.2.3 did not explicitly identify the methodology to be used to determine the pipe whip dynamic response. In its response to **RAI 71-986, Question 03.06.02-9**, the applicant clarified that the energy balance method is used to characterize the whip restraint response to a pipe break as described in DCD Tier 2, Subsection 3.6.2.4.2.2. Based on its review of DCD Tier 2, Subsection 3.6.2.4.2.2, the staff found that the applicant's methodology for determining the pipe whip dynamic response is consistent with the pertinent SRP Section 3.6.2 guideline.

Furthermore, in its amended response to **RAI 71-986, Question 03.06.02-9**, dated June 20, 2012, the applicant proposed to revise DCD Tier 2, Section 3.6.2.1.3.1 to state that, for identifying potential targets resulting from the postulated pipe rupture, the direction of the fluid jet is based on the arrested position of the broken pipe end, including variation of the broken pipe end movement during the blowdown. This is consistent with the staff guideline for identifying potential targets resulting from the whipping movement of the postulated broken pipe end during the blowdown and is, therefore, acceptable. Therefore, the staff finds the amended response to **RAI 71-986, Question 03.06.02-9**, to be acceptable. **RAI 71-986, Question 03.06.02-9 is being tracked as a Confirmatory Item** until the applicant updates the DCD to incorporate the amended response.

DCD Tier 2, Section 3.6.2.3 did not address potential feedback between the jet and any nearby reflecting surface(s). This feedback can substantially increase (through resonance) the dynamic jet forces impinging on the nearby target components and the dynamic thrust blowdown forces on the ruptured pipe. In **RAI 71-986, Question 03.06.02-9** and **RAI 459-3331, Question 03.06.02-28**, the staff requested the applicant to address the jet feedback issue. In its response to **RAI 71-986, Question 03.06.02-9**, the applicant deferred this part of the RAI response to **RAI 459-3331, Questions 03.06.02-13 and 03.06.02-14**. Furthermore, in its amended response to **RAI 459-3331, Question 03.06.02-28,** the applicant referred to its response to **RAI 636-4732,**

Question 03.06.02-44, which is a follow-up to **RAI 459-3331, Question 03.06.02-13**. It should be noted that the staff's evaluations of the applicant's responses to **RAI 459-3331, Questions 03.06.02-13 and 03.06.02-14** and their associated follow-up RAIs are addressed in Section 3.6.2.4.4.1 of this report. The staff concurs with the applicant that the concerns in **RAI 459-3331, Question 03.06.02-28** may be addressed by **RAI 636-4732, Question 03.06.02-44**; therefore, the staff considers **RAI 459-3331, Question 03.06.02-28 resolved.** The staff's evaluation of **RAI 636-4732, Question 03.06.02-44** is discussed in Section 3.6.2.4.4.1 of this report under the heading "Jet Dynamic Loading Including Potential Feedback Amplification and Resonance Effects."

Based on its review of the applicant's responses as described above, the staff concluded that with the exception of the potential jet feedback issue which is deferred to **RAI 459-3331, Questions 03.06.02-13 and 03.06.02-14** and their associated follow-up RAIs addressed in Section 3.6.2.4.4.1 of this report, the applicant has adequately addressed the staff's concerns included in **RAI 71-986, Question 03.06.02-9** and its follow-up RAIs, **RAI 459-3331, Question 03.06.02-27** and **RAI 459-3331, Question 03.06.02-28**. Therefore, the staff found the applicant's responses to the follow-up RAIs acceptable. **RAI 71-986, Question 03.06.02-9 is being tracked as a Confirmatory Item** pending the applicant updating DCD Tier 2, Section 3.6.2.1.3.1**.** In addition, as described above, **RAI 459-3331, Question 03.06.02-27 and RAI 459-3331, Question 03.06.02-28 are resolved.**

3.6.2.4.4 Dynamic Analysis Methods to Verify Integrity and Operability

In DCD Tier 2, Section 3.6.2.4, the applicant provides the methods to perform dynamic analyses of a ruptured pipe and its jet impingement and whipping effects on safety-related SSCs. Time-dependent and steady-state thrust reaction loads caused by saturated or superheated steam, saturated or sub-cooled water, and cold water (non-flashing) fluid from a ruptured pipe are used in the design against the dynamic effects of a pipe break.

3.6.2.4.4.1 Jet Impingement Loading on Safety-Related Structures, Systems, and Components

SRP Section 3.6.2 describes currently acceptable procedures for assessing the forces induced by jets emanating from postulated piping breaks on neighboring SSCs, along with acceptable means of modeling jet expansion (which determine the spatial zones of influence of the loads within expanding jets). SRP Section 3.6.2 Section III.3F states that expansion models may be used for jet shapes when substantiated by test or analysis, but only for steam and water/steam mixtures; jet expansion should not be applied to cases of saturated water or subcooled water blowdown. Until recently, industry commonly used the ANSI/ANS Standard 58.2-1988, "Design-Basis for Protection of Light Water Nuclear Power Plants Against Effects of Postulated Pipe Rupture," (also referred to as ANS 58.2), Appendices C and D, for estimating jet plume geometries and loads based on the fluid conditions internal and external to the piping, and ANS 58.2 Standard had been accepted by the NRC staff.

The originally submitted DCD references the use of ANS 58.2 Appendices C and D to assess which SSCs might be loaded by jets emanating from postulated pipe breaks, and to assess resulting jet impingement loads on the impacted SSCs. The staff reviewed the ANS 58.2 Standard and Appendices, and DCD Tier 2, Section 3.6.2. The staff also considered the recent determination described in SRP Section 3.6.2, Section III.3 and further discussed below that several inaccuracies may lead to non-conservative assessments of the strength, zone of influence, and space and time-varying nature of the loading effects of supersonic expanding jets on neighboring structures. In addition, initial blast waves are unaccounted for in the standard.

Review of the ANS 58.2 jet models was motivated by GSI 191, which addresses the blockage of strainers upstream of emergency sump pumps by particulate. The particulate is formed by fibrous ceramic insulation, which can be broken loose by blast waves and/or jets emanating from nearby pipe ruptures. Letters from the Advisory Committee on Reactor Safeguards (ACRS) on this topic are cited in SRP Section 3.6.2, Section III.3. Also, examples of inconsistencies among existing standards for simulating the effects of LOCAs on neighboring structures are given in the Nuclear Energy Agency/Committee on the Safety of Nuclear Installations (NEA/CSNI) report NEA/CSNI/R(95)11, "Knowledge Base for Emergency Core Cooling System Recirculation Reliability," dated February 1996. Although the focus of the ACRS letters and the NEA/CSNI report was on debris generation and sump blockage, their comments directly affect the assessments of postulated pipe breaks on neighboring SSCs. The applicant was advised by the staff that, as stated in SRP Section 3.6.2 Section III.3, the ANS 58.2 Standard is no longer universally acceptable for modeling jet expansion in nuclear power plants. As a result, RAIs (**RAI 71-986, Questions 03.06.02-10 through 03.06.02-15**) and associated follow-up RAIs (**RAI 459-3331, Questions 03.06.02-28 through 03.06.02-35**, **RAI 636-4732, Questions 03.06.02-40 through 03.06.02-46**) related to the possible non-conservatisms in ANS 58.2 were issued to the applicant.

To resolve the above RAIs, the applicant revised pertinent parts of DCD Tier 2, Section 3.6.2 and issued two technical reports, MUAP-10017-P, Revision 3, and MUAP-10022-P, Revision 2. The staff's evaluations of the final applicant responses, the pertinent DCD Tier 2, Section 3.6.2 revision, and the two applicant technical reports follow.

<u>Blast Waves</u>

In the event of a high pressure pipe rupture, the first significant fluid load on surrounding structures would be induced by a blast wave. A spherically expanding blast wave is reasonably approximated to be a short duration transient and analyzed independently of any subsequent jet formation. Since the blast wave is not considered in the ANS 58.2 Standard or the DCD evaluation of the dynamic effects associated with the postulated pipe rupture, omission of blast wave considerations is clearly non-conservative. Therefore, in **RAI 71-986, Question 03.06.02-10** and two follow-up RAIs, **RAI 459-3331, Question 03.06.02-29** and **RAI 636-4732, Question 03.06.02-40** the staff requested the applicant to explain how the effects of blast loads on neighboring safety-related SSCs will be accounted for.

The applicant provided its response to **RAI 71-986, Question 03.06.02-10** dated November 7, 2008. The applicant later amended that response by letters dated December 27, 2011, and June 20, 2012. The applicant provided its response to **RAI 459-3331, Question 03.06.02-29** by a letter dated December 1, 2009, and subsequently amended that response by a letter dated June 20, 2012. The applicant responded to **RAI 636-4732, Question 03.06.02-40** by a letter dated November 24, 2010. By letters dated December 15, 2010 and June 20, 2012, the applicant subsequently amended its responses to **RAI 636-4732, Question 03.06.02-40**. The applicant's responses to the staff's concerns are described below.

The applicant responded to these RAIs by adding DCD Tier 2, Section 3.6.2.4.1.1, "Blast Wave Assessing Procedure," to the DCD. In this revised DCD section, the applicant stated that in the US-APWR design, blast waves may only emanate from steam line breaks in the pressurizer compartment. In its response, the applicant cited computational fluid dynamics (CFD) analyses which are described in MUAP-10022-P, Section 2. The analysis results show that the blast waves are attenuated significantly over the large distances between postulated pipe breaks and targets. The applicant further states that in the event the US-APWR design is changed, the methods described in MUAP-10022-P, Revision 2, will be used to evaluate blast wave effects.

In Section 2 of MUAP-10022-P, the applicant explains that blast waves cannot occur from ruptures of sub-cooled water piping, since the high-density water or two-phase discharge flow may not be accelerated to more than the speed of sound in the outside air. Based on its evaluation, the staff determined that the applicant's assertion is consistent with the March 31, 2010, letter from J. Rowley, US NRC, to A. Nowinowski, Westinghouse Electric Company, "Nuclear Regulatory Commission Conclusions Regarding Pressurized Water Reactor Owners Group Response to Request for Additional Information Dated January 25, 2010, Regarding Licensee Debris Generation Assumptions For GSI-191," which concluded that a blast wave would be insignificant for a sub-cooled liquid. Since blast waves from steam piping, however, may occur, the applicant provided simplified analytic (described in Appendix 1 of MUAP-10022-P), as well as CFD methods (described in Appendix 2 of MUAP-10022-P) for assessing blast wave pressures on postulated targets in the pressurizer compartment.

Furthermore, the applicant compared generic blast wave pressures for free blast waves, and blast waves that are amplified because of reflection against floors, walls, and ceilings, showing that CFD analyses provide more conservative (higher) free-blast wave loads than simplified analytic models, but that the amplification due to reflections may be modeled adequately using either CFD or simple analytic models. The applicant also provided a sample CFD analysis of a break from a nozzle on a safety valve, showing the closest target to be about 6.6 piping diameters from the break, and that all reflecting surfaces are far enough from the break so that (a) their loads on compartment boundaries are much smaller than those used to design the boundaries, and (b) their amplifying effects on neighboring targets are insignificant. The final loads on the target piping are shown to be small, producing one eighth of the allowable stress in the structural material. Furthermore, in its June 20, 2012, amended response to **RAI 636-4732, Question 03.06.02-40**, the applicant stated that DCD Tier 2, Section 3.6.2.4.1.1 will be revised to make a reference to MUAP-10022-P, Section 2 for the details of the evaluation and the design procedure for assessing a blast wave resulting from a postulated steam pipe break. Moreover, the applicant will revise DCD Tier 2, References 3.6-25 and 3.6-32. Specifically, Reference 25, MUAP-10017, will be revised from Revision 1 to Revision 3 and Reference 32, MUAP-10022, will be revised from Revision 0 to Revision 2.

Based on its review of the above information, the staff determined that the applicant has adequately addressed the staff's concerns and finds the response to **RAI 636-4732, Question 03.06.02-40** acceptable. Specifically, the applicant has provided acceptable blast wave analysis methods (in Section 2 of MUAP-10022-P) that show that the only potential blast waves in the US-APWR are in a large compartment, such that blast wave pressures are attenuated significantly before impinging on target boundaries or structures. These methods will be used for evaluating blast wave effects in the event the US-APWR design is modified. Therefore, since follow-up RAIs were necessary, **RAI 71-986 Question 03.06.02-10 and follow-up RAI 459-3331, Question 03.06.02-29 are closed and unresolved.** The last follow-up **RAI 636-4732, Question 03.06.02-40 is being tracked as a Confirmatory Item** until the applicant updates DCD Tier 2, Section 3.6 to incorporate the changes described above.

<u>Jet Plume Expansion and Zone of Influence</u>

It should be noted that in the characterization of supersonic jets given by ANS 58.2, some physically incorrect assumptions underlie the approximating methodology. The model of the supersonic jet itself is given in Figures C-1 and C-2 of the Standard. A fundamental problem of this model is the assumption that a jet issuing from a high pressure pipe break will always spread with a fixed 45 degree angle up to an asymptotic plane and subsequently spread at a constant 10 degree angle. The characteristics of the jet, however, are not universal. The initial jet spreading rate is highly dependent on the ratio of the total conditions of the source flow to the ambient conditions. Subsequent spreading rates depend, at a given axial position, on the ratio of the static pressure in the outermost jet flow region to the ambient static pressure. In ANS 58.2, the asymptotic plane is described as the point at which the jet begins to interact with the surrounding environment. However, the jet is highly dependent on the conditions in the surrounding medium and, at a given distance from the issuing break, will spread or contract at a rate depending on the local jet conditions relative to the surrounding fluid pressure.

In addition, supersonic jet behavior can persist over distances from the break far longer than those estimated by the standard, extending the zone of influence of the jet, and the number of safety-related SSCs that could be impacted by a supersonic jet. As described in NEA/CSNI/R(95)11, tests in the Seimens-KWU facility in Karlstein, Germany showed that significant damage from steam jets can occur as far as 25 pipe diameters from a rupture. Therefore, in **RAI 71-876, Question 03.06.02-11 and two follow-up RAIs, RAI 459-3331, Question 03.06.02-30, and RAI 636-4732, Question 03.06.02-41,** the staff requested the applicant to explain what analysis and/or testing has been used to substantiate the use of the ANS 58.2 Appendices C and D for defining conservatively which safety-related SSCs are in jet paths and the subsequent loading areas on the safety-related SSCs. In addition, in light of the findings in NEA/CSNI/R(95)11 that steam jets can cause significant damage at distances up to 25 pipe diameters, the staff requested the applicant to substantiate its assumption that components beyond 10 pipe diameter range are considered as undamaged and functional as included in DCD Subsection 3.6.1.1, Item L.2.

The applicant provided its response to **RAI 71-986, Question 03.06.02-11** by a letter dated November 7, 2008. The applicant provided its response to **RAI 459-3331, Question 03.06.02-30** by a letter dated December 1, 2009. The applicant provided its response to **RAI 636-4732, Question 03.06.02-41** by a letter dated November 24, 2010, and subsequently revised that response by a letter dated December 15, 2010. The applicant's responses to the staff's concerns are described below.

In response to the staff's concerns as described in the above RAIs, the applicant stated that the US-APWR design no longer relies on ANS 58.2 to define which SSCs are in jet paths and subsequent loading areas. Instead, the applicant has submitted MUAP-10017-P to define jet impingement loading. Specifically, in MUAP-10017-P, the applicant references six experimental studies performed in Japan that measured jet pressure distributions and magnitudes, along with three representative critical flow modeling approaches. The applicant uses the models in these references, substantiated by the measurements in the references, to define jet shapes and accompanying static jet impingement loads. In MUAP-10017-P the applicant also references MUAP-10022-P, for any design calculations that may be required to assess the effects of blast waves and oscillatory jet loading. Based on a review of the responses, the staff determined that the applicant has adequately addressed the staff's concerns. Specifically, the applicant has provided updated methodologies for determining which SSCs are in jet paths and resultant loading areas, all of which are substantiated by experiment. In addition, the staff has confirmed that the applicant has included in DCD Tier 2, Revision 3, Section 3.6.1.1 Item L.2 a statement that the guidance included in NUREG/CR 2913, "Two-Phase Jet Loads," January 1983, will be used to properly evaluate the jet effects on components beyond 10 pipe diameters of a postulated break. Therefore, since follow-up RAIs were necessary, **RAI 71-986, Question 03.06.02-11** and follow-up **RAI, RAI 459-3331, Question 03.06.02-30 are closed and unresolved.** Follow-up **RAI 636-4732, Question 03.06.02-41 is resolved.**

Distribution of Pressure Within the Jet Plume

Another assumption in ANS 58.2 that may lead to potential non-conservatism is related to the ANS 58.2 formulas for the spatial distribution of pressure through a jet cross-section. In some cases, the ANS 58.2 Standard's assumption that the pressure within a jet cross section is maximum at the jet centerline is correct (near the break, for instance). Far from the break, however, the pressure variation is quite different, often peaking near the outer edges of the jet. Applying the ANS 58.2 Standard's formulas could lead to non-conservative pressures away from the jet centerline. Therefore, in **RAI 71-986, Question 03.06.02-12 and four follow-up RAIs (RAI 459-3331, Question 03.06.02-31; RAI 459-3331, Question 03.06.02-32; RAI 636-4732, Question 03.06.02-42; and RAI 636-4732, Question 03.06.02-43),** the staff requested the applicant to explain what analysis and/or testing has been used to substantiate use of ANS 58.2 Appendix D for defining conservatively the net jet impingement loading on SSCs in light concerns about the accuracy of the pressure distribution models presented in ANS 58.2. In addition, the staff requested the applicant to expand DCD Tier 2, Table 3.6-2, to include the properties of the fluid internal and external to the ruptured pipe. The table should specify what type of jet the applicant assumes will emanate from each pipe break (incompressible nonexpanding jet, or compressible supersonic expanding jet) along with how impingement forces will be calculated for each jet. In addition, specific examples of jet impingement loading calculations made using the ANS 58.2 Standard for the postulated piping breaks in the US-APWR design should be given, along with proof that the calculations lead to conservative impingement loads in spite of the cited inaccuracies and omissions in the ANS 58.2 models.

The applicant provided its response to **RAI 71-986, Question 03.06.02-12** by a letter dated November 7, 2008. The applicant provided its response to **RAI 459-3331, Questions 03.06.02-31 and 03.06.02-32** by a letter dated December 1, 2009. In addition, the applicant provided its responses to **RAI 636-4732, Questions 03.06.02-42 and 03.06.02-43** by a letter dated November 24, 2010, and subsequently revised these responses by a letter dated December 15, 2010. The applicant's responses to the staff's concerns are described below.

In response to the staff's concerns as described in the above RAIs, the applicant stated that the US-APWR design no longer relies on ANS 58.2 to define net static jet impingement loading. Instead, the applicant has submitted MUAP-10017-P to define jet impingement loading. Also, in DCD Tier 2, Section 3.6.2.4.1, "Jet Impingement Loading on Safety-Related Components," the applicant explains that the maximum pressure in any impinging jet non-uniform pressure distribution is applied conservatively as a uniform pressure over a target. Finally, since the static jet impingement is applied as a near-step function with a 1 millisecond (ms) rise time, a dynamic load factor (DLF) of 2.0 is applied to the short-term load to account for any dynamic responses in the target structure. In addition, no DLF is applied for long-term static response calculations later during blowdown. The applicant also discusses certain cases for which a DLF of 2.0 is not used in assessing structural response to oscillatory jet in its responses to **RAI 71-986, Question 03.06.02-13** and its follow-up RAIs, **RAI 459-3331, Question 03.06.02-33** and **RAI 636-4732, Question 03.06.02-44**. The staff evaluation of the applicant responses to these RAIs is discussed later in this section under the heading "Jet Dynamic Loading Including Potential Feedback Amplification and Resonance Effects."

In addition, the applicant has provided in DCD Tier 2, Table 3.6-2, which includes a list of high energy lines for pipe break hazard analysis, including properties of internal and external fluids. Also, since the applicant no longer uses ANS 58.2 for their calculation methods, there is no need for them to provide examples of its calculations using ANS 58.2 as originally requested in the RAI. Moreover, the applicant discussed its updated calculation methods in MUAP-10017-P and these updated calculation methods are substantiated by the referenced measurements.

Based on a review of the above information, the staff determined that the applicant has adequately addressed the staff's concerns because that the applicant conservatively applies the highest pressure in a jet loading profile as a uniform loading profile, has included the requested table of postulated high energy line breaks, and has substantiated their loading calculation approach with several references to measurements. Therefore, since follow-up RAIs were necessary, **RAI 71-986, Question 03.06.02-12; follow-up RAI 459-3331, Question 03.06.02-31;** and follow-up **RAI 459-3331, Question 03.06.02-32) are closed and unresolved.** Follow-up **RAI 636-4732, Question 03.06.02-42** and follow-up **RAI 636-4732, Question 03.06.02-43) are resolved.**

Jet Dynamic Loading Including Potential Feedback Amplification and Resonance Effects

In DCD Tier 2, Section 3.6.2.4.1, the applicant stated that "structural integrity of safety-related SSCs against jet impingement load caused by pipe break is evaluated based on steady state jet force." The applicant also stated that the dynamic component of jet loading is considered independently from the static component, and that when static analysis methods are used to assess short term dynamic jet loads, the results are to be multiplied by a factor of two.

Supersonic expanding jets are unsteady, however, particularly those impinging on nearby structures. As described in NEA/CSNI/R(95)11, tests in Germany's Heissdampfreaktor showed high dynamic oscillating loads in the immediate vicinity of breaks. In the case of supersonic jets, their strong unsteadiness will tend to propagate in the shear layer and induce unsteady time-varying oscillatory loads on obstacles in the flow path. Pressures and densities vary non-monotonically with distance along the axis of a typical supersonic jet and this in turn feeds and interacts with shear layer unsteadiness. In addition, for a typical supersonic jet, interaction with obstructions will lead to backward-propagating transient shock and expansion waves that will cause further unsteadiness in downstream shear layers.

Moreover, in some cases, synchronization of the transient waves with the shear layer vortices emanating from the jet break can lead to significant amplification of the jet pressures and forces (a form of resonance) that is not considered in either ANS 58.2 or initially in the DCD. If the dynamic response of the neighboring structure also synchronizes with the jet loading time scales, further amplification of the loading can occur, including that at the source of the impinging jet. These feedback phenomena are well-known to those in the aerospace industry who work with aircraft that use jets to lift off and land vertically.[1] Some general observations by past investigators are that strong discrete frequency loads are observed when the impingement surface is within 10 diameters of the jet opening, and that when resonance within the jet occurs, significant amplification of impingement loads can result (Ho and Nosseir show a factor of 2-3 increase in pressure fluctuations at the frequency of the resonance, but this has not been shown to be a limiting value).

[1] Ho, C.M., and Nosseir, N.S., Dynamics of an Impinging Jet. Part 1, The Feedback Phenomenon, *Journal of Fluid Mechanics*, Vol.105, pp. 119-142, 1981.

Therefore, in **RAI 71-986, Question 03.06.02-13** and two follow-up RAIs, **RAI 459-3331, Question 03.06.02-33 and RAI 636-4732, Question 03.06.02-44,** the staff requested the applicant to provide information that establishes that the applicant's interpretation of the jet impingement force as static is conservative. In addition, the staff requested the applicant to explain whether any postulated pipe break locations are within 10 diameters of a neighboring SSC (or barrier/shield), and if so, how jet feedback/resonance and resulting dynamic load amplification are accounted for. Furthermore, the staff requested the applicant to clarify whether dynamic jet loads are to be considered, and if so, using what methods. Also, should the dynamic loading include strong excitation at discrete frequencies corresponding to resonance frequencies of the SSC impinged upon, provide the basis for assuming a static analysis with a DLF of 2.0 is conservative.

The applicant provided its response to **RAI 71-986, Question 03.06.02-13** by a letter dated November 7, 2008, and later amended that response by a letter dated June 20, 2012. The applicant provided its response to **RAI 459-3331, Question 03.06.02-33** by a letter dated December 1, 2009, and subsequently amended this response by a letter dated June 20, 2012. The applicant provided its response to **RAI 636-4732, Questions 03.06.02-44** by a letter dated November 24, 2010, and subsequently amended this response by letters dated December 15, 2010 and June 20, 2012. By letters dated December 28, 2010 and February 9, 2011, the applicant provided references cited in the December 15, 2010 amended response. The applicant's responses to the staff's concerns are described below.

As stated in Revision 3, DCD Tier 2, Section 3.6.2.4.1, the applicant considers the effects of oscillatory jet impingement loadings for selected steam line breaks or piping breaks with source fluid that may flash to steam during blowdown. These calculations complement those performed for static jet loads. The applicant provides its analysis methodology in MUAP-10022-P. In that technical report, the applicant provides a series of calculations and examples from the literature, which reveal that oscillatory loads due to jet feedback/resonance can only occur when the jet impingement surface pressure distribution is concave, with pressure peaks located along the outer radii of the impingement circle. The applicant then evaluates subcooled water and steam source fluids, revealing that subcooled water jets can never realize a concave surface pressure distribution (and hence no strong oscillatory loads can occur). However, for steam jets, or jets which flash to steam, concave conditions can occur later during blowdown at lower source-to-ambient fluid pressure ratios. At high pressure ratios near the beginning of blowdown, the jet expansion is so severe that reflected waves cannot interact strongly with the jet flow, preventing significant pressure oscillations on the target structure.

Furthermore, blowdown analysis (used to define jet source conditions) was performed on US-APWR Reactor Coolant Loop (RCL) piping as described in Appendix 3 of MUAP-10022-P. Thirteen piping systems with diameters ranging from 2" to 10" were evaluated. Short time periods during blowdown when the jet becomes pure steam are identified for nine of the piping systems, and later evaluated for oscillatory jet impingement pressures.

Once a time period is identified which requires oscillatory jet pressure and structural response calculations, the applicant applies an extremely conservative analysis procedure. The lowest possible jet resonance frequencies are computed as described in Appendix 5 of MUAP-10022-P. Instead of using CFD analysis to assess oscillatory jet load amplitudes under jet resonance, the applicant simply assumes that the jet impingement pressures oscillate between 0 and twice the static pressure (where the static pressure is computed using the methods described in MUAP-10017-P). The oscillation frequency is conservatively assumed to be identical to the resonance frequencies of the target structure, maximizing the structural response. This is the highest oscillatory jet loading that can occur on a target structure, and is extremely conservative.

For dynamic structural response calculations due to oscillatory jet loading, the applicant does not assume a DLF of 2.0 and apply it to a structural static analysis. Instead, the applicant assumes a damping ratio of 1 percent and coincident loading and structural resonance frequencies, leading to a DLF of 50, which is conservative. The applicant applies various structural modeling methods depending on the target and its size and orientation in relation to the jet. For cases where no simplification is justified, a finite-element (FE) model is built and analyzed to determine resonances and structural response. For complex piping systems, three-dimensional beam models are used. In some cases, a hand calculation assuming a pipe section is conservatively represented as a simply supported beam is performed. The applicant substantiates its approaches in Appendix 6 of MUAP-10022-P. Moreover, the applicant provides summary tables showing that for cases where oscillatory jet loading is applied, the net effects are higher than those of static loading by factors ranging from 1 to 4.7. Finally, in its June 20, 2012, amended response to **RAI 636-4732, Question 03.06.02-44**, the applicant proposes to revise DCD Tier 2, Section 3.6.2.4.1.2, “Jet Pressure Oscillation Assessing Procedure,” to state that the load of jet pressure oscillation is evaluated in the later stage of the blowdown process when the water flashes to steam. The revised DCD Tier 2 Section 3.6.2.4.1.2 will also refer to Section 3 of MUAP-10022-P for the details of the evaluation and the design procedures for assessing jet pressure oscillation from a postulated steam pipe break.

Based on its review of the above information, the staff determined that MHI has adequately addressed the staff’s concerns. Specifically, the applicant’s methodology is to screen piping breaks for conditions where jet resonance and strong oscillatory pressures may occur, and then conservatively apply oscillatory loading to target structures. Therefore, since follow-up RAIs were necessary, **RAI 71-986, Question 03.06.02-13** and follow-up **RAI 459-3331, Question 03.06.02-33 are closed and unresolved.** The follow-up **RAI 636-4732, Question 03.06.02-44 is being tracked as a Confirmatory Item** until the applicant updates DCD Tier 2, Section 3.6.2.4.1.2, as described above.

Based on its review of the DCD Tier 2, Section 3.6.2, the staff noted that the applicant did not address the effects of jets being reflected by neighboring structures in the US-APWR design. Therefore, in **RAI 71-986, Question 03.06.02-14** and two follow-up RAIs, **RAI 459-3331, Question 03.06.02-34 and RAI 636-4732, Question 03.06.02-45,** the staff requested the applicant to explain quantitatively how reflections of jets by neighboring structures will be considered for the US-APWR design.

The applicant provided its response to **RAI 71-986, Question 03.06.02-14** by a letter dated November 7, 2008, and later amended that response by a letter dated June 20, 2012. The applicant provided its response to **RAI 459-3331, Questions 03.06.02-34** by a letter dated December 1, 2009, and subsequently amended this response by a letter dated June 20, 2012. The applicant provided its response to **RAI 636-4732, Question 03.06.02-45** by a letter dated November 24, 2010, and subsequently amended this response by letters dated December 15, 2010, and June 20, 2012. The applicant provided references cited in the December 15, 2010 amended response by letter dated February 9, 2011. The applicant’s responses to the staff’s concerns are described below.

In responding to staff’s concerns, the applicant provided the details of its evaluation and the design procedure for assessing the effects of reflected jets on neighboring structures in Section 4 of MUAP-10022-P. The applicant assumes that jets continue to propagate parallel to an impingement surface, with jet velocity decreasing with the distance from the impingement region. An additional zone of influence for the jet reflection is computed for pressures of 0.01 MPa (1 psi) or more, where reflected pressures decrease with the square of distance from the impingement region, and slightly again as the reflected jet expands. The applicant performs a sample calculation of a perpendicularly aligned impinging jet using CFD calculations and confirms the conservatism of its zone of influence calculation approach. In its June 20, 2012, amended response to **RAI 636-4732, Question 03.06.02-45**, the applicant also proposed to revise DCD Tier 2, Section 3.6.2.4.1.3, "Jet Reflection Assessing Procedure," to state that there may be a case that a jet impinges upon an oblique wall; the effect of jet reflection is considered outside of the zone of influence. The applicant further states that the details of the evaluation and the design procedure are contained in Section 4 of MUAP-10022-P.

Based on its review of the above information, the staff determined that the applicant has adequately addressed the staff's concerns. Specifically, the applicant has provided a conservative approach for assessing reflected jets and subsequent loading on target structures in Section 4 of MUAP-10022-P. The applicant also proposed in its response to **RAI 636-4732, Questions 03.06.02-45** to include in DCD Tier 2, Section 3.6.2.4.1.3 the conditions under which reflections are considered, and to reference Section 4 of MUAP-10022-P for the methodologies used to evaluate the effects of reflected jets. Therefore, since follow-up RAIs were necessary, **RAI 71-986, Question 03.06.02-14 and follow-up RAI 459-3331, Question 03.06.02-34 are closed and unresolved**. The follow-up **RAI 636-4732, Question 03.06.02-45 is being tracked as a Confirmatory Item** until the applicant updates the DCD Tier 2, Section 3.6.2.4.1.3 as described above.

In DCD Tier 2, Section 3.6.2.4.4.2, "Jet Impingement Barriers and Shields," the applicant stated that in some cases, barriers or shields around essential equipment and instrumentation will be specified. However, the staff is concerned that if the barriers or shields are close to postulated jets, these nearby surfaces can induce feedback and resonance within the jets, potentially destroying the barrier or shield. Therefore, in **RAI 71-986, Question 03.06.02-15** and two follow-up RAIs, **RAI 459-3331, Question 03.06.02-35 and RAI 636-4732, Question 03.06.02-46,** the staff requested the applicant to explain how the barriers or shields will be designed so that they will not be damaged or destroyed by dynamic jet resonant loading.

The applicant provided its response to **RAI 71-986, Question 03.06.02-15** by a letter dated November 7, 2008, and later amended that response by a letter dated June 20, 2012. The applicant provided its response to **RAI 459-3331, Question 03.06.02-35** by a letter dated December 1, 2009, and subsequently amended this response by a letter dated June 20, 2012. The applicant provided its response to **RAI 636-4732, Question 03.06.02-46** by a letter dated November 24, 2010, and subsequently amended this response by letters dated December 15, 2010, and June 20, 2012. The applicant's responses to the staff's concerns are described below.

In responding to staff's concerns, the applicant provided an example of a barrier structural integrity analysis in response to resonant oscillatory jet loading in Section 3 of Appendix 6 of MUAP-10022-P. The applicant applies the methods for determining the oscillatory jet load described in their responses to **RAI 71-986, Question 03.06.02-1**3 and two follow-up RAIs, **RAI 459-3331, Question 03.06.02-33** and **RAI 636-4732, Question 03.06.02-44**. The applicant then applies worst-case oscillatory jet loading at the lowest structural resonance frequency of the barrier and conservatively assumes one-percent damping for the structural response. A worst-case stress is computed and compared to meet the applicable ASME Code, Section III Service Level D standards. In its June 20, 2012, amended response to **RAI 636-4732,**

Question 03.06.02-46, the applicant also proposed to revise DCD Tier 2, Section 3.6.2.4.4.2 to state that the design procedure for jet pressure oscillation is contained in Section 4 of MUAP-10022-P.

Based on its review of the above information, the staff determined that the applicant has adequately addressed the staff's concerns. Specifically, the applicant has provided an example of a conservative approach for assessing the effects of jets on a barrier and the resulting stress is to meet the applicable ASME stress limits. Therefore, since follow-up RAIs were necessary, **RAI 71-986, Question 03.06.02-15** and follow-up **RAI 459-3331, Question 03.06.02-35 are closed and unresolved. The follow-up RAI 636-4732, Question 03.06.02-46 is being tracked as a Confirmatory Item** until the applicant updates DCD Tier 2, Section 3.6.2.4.4.2 as described above**.**

3.6.2.4.4.2 Dynamic Analysis for Piping Systems

In DCD Tier 2, Subsection 3.6.2.4.2, "Dynamic Analysis for Piping Systems," the applicant provides methods for evaluating the dynamic effects of pipe whipping action resulting from postulated pipe failure of high-energy fluid system including RCL piping. As stated in DCD Tier 2, Subsection 3.6.2.4.2.1, "RCL Piping," Appendix 3C, "Reactor Coolant Loop Analysis Methods," provides methods used in the dynamic analysis of the RCL piping. Loads generated by postulated breaks from branch lines are applied to determine the structural response of the RCL piping. There are two models: one coupled with the supporting structures and another involving the piping and its support systems without the supporting structure. DCD Tier 2, Subsection 3.6.2.4.2.2, "Piping Other Than RCL Piping," describes methods used in the dynamic analysis for high-energy-fluid system piping other than RCL piping. The applicant states that in evaluating the dynamic analysis for high-energy-fluid system piping, possible break locations and break configurations are first established based on DCD Tier 2, Section 3.6.2.1 and the effects of pipe whipping are then evaluated as described in DCD Tier 2, Subsection 3.6.2.4.4. The applicant further stated that the dynamic effects of a pipe break on the pipe and pipe whip restraint are evaluated by the energy balance method and the design methodology for pipe whip restraints are described in DCD Tier 2, Subsection 3.6.2.4.4.1 "Pipe Whip Restraints." The staff's evaluations of the design methodology for pipe whip restraints are described in Section 3.6.2.4.4.4 of this report.

Moreover, in DCD Tier 2, Subsection 3.6.2.4.2.3, "Closure of the Feedwater Check Valve," the applicant discusses the loading from closure of a feedwater check valve, specifically rapidly traveling pressure waves in piping systems connected to the broken piping system. The closure of the feedwater check valve due to a postulated pipe rupture upstream of the valve can increase the magnitude of these loads. For piping systems with closing check valves, the magnitude of the loadings depends on the valve closure time, with shorter closing times generally causing higher loadings. The maximum internal pressure and the kinetic energy of the valve disc at the time of closure are used to assure the pressure boundary integrity of the piping. The thermo-hydraulic computer code RELAP-5 is used to calculate the pressure and kinetic energy.

Based on its review of the above information as included in DCD Tier 2, Subsections 3.6.2.4.2.1, 3.6.2.4.2.2, and 3.6.2.4.2.3, the staff determined that some areas related to the applicant's analysis methods for evaluating the dynamic effects of postulated pipe ruptures need to be clarified. In **RAI 71-986, Question 03.06.02-16** and two follow-up RAIs, **RAI 459-3331, Questions 03.06.02-36 and 03.06.02-37,** the staff requested the applicant to clarify how the applicant will ensure the operability of pipe-mounted safety-related components on the ruptured pipe and how supports including pipe whip restraints are modeled in the detailed evaluation of ruptured piping system.

The applicant provided its response to **RAI 71-986, Question 03.06.02-16** by a letter dated October 7, 2008. The applicant provided its responses to **RAI 459-3331, Questions 03.06.02-36 and 03.06.02-37** by a letter dated October 19, 2009. The applicant's responses to the staff's concerns are described below.

In responding to the staff's concern about how the applicant will ensure the operability of pipe-mounted safety-related components on the ruptured pipe, the applicant stated that a five-way restraint is installed for MS piping and feedwater piping outside of the PCCV to prevent a load from being applied to the PCCV isolation valve due to a postulated pipe break outside of break exclusion zone, as described in DCD Tier 2, Subsection 3.6.2.1.1.1. In other cases, the subject valve is installed sufficiently away from a postulated break location to prevent dynamic effects. Furthermore, the pipe stress in the vicinity of the valve is validated as very small by using a static force displacement methodology for the pipe displacement at the break location. The applicant also stated that the analysis of the broken pipe identifies the loading applied at the end of the pipe-mounted safety-related component, and the analysis will ensure that the loading meets the loading specified in the specification of that safety-related component. In addition, the applicant refers to DCD Tier 2, Section 3.6.2.1.1.1, Item (3) which states that the operability of these valves is assured in accordance with the guidelines in SRP Section 3.9.3, "ASME Code Class 1, 2, and 3 Components, and Component Supports, and Core Support Structures," Revision 2, March 2007.

Based on its evaluation of the above information, the staff determined that the applicant's responses adequately addressed the staff's concerns on the operability of pipe mounted components and, therefore, found the responses to the follow-up RAIs acceptable. In responding to the staff's concerns related to how the supports including the pipe whip restraints are modeled in the detailed evaluation of the ruptured piping system, the applicant proposed a modification of DCD Tier 2, Section 3.6.2.4.2.2 to state that when analyzing the dynamic effects associated with pipe rupture events on the broken pipe, a non-linear elastic-plastic analysis is performed. In this model, pipe whip restraints specifically designed to prevent pipe whip are included. The normal supports that act during plant operational loads, including seismic events, to maintain the integrity of the unbroken pipe are not considered unless they are capable of withstanding pipe rupture loads based on a broken pipe analysis. The staff verified that the Revision 3 of DCD Tier 2, Subsection 3.6.2.4.2.2 has been appropriately revised to include this statement. Therefore, the staff determined that the applicant has sufficiently clarified how the supports including the whip restraints are modeled in the detailed evaluation of the ruptured piping system, and, therefore, found the responses to the follow-up RAIs acceptable. Therefore, since follow-up RAIs were necessary, **RAI 71-986, Question 03.06.02-16 is closed and unresolved.** The two follow-up RAIs, **RAI 459-3331, Questions 03.06.02-36 and 03.06.02-37 are resolved.**

3.6.2.4.4.3 Subcomponent Pressure Forces

In DCD Tier 2, Subsection 3.6.2.4.3, "Subcompartment Pressure Forces," the applicant states that subcompartment pressure forces should be considered in the evaluation of structures and components. It also states that the code GOTHIC may be used to calculate the pressure transients in the building subcompartments. The staff's review of subcompartment pressurization is within the scope of SRP 6.2.1.2. The staff's evaluation is addressed in Section 6.2.1.2 of this report and is not otherwise addressed here.

3.6.2.4.4.4 Pipe Whip Restraints, Barriers and Shields

In DCD Tier 2, Subsection 3.6.2.4.4, the applicant provides analytical methods used to verify integrity and operability of the safety-related SSCs needed to safely shutdown the plant, that are nearby the postulated pipe breaks. The analytical methods apply to the design of whip restraints and jet impingement barriers and shields.

In DCD Tier 2, Subsection 3.6.2.4.4.1, the applicant describes two types of whip restraints. Each type of whip restraint typically consists of a structural steel frame or truss and an energy absorbing element as described below:

- U-Bar (One-Dimensional Restraint): This is a U-shaped rod or flat plate, usually of carbon steel, looped around the pipe but not in contact with the pipe to allow unimpeded pipe movement during normal operation and a seismic event. At rupture, the pipe impacts the U-Bar(s), and the U-Bar absorbs the kinetic energy of the pipe by yielding plastically.

- Structural Steel (Two-Dimensional Restraint): This is a structural steel frame assembly enveloping the pipe but not in contact with the pipe allowing unimpeded pipe motion during normal operation and a seismic event. At rupture, the pipe impacts the structural steel frame; the frame deflects plastically, absorbing the kinetic energy of the pipe.

Each restraint is either a combination of an energy absorbing element and a restraining structure suitable for the geometry required to transfer the load from the whipping pipe to the main building structure, or a relatively rigid steel frame to restrain the whipping pipe. Both designs allow normal movement of the piping and yield plastically at rupture absorbing the pipe’s kinetic energy. Factors affecting the selection of a whip restraint include the available space, allowable building structure reaction, permissible pipe deflection, and equipment operability.

The following criteria are used in the analysis and design of the pipe whip restraints for US-APWR design:

- Pipe whip restraints are designed based on the principle of energy absorption by considering the behavior of the material’s elasticity/plasticity and strain hardening.
- A coefficient of rebound of 1.1 is applied to jet thrust forces.
- Energy absorption by the broken pipe is assumed to be zero, except in the case of calculating to check the formation of a plastic hinge. The developed thrust force is applied to move a broken pipe directly, and is not reduced by the forces required to bend pipes.
- In the elasticity/plasticity design, the kinetic energy of the pipe is absorbed by the restraint by yielding plastically.

In DCD Tier 2, Subsection 3.6.2.4.4.2, the applicant also states that barriers or shields are provided to protect essential equipment, including instrumentation, from the effects of jet forces resulting from postulated pipe breaks. Generally the protection requirements are met through the protection provided by walls, floors, and columns. Loading combinations and design criteria for barriers and shields are described in DCD Tier 2, Section 3.8, "Design of Category I Structures," for the design of major structures.

Moreover, in DCD Tier 2, Subsection 3.6.2.4.4.3, "Pipe Whip Impact on Structures," the applicant provides criteria describing the impact of pipe whip on structures. For surrounding structures, the level of energy in the whipping pipe may be determined by calculating work quantities using simplified methods. As the impact occurs on concrete targets, the section of the pipe near the impact area is rapidly decelerated and crushed. The magnitude and the duration of the impact loading are determined by characteristics of both the whipping pipe and the concrete barrier. In the evaluation of the target, both local and overall responses are considered. For piping systems near the whipping pipe, the applicant assumes that an unrestrained whipping pipe is considered capable of causing circumferential and longitudinal breaks, individually, in impacted pipes of smaller nominal pipe size, and of developing leakage cracks in equal or larger nominal pipe sizes with thinner wall thickness, except where analytical or experimental, or both, data for the expected range of impact energies demonstrate the capability to withstand the impact without rupture.

Based on a review of the above information, the staff found that the applicant's criteria are consistent with the staff guidelines for the analysis and design of the pipe whip restraints. However, the staff also found that it is not clear how the seismic load is considered in the pipe whip restraints design. Therefore, in **RAI 71-986, Question 03.06.02-17** and its follow-up **RAI 459-3331, Question 03.06.02-38,** the staff requested the applicant to clarify how the seismic load is considered in the pipe whip restraints design.

The applicant provided its response to **RAI 71-986, Question 03.06.02-17** by a letter dated October 7, 2008. The applicant provided its response to **RAI 459-3331, Question 03.06.02-38** by a letter dated October 19, 2009. In response to staff's question, the applicant stated that pipe whip restraints used to protect SSCs are designed as seismic Category I structures. The applicant further stated that seismic loads are independently considered to confirm the structural integrity of the pipe whip restraints. In addition, the applicant proposed to revise the DCD to clarify that pipe whip restraints are designed as seismic Category I structures. The staff found the applicant's proposed DCD revision acceptable because it adequately addressed the staff's concerns. In addition, the staff verified that DCD Tier 2, Subsection 3.6.2.4.4.1 Revision 2 has been revised accordingly. Therefore, since a follow-up RAI was necessary **RAI 71-986, Question 03.06.02-17 is closed and unresolved.** Its follow-up **RAI 459-3331, Question 03.06.02-38 is resolved.**

In DCD Tier 2, Subsection 3.6.2.4.4.2, the applicant states that barriers or shields are provided to protect essential equipment, including instrumentation, from the effects of jet forces resulting from postulated pipe breaks. The applicant also states that walls, floors, and columns are provided for protection, and that the loading combinations and design criteria for barriers and shields are described in DCD Tier 2, Section 3.8. The staff's evaluation of DCD Tier 2, Section 3.8 is addressed in Section 3.8 of this report.

Moreover, in DCD Tier 2, Subsection 3.6.2.4.4.3, the applicant provides criteria for evaluating structures that are impacted by whipping pipes using the energy balance method. The applicant also states that an unrestrained whipping pipe is considered capable of causing circumferential and longitudinal breaks, individually, in impacted pipes of smaller nominal pipe size, and of developing leakage cracks in equal or larger nominal pipe sizes with thinner wall thickness, except where analytical and/or experimental data for the expected range of impact energies demonstrate the capability to withstand the impact without rupture. The applicant's criteria as described are consistent with the pertinent SRP Section 3.6.2 guidelines and, therefore, the staff found them acceptable.

3.6.2.4.5 Implementation of Criteria Dealing with Special Features

DCD Tier 2, Section 3.6.2.5 states that special features such as pipe whip restraints, barriers, and shields are discussed in DCD Tier 2, Subsection 3.6.2.4.4. The staff's evaluation of the information included in DCD Tier 2, Subsection 3.6.2.4.4 is discussed in Section 3.6.2.4.4.4 of this report.

3.6.2.4.6 Outline of Pipe Break Hazard Analysis Report(s)

10 CFR 52.47(b)(1) requires that a design certificate 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 has been constructed and will operate with the DC, the provisions of the Atomic Energy Act of 1954, and the NRC's regulations. As described in SRP Section 14.3.3, "Piping Systems and Components – ITAAC," ITAAC should be included regarding the completion of a pipe break evaluation report that documents that SSCs that are required to be functional during and following a SSE have adequate high-energy pipe break mitigation features. The design description should discuss the criteria used to postulate pipe breaks, the analytical methods used to performed pipe breaks, and the methods to confirm the adequacy of the results of the pipe break analyses. The design description should be verified in a pipe break hazard analysis report that provides assurance that the postulated pipe break analyses have been completed. The pipe break hazard analysis report shall conclude that, for each postulated pipe failure, the reactor can be shut down safely and maintained in a safe, cold shutdown condition without offsite power. Detailed information that supports this ITAAC should be contained in DCD Tier 2, Chapter 3. Furthermore, an as-built reconciliation review of this pipe break hazard analysis report should be included in the ITAAC. It should be noted that the pipe break hazard analysis report cannot be completed until the piping design is completed. Since the piping design for the US-APWR will not be completed for the DC, an ITAAC for pipe break hazard analysis is therefore needed.

Based on its review of the DCD, the staff found that the applicant did not initially provide sufficient information related to the pipe break analysis report as described above. Specifically, DCD Tier 2, Section 3.6.2 did not contain a section that lists or summarizes the specific information that will be included in the pipe break hazard analysis report. Moreover, the DCD did not address specifically when the design aspect of the pipe break hazard analysis report will be made available for the staff's review. The staff's position is that if the as-designed pipe break hazard analysis will not be completed within the DC review phase, the applicant is to propose an ITAAC to address the as-designed (in addition to the as-built) pipe break hazard analysis. Therefore, in **RAI 71-986, Question 03.06.02-18** and two follow-up RAIs, **RAI 459-3331, Question 03.06.02-39 and RAI 636-4732, Question 03.06.02-48,** the staff requested the applicant to address these concerns.

The applicant provided its response to **RAI 71-986, Question 03.06.02-18** by a letter dated October 7, 2008. The applicant provided its response to **RAI 459-3331, Question 03.06.02-39** by a letter dated December 1, 2009. The applicant provided its response to **RAI 636-4732, Question 03.06.02-48** by a letter dated November 24, 2010, and later revised that response by a letter dated December 15, 2010. The applicant's responses to the staff's concerns are described below.

In addressing the staff's concern, the applicant added DCD Tier 2, Section 3.6.2.6 to Revision 3 of the DCD; this section provides an outline of the pipe break hazard analysis report. Specifically, the following activities will be completed for high and moderate energy piping systems (including the nonsafety class piping) identified in DCD Tier 1, Section 2.3, Table 2.3-1, "High and Moderate Energy Piping System Considered for Protection of Essential Systems." These activities will enable the closure of ITAAC in DCD Tier 1, Section 2.3, Table 2.3-2 related to the pipe break hazard analysis report (as-designed and as-built).

- Identification of pipe break locations in high energy piping.
- Identification of leakage crack locations in high and moderate energy piping.
- Identification of SSCs that are safety-related or required for safe shutdown.
- Evaluation of consequences of pipe whip and jet impingement.
- Evaluation of consequences of spray wetting, flooding, and environmental conditions.
- Design and location of protective barriers, restraints, and enclosures.

In DCD Tier 2, Section 3.6.2.6, the applicant also notes that DCD Tier 2, Table 3.6-2 provides list of high energy lines for the pipe break hazard analysis, including properties of internal and external fluids. In addition, all the SSCs that are safety-related or required for safe shutdown in close proximity to the postulated pipe rupture will be identified in the pipe break hazard analysis report.

Furthermore, the applicant added an as-designed pipe break hazard analysis ITAAC item (in addition to as-built) to Revision 3 of DCD Tier 1, Section 2.3, Table 2.3-2, which is consistent with the staff's guidelines in SRP Section 14.3.3.

Based on a review of the above information, the staff determined that the applicant has adequately addressed the staff's concerns. Specifically, the outline of the pipe break hazard analysis report as described in DCD Tier 2, Section 3.6.2.6 is acceptable because it ensures that the report will provide sufficient information to demonstrate the acceptability and the adequacy of the applicant's pipe break hazard analysis for US-APWR design. In addition, DCD Tier 1, Section 2.3, Table 2.3-2 now addresses both aspects (as-designed and as-built) of the pipe break hazard analysis report as the staff requested. Therefore, since follow-up RAIs were necessary, **RAI 71-986, Question 03.06.02-18** and follow-up **RAI 459-3331, Question 03.06.02-39 are closed and unresolved. RAI 636-4732, Question 03.06.02-48, is resolved.** It should be noted that the staff's full evaluation of the acceptability of DCD Tier 1. Table 2.3-2, and thus compliance with 10 CFR 52.47(b)(1) for the pipe hazard analysis issue is addressed in Section 14.3.3 of this report.

3.6.2.5 Combined License Information Items

The following is a list of COL item numbers and descriptions from Table 1.8-2 of the DCD related to determination of rupture locations and dynamic effects associated with the postulated rupture of piping:

Table 3.6.2-1
US-APWR Combined License Information Items

Item No.	Description	Section

Table 3.6.2-1 US-APWR Combined License Information Items		
Item No.	**Description**	**Section**
COL 3.6(1)	The COL Applicant is to identify the site-specific systems or components that are safety-related or required for safe shutdown that are located near high-energy or moderate-energy piping systems, and are susceptible to the consequences of these piping failures. The COL Applicant is to provide a list of site-specific high-energy and moderate-energy piping systems, which includes a description of the layout of all piping systems where physical arrangement of the piping systems provides the required protection, the design-basis of structures and compartments used to protect nearby essential systems or components, or the arrangements to assure the operability of safety-related features where neither separation nor protective enclosures are practical. Additionally, the COL Applicant is to provide the failure modes and effect analyses that verifies the consequences of failures in site-specific high-energy and moderate-energy piping does not affect the ability to safely shut down the plant. The COL Applicant is to update the as-design pipe hazards analysis report to include the impact of all site-specific high and moderate piping systems,	3.6.1.3
COL 3.6(4)	The COL applicant is to implement the criteria for defining break and crack locations and configurations for site-specific high-energy and moderate energy piping systems. The COL applicant is to identify the postulated rupture orientation of each postulated break location for site-specific high-energy and moderate -energy piping systems. The COL applicant is to implement the appropriate methods to assure that as-built configuration of site-specific high-energy and moderate-energy piping systems is consistent with the design intent and provide as-built drawings showing component locations and support locations and types that confirms this consistency.	3.6.2.1

It should be noted that the actions described in COL Information Item 3.6(4) are enveloped by the actions items for COL Information Item 3.6(1) and are a part of the pipe break and crack evaluations for site-specific high energy and moderate energy piping systems. Therefore, the staff's evaluation of the adequacy and acceptability of the above listed COL Information Items 3.6(1) and 3.6(4) concerning the pipe break and crack evaluations for site-specific high-energy and moderate-energy piping systems is addressed in Section 3.6.1 of this report and is not otherwise addressed here. As discussed above in Section 3.6.1.4 of this report, the applicant modified COL Information Item 3.6(1) in response to **RAI 795-5884, Question 03.06.01-9, which is being tracked as a Confirmatory Item.**

3.6.2.6 Conclusions

Based on its review, the staff concludes that the criteria for postulating pipe rupture and crack locations and the methodology for evaluating the subsequent environmental and dynamic effects on safety-related SSCs resulting from these ruptures are generally consistent with the guidelines in Section 3.6.2 of the SRP and meet the requirements of GDC 4 and, therefore, are acceptable for ensuring that the US-APWR design of essential SSCs is adequately protected against the effects of postulated pipe breaks.

The proposed pipe rupture locations will be adequately determined using the staff-approved criteria and guidelines given in BTP 3-4. The applicant has sufficiently and adequately defined the design methods for high-energy mitigation devices and the measures to deal with the subsequent dynamic effects of pipe whip and jet impingement to provide reasonable assurance that, upon completion of the high-energy line break analyses as part of the ITAAC process [DCD Tier 1 (Revision. 3) Table 2.3-2, item 4], the ability of safety-related SSCs to perform their safety functions will not be impaired by the postulated pipe ruptures.

The provisions for protection against the dynamic effects associated with pipe ruptures of the RCPB inside the containment and the resulting discharging fluid provides reasonable assurance that design-basis LOCAs will not be aggravated by the sequential failures of safety-related piping and that the performance of the ECCS will not be degraded as a result of these dynamic effects.

The arrangement of piping and restraints and the final design considerations for high- and moderate-energy fluid systems inside and outside the containment, including the RCPB, will be the responsibility of the COL applicant. These staff-approved high-energy and moderate-energy line break criteria and guidelines will be used to assure that the SSCs important to safety that are in close proximity to the postulated pipe ruptures will be adequately protected. Using these criteria and guidelines will ensure that the consequences of pipe ruptures will be adequately mitigated so that the reactor can safely be shut down and maintained in a safe-shutdown condition in the event of a postulated rupture of a high- or moderate-energy piping system inside or outside the containment.

3.6.3 Leak-Before-Break Evaluation Procedures

3.6.3.1 Introduction

This section describes and evaluates the design provisions and basis for the use of LBB methods for the US-APWR as they relate to eliminating from the design-basis the dynamic effects of pipe rupture for selected piping systems. The LBB method is used in the design of the US-APWR RCL to ensure that a pipe leak would be detected at load conditions that are well below those that would cause pipe rupture. A prerequisite to applying the LBB methodology is that the piping and weld material must be selected and the water chemistry controlled such that passive failure mechanisms such as primary water stress corrosion cracking (PWSCC) will not cause cracking. As such, the LBB method requires that crack growth be predictable based on fatigue analysis of the cyclic operating loads. It is also assumed that the highest probability of cracking occurs at the welds, and therefore, the LBB evaluation is performed for all the weld locations in the RCL piping. The staff’s review ensures that consideration has been given to piping failure mechanisms and degradation mechanisms that could adversely challenge the integrity of the piping.

3.6.3.2 Summary of Application

DCD Tier 1: The Tier 1 information associated with this section is found in DCD Tier 1, Sections 2.3, "Piping Systems and Components," 2.4.2, "Reactor Coolant System (RCS)," 2.4.4, "Emergency Core Cooling System (ECCS)," 2.4.5, "Residual Heat Removal System (RHRS)" and 2.7.1.2, "Main Steam Supply System (MSS)."

DCD Tier 1, Section 2.3.1, "Design Description," addresses the design description of piping, systems and components including requirements specific to LBB. For applicable high-energy piping, the US-APWR design performs a LBB evaluation so that the dynamic effects of pipe rupture can be eliminated. The descriptions of DCD Tier 1, Chapter 2, "Design Descriptions and ITAAC," address LBB design requirements for the applicable systems.

DCD Tier 1, Table 2.3-2, "Piping Systems and Components Inspections, Tests, Analyses, and Acceptance Criteria," Item 2.a has a design commitment stating, "Reactor coolant piping, pressurizer surge line piping and MS piping in the PCCV, for systems identified in Table 2.3-3, are designed in accordance with the LBB method." DCD Tier 1, Table 2.3-2, Item 2.a also has corresponding ITAAC that will ensure that a LBB analysis using the LBB methods will be performed and the results of the LBB analysis will conclude that the stress values conform to the LBB acceptance criteria using the LBB assumptions. DCD Tier 1, Table 2.3-2, Item 2.b has the same design commitments and ITAAC for the remaining piping designed in accordance with LBB methods. DCD Tier 1, Tables 2.4.2-3, "Reactor Coolant System Piping Characteristics," 2.4.4-3, "Emergency Core Cooling System Piping Characteristics," 2.4.5-3, "Residual Heat Removal System Piping Characteristics," and 2.7.1.2-3, "Main Steam Supply System Piping Characteristics," identify the Reactor Coolant System, Emergency Core Cooling System, Residual Heat Removal System and MS Supply System piping as designed using the LBB methods.

The following DCD Tier 1, Tables address ITAAC for the as built piping for the indicated systems to ensure the systems will meet the LBB acceptance criteria, or protection is provided for dynamic effects of piping breaks:

- DCD Tier 1, Table 2.4.2-5, "Reactor Coolant System Inspections, Tests, Analyses, and Acceptance Criteria," Item 16 addresses the as built piping for the Reactor Coolant System.

- DCD Tier 1, Table 2.4.4-5, "Emergency Core Cooling System Inspections, Tests, Analyses, and Acceptance Criteria," Item 13 addresses the as built piping for the Emergency Core Cooling System.

- DCD Tier 1, Table 2.4.5-5, "Residual Heat Removal System Inspections, Tests, Analyses, and Acceptance Criteria," Item 14 addresses the as built piping for the Residual Heat Removal System.

- DCD Tier 1, Table 2.7.1.2-5, "Main Steam Supply System Inspections, Tests, Analyses, and Acceptance Criteria," Item 12 addresses the as built piping for the MS System.

DCD Tier 2: The applicant has provided a DCD Tier 2 description in Section 3.6.3, "LBB Evaluation Procedures," summarized here in part, as follows:

The section describes how GDC 4 is applied, which allows the use of staff approved LBB analyses to eliminate from the design-basis dynamic effects associated with pipe rupture.

Information provided in DCD Tier 2, Section 3.6.3 stated that, “this subsection describes the design-basis to eliminate the dynamic effects of pipe rupture (Subsection 3.6.2) for the selected high-energy piping systems of RCL piping, RCL branch piping, and main steam piping.” The applicant stated, “The LBB evaluation is performed in accordance with SRP 3.6.3.”

The applicant stated that the LBB analysis combines normal and abnormal (including seismic) loads to determine a critical crack size for a postulated pipe break, that is then compared to the size of a leakage crack for which detection is certain. If the leakage crack size is sufficiently smaller than the critical crack size, then the LBB requirements are satisfied. The applicant stated that for piping where LBB is demonstrated the evaluation of environmental effects for spray wetting and flooding is still performed in accordance with DCD Tier 2, Section 3.6.2, “Determination of Rupture Locations and Dynamic Effects Associated with the Postulated Rupture of Piping.”

The applicant stated that the COL applicant is to identify the types of as-built materials and material specification used for base metal welds, weldments, and safe ends for piping evaluated for LBB. Additionally, information is to be provided related to as-built material and material specifications for piping including toughness (J-R curves) and tensile strength (stress-strain curves), yield and ultimate strength, welding process/methods used, provide confirmation that the actual plant-specific stress analysis based on final as-built plant piping layout and material properties and welds satisfy the bounding LBB analysis, and provide confirmation that the final bounding LBB analysis addresses all plant-specific and generic degradation mechanisms in the as-built piping systems. The applicant has committed to ITAAC described in DCD Tier 1, Table 2.3-2 to address this issue. The applicant established COL Information Item 3.6(10) to address water hammer protection.

Each of the main areas of review for the LBB analysis is addressed below:

3.6.3.2.1 Application of Leak-Before-Break, Design Criteria for Leak-Before-Break, and Potential Failure Mechanisms for Piping

The application of LBB is limited to high energy, ASME Code Class 1 or 2 piping or the equivalent. The application establishes specific design criteria for application of LBB methods. Also, the application describes the applicant’s bases that specific potential failure mechanisms are not credible sources of potential pipe rupture. In DCD Tier 2, Section 3.6.3.1 through 3.6.3.3, the applicant indicated the following:

- DCD Tier 2, Section 3.6.3.1, “Application of LBB Criteria,” addresses the piping systems to which LBB criteria are applied. The LBB method is applied to high-energy systems with well-defined loading combinations including: RCL piping, RCL branch piping with nominal diameter of 6 in. (15 cm) or larger (except for piping with steam i.e. the pressurizer safety valve and power operated relief valve), and the MS pipe in the PCCV.

- DCD Tier 2, Section 3.6.3.2, “Design Criteria for LBB,” addresses the design criteria for LBB. Specifically, the applicant includes a list of design features addressing preservice inspection, vibration fatigue, material toughness, leak detection systems that meet the guidelines of RG 1.45, "Guidance on Monitoring and Responding to Reactor Coolant System Leakage," Revision 1, May 2008, stresses within the bounding analysis curves (BACs), verification of final as-built piping conditions, ISI and testing of snubbers, evaluation of erosion, erosion/corrosion and erosion cavitation, and evaluation of adjacent structures and components designed for the SSE event to assure low probability of indirect piping failure.

- DCD Tier 2, Section 3.6.3.3, "Potential Failure Mechanism for Piping," addresses the evaluation of potential failure mechanisms for piping and other degradation sources to assure acceptability of the LBB criteria. The evaluation is based on the guidance in SRP Section 3.6.3, "Leak-Before-Break Evaluation Procedures," and includes a detailed evaluation of failure mechanisms and degradation sources that could challenge the integrity of the piping to ensure that failure by these mechanisms is not credible. The following failure mechanisms and degradation sources were evaluated: water hammer, creep damage, wall thinning induced by erosion/corrosion, stress corrosion cracking (SCC), fatigue, thermal aging and other mechanisms. The other mechanisms evaluated included failures at piping cleavages, surrounding SSC failures and damage from missiles.

3.6.3.2.2 Analytical Methods and Criteria

In DCD Tier 2, Section 3.6.3.4, "Analytical Methods and Criteria," the applicant addresses the methods and criteria used for LBB analysis and states they are consistent with NUREG-1061, Evaluation of Potential Pipe Breaks," November 1984, and SRP Section 3.6.3. The applicant stated that LBB BACs are prepared for each applicable piping system. These curves provide the design guidelines for meeting the allowable standards for stress limits and LBB acceptance criteria. The maximum stresses in the piping must be on or below the BAC to satisfy the LBB criteria. The applicant states that the LBB evaluation is based on the fracture mechanics of cracks and analysis of break mechanism, which compares the selected leakage cracks with critical crack sizes.

In DCD Tier 2, Subsection 3.6.3.4.1, "Leak Detection Capability," the applicant states that the sizes of the postulated leakage flaws are sufficiently large so that leaks can be detected by a sufficient margin. A leak rate of 10 times the capability of the leak detector is postulated for normal operating load combinations. The rated detection capability for the leak detector for reactor coolant in the containment is 0.5 gpm (2 L/min) within one hour of detector response time. In DCD Tier 2, Subsection 3.6.3.4.2, "Stability and Critical Crack Sizes," the applicant addresses the stability of the critical crack size and the methods used to determine the critical crack size. In DCD Tier 2, Subsection 3.6.3.4.3, "Allowable Standards," the applicant states that the critical crack size is determined by adding maximum individual loads by absolute summation and that a margin of 1.0 on the load is used, since the loads are added by absolute sum. In the same subsection, the applicant also states that a margin of two applies to the margin between the critical flaw size and the leakage crack size. The leakage from the flaw during normal operation is 10 times greater than the minimum leakage the detection system is capable of detecting.

DCD Tier 2, Section 3.6.3.4.1 describes the leak detection methods for supporting LBB. The methods or indications used for detecting the reactor coolant leak are the containment sump water levels, inventory balance, and the radiation in the environment of containment. DCD Tier

2, Section 3.6.3.2 states that leak detection systems meet the requirements of RG 1.45. DCD Tier 2, Section 5.2.5, “RCPB Leakage Detection,” describes the RCPB leakage detection instruments supporting the LBB application for RCPB. In DCD Tier 2, Section 5.2.5, “Reactor Coolant Pressure Boundary (RCPB) Leakage Detection,” the applicant indicates that the RCPB leakage detection methods have the capability to detect RCS leakage of 0.5 gpm (2 L/min) within one hour.

In the DCD Tier 2, Subsections 3.6.3.4.4, “Bounding Analysis Methods,” through 3.6.3.4.10, “Bounding Analysis Results,” and Appendix 3B, “Bounding Analysis Curve Development for Leak Before Break Evaluation of High-Energy Piping for US-APWR,” the applicant lays out the methods, procedures, data, properties, and calculation steps (DCD Tier 2, Figure 3.6-4, “LBB Evaluation Procedure,”) used in developing the BACs. The evaluation of piping and the LBB results are contained in the applicant Technical Reports MUAP-09010-P, “Summary of Stress Analysis Results for the Reactor Coolant Loop Piping,” Revision 3, March 2011; MUAP-09011-P, “Summary of Stress Analysis Results for Reactor Coolant Loop Branch Piping,” Revision 2, December 2010; MUAP-09013-P, “Summary of Stress Analysis Results for Main Steam Piping Inside Containment Vessel.” Revision 2, March 2011; and MUAP-11003-P, “Summary of Stress Analysis Results for the US-APWR Pressurizer Surge Line,” Revision 1, March 2011.” DCD Tier 2, Appendix 3B provides details on the PICEP: Pipe Crack Evaluation Program, Revision 1, 1987, software being used in the LBB analysis.

In DCD Tier 2, Subsection 3.6.3.4.11,”Differences in Inspection Criteria for Class 1 and 2 Systems,” the applicant addresses the differences in ISI inspection criteria between ASME Code Class 1 and Class 2 systems and clarifies that the LBB evaluations that are performed are the same for Class 1 and Class 2 systems and are based on the ability to detect a potential leaking crack; not the ability to find the cracks by ISI. In DCD Tier 2, Subsection 3.6.3.4.12, “Differences in Fabrication Requirements of ASME Code, Section III Class 1 and Class 2 Piping,” the applicant addresses the differences in the fabrication and nondestructive examination requirements for ASME Code Class 1 and 2 systems and states that the differences between Class 1 and Class 2 systems do not affect the LBB analyses assumptions, criteria, or methods.

ITAAC: The ITAAC associated with DCD Tier 2, Section 3.6.3 are delineated in DCD Tier 1, Sections 2.3, 2.4.2, 2.4.4, 2.4.5 and 2.7.1.2. DCD Tier 1, Section 2.3.1 addresses the design description of piping systems and components including ITAAC requirements specific to LBB. There are several ITAAC tables and items for high-energy and moderate-energy systems outside containment.

TS: There are no LBB specific requirements in the TS. However, operability of the RCS leak detection instrumentation assumed in the LBB analysis is covered under TS LCO 3.4.15, “RCS Leakage Detection Instrumentation.” The TS are found in the DCD Tier 2, Chapter 16, “Technical Specifications.”

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

Technical Reports: Technical reports associated with DCD Tier 2, Section 3.6.3 are as follows:

1. MUAP-09010-P, “Summary of Stress Analysis Results for the Reactor Coolant Loop Piping,” Revision 0, issued March 2009.

2. MUAP-09010-P, “Summary of Stress Analysis Results for the Reactor Coolant Loop Piping,” Revision 1, issued May 2009.

3. MUAP-09010-P, “Summary of Stress Analysis Results for the Reactor Coolant Loop Piping,” Revision 3, issued March 2011.

4. MUAP-09011-P, “Summary of Stress Analysis Results for Reactor Coolant Loop Branch Piping,” Revision 0, issued March 2009.

5. MUAP-09011-P, “Summary of Stress Analysis Results for Reactor Coolant Loop Branch Piping,” Revision 2, issued December 2010.

6. MUAP-09013-P, “Summary of Stress Analysis Results for Main Steam Piping Inside Containment Vessel.” Revision 0, issued March 2009.

7. MUAP-09013-P, “Summary of Stress Analysis Results for Main Steam Piping Inside Containment Vessel.” Revision 2, issued March 2011.

8. MUAP-11003-P, “Summary of Stress Analysis Results for the US-APWR Pressurizer Surge Line,” Revision 1, issued March 2011.

Cross-cutting Requirements (TMI, USI/ GSI, Op Ex): There are no cross-cutting requirements for this area of review.

US-APWR Interface Issues Identified in the DCD: See DCD Tier 2, Table 1.8-2, “Compilation of All Combined License Applicant Items for Chapters 1-19.”

Site Interface Issues Identified in the DCD: See DCD Tier 2, Table 1.8-1, “Significant Site Specific Interfaces with the Standard US-APWR Design.”

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

3.6.3.3 Regulatory Basis

The relevant requirements of the Commission’s regulations for this area of review, and the associated acceptance criteria, are given in Section 3.6.3, “Leak-Before-Break Evaluation Procedures,” Revision 1, issued March 2007 of NUREG-0800 and are summarized below. Review interfaces with other SRP sections can be found in Section 3.6.3 of NUREG-0800.

1. GDC 4, as it relates to the exclusion of dynamic effects of the pipe ruptures that are postulated in SRP Section 3.6.2, “Determination of Rupture Locations and Dynamic Effects Associated with the Postulated Rupture of Piping.” The design-basis for the piping means those conditions specified in the safety analysis report (SAR), as amended, and, which may include regulations in 10 CFR Part 50, applicable sections of NUREG-0800, RGs, and industry standards such as the ASME Code.

2. 10 CFR 52.47(b)(1), which requires that a DC application address 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 has been constructed 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. Compliance with GDC 4 requires that components important to safety be designed to accommodate the effects of, and be compatible with, environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including LOCAs. Safety-related components should be protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids that may result from equipment failure or events and conditions outside the nuclear power unit.

   Meeting the requirements of GDC 4 provides assurance that SSCs important to safety will be protected from the dynamic effects of pipe rupture and capable of performing their intended safety function.

2. LBB analyses should demonstrate that the probability of pipe rupture is extremely low under conditions consistent with the design-basis for the piping. A deterministic evaluation of the piping system that demonstrates sufficient margins against failure, including verified design and fabrication and an adequate ISI program, can be assumed to satisfy the extremely low probability criterion.

3. LBB should only be applied to high-energy, ASME Code Class 1 or 2 piping or the equivalent. Applications to other high-energy piping will be considered based on an evaluation of the proposed design and ISI requirements as compared to ASME Code Class 1 and 2 requirements.

4. Approval of the elimination of dynamic effects from postulated pipe ruptures is obtained individually for particular piping systems at specific nuclear power units. LBB is applicable only to an entire piping system or analyzable portion thereof. LBB cannot be applied to individual welded joints or other discrete locations. Analyzable portions are typically segments located between piping anchor points. When LBB technology is applied, all potential pipe rupture locations are examined. The examination is not limited to those postulated pipe rupture locations determined from NUREG-0800, Section 3.6.2.

3.6.3.4 Technical Evaluation

The staff focused its review of DCD Tier 2, Section 3.6.3 on the LBB analyses and evaluations to be used for the US- APWR reactor. The staff used the applicable guidance and requirements found in SRP Section 3.6.3 to complete its review. The results and conclusions reached are as follows.

The staff's review examined the sections of the DCD that discussed and defined the LBB evaluation procedures and analyses and judged if the proposed methods are sufficiently comprehensive and complete to demonstrate compliance with the applicable regulations, requirements, standards, and guidance. An internal consistency check was performed by the staff to verify that the LBB requirements and commitments were consistent through the various sections of the DCD including Tier 2, Section 3.6.3; Tier 2, Appendix 3B; and other referenced sections.

COL applicants referencing the US-APWR design need to fully describe the site-specific piping system configuration. The as built conditions must be verified to assure that the design methods used for LBB evaluations are consistent with the final as-built conditions including material specifications, pipe geometry, support conditions and locations and weights of components such as valves.

Information provided in DCD Tier 2, Section 3.6.3 states that the LBB evaluation is performed in accordance with SRP Section 3.6.3. The staff has reviewed the submitted information and finds that it meets the requirements and criteria on LBB and is acceptable. The staff also found that the information provided by the applicant for resolution of the applicable RAIs was acceptable. The staff developed multiple questions during its evaluation of SRP Section 3.6.3. These requests for RAIs were sent to the applicant. The applicant's response and the staff disposition of the RAIs are documented below. The staff confirmed that Revision 3 of the DCD is consistent with all of the documented RAI dispositions.

3.6.3.4.1 Leak-Before-Break Inspections, Tests and Analyses and the Acceptance Criteria

In **RAI 210-1948, Question 03.06.03-1**, the staff completed a review of the ITAAC for LBB contained in DCD Tier 1, Tables 2.4.2-5, 2.4.4.-5, 2.4.5-5 and 2.7.1.2-5 and identified the need for clarification of the ITAAC called out in the tables.

In its response to **RAI 210-1948, Question 03.06.03-1**, dated April 9, 2009**,** the applicant indicated that the inspections, tests, analyses (ITA) for each of the above items will be changed in Revision 2 of the DCD to clarify that the criteria stated in these items is in reference to inspections of the as-built piping to confirm that the installation is consistent with the evaluation report for LBB, or the inspection is to confirm the protection from dynamic effects of a pipe break. The applicant also clarified that it recognizes that the as-built piping may exceed tolerances thereby requiring an as-built analysis in the LBB evaluation report. The applicant went on to clarify that DCD Tier 2, Subsection 3.6.3.4, states that LBB BACs are prepared for each applicable piping system to determine the critical location having the highest stress point from piping analysis. The BACs are therefore a part of the LBB evaluation report. The applicant clarified that the ITAACs are to confirm the as-built piping configuration is consistent with the LBB evaluation report, or to confirm the protection from dynamic effects of a pipe break. The acceptance criteria to verify the as-built piping configuration and material are not dependent on the analytical location of the highest stress point. If the as-built configuration is determined by inspection to be outside the LBB acceptance criteria, the BAC will be prepared as part of the updated LBB evaluation report. Therefore, the acceptance criteria as stated are consistent with the design commitment that the as-built piping or the protection for the dynamic effects of the piping break meets the LBB acceptance criteria.

The applicant committed to change the ITA for the items listed above to read, "Inspections of the as-built piping will be performed based on the evaluation report for LBB or the protection from dynamic effects of a pipe break, as specified in Section 2.3." These revisions are acceptable because they ensure that the LBB acceptance criteria will be met for the as-built piping. The applicant in its response committed to changes to the DCD that resolve the staff's concerns. The staff verified that the applicant revised each of the ITAAC for the piping subject to LBB analyses to include the clarifications and wording described above. Based on the above, the staff found the applicant's response acceptable. Accordingly, **RAI 210-1948, Question 03.06.03-1, is resolved**.

In **RAI 849-6109, Question 03.06.03-26**, the staff issued a follow-up ITAAC question to clarify the wording in DCD Tier 1, Table 2.3-2, Item 2.a. SRP Section 3.6.3 calls for verification that a LBB evaluation exists which documents that the LBB acceptance criteria for the as built LBB evaluation is bounded by the design LBB analysis. The staff notes that the LBB acceptance criteria are described in DCD Tier 2, Section 3.6.3.4 and Appendix 3B and are evaluated in Section 3.6.3.4.8 of this report.. In its response to **RAI 849-6109, Question 03.06.03-26**, dated November 21, 2011, the applicant clarified that DCD Tier 1, Table 2.3-2 provides design ITAAC for piping systems and components. The applicant stated that the inspection ITAAC of as-built piping systems and components for LBB is described in DCD Tier 1, Table 2.4.2-5 for the RCS, DCD Tier 1, Table 2.4.4-5 for the ECCS, DCD Tier 1, Table 2.4.5-5 for the RHRS and DCD Tier 1, Table 2.7.1.2-5 for the MS system (MSS). The staff verified that Revision 3 of the DCD does contain the correct wording for the Inspection, Tests and Analyses and the Acceptance Criteria ITAAC for each of the ITAAC tables cited. The staff found the applicant's response acceptable since it clarified the location of the as-built ITAAC for LBB. Accordingly, **RAI 849-6109, Question 03.06.03-26, is resolved**.

3.6.3.4.2 Wall Thinning in Leak-Before-Break Piping

SRP Section 3.6.3 Section III.2 provides guidelines for the evaluation of wall thinning in LBB piping. The last paragraph of DCD Tier 2, Section 3.6.3.1 addresses wall thinning in general terms.

Wall thinning is addressed in more detail in DCD Tier 2, Section 3.6.3.3.3, "Wall Thinning Induced by the Effects of Erosion/Corrosion." The discussion covers RCL, RCL branch piping, and MS piping. The applicant properly concluded that wall thinning is not credible for the RCL and RCL branch piping based on the use of stainless steels for this piping, and industry experience that has shown no cases of wall thinning.

The applicant also addressed wall thinning for the SA-333 Grade 6 carbon steel of the MS piping. The applicant states that wall thinning "is not anticipated" due the high temperatures and high quality steam for this piping, and goes on to state that wall thinning due to erosion is not a credible failure mechanism. There was no discussion on flow accelerated corrosion (FAC) or on elbows and other fittings to ensure ASME Code minimum wall thicknesses requirements are met. In **RAI 210-1948, Question 03.06.03-2**, the staff requested that the applicant provide additional information on FAC and also address elbows and fittings for the MS piping. The applicant was also asked to provide more detail on the information in the DCD used to conclude that wall-thinning is not credible and the wall thickness of MS piping will not be reduced below ASME Code minimum wall-thicknesses.

In its response to **RAI 210-1948, Question 03.06.03-2**, dated April 23, 2009, the applicant clarified that DCD Tier 2, Subsection 10.3.6.3, "Flow-Accelerated Corrosion (FAC)," the applicant describes (FAC), including elbows for the MS piping. The applicant indicated that DCD Tier 2, Subsection 10.3.6.3 lists five design controls that have been used to address FAC:

1. Selection of corrosion resistant materials.

2. Chemistry controls to minimize corrosion.

3. Pipe schedule and wall thickness for design life of the plant.

4. Corrosion allowances that meet ASME Code requirements, as applicable.

5. Flow velocities with industry standards.

The applicant also stated that the COL applicant will be responsible for implementing a FAC monitoring program. The staff finds that the applicant's response has provided the additional information and clarifications necessary to conclude that FAC has been addressed through the indicated design controls and that wall thinning of the MS piping is not credible and that the wall thickness will not be reduced below the ASME Code requirements. Therefore, the staff found the applicant's response acceptable. Accordingly, **RAI 210-1948, Question 03.06.03-2, is resolved**.

3.6.3.4.3 Stress Corrosion Cracking in Leak-Before-Break Piping

The staff reviewed the evaluations to demonstrate that SCC will not impact the structural integrity of piping. Note that from Revision 1 to Revision 2, the applicant added and removed several figures in DCD Tier 2, Appendix 3B and then renumbered the figures in Revision 2. Therefore the DCD revision number will be used for each discussion of these figures for clarity purposes. The DCD Tier 2 Revision 1, Appendix 3B, Figures 3B-7, "US-APWR BAC for Primary Loop Hot Leg (SA182 F316LN)," through 3B-10, "US-APWR BAC for Primary Loop Crossover Leg (SA182 F316LN)," indicate that the RCL pipe materials are SA-182 F316LN and SA-182 F316. The specific weld alloys that will be used in the US-APWR and the potential for weld cracking were not identified in the LBB evaluation. DCD Tier 2, Subsection 3.6.3.3.4, "SCC," evaluates the SCC of stainless steel piping and the SA-333 Grade 6 carbon steel of the MS piping. The applicant concludes that SCC is not a credible mechanism for stainless steels of the RCL, RCL branch piping and the ferritic steels of the MS piping. The applicant states that cracking "is not anticipated" due to favorable water chemistries that will be maintained for the MS piping. The applicant addressed SCC but did not specifically address PWSCC. In **RAI 210-1948, Question 03.06.03-3,** the staff requested the applicant to provide additional information to support the conclusion in DCD Tier 2, Subsection 3.6.3.3.4 and to address the following questions:

1. Provide additional information and evaluations on why PWSCC is not a potential source of pipe rupture and the selection of pipe material grades and weld alloys that are resistant to cracking by PWSCC. Clarify in the LBB evaluations which pipe material grades and weld alloys will be used in the different piping systems in the US-APWR.

2. Provide information on the weld alloys used and the potential for cracking in welds by SCC or PWSCC. Provide detailed information on what weld practices will be used to ensure PWSCC is not a concern due to chromium content, dilution effects, cleaning methods, weld qualifications and environmental effects on crack growth in Alloy 690.

3. Provide additional information on the guidelines the applicant will be following to maintain the favorable water chemistries for the MS piping.

In its response to **RAI 210-1948, Question 03.06.03-3**, dated April 23, 2009, the applicant addressed the three questions identified by the staff. Regarding the first question, the applicant stated that the wrought stainless steel alloys SA-182 F316LN and SA-182 F316 are used for the piping. Compatible stainless steel gas tungsten arc (GTA) and shielded metal arc (SMA) weld filler materials will generally be used for all welds between the stainless steel piping and between the piping and the reactor coolant pumps, SGs and the RV. The dissimilar metal welds joining the piping and ferritic nozzles will be constructed with Alloy 52M/152 nickel-based weld filler metal. The applicant noted that Alloy 52M and Alloy 152 have similar alloy content to Alloy 690 wrought material. The weldments typically would consist of a butter application to the end of the ferritic nozzle and then joined by the same filler material directly to the stainless steel pipe or safe end following the vessel post weld heat treatment (PWHT). This fabrication sequence avoids sensitization of the stainless steel weld heat affected zone due to the PWHT.

Regarding the second question, the applicant clarified that limiting the amount of carbon and welding heat input reduces susceptibility of stainless steel weld material in PWR coolant to PWSCC. The stainless steel materials, including the welds, have not shown susceptibility to PWSCC in a PWR primary water environment. The only material in existing PWR plants that has exhibited susceptibility to PWSCC is Alloy 600 and its compatible weld filler metals, Alloy 82/182. These materials are found in the dissimilar metal welds of operating nuclear reactors joining the ferritic nozzles to the stainless steel piping or safe-ends and will not be used in the US-APWR plant. For Alloy 182, the Cr content can be as low as 14 percent. This is well below the accepted threshold for high resistance to PWSCC (24 percent Cr). Both Alloy filler metals 52M and 152 contain nominally 30 percent Cr, which is well above the accepted threshold that has been qualified by testing as being highly resistant to PWSCC. This qualification is documented in Electric Power Research Institute (EPRI) MRP-139, "Material Reliability Program: Primary System Piping Butt Weld Inspection and Evaluation Guideline," Revision 1, issued December 2008, where PWSCC resistant materials are identified as including high nickel materials and resistant welding materials, including Alloy 52M for GTA welding (GTAW) and Alloy 152 for SMA welding (SMAW), having nominally 30 percent chromium. The applicant stated that to assure that PWSCC will not be a concern for the dissimilar metal welds; the following weld practices will be followed:

1. The weld material will be procured to an ASME specification that requires that the material meet the requirements for ERNiCrFe7A (Alloy 52M) or ENiCrFe7A (Alloy 152) or equivalents. The composition of each lot of weld filler metal shall be certified by the manufacturer to meet the ASME chemical requirements for those weld filler materials.

2. All dissimilar metal welds joining austenitic stainless steel pipe to ferritic steel nozzles, shall be qualified to establish that the filler materials and welding procedure will produce a chromium level in the initial layers adjacent to the dissimilar metals to a level equal to or greater than 24 percent Cr. This requirement shall be validated by a representative mock up, welding procedure qualification test, or both. Machine GTAW shall be limited to a maximum heat input of 45 kJ/in (43 BTU/in.) and shall establish an effective Power Ratio range (as defined by ASME Section IX) to control weld dilution and provide quality weld deposits. SMAW may be controlled by heat input limits only.

3. Cleaning and other preparation methods shall use only materials that are compatible with the nickel based material and are designed so as to not contaminate these materials with regards to weldability issues and/or service corrosion performance.

4. As required by Section IX of the ASME Code, weld qualification tests and procedure qualifications shall be performed, as required, to verify the procedure

is capable of producing welds that meet the required mechanical and chemical properties for the material specifications. All field welds shall be designed to meet these requirements.

Regarding the third question, the applicant also clarified that DCD Tier 2, Subsection 10.3.5.1, "Chemistry Control Basis," describes water chemistry and that this information will be used as guidelines by the COL applicant.

The staff finds that the applicant's response has addressed why SCC and PWSCC are not a potential source of pipe rupture and has provided additional information on the selection of pipe material grades and weld alloys that are resistant to cracking by PWSCC. The applicant has clarified which pipe material grades and weld alloys will be used in the different piping systems in the US-APWR. The applicant also provided information on the weld alloys used and the potential for cracking in welds by SCC or PWSCC. The applicant provided detailed information on what weld practices will be used to ensure PWSCC is not a concern due to the effects described. The applicant clarified that the guidelines to maintain the favorable water chemistries for the MS piping can be found in DCD Tier 2, Section 10.3.5.1. The staff finds that these water chemistry controls for the MS system are consistent with previous industry practice and will ensure favorable water chemistry to mitigate PWSCC.

In follow-up **RAI 485-3825, Question 03.06.03-18,** the staff requested the applicant to clarify a statement concerning material in dissimilar metal welds. In its response to follow-up **RAI 485-3825, Question 03.06.03-18**, dated January 19, 2010, the applicant clarified that materials susceptible to PWSCC such as Alloy 600 and its compatible weld filler metals, Alloy 82/182 are not found in the dissimilar metal welds joining the ferritic nozzles to the stainless steel piping or safe-ends in the US-APWR. This is consistent with the first paragraph of the applicant's original response that indicate that dissimilar metal welds joining the piping and ferritic nozzles will be constructed with Alloy 52M/152 nickel-based weld filler metal. The staff finds that the applicant's response is acceptable, as it has clarified the inconsistent statement regarding the materials used in the dissimilar metal welds joining the ferritic nozzles to the stainless steel piping. Accordingly, **RAI 210-1948, Question 03.06.03-3** and follow-up **RAI 485-3825, Question 03.06.03-18 are resolved**.

3.6.3.4.4 Water Hammer in Leak-Before-Break Piping

In **RAI 210-1948, Question 03.06.03-4**, the staff indicated that the applicant in DCD Tier 2, Section 3.6.3.3 addresses the evaluation of potential failure mechanisms including: water hammer, creep damage, wall thinning induced by erosion/corrosion, SCC, fatigue, and thermal aging. The last sentence of DCD Tier 2, Section 3.6.3.3 states that each failure mechanism and degradation source is evaluated below and confirmed as credible, thereby confirming LBB eligibility. The staff pointed out that the approach in SRP Section 3.6.3 is to evaluate these failure mechanisms to confirm that they are not credible. The staff asked the applicant if this sentence should have read, "…confirmed as not being credible, thereby confirming LBB eligibility." In its response to **RAI 210-1948, Question 03.06.03-4**, dated April 9, 2009, the applicant confirmed that the statement was incorrect and corrected it in Revision 2 of the DCD. The staff confirmed that DCD Tier 2, Section 3.6.3.3, Revision 2 reads, "Each failure mechanism and degradation source is evaluated below and confirmed as not credible, thereby confirming LBB eligibility." The staff found the response and the revisions to the DCD acceptable since they clarified LBB eligibility. Accordingly, **RAI 210-1948, Question 03.06.03-4, is resolved**.

In **RAI 210-1948, Question 03.06.03-5**, the staff requested the applicant to provide additional information on the design features and operation/maintenance controls that will be in place to prevent water hammer. The applicant addresses water hammer in DCD Tier 2, Section 3.6.3.3 and Subsection 3.6.3.3.1, "Water Hammer." When implementing operational or other means of control to abate water hammer events, the applicant should indicate that selected measures will be effective for the life of the plant. The applicant indicated that water hammer has been reported for ECCS piping in the past and operational controls will be applied in a way that avoids water hammer. No information was provided on how these controls will be maintained over the life of the plant.

The applicant addresses water hammer for RCL branch piping in DCD Tier 2, Subsection 3.6.3.3.1 and states the following: "That water hammer has been reported in ECCS piping in the past. In US-APWR, however, operational control is applied in a way that avoids water hammer." The applicant also addresses water hammer in the MSS Piping and states that protection against water hammer is provided through operations and maintenance procedures and proper draining. Guidance in SRP Section 3.6.3, Section III.5 allows for the use of historical frequencies, operating procedures and design configurations to demonstrate that water hammer will not be a significant contributor. The SRP also states that measures needed to abate water hammer frequency and magnitude will be effective for the life of the plant. The US-APWR will be relying on operational controls, maintenance procedures and design configurations in a way that will avoid water hammer for the ECCS piping and MS Piping. The staff requested the applicant to provide additional information in the DCD regarding the design features that will be in place to ensure that water hammer will not be a concern for the life of the plant. The staff also requested the applicant to determine if a COL item is required to address the operational and maintenance controls described above that ensure water hammer will be avoided.

In its response **to RAI 210-1948, Question 03.06.03-5**, dated April 9, 2009, the applicant stated that water hammer prevention and mitigation of RCL branch piping is implemented in accordance with NUREG-0927, "Evaluation of Water Hammer Occurrence in Nuclear Power Plants," Revision 1, issued March 1984. The applicant listed the items NUREG-0927 recommends to be included in operating and maintenance procedures including: A) Prevention of rapid valve motion, B) Proper filling and venting of water-filled lines and components, C) Introduction of voids into water-filled lines and components, D) Introduction of steam or heated water that can flash into water-filled lines, E) Introduction of water into steam-filled lines or components, F) Proper warm-up of steam-filled lines, G) Proper drainage of steam-filled lines, and H) The effects of valve alignments on line conditions. The applicant clarified that item B is the item corresponding to operating and maintenance procedures. The applicant clarified that venting will be required prior to system operation and that venting before system operation and confirmation of system line up will be specified in the operation manual. Therefore, the applicant indicated that it is unnecessary to add a new COL item.

The applicant also provided additional information on how each of the items A) through G) is or is not implemented. In item A) the applicant indicated that in relation to the rapid closure of a valve, at the stage of plant construction, water hammer load accompanying valve rapid closure will be evaluated by analysis, and it will be reflected in the design of the valve and associated supports. In item B) the applicant stated vent valves are installed such that venting of piping or equipment can be performed properly. Also strict venting requirements before system operation will be specified in the operation manual. For item C) the applicant stated that because the system is vented appropriately before operation, there is little possibility of void introduction into the system during plant start-up, shutdown, and power operation. In addition, water hammer due to voids generated by a pressure rise inside the piping accompanying pump activation can be prevented by venting before system operation, which includes normal start-up as well as restoring after a plant trip. Venting before system operation will be specified in the operation manual as is described in B. Therefore, the applicant concluded that the possibility of water hammer due to void introduction into water-filled lines and components is extremely low. For item D) the applicant stated that there is very little possibility that pressurizer steam can flow into the pressurizer spray line or surge line to the main coolant piping and cause water hammer. For item E) the applicant clarified that pressurizer spray is included in this item, but that pressurizer spray does not cause water hammer. For items F) and G) the applicant clarified that these two items are not applicable to RCL branch lines. For item H) the applicant stated that elevations of valves are considered in the layout design. The applicant concluded that with these clarifications and additional information and with the requirements that will be included in the operation manual no additional changes to the DCD or a COL item is necessary. The staff agrees that the information provided is consistent with the guidelines necessary to prevent and mitigate water hammer and that including strict venting requirements prior to operation in the operations manual will help to prevent water hammer. However, the applicant stated that these venting requirements are items that will be specified in the operations manual indicating they currently do not exist. During the DC stage, there is no means for the staff to confirm that these venting requirements have been included in the operations manual, which will be written later.

Therefore, in follow-up **RAI 485-3825, Question 03.06.03-19**, the staff requested the applicant to add an appropriate COL item to insure that instructions to prevent and mitigate water hammer for the RCL branch piping and MS lines included in the LBB analyses will be included in the operations manual. Additionally, the applicant should provide a discussion or commitment to minimize the use of elbows and miters to reduce the effects of steam and water hammer. In its response to **RAI 485-3825, Question 03.06.03-19**, dated January 19, 2010, the applicant stated that COL item 10.3(3) was added in Revision 2 of the DCD to address the operating and maintenance procedures to prevent water hammer in the MS lines. The staff confirmed that Revision 3 of the DCD includes COL item 10.3(3). In DCD Tier 2, Subsection 10.3.2.3.1, "Main Steam Piping," Revision 3, the applicant stated that the MS piping to the turbine is routed to minimize bends and elbows. The applicant also stated in DCD Tier 2, Subsection 10.3.2.4.3, "Water (Steam) Hammer Prevention," Revision 3, that the COL applicant is to provide operating and maintenance procedures including adequate precautions to prevent water (steam) hammer, relief valve discharge loads and water entrainment effects in accordance with NUREG-0927 and a milestone schedule for implementation. In its response, the applicant also committed to revise DCD Tier 2, Subsection 3.6.3.3.1 to add a discussion on the operating and maintenance procedures to prevent water hammer and to add a COL item to DCD Tier 2, Section 3.6.4, "Combined License Information." The COL item will require that the COL Applicant develop a milestone schedule for implementation of the operating and maintenance procedures for prevention of water hammer. The applicant also committed to revise DCD Tier 2, Section 1.8, "Interfaces for Standard Design," to add this COL item.

The staff confirmed that the applicant implemented the response to follow-up **RAI 485-3825, Question 03.06.03-19,** by adding the following discussion to Revision 3 of the DCD Tier 2, Section 3.6.3.3.1: "Also, proper operating and maintenance procedures will be performed to prevent water hammer. The COL applicant is to develop a milestone schedule for implementation of the operating and maintenance procedures for prevention of water hammer. The procedures should address the plant operating and maintenance procedures for adequate measures to avoid water hammer due to a voided line condition." The applicant also added the following COL item to DCD Tier 2, Section 3.6.4: "COL 3.6(10) The COL applicant is to develop a milestone schedule for implementation of the operating and maintenance procedures for prevention of water hammer." The staff finds that the applicant's response is acceptable, as the appropriate requirements to prevent water (steam) hammer have been incorporated in DCD Tier 2, Sections 10.3, “Main Steam Supply System,” and 3.6.3 and the applicant also included the necessary requirements for the COL applicant to develop procedures to minimize the potential for water hammer in COL items 10.3(3) and 3.6(10). Accordingly, **RAI 210-1948, Question 03.06.03-5** and follow-up **RAI 485-3825, Question 03.06.03-19, are resolved**.

3.6.3.4.5 Creep Damage in Leak-Before-Break Piping

Creep damage is addressed in DCD Tier 2, Section 3.6.3.3.2, “Creep Damage,” The applicant states materials are selected and operational temperature limits established not to exceed 700 °F (371.1 °C) for ferritic steel and not to exceed 800 °F (426.7 °C) for austenitic stainless steel. Therefore, the applicant states that piping is not susceptible to creep and creep-fatigue. The staff agrees that piping is not susceptible to creep and creep-fatigue with these design criteria.

3.6.3.4.6 Brittle Failure in Leak-Before-Break Piping

The applicant addresses brittle cleavage-type failures over the range of operating temperatures in DCD Tier 2, Subsection 3.6.3.3.7, “Other Mechanisms.” The applicant states that material fracture toughness tests demonstrate the ability of selected piping to perform under the defined operating conditions. They also conclude that these types of failures are not credible in stainless steel piping systems operated within the thermal parameters defined for the RCS. The applicant also indicates that the carbon steel used for the MS piping is resistant to cleavage failure at the defined operating temperatures.

3.6.3.4.7 Fatigue Failures in Leak-Before-Break Piping

In regard to fatigue, DCD Tier 2, Subsection 3.6.3.3.5, ”Fatigue,” addresses LCF and HCF. The applicant states that the Class I piping satisfies the requirements of the ASME Code, Section III and are designed for LCF including thermal stratification. The applicant addresses the high cycle vibrations caused by the RCPs and addresses the operational controls established to minimize vibrations. Monitoring is used to detect and alarm when vibrations exceed acceptable limits. These controls address the potential for vibration-induced fatigue cracking or failure.

The applicant concluded that LCF is not a potential failure mechanism because the piping is designed for fatigue loadings by application of the ASME Code, Section III for normal operation and anticipated transients. The applicant states that thermal stratification is addressed. HCF failures are also considered to have a very low probability due to measures such as monitoring for vibration of small diameter piping.

The applicant addresses the issues described under SRP Section 3.6.3, Section III.10 except for the need for more details regarding vibration fatigue. Although vibration fatigue failures would not contribute to the dynamic loadings of most concern to LBB, such failures could contribute to plant risk by virtue of water spray onto sensitive components. The staff confirmed that the US-APWR design includes good practices as learned from past plant operational experience.

DCD Tier 2, Subsection 3.6.3.3.5 addresses LCF and HCF and states that the US-APWR is designed to address the potential for fatigue failures. In **RAI 210-1948, Question 03.06.03-6**, the staff requested the applicant to identify what specific design features are used to reduce the potential for fatigue failures in addition to the application of the ASME Code, Section III. The staff also requested the applicant to clarify what operational controls are in place for vibration induced fatigue for the US-APWR certified design and to provide additional information on the methods used to mitigate the potential for fatigue failures. In its response to **RAI 210-1948, Question 03.06.03-6**, dated April 9, 2009, the applicant stated that to reduce the potential for LCF failures, the following design features are used: 1) water solid condition during startup and cool down, and 2) application of butt weld instead of socket weld.

The applicant also stated that because RCP vibration induces RCS piping vibration, design features are used to monitor vibrations at shaft, lower and upper frames of the RCP and to alarm when vibration exceeds the limit. This vibration monitoring and associated limits will be specified and controlled in the RCP operating instruction manual. The staff found the overall direction of the applicant's response acceptable but the applicant's response was incomplete. The applicant's statement "water solid condition during startup and cool down" could be described as an operational control/requirement and the brevity of the statement does not make it clear how this reduces the potential for fatigue failures. The applicant did not present a thorough description of the design features used to reduce the potential for fatigue failures and did not go into detail on the design features used to monitor vibrations (other than the RCPs) in the RCS piping. In addition, the applicant should also describe in more detail what measures will be taken to reduce thermal stratification effects in the MS line and the surge line piping. In follow-up **RAI 485-3825, Question 03.06.03-20**, the staff requested the applicant to augment and clarify their original statements and to provide a more thorough response.

In its response to **RAI 485-3825, Question 03.06.03-20**, dated January 19, 2010, the applicant stated that at the horizontal portion of the pressurizer surge line piping thermal stratification is expected to occur and disappear cyclically due to alternate in-surge flow and out-surge flow. This will cause cyclic stress on the pressurizer surge line piping and nozzle. This stress becomes larger if the temperature difference between the pressurizer and RCL hot leg becomes larger, such as during a plant heat-up and cool-down. The applicant clarified that the US-APWR will use a water solid mode at plant heat-up and cool-down, which will result in decreased cyclic stress. The applicant clarified that there are two kinds of operating modes at plant heat-up and cool-down: (1) Steam Bubble Mode: A Pressurizer steam region bubble is formed at the beginning of the startup and disappears at the end of the cool-down in order to prevent low temperature overpressure. The temperature difference between the pressurizer and RCL hot leg becomes relatively large at this operating mode (over 300 °F (149 °C)) and (2) Water Solid Mode: A Pressurizer steam region bubble is formed after RCS temperature is increased to about 350 °F (177 °C) at plant startup and disappears after RCS temperature is decreased to about 350 °F (177 °C) during plant cool down. The temperature difference between the pressurizer and RCL hot leg becomes relatively small during this operating mode (about 100 °F (38 °C)). The applicant clarified the details for the water solid condition and how this approach reduces the cyclic stress, which reduces the potential for fatigue failures.

The applicant clarified that for the MS line; no thermal stratification is expected to occur. For the surge line piping, the applicant clarified that it is hard to prevent thermal stratification since it will occur due to in-surge flow and out-surge flow. Therefore, the US-APWR considers thermal stratification in the stress evaluation of the surge line piping and nozzle. The applicant confirmed that the cumulative usage factor of these portions of the piping is less than one. Given that the applicant has clarified the design features used to reduce the potential for fatigue failures and that the fatigue cumulative usage factor is less than one on these sections of piping the staff finds this response acceptable. The applicant also clarified that in addition to the RCP vibration monitoring, vibration testing will be conducted for the RCL piping system as part of the initial test program (ITP) as described in DCD Tier 2, Section 3.9.2.1, "Piping Vibration, Thermal Expansion, and Dynamic Effects." Vibration testing is implemented to verify that the RCL piping will remain within acceptable limits when subjected to piping vibrations. Therefore, the staff found the response regarding fatigue design features and vibration monitoring and testing acceptable. Accordingly, **RAI 210-1948, Question 03.06.03-6** and follow-up **RAI 485-3825, Question 03.06.03-20, are resolved**.

3.6.3.4.8 Bounding Analysis Methods for Leak-Before-Break Evaluation

The bounding analysis methods used for the LBB analysis are described in DCD Tier 2, Subsection 3.6.3.4.4. The applicant states that the methods and criteria used are consistent with the guidelines in NUREG-1061 and SRP Section 3.6.3. The bounding analysis methods are described in detail in DCD Tier 2, Appendix 3B. The calculation methods and criteria methods described in DCD Tier 2, Section 3.6.3.4 appear to be fully consistent with the leak-before-break evaluation procedures of the Standard Review Plan 3.6.3. The LBB evaluation therefore appears to be acceptable based on the review that was performed following the guidance given in the SRP. However, many details of how these procedures have been (or will be) implemented could not be completely reviewed for the initial DCD Revision 1 submittal because:

- The initial submittal made reference to a detailed report to be prepared later that will fully describe LBB evaluations for the US-APWR (**RAI 210-1948, Question 03.06.03-7**).

- The initial submittal presented the results of LBB calculations in terms of "BAC." A complete numerical verification of the curves of DCD Tier 2, Revision 1, Figures 3B-7 through 3B-17, "US-APWR BAC for Main Steam Line," was beyond the scope of the initial review of the US-APWR DCD. Nevertheless, the curves appeared to have reasonable data and trends. (**RAI 210-1948, Questions 03.06.03-9 through 03.06.03-12**).

- Much of the initial evaluation procedure (including inputs for critical material and fracture properties) made reference to Revision 2 of the DCD for the Economic Simplified Boiling Water Reactor (ESBWR) plant design. The LBB portion of the ESBWR was subsequently deleted in Revision 4 of the ESBWR DCD. It was not clear if NRC staff ever found the material properties and calculation procedures of the ESBWR to be fully acceptable and consistent with SRP Section 3.6.3 and thereby suitable for uncritical use for the US-APWR DCD. (**RAI 210-1948, Question 03.06.03-9**).

- Details regarding as-built piping configurations, materials/welding, and calculated stresses will become available at a later date during the COL process.

The staff notes that **RAI 210-1948, Questions 03.06.03-7 through 03.06.03-12** are discussed in the next several subsections of this report and have been resolved.

Specific items of the US- APWR DCD were reviewed and found to be acceptable and consistent with SRP Section 3.6.3 as follows:

- The fracture mechanics equations that were used to calculated critical crack lengths are the same as those recommended in SRP Section 3.6.3.

- Consistent with SRP Section 3.6.3, the evaluation for LBB uses the Z-factor approach to account for the relatively low toughness of stainless steel flux type welds as compared to the toughness of stainless steel parent metal.

- The DCD is consistent with SRP Section 3.6.3 in using lower bound values for yield and ultimate strengths taken from ASME Code, Section III for application in limit load calculations for critical flaw sizes in stainless steel piping.

- The DCD is consistent with SRP Section 3.6.3 as it conservatively combines loads (pressure, dead weight, thermal expansion and seismic) in an absolute sense for purposes of calculating critical crack lengths. The use of a safety factor of 1.0 rather than 1.4 is acceptable given the use of the absolute sum approach. The DCD I is consistent with SRP Section 3.6.3 in combining loads as an algebraic sum for purposes of calculating the crack opening areas that enter into leak rate calculations.

- The DCD correctly applies a tearing instability method for the ferritic steel of the MS piping. Also consistent with SRP Section 3.6.3, the calculations conservatively evaluate critical crack lengths on the basis of fracture toughness of the weld metal in combination with the lower stress strain curves for the adjacent base metal.

- The DCD appropriately calculates leak rates for through-wall cracks making use of the PICEP computer code.

RAI 210-1948, Question 03.06.03-7 through 03.06.03-14 address issues identified during the staff review of the LBB analysis methods used for the US-APWR and are discussed in the several following sections.

In **RAI 210-1948, Question 03.06.03-7**, the staff inquired about DCD Tier 2, Subsection 3.6.3.4.10, which states that the bounding analysis results will be provided in the technical report (Reference 3.6-24). Reference 3.6-24 was originally titled, "US-APWR Leak-Before-Break Evaluation, MHI Technical Report, Later," which is not a usable reference. The staff requested the applicant to clarify when this report will be provided to the NRC. The staff also requested the applicant to provide additional information and indicate if this report is part of the COL applicant's responsibility or if this report is to be provided as part of the DCD documentation. If the technical report is part of the DCD documentation, the applicant was asked to provide a copy for the staff to review. In addition, the staff requested the applicant to provide copies of a sample of LBB calculations. The staff requested that the sample calculations include those for the MS Line, the Pressurizer Surge Line and RCS Loops and should include both high stress and low stress conditions. In its response to **RAI 210-1948, Question 03.06.03-7**, dated April 9, 2009, the applicant stated that three technical reports were issued as Revision 0 documents on March 31, 2009, which included the LBB evaluations. The technical reports are MUAP-09010-P, MUAP-09011-P, and MUAP-09013-P, which provide summaries of stress analysis results for RCL piping, RCL branch piping, and MS piping, respectively. The technical reports were provided by letter dated March 31, 2009, and were reviewed by the staff as described below.

3.6.3.4.9 Confirmatory Leak-Before-Break Analysis

The staff reviewed the LBB analysis methods and its contractors from Pacific Northwest National Laboratory performed a confirmatory LBB analysis for the RCL piping (Johnson and Gates, 2012. “Confirmatory Leak Before Break Evaluation of the US-APWR Reactor Coolant Loop Piping, PNNL-21242.”). The analysis included the leakage crack size and limit load calculations necessary to construct the BAC plus to calculate the operating stress versus maximum stress points for comparison with the BAC. The LBB confirmatory analysis (Johnson and Gates 2012) compares very closely with the LBB results documented by the applicant in MUAP-09010-P, Revision 1. Therefore, the staff agrees with the LBB methodology used, and that the applicant has correctly applied the methodology.

The LBB analysis of the RCL piping was updated in MUAP-09010-P, Revision 3. Review of MUAP-09010-P, Revision 3 shows that the BACs have higher maximum stresses than in MUAP-09010-P, Revision 1. This occurs due to the change in material specification for the RCL piping. MUAP-09010-P, Revision 3 specifies SA-336 grade F316 piping material, which has higher yield and ultimate tensile strengths at the higher temperatures than the SA-182 and SA-336 grade F316LN material specified in MUAP-09010-P, Revision 1. This material change results in higher flow stresses in the BAC calculations. Staff review of MUAP-09010-P, Revision 3, Tables 7-5-1, “Member Load of Hot Leg,” 7-5-2, “Member Load of Crossover Leg,” and 7-5-3, “Member Load of Cold Leg,” also shows that section force and moment loads have also been revised. The LBB results in MUAP-09010-P, Revision 3, Figures 11-1, “Hot Leg LBB Evaluation Result,” 11-2, “Crossover Leg LBB Evaluation Result,” and 11-3, “Cold Leg LBB Evaluation Result,” show that the LBB criteria are met with the updated material and applied loads.

Based on the detailed results of the staff review and independent confirmatory analyses, the staff has confirmed that the applicant’s LBB analyses and calculations meet the LBB requirements. The staff concluded the applicant’s LBB analyses and calculations are acceptable. Accordingly, **RAI 210-1948, Question 03.06.03-7, is resolved.**

3.6.3.4.10 Review of Ramberg-Osgood and J-T Curves

In **RAI 210-1948, Question 03.06.03-8**, the staff commented that the DCD Tier 2, Figure 3.6-4 is not consistent with a similar figure in Appendix 3B. DCD Tier 2 Revision 1, Figure 3B-6, “LBB Evaluation Procedure of the U.S. Plant,” in Appendix 3B uses “Break: Restraints” for the no path and “Leak: No restraints” for the yes path. Figure 3.6-4 uses “Not Qualified for LBB” for the no path and “Qualified for LBB” for the yes path. The staff requested the applicant to correct these figures so they are consistent or provide justification for why they should be different. In its response to **RAI 210-1948, Question 03.06.03-8**, dated April 9, 2009, the applicant clarified that DCD Tier 2, Figure 3.6-4 is correct and that DCD Tier 2, Figure 3B-6 will be corrected to be consistent with DCD Tier 2, Figure 3.6-4. The applicant provided a markup of the DCD Tier 2, Section 3.6 revisions that will be included in Revision 2 of the DCD. The markup states that the last two boxes of figure 3B-6 will be changed to “Not qualified for LBB” for the “No” condition and “Qualified for LBB” for the “Yes” condition. This response is acceptable as it corrects the inconsistency between these two figures. The staff verified that the applicant completed the above changes and notes that in DCD Revision 2 that DCD Tier 2, Figure 3B-6 was renumbered to be DCD Tier 2, Figure 3B-5, “LBB Evaluation Procedure of the U.S. Plant.” Accordingly, **RAI 210-1948, Question 03.06.03-8, is resolved**.

In **RAI 210-1948, Question 03.06.03-9** the staff commented on the reference the applicant cited as the source for the Ramberg-Osgood stress-strain curve and the J-T curve used for the LBB evaluation of the MS piping. DCD Tier 2, Appendix 3B cites Reference 3B-16 titled “ESBWR Design Control Document, 26A6642AL Revision 2, 2006” Appendix 3E. This document is the source for the Ramberg-Osgood stress-strain curve and the J-T curve used for the evaluation of LBB for the MS piping. Revision 4 of the ESBWR DCD removed Appendix 3E because LBB will not be used for the ESBWR. In **RAI 210-1948, Question 03.06.03-9,** the staff requested the applicant to provide an alternative reference as an appropriate source for the Ramberg-Osgood and J-T curves that will be applied to develop BAC curves for the US-APWR MS piping. In its response to **RAI 210-1948, Question 03.06.03-9**, dated April 9, 2009, the applicant stated:

> MHI has recognized that arbitrary specification of the material properties based on the ESBWR DCD and requiring final verification that the properties are met at the time of plant construction, represents a risk. As a result, an alternate approach has been developed whereas minimum acceptable material properties will be specified for procurement of main steam line piping materials and qualification of the welding processes. Fracture mechanics instability analysis has been conducted for a range of material yield and ultimate tensile strength properties based on the stress-strain curve shape using ASME Code minimum properties. From this analysis, a minimum required J-T curve is developed that is a function of the actual measured base metal yield and ultimate tensile strengths at the time of piping procurement. The procedure has been evaluated against ferritic piping and weld material data that are published in the “Pipe Fracture Encyclopedia, Test Data - Volume 3” (USNRC, December 1997) and summarized in Appendix B of NUREG/CR-6004, where it was shown that many of the materials tested would have been acceptable (except that many of the specimens tested were much thinner than the testing to be specified for the main steam line procurement and the J-R curves were quite limiting due to the small specimen size). The outline of this procedure is presented in MHI Technical Report MUAP-09013 (RO) "Summary of Stress Analysis Results for the US-APWR Main Steam Piping inside Containment Vessel" submitted in March 2009.

The staff finds that the overall direction of the applicant’s response was acceptable as they did provide clarification, a better approach and alternate references for acceptable material properties for the MS line piping. But the applicant’s response did not address necessary corrections to the DCD. There was no discussion on revising the text and deleting the reference to the ESBWR DCD in Appendix 3B of the US- APWR DCD. In follow-up **RAI 485-3825, Question 03.06.03-21**, the staff requested the applicant to revise US- APWR DCD Appendix 3B to include the approach described in their response to **RAI 210-1948, Question 03.06.03-9** and to add references to the, “Pipe Fracture Encyclopedia, Test Data – Volume 3,” US NRC, issued December 1997 and to NUREG/CR-6004, “Probabilistic Pipe Fracture Evaluations for Leak-Rate Detection Applications,” Appendix B, issued April 1995.

In its response to **RAI 485-3825, Question 03.06.03-21**, dated January 19, 2010, the applicant stated that Revision 2 of DCD Tier 2, Appendix 3B now includes the approach and acceptable material properties as described in NUREG/CR-6004 and the Pipe Fracture Encyclopedia, Test Data – Volume 3 US NRC. The staff also confirmed that the applicant revised DCD Tier 2, Appendix 3B to describe the methodology. The J-T methodology is one of elastic-plastic fracture mechanics consistent with the methods described in NUREG-1061 and SRP Section 3.6.3. The actual material data is to be obtained from the same material to be used, but has not yet been produced. Therefore, the applicant used a conservative approach to perform the analysis using ASME Code minimum properties based on ASME Section II, Part D. The required fracture toughness properties for the base metal and welding will be provided to suppliers to assure that material properties will meet the requirements and to ensure the BAC requirements will be met. The applicant has described an acceptable fracture mechanics approach and has provided references and comparisons with acceptable methods and experimental data as described in NUREG-1061, the ASME Code and SRP Section 3.6.3. The applicant also clarified that the reference number for NUREG/CR-6004 will be corrected to Reference 3B-13 and DCD Tier 2, Subsection 3B.2.2.2, "Fracture Mechanics Analysis," will directly reference Appendix B of NUREG/CR-6004 and the Pipe Fracture Encyclopedia, Test Data – Volume 3. The Pipe Fracture Encyclopedia, Test Data – Volume 3 will be added as Reference 3B-14. The staff confirmed that Revision 3 of DCD Tier 2, Subsection 3B.2.2.2 incorporates a review of the curves against actual test data from Appendix B of NUREG/CR-6004 [Reference 3B-13] and Pipe Fracture Encyclopedia, Test Data – Volume 3 [Reference 3B-14] and the applicant has shown that the J-T curves should be achievable. The staff confirmed that Revision 3 of the DCD includes the correct references for the conservative material properties used in the J-T fracture analysis of the MS piping. Based on the above, the staff found the applicant's responses acceptable. Accordingly, **RAI 210-1948, Question 03.06.03-9** and follow-up **RAI 485-3825, Question 03.06.03-21, are resolved**.

3.6.3.4.11 Review of Bounding Analysis Curves

In **RAI 210-1948, Questions 03.06.03-10**, the staff asked the applicant about the BAC curves of DCD Tier 2, Revision 1, Figures 3B-11, "US-APWR BAC for Surge Line," through 3B-17, which show a lower cutoff on the normal stress axis that serves as a minimum value of normal stress for the curve. DCD Tier 2, Section 3B.3.1.1, "BAC Methodology," describes the steps used to construct the BAC plots and cites "Case 1" that equates this minimum stress value to the membrane stress P_m due to internal pressure under normal operation. Most of the curves of DCD Tier 2, Revision 1, Figures 3B-11 through 3B-17 appear to be consistent with this definition. However, the staff identified that some of the Figures appeared to be inconsistent. For example, DCD Tier 2, Figure 3B-13, "US–APWR BAC for RHRS," shows a lower limit value of about 3.9 ksi whereas the NRC review has calculated a value of 5.3 ksi ($P_m = pD_o/4t = 2235 \times 10.74/4 \times 1.125 = 5.3$ ksi). In **RAI 210-1948, Question 03.06.03-10,** the staff requested the applicant to provide additional information and resolve the source of this apparent inconsistency.

In its response to **RAI 210-1948, Questions 03.06.03-10**, dated April 9, 2009, the applicant stated that the membrane stress due to internal pressure is obtained by the equation for the axial direction stress of a thick cylinder, $Pm=p/(k^2-1)$ where $k=b/a$, b is the outer radius, and a is the inner radius. The applicant stated that this equation is derived from the force equilibrium under closed conditions at both axial ends simulating a continuous pipe configuration. The staff concurs that this approach yields a more realistic but smaller stress value. Also, the applicant indicated that for the BAC curves, when the operational stress (the horizontal axis) becomes small, the corresponding allowable maximum stress (the vertical axis) also becomes small. Therefore, the range of the achievable LBB condition does not change. The applicant also indicated that for the piping design, the membrane stress is conservatively obtained by the equation included within the associated NRC question. The applicant's response did not appear to be consistent. The applicant clarified that they used an equation that results in a more realistic but smaller stress value. However, use of a more realistic smaller stress value does not appear to be conservative when considering the BAC methodology. As stated in DCD Tier 2, Appendix 3B, "The area below the BAC is a leak mode and that beyond the BAC is a failure mode." Using an approach that results in a smaller stress value would appear to reduce the "failure mode" area, which may not be conservative in all cases. The applicant also stated, "For the piping design, the membrane stress is conservatively obtained by the equation included within the associated NRC question" which is not consistent with the earlier discussion that a more realistic smaller stress value using the closed end assumptions were used.

In follow-up **RAI 485-3825, Question 03.06.03-22**, the staff requested the applicant to provide additional information to resolve these apparent inconsistencies in their response and to define the most conservative approach for calculating the membrane stress. In its response to **RAI 485-3825, Question 03.06.03-22**, dated January 19, 2010, the applicant states that for the BACs of DCD Tier 2, Revision 1, Figures 3B-11 through 3B-17, the maximum stress and the normal stress are not independent. The applicant clarified that using a lower normal stress increases the leakage crack length, which in turn reduces the maximum allowable stress. Therefore, for a lower normal stress, the BAC shifts to the left. The applicant stated that this actually changes the domain of stress combinations where LBB can be applied rather than reducing the failure area for a given range of normal stress. This clarifies the staff's concern regarding a conservative selection of normal stresses. The applicant also clarified that the BAC is based on the smaller realistic stress for both the normal stress case and the maximum stress case, thus resulting in the "failure mode" area that incorporates the minimum normal stress that can exist for a given system (the heat up/cool down surge line BAC in DCD Tier 2 Revision 1, Figure 3B-13). The applicant also clarified that the statement that the membrane stress was calculated using the more conservative Code method only applies to the ASME Code piping analysis, not the LBB analysis. The applicant also clarified that the BAC curves are based on nominal wall thickness, whereas the piping assessment is based on the minimum wall thickness providing further conservatism. The applicant's response is acceptable as it clearly indicates that they use a minimum normal stress, which increases the leakage crack length, which in turn reduces the maximum allowable stress for the critical crack and is a conservative approach. Based on the above, the staff found the applicant's clarifications acceptable. Accordingly, **RAI 210-1948, Question 03.06.03-10** and **follow up RAI 485-3825, Question 03.06.03-22, are resolved**.

3.6.3.4.12 Review of Leak-Before-Break for Main Steam Line Piping

In **RAI 210-1948, Question 03.06.03-11**, the staff asked applicant the about the basis for performing the LBB evaluation on the MS piping. The BAC curve for the MS piping addresses the operating temperature of 535 °F (279 °C). SRP Section 3.6.3 (Section III.11.B.iv) cites the need for calculations to address possible fracture at temperatures lower than the temperature of normal operation (e.g. hot standby). These calculations would account for the possibility of reduced toughness at the lower temperatures. In **RAI 210-1948, Question 03.06.03-11**, the staff requested the applicant to provide additional information on the basis for performing the LBB evaluation for the MS piping only for the normal operating temperature.

In its response to **RAI 210-1948, Question 03.06.03-11**, dated April 9, 2009, the applicant stated,

> The requirements from SRP Section 3.6.3 assure that the resistance to fracture at lower temperatures where conditions (e.g. hot standby) may exist that would present safety concerns similar to normal operation. However, the main steam line always operates at saturated conditions. The following provides the thermodynamically possible temperature-pressure conditions for reduced operating temperatures, along with the ratios of the pressure and thermal expansion moment for reduced temperature conditions:"

T, °F	P, psig	P/Pmax	Mt/Mtmax*
535	907	1.0	1.0
500	665	0.73	0.92

400	232	0.26	0.71
300	50	0.06	0.48
200	0	0.00	0.28

*Ratio of reduced temperature thermal expansion moment to maximum thermal expansion moment

This shows that the loads on the piping will significantly reduce for reduced temperature conditions. This is not necessarily the case for reactor coolant piping, where the plant may operate at elevated pressure (based on elevated pressurizer temperature) for conditions of low temperature in the piping. This provides justification for only evaluating the main steam line J resistance curves at the maximum operating conditions.

The staff found that the overall direction of the applicant's response acceptable but the applicant's response was incomplete. The applicant clarified the reduction in the loads for the MS line but did not address the question of material toughness. Specifically, the reduced toughness of the MS line material at lower temperatures. It is understood that the pressures and hence the loads will be reduced, however, the applicant should clarify whether or not the reduced material toughness for lower temperature conditions would be a concern even under the lower loads (is the toughness decreasing faster than the loads to be a concern)? Also the response does not make clear the basis for Pmax and Mtmax, is it material toughness or some other material limitation.

In follow-up **RAI 485-3825, Question 03.06.03-23**, the staff requested the applicant to clarify their response regarding the MS line material toughness to ensure SRP Section 3.6.3, Subsection III.11.B.iv has been properly addressed. In its response to **RAI 485-3825, Question 03.06.03-23**, dated January 19, 2010, the applicant clarified that Pmax and Mtmax are the maximum pressure and maximum thermal expansion moment at the lower temperature operating conditions. These terms relate to the loading, not to material properties. The applicant clarified that the dead weight moment stress, normally quite small, remains the same with reduced operating temperatures. The applicant clarified that the load reduction would be significant and it is expected that the reduction in toughness would not be nearly as significant. In its response, the applicant provided data from the Pipe Fracture Encyclopedia, Test Data – Volume 3, US NRC, issued December 1997 for SA-333 Grade 6 carbon steel that shows that the decrease in absorbed energy in a Charpy V-Notch impact test (a measure of fracture toughness) is small except at sub-zero temperatures. The staff found the applicant's response acceptable as the applicant demonstrated for the MSline material that the decrease in fracture toughness with reducing temperature and loads is not a concern. Accordingly, **RAI 210-1948, Question 03.06.03-11** and **follow-up RAI 485-3825, Question 03.06.03-23, are resolved**.

3.6.3.4.13 Evaluation of Bounding Analysis Curve for Pressurizer Surge Line

In **RAI 210-1948, Question 03.06.03-12**, the staff identified that two BAC plots are provided in DCD Tier 2, Section 3.6.3 for the surge line. DCD Tier 2 Revision 1, Figure 3B-11 is for normal operation at a pressure of 2235 psi (15.41 MPa) and a temperature of 653° F (345 °C), whereas DCD Tier 2 Revision 1, Figure 3B-12, "US–APWR BAC for Surge Line," is for a pressure of 400 psi (2.76 MPa) and a temperature of 449 °F (232 °C). In **RAI 210-1948, Question 03.06.03-12**, the staff asked the applicant about the significance of the loading condition at the lower temperature and pressure used for DCD Tier 2 Revision 1, Figure 3B-12. The staff requested the applicant to provide additional information and the rationale for selecting this loading condition. In its response to **RAI 210-1948, Question 03.06.03-12**, dated April 9, 2009, the applicant stated that this condition represents the time during heatup and cooldown of the plant, where the pressurizer is heated (the saturation pressure at 449 °F (232 °C) is 418 psia (2.88 MPa)) and the RCL piping temperature may be relatively low (especially during heatup). For this condition, the maximum thermal stratification can exist in the surge line piping. Thus, the maximum stress condition represents the combination of the stated pressure condition with the associated dead weight and thermal expansion loads (including thermal stratification). The curve in the DCD evaluated the normal stresses for this same condition assuming that the leakage would be detectable for this low temperature condition. The applicant stated this curve is in the process of being revised to reflect that the normal operating stress will be that associated with normal plant operation, since it is during the normal plant operating conditions that the leakage must be detected, not taking credit for the stratification stresses that will occur at the low temperature conditions. The staff found that the direction of the applicant's response was acceptable as they did clarify the load conditions but the response needed to be clarified to ensure that they will be selecting load combinations that will result in the least favorable conditions for normal operations. Also, consistent with the SRP Section 3.6.3, Section III.11.B.iv, the applicant must use material properties (toughness) that reflect the least favorable conditions considering both normal operation and conditions like hot standby where pipe break would present safety concerns similar to normal operation. The applicant stated that they are in the process of revising the BAC curve to reflect the normal operating stresses.

In follow-up **RAI 485-3825, Question 03.06.03-24**, the staff requested the applicant to provide additional information and clarify that the revised BAC curve will reflect the least favorable conditions for stresses and material properties. In its response to **RAI 485-3825, Question 03.06.03-24**, dated January 19, 2010, the applicant stated that the condition chosen for normal operation, with no stratification, is the condition where minimum leakage would occur during plant normal operation. This loading combination considers only the pressure, thermal expansion and dead weight stresses in the piping, leading to the minimum crack opening displacement for purposes of leak detection. The applicant clarified that the BAC curves had been revised as those submitted were based on a leakage determination at a transient condition at hot standby. The applicant stated that the revised BAC for normal operation in DCD Tier 2 Revision 2, Figure 3B-12, "US–APWR BAC for Surge Line (Normal Operation)," at 653 ºF (345 ºC) is more conservative than the BAC for the thermal stratification case as the material properties become less favorable at higher temperatures. The applicant chose to remove the lower temperature DCD Tier 2 Revision 2, Figure 3B-13, "US-APWR BAC for Surge Line (Heatup/Cooldown)," from the DCD and use only DCD Tier 2 Revision 2, Figure 3B-12 for the surge line BAC for all load combinations (with or without thermal stratification).

The staff concurs with the applicant's conclusion that the above approach will result in lower bound tensile properties (yield and ultimate strength) when evaluating the surge line with and without thermal stratification as the tensile properties will be lower at the higher temperatures of DCD Tier 2 Revision 2, Figure 3B-12 (653 ºF (345 ºC) vs. 449 ºF (232 ºC)). The staff concurs with the applicant's conclusion that the yield and ultimate tensile strength for stainless steel do not decrease at lower temperature. However, the staff took issue with the applicant's contention that SRP Section 3.6.3, Section 11.B.iv is not applicable to stainless steel materials. In fact this section of the SRP clearly states, "Industry generic data bases must provide a reasonable lower bound for the population of material tensile and toughness properties associated with any individual specification (e.g. A106, Grade B), material type (e.g., austenitic stainless steel) or welding procedures. The staff agrees that the applicant in their response has provided a reasonable approach for the lower bound for tensile properties. However, the applicant did not address toughness properties, which can decrease with decreasing temperature. In a follow-up **RAI 485-3825, Question 03.06.03-24** (original **RAI 210-1948, Question 03.06.03-12**) the staff requested that the applicant re-assess its response to state clearly that lower bound toughness properties will be used when performing the LBB evaluation on the pressurizer surge line; including load combinations with and without thermal stratification.

The staff reviewed the pressurizer surge line LBB analysis in MUAP-09011-P, Revision 2. MUAP-09011-P, Revision 2, Table A1-1-1-1, "Block division and piping specifications," (lists the pressurizer surge line piping material as austenitic stainless steel grade SA-312 (seamless) TP316. MUAP-09011-P, Revision 2, Section A2.2.2, "Fracture Mechanics Analysis," states that because austenitic stainless steel has a high fracture toughness, limit load methodology can be applied to evaluate the fracture behavior of the piping. Limit load analysis is based on the flow stress at temperature rather than the fracture toughness. The toughness of austenitic stainless steels is high enough that the limiting condition is net section plastic flow rather than fracture. The applicant conservatively calculates the flow stress as the average of the yield and ultimate strengths at temperature. The applicant also uses the yield and ultimate strengths at 653 °F (345 ºC), which are lower than the strengths at 449 °F (232 ºC). Therefore, the BAC is calculated using a lower bound strength criterion, which is more limiting than a fracture based criterion. The staff agrees with the applicant's use of the lower strengths corresponding to the higher temperature. This produces a conservative BAC, which the applicant uses for the LBB analysis of all loading conditions, including combinations with and without stratification, over the range of temperature. The staff confirmed that the applicant is using a conservative limit load criterion for the pressurizer surge line LBB analysis. The criterion is based on net section plastic flow rather than on fracture mechanics. The staff found the applicant's response acceptable. Accordingly, **RAI 210-1948, Question 03.06.03-12** and **follow-up RAI 485-3825, Question 03.06.03-24, are resolved**.

In **RAI 210-1948, Question 03.06.03-13**, the staff requested the applicant to provide a single table that lists all piping that is to be addressed by LBB evaluations giving key parameters including systems, pipe diameters/wall thicknesses, piping materials, normal operating pressures, and normal operating temperatures. In its response to **RAI 210-1948, Question 03.06.03-13**, dated April 9, 2009, the applicant stated that DCD Tier 2, Revision 1, Appendix 3B Table 3B-2, "List of BAC for LBB Evaluation," lists the piping systems selected for LBB evaluations and associated BACs. The staff finds this response acceptable as DCD Tier 2, Revision 1, Table 3B-2, provides the key parameters including systems, pipe diameters/wall thicknesses, piping materials, normal operating pressures and normal operating temperatures used in the analyses resulting in the detailed BACs shown in DCD Tier 2, Revision 1, Figures 3B-7 through 3B-17. Therefore, the staff found the applicant's response acceptable. Accordingly, **RAI 210-1948, Question 03.06.03-13, is resolved**.

3.6.3.4.14 Leak-Before-Break Evaluation of Ferritic Steel Piping Systems

In **RAI 210-1948, Question 03.06.03-14**, the staff asked the applicant about DCD Tier 2, Revision 1, Table 3B-1, "Ramberg-Osgood Parameters at 550°F Material Constant," and Figure 3B-5, "$(J\text{-}T)_{mat}$ Curve at 550° F (Reference 3B-16)," which provide critical inputs to the LBB evaluation for the MS piping. This is the only piping in the submittal that is constructed of ferritic steel and which therefore requires the more complex tearing instability calculations. The applicant was asked to justify that these inputs provide a conservative or bounding basis for the LBB calculations. The applicant was also asked to provide additional information and the steps that will be taken to verify that the selected Ramberg-Osgood stress strain curve and $(J\text{-}T)_{mat}$ curve are suitable bounds for the properties of the as-built MS piping.

In its response to **RAI 210-1948, Question 03.06.03-14**, dated April 9, 2009, the applicant stated:

> MHI has recognized that arbitrary specification of the material properties based on the ESBWR DCD and requiring final verification that the properties are met at the time of plant construction, represents a risk. As a result, an alternate approach has been developed whereas minimum acceptable material properties will be specified for procurement of main steam line piping materials and qualification of the welding processes. Fracture mechanics instability analysis has been conducted for a range of material yield and ultimate tensile strength properties based on the stress-strain curve shape using ASME Code minimum properties. From this analysis, a minimum required J-T curve is developed that is a function of the actual measured base metal yield and ultimate tensile strengths at the time of piping procurement. The procedure has been evaluated against ferritic piping and weld material data that are published in the Pipe Fracture Encyclopedia, Test Data - Volume 3 (USNRC, December 1997) and summarized in Appendix B of NUREG/CR-6004, where it was shown that many of the materials tested would have been acceptable (except that many of the specimens tested were much thinner than the testing to be specified for the main steam line procurement and the J-R curves were quite limiting due to the small specimen size). The outline of this procedure is presented in MHI Technical Report MUAP-09013-P (R0) "Summary of Stress Analysis Results for the US-APWR Main Steam Piping inside Containment Vessel" submitted in March 2009.

The staff found that the overall direction of the applicant's response was acceptable as they did provide clarification, a better approach and alternate references for acceptable material properties for the MS line piping. But the applicant's response did not address necessary corrections to the DCD. There was no discussion on revising the text and deleting the reference to the ESBWR DCD in Appendix 3B of the US-APWR DCD. In **follow-up RAI 485-3825, Question 03.06.03-25,** the staff requested the applicant to revise DCD Tier 2, Appendix 3B to include the approach described in its response to **RAI 210-1948, Question 03.06.03-14** and to add references to the Pipe Fracture Encyclopedia, Test Data – Volume 3 US NRC, issued December 1997 and to NUREG/CR-6004 Appendix B. In its response to **RAI 485-3825, Question 03.06.03-25**, dated January 19, 2010, the applicant referred to the response to **RAI 485-3825, Question 03.06.03-21**. In its response to **RAI 485-3825, Question 03.06.03-21**, the applicant provided the changes requested by the staff that the staff found acceptable, as described above. The staff confirmed that DCD Tier 2, Revision 3, Section 3B.2.2.2 incorporates a review of the curves against actual test data from Appendix B of NUREG/CR-6004. Accordingly, **RAI 210-1948, Question 03.06.03-14** and **follow-up RAI 485-3825, Question 03.06.03-25, are resolved**.

3.6.3.4.15 Leak Detection Systems for Leak-Before-Break

The applicant addresses the leak detection systems in DCD Tier 2, Section 3.6.3.2 and states that the leak detection systems meet the guidelines of RG 1.45. This is consistent with SRP Section 3.6.3 which states that the specifications for plant-specific leak detection systems inside containment should be equivalent to those in RG 1.45. The applicant does not apply LBB methods to piping outside of containment so no leak detection systems are necessary outside containment to support the LBB analyses. The staff issued three RAIs to address issues regarding leak detection. With incorporation of the applicant responses to the RAIs into DCD

Revision 3 as described below, the applicant's approach in DCD Revision 3 for leak detection systems is consistent with the SRP guidance.

In **RAI 217-2025, Question 03.06.03-15**, the staff requested the applicant demonstrate that the design of leak detection capability is adequate to support the margin of 10 to the postulated leakage rate for LBB application in RCL branch piping and MS piping. In its response to **RAI 217-2025, Question 03.06.03-15**, dated March 24, 2009, the applicant acknowledged that a leak detector capability of one gpm (4 L/min) is not adequate to satisfy the LBB margin of 10 from analysis results of LBB application in the RCL branch piping and MS piping. To satisfy the LBB margin of 10, the applicant proposed to change the required leak rate and required leak detection capability to 5 gpm (20 L/min) and 0.5 gpm (2 L/min) respectively, as the leak detection instruments have a capability to detect leakage lower than 0.5 gpm (2 L/min) within one hour of detector response time. The affected DCD sections included Tier 2, Section 5.2.5, "Reactor Coolant Pressure Boundary (RCPB) Leakage Detection," Chapter 16, and Section 3.6. The staff reviewed the proposed changes regarding their application to the RCPB leakage detection system in DCD Tier 2 Section 5.2.5 and associated TSs in Chapter 16, and determined that the applicant's response and proposed changes to the DCD meet the guidelines of RG 1.45. The staff finds the 0.5 gpm (2 L/min) detection capability acceptable, as described in Section 5.2.5 of this report. The leak detection capability of 0.5 gpm (2 L/min) also is adequate to satisfy the LBB margin of 10 for the leak rate of 5 gpm (20 L/min). The staff confirmed that DCD Revision 3 incorporates the DCD changes identified in the response to **RAI 217-2025, Question 03.06.03-15** relative to 0.5 gpm (2 L/min) leak detection capability. Accordingly, **RAI 217-2025, Question 03.06.03-15, is resolved**.

In its review of DCD Tier 2, Revision 1, Section 3.6.3.4.1, the staff found that the leak detection methods identified for the RCL piping may not be applicable for the MS piping. DCD Tier 2, Subsection 3.6.3.4.1 states that the method to detect leaks from the MS pipe in containment relies on detecting any increase in the containment sump water level. The condensate water flow rate of containment air cooler, containment pressure, and containment temperatures also provide indication of possible leakage. The containment sump level is qualified for seismic events not requiring a plant shutdown. DCD Tier 2, Section 3.6.3.2 states that leak detection systems meet the guidelines of RG 1.45, Revision 1. The staff reviewed the above information relative to RG 1.45 and identified several issues that did not fully address the guidance. The staff found that there was only one quantitative detection method, containment sump level, to support the MS piping LBB. Further, the seismic qualification of the containment sump level detector was not listed in DCD Tier 2, Table 3.2-2, "Classification of Mechanical and Fluid Systems, Components, and Equipment." In addition, there was no TS for the MS leak detection. In **RAI 217-2025, Question 03.06.03-16**, the staff requested the applicant demonstrate how the methods for the MS leakage detection meet the guidelines of RG 1.45 or its equivalent in terms of instrument capability, sensitivity, diversity (and/or redundancy), seismic qualification, response time, and TS operability requirements.

In its response to **RAI 217-2025, Question 03.06.03-16**, dated April 23, 2009, the applicant provided a markup of the proposed changes to be made to various sections in DCD Tier 2. Specifically, to address the concern of seismic qualification, the applicant stated that the seismic qualification of containment sump level will be changed to seismic Category I. In addition, to address the concern of lacking a TS limit for steam line leakage, the applicant added a new TS, Limiting Condition for Operation (LCO) 3.7.15, "Main Steam Line Leakage," which limits the MS line leakage in the containment to 0.5 gpm (2 L/min). New TS bases for LCO 3.7.15 were provided as well. To address the concern of diversity for MS leakage detection, the applicant modified the description of the leak detection capability in DCD Tier 2, Subsection 3.6.3.4.1, clarifying that "leaks can also be detected by the condensate water flow rate of containment air cooler" and other means, such as containment pressure and temperature measurements suggest the possibility of leakage. The staff reviewed these changes and determined that the MS leakage detection meets the guidelines of RG 1.45 in terms of instrument capability, sensitivity, diversity, response time, and TS requirements. Therefore, **RAI 217-2025, Question 03.06.03-16** was resolved with one exception discussed below.

The applicant stated that the containment sump level monitor seismic qualification will be changed to seismic Category I. However, the applicant did not provide a markup in the classification table of the DCD as part of its response to **RAI 217-2025, Question 03.06.03-16**. In follow-up **RAI 414-3078 Question 03.06.03-17**, the staff requested that the applicant include the containment sump level monitoring system in the DCD for the seismic category classification. In its response to **RAI 414-3078 Question 03.06.03-17**, dated August 3, 2009, the applicant committed to a revision of DCD Tier 2, Table 3D-2, "US-APWR Environmental Qualification Equipment List," to add the containment sump water level instrument to be seismic Category I. The staff found the changes to the DCD acceptable since the applicant addressed the seismic classification of the containment sump level monitor. The staff confirmed all of the DCD changes identified in the responses to **RAI 217-2025, Question 03.06.03-16** and **RAI 414-3078, Question 03.06.03-17** has been completed. Accordingly, **RAI 217-2025, Question 03.06.03-16** and **follow-up RAI 414-3078, Question 03.06.03-17, are resolved**.

3.6.3.5 Combined License Information Items

The following is a list of COL item numbers and descriptions from Table 1.8-2 of the DCD related to leak-before-break evaluation procedures:

Table 3.6.3-1
US-APWR Combined License Information Items

Item No.	Description	Section
COL 3.6(10)	The COL Applicant is to develop a milestone schedule for implementation of the operating and maintenance procedures for water hammer protection.	3.6.3.3.1

The staff's evaluation of the adequacy and acceptability of the above listed COL Information Item 3.6(10) is addressed in Section 3.6.3.4.4 of this report above.

3.6.3.6 Conclusions

The staff concludes that the LBB analyses and results adequately demonstrate that the probability of pipe rupture is extremely low for the systems selected and satisfy the requirements of GDC 4. The staff determination is based on the following:

1. The probability of failures to water hammer, corrosion creep, fatigue, erosion, environmental conditions, and indirect sources are extremely low. The extremely low probability is based on deterministic evaluations that demonstrate structures and components are adequately engineered and meet the applicable regulations and NRC-endorsed industry codes.

2. A deterministic fracture mechanics evaluation has been completed, verified, and approved by the staff.

3. Leak detection systems are sufficiently reliable, redundant, diverse and sensitive, and that sufficient margin exists to detect the through-wall flaw used in the deterministic fracture mechanics evaluation.

The staff concludes for the systems selected that the LBB acceptance criteria have been satisfied, and, that the dynamic effects of pipe rupture may be excluded from the design-basis for these piping systems.

3.7 Seismic Design

[This section is still under review and the SE will be provided at a later time]

3.8 Design of Category I Structures

[This section is still under review and the SE will be provided at a later time]

3.9 Mechanical Systems and Components

3.9.1 Special Topics for Mechanical Components

3.9.1.1 Introduction

This report provides the staff's evaluation of the applicant's special topics for mechanical components for the US-APWR DC. The staff's evaluation considered if it meets the requirements, codes and standards, and the regulatory guidance on the methods of analysis for seismic Category I components and supports, including both those designated as Class 1, 2, 3, or core support by the ASME B&PV Code, Section III, and those not covered by the Code. This section describes design transients for Code Class 1 and core support components and supports.

3.9.1.2 Summary of Application

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

DCD Tier 2: The applicant has provided a DCD Tier 2 description in Section 3.9.1, "Special Topics for Mechanical Components," summarized here in part, as follows:

DCD Tier 2, Section 3.9.1.1, "Design Transients," describes the design transients for each of five service or test conditions defined in ASME Code, Section III and the frequencies (number of cycles) for each transient assumed in the Code design and fatigue analyses of RCS Class 1 components, auxiliary Class 1 components, RCS component supports, and reactor internals. The number of cycles assumed for each design transient was based on a 60 year design life.

DCD Tier 2, Section 3.9.1.2, "Computer Programs Used in Analyses," identifies the computer programs that are used for static, dynamic, and hydraulic transient analyses of mechanical system components.

DCD Tier 2, Section 3.9.1.3, "Experimental Stress Analysis," states that experimental stress analysis is not used for the US-APWR.

DCD Tier 2, Section 3.9.1.4, “Considerations for the Evaluation for the Faulted Condition,” states that analytical methods used to evaluate faulted condition (Level D loading) for ASME Code, Section III, Class 1, 2, and 3 components are described in Subsection 3.9.3.

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

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

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

Technical Reports: Technical reports associated with DCD Tier 2, Section 3.9.1 are as follows:

1. MUAP-07012-P, “LOCA Mass and Energy Release Analysis Code Applicability Report for US-APWR,” Revision 2, issued May 2008.

2. MUAP-07013P, “Small Break LOCA Methodology for US-APWR,” Revision 2, issued October 2010.

3. MUAP-09001-P, “Summary of Design Transient,” Revision 0, issued January 2009.

4. MUAP-09002-P, “Summary of Seismic and Accident Load Conditions for Primary Components and Piping,” Revision 0, issued January 2009.

5. MUAP-09004-P, “Summary of Stress Analysis Results for Core Support Structures,” Revision 1, issued January 2011.

6. MUAP-09005-P, “Summary of Stress Analysis Results for Reactor Vessel,” Revision 2, issued March 2011.

7. MUAP-09006-P, “Summary of Stress Analysis Results for Steam Generator,” Revision 1, issued March 2011.

8. MUAP-09007-P, “Summary of Stress Analysis Results for Pressurizer,” Revision 1, issued March 2011.

9. MUAP-09008-P, “Summary of Stress Analysis Results for Reactor Coolant Pump,” Revision 2, issued March 2011.

10. MUAP-09009-P, “Summary of Stress Analysis Results for Control Rod Drive Mechanism,” Revision 1, issued February 2011.

11. MUAP-09010-P, “Summary of Stress Analysis Results for Reactor Coolant Loop Piping,” Revision 3, issued March 2011.

12. MUAP-09011-P, “Summary of Stress Analysis Results for Reactor Coolant Loop Branch Piping,” Revision 2, issued December 2010.

13. MUAP-09012-P, “Summary of Stress Analysis Results for Accumulator,” Revision 1, issued January 2011.

14. MUAP-09013-P, “Summary of Stress Analysis Results for Main Steam Piping inside Containment Vessel,” Revision 2, issued March 2011.

15. MUAP-11003-P, “Summary of Stress Analysis Results for Pressurizer Surge Line,” Revision 1, issued March 2011.

Cross-cutting Requirements (TMI, USI/ GSI, Op Ex): There are no cross-cutting requirements for this area of review.

US-APWR Interface Issues Identified in the DCD: See DCD Tier 2, Table 1.8-2, “Compilation of All Combined License Applicant Items for Chapters 1-19.”

Site Interface Issues Identified in the DCD: See DCD Tier 2, Table 1.8-1, “Significant Site Specific Interfaces with the Standard US-APWR Design.”

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

3.9.1.3 Regulatory Basis

The relevant requirements of the Commission’s regulations for this area of review, and the associated acceptance criteria, are given in Section 3.9.1, “Special Topics for Mechanical Components,” Revision 3, issued March 2007, of NUREG-0800, the SRP, and are summarized below. Review interfaces with other SRP sections also can be found in Section 3.9.1 of NUREG-0800.

1. GDC 1, which requires, in part, that components important to safety be designed, fabricated, erected, and, tested to quality standards commensurate with the importance of the safety functions to be performed.

2. GDC 2, which requires, in part, that components important to safety be designed to withstand seismic events without loss of capability to perform their safety functions.

3. GDC 14, which requires that the RCPB be designed, fabricated, erected and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture.

4. GDC 15, which requires that the reactor coolant system and associated auxiliary, control and protection systems be designed with sufficient margin to assure that the design conditions of the RCPB are not exceeded during any condition of normal operation, including anticipated operational occurrences (AOOs).

5. 10 CFR Part 50, Appendix B, Criterion III, as it relates to quality of design control.

6. 10 CFR Part 50, Appendix S, as it relates to the suitability of the plant design bases for mechanical components established in consideration of site seismic characteristics.

Acceptance criteria adequate to meet the above requirements include:

1. To meet the requirements of GDCs 1, 2, 14, 15, and 10 CFR Part 50, Appendix S, the applicant should provide a complete list of transients to be used in the design and fatigue analysis of all ASME Code Class 1 and core support components, supports, and reactor internals within the RCPB. The number of events for each transient and the number of load and stress cycles per event and for events in combination should be included.

2. To meet the requirements of 10 CFR Part 50, Appendix B, and GDC 1, a list of computer programs to be used in dynamic and static analyses to determine the structural and functional integrity of seismic Category I ASME Code and non-Code items and the analyses to determine stresses should be provided.

3. To meet the requirements of GDCs 1, 14, and 15, if experimental stress analysis methods are used in lieu of analytical methods for any seismic Category I ASME Code or non-ASME Code items, the section of the SAR addressing the experimental stress analysis methods is acceptable if the information meets the provisions of Appendix II to ASME Code, Section III, Division 1 and, as in the case of analytical methods, if the information is sufficiently detailed to show the design meeting the provisions of the ASME Code-required "Design Specifications."

4. To meet the requirements of GDCs 1, 14, and 15 when Service Level D limits are specified by the applicant for ASME Code Class 1 and core support components and for supports, reactor internals, and other non-ASME Code items, the methods of analysis to calculate the stresses and deformations should conform to the methods outlined in Appendix F to ASME Code, Section III, Division 1, subject to the conditions addressed in SRP Section 3.9.1, Section III.4.

3.9.1.4 Technical Evaluation

The applicant's submittal related to SRP Section 3.9.1 is acceptable if it meets the requirements, codes and standards, and the regulatory guidance on the methods of analysis for seismic Category I components and supports, including both those designated as Class 1, 2, 3, or core support by the ASME B&PV Code, Section III, and those not covered by the Code. The applicant's submittal was reviewed to ensure information on design transients for Code Class 1 and core support components and supports was provided. Specific areas of the staff's review included:

- Transients used in the design and fatigue analyses of all Code Class 1 and core support components, supports, and reactor internals.
- Description and verification of all computer programs to be used in analyses of seismic Category I Code and non-Code items.
- Descriptions of any experimental stress analysis programs to be used in lieu of theoretical stress analyses.
- Descriptions of the analysis methods to be used if the applicant elects to use elastic-plastic stress analysis methods in the design of any components.

- The environmental conditions to which all safety-related components will be exposed over the life of the plant.

3.9.1.4.1 Design Transients

To meet the requirements of GDCs 1, 2, 14, 15, and 10 CFR Part 50, Appendix S, this section evaluates the acceptability of the transients including the number of cycles and events expected over the service lifetime of the plant for use in the design and fatigue analysis of ASME Code Class 1 and core support components, supports, and reactor internals within the RCPB. The number of events for each transient and the number of load and stress cycles per event and for events in combination is evaluated.

DCD Tier 2, Subsection 3.9.1.1.2, "Level B Service Conditions (Upset Conditions)," states that plant heat-up operations are conservatively represented by uniform ramp temperature changes of 50 °F (28 ºC) per hour, which bounds the heat-up rate resulting from RCP operation. SRP Section 3.9.1 Subsection III.1 states that any deviation from previous accepted practice should be justified. Standard design practices for maximum design heat-up rates have typically used a value of 100 °F (56 ºC) per hour. Given an expected heat-up rate of approximately 50 °F (28 ºC) per hour, this value provides a margin of two to the design limit.

In **RAI 296-2254, Question 03.09.01-1, Item 1**, the staff requested the applicant to provide additional information and clarify the margins between the maximum design heat up rate and the expected normal heat up rate. In its response to **RAI 296-2254, Question 03.09.01-1, Item 1** dated May 14, 2009, the applicant clarified that the US-APWR employs a pressurizer water-solid mode of operation during RCS low-pressure heat-up and cool-down. The four RCPs and pressurizer heaters provide the heat inputs for plant heat-up. The maximum heat-up rate expected to occur at RCS low temperature is estimated at approximately 40 °F (22 ºC) per hour, which is bounded by 50 °F (28 ºC) per hour.

The applicant stated that the conditions for the estimate are as follows:

- Four RCPs and pressurizer heaters are operating.
- Heat loss of components is assumed to be zero.
- Heat loss caused by letdown is assumed to be zero.

The applicant also clarified that the RCS heat-up rate decreases at higher RCS temperatures since heat loss increases. Further, the applicant clarified that operating data from pressurized-water reactors (PWRs) that employ a water-solid mode of operation demonstrates that the actual heat up rate is bounded by 50 °F (28 ºC) per hour.

The staff finds that this response is acceptable as it provided the clarification requested on the heat up limits used and the associated margins and is based on operating data. Accordingly, **RAI 296-2254, Question 03.09.01-1, Item 1 is resolved**.

DCD Tier 2, Subsection 3.9.1.1.1.2, "Plant Heat-up and Cooldown," states that plant heat-up and cool-down operations are assumed to each occur 120 times during the plant design life. SRP Section 3.9.1 Subsection III.1 states that any deviation from previous accepted practice be justified. Previously, a typical four-loop PWR would postulate five heat-up and cool-down cycles per year for a total of 300 cycles for a 60 year design life.

In **RAI 296-2254, Question 03.09.01-1, Item 2**, the staff requested the applicant to provide additional information and justification for assuming 120 cycles for a 60 year design. In its response to **RAI 296-2254, Question 03.09.01-1, Item 2**, dated May 14, 2009, the applicant stated that the US-APWR shutdown design transient frequency is two cool downs per year, one planned and one unplanned. The same frequency is used for heat-ups. The applicant clarified that the following considerations were used:

- The US-APWR uses a 24-month fuel cycle, which means the actual planned shutdown frequency will be 0.5 heat-ups and cool downs per year.

- Based on current PWR operating data, plant heat-up and cool-down frequency rarely exceeds two per year.

The staff finds that this response is acceptable as it provides the basis for the 120 cycles for a 60-year design life and is also based on current PWR operating data which has been demonstrated through the license renewal applications that no plant heat-up and cool-down frequency exceeds two cycles per year. Accordingly, **RAI 296-2254, Question 03.09.01-1, Item 2 is resolved**.

DCD Tier 2, Subsections 3.9.1.1.1.4, "Ramp Load Increase and Decrease at Five Percent of Full Power per Minute," 3.9.1.1.1.5, "Step Load Increase and Decrease of Percent of Full Power," and 3.9.1.1.1.6, "Large Step Load Decrease with Turbine Bypass," discuss ramp load increases and decreases between specified power levels and step load increases and decreases. The numbers of occurrences over 60 years are also described in DCD Tier 2, Table 3.9-1, "RCS Design Transients." In **RAI 296-2254, Question 03.09.01-1, Item 3**, the staff identified the following issues with respect to occurrences over 60 years:

- In DCD Tier 2, Subsection 3.9.1.1.1.4, the basis for selecting 600 occurrences for the ramp load increase and decrease of five percent of full power per minute is not clear. Previously accepted practice for a standard four-loop PWR uses a greater number of occurrences. The staff requested the applicant to provide additional information and justification for assuming 600 cycles. In DCD Tier 2, Subsection 3.9.1.1.1.5, the basis for selecting 600 occurrences for the step load increase and decrease of 10 percent of full power per minute is not clear. Previously accepted practice for a standard four-loop PWR uses a greater number of occurrences, typically 50 per year. The staff requested the applicant to provide additional information and justification for assuming 600 cycles. In DCD Tier 2, Subsection 3.9.1.1.1.6, the basis for selecting 60 occurrences for the large step load decrease with turbine bypass is not clear. Previously accepted practice for a standard four-loop PWR uses a greater number of occurrences, typically five per year. The staff requested the applicant to provide additional information and justification for assuming 60 cycles.

In its response to **RAI 296-2254, Question 03.09.01-1, Item 3**, dated May 14, 2009, the applicant clarified that the number of ramp load increase is the sum of the following events that lead to hot standby conditions or low reactor power and require an increase in power to reach normal operating conditions.

Transient Events	Number
Plant heat-up and cool down	120

Large step load decrease with turbine bypass	60
Loss of load	60
Loss of offsite power	60
Reactor trip from full power	
• With no inadvertent cool down	60
• With cool down and no safety injection	30
• With cool down and safety injection	10
Control rod drop	30
Inadvertent safeguards actuation	30
Partial loss of reactor coolant flow	30
Inadvertent RCS depressurization, umbrella case	30
Partial loss of emergency feed water	30
Sum	550

The applicant clarified that based on the selection of 600 increases is designed to provide margin beyond the estimate above. The number of ramp load decreases is 600 and is based on the number of ramp load increases. The applicant also clarified that the numbers are based on data from operating PWRs, and the number of occurrences is conservative. The staff finds that this response is acceptable as it provides the basis for the 600 ramp load increases and decreases for a 60 year design life and is also based on current PWR operating data. Accordingly, the **first two issues for RAI 296-2254, Question 03.09.01-1, Item 3 are resolved**.

In its response to the third issue for **RAI 296-2254, Question 03.09.01-1, Item 3**, dated May 14, 2009, the applicant stated that they assumed the frequency of a large step load decrease with turbine bypass was the same frequency as an electrical disturbance at one per year. The applicant also clarified that for step load increases and decreases of 10 percent of full power, the applicant assumed the frequency would be 10 times larger than large step load decrease with turbine bypass. The applicant clarified that both are based on operating data of current PWRs and these numbers are conservative. The staff finds that this response is acceptable as it provides the basis for the transients for a 60 year design life and is also based on current PWR operating data. Accordingly, the **third issue for RAI 296-2254, Question 3.09.01-1, Item 3 is resolved**.

DCD Tier 2, Section 3.9.1.1.1.7, “Steady-State Fluctuations and Load Regulation,” addresses steady state fluctuations and load regulation. Previously accepted practice for a standard four loop PWR defines the magnitude of these transients including temperature and pressure variations and duration. The description of these fluctuations was not complete in the initial submittal. In **RAI 296-2254, Question 03.09.01-1, Item 4**, the staff requested the applicant to provide additional information and justification for these steady state fluctuations.

In its response to **RAI 296-2254, Question 03.09.01-1, Item 4**, dated May 14, 2009, the applicant provided the following steady-state temperature and pressure variations and durations for these transients.

	Steady-State Fluctuations	Load Regulation
Temperature Variation	±3.1 F for T_{avg}	±4 F for T_{hot} ±6 F for T_{cold}
Pressure Variation	±50 psi	±50 psi
Duration	60 sec	2500 sec

The applicant stated that these values are based on operating plant data that shows few steady-state fluctuations at normal operation, and that the transient condition noted above is expected to be conservative. The applicant also clarified that for load regulation; the transient condition is determined conservatively based on load regulation analysis. The staff finds that this response is acceptable as it provides the basis for the steady state fluctuations and load regulation and is also based on current PWR operating data. Accordingly, **RAI 296-2254, Question 03.09.01-1, Item 4 is resolved**.

DCD Tier 2, Subsections 3.9.1.1.1.9, "Main Feedwater Cycling," and 3.9.1.1.2.11, "Emergency Feedwater Cycling," address main and emergency feed water cycling. Industry experience described in NRC Bulletin 88-08, "Thermal Stresses in Piping Connected to Reactor Cooling Systems," and its two supplements, dated June 22, 1998, June 24, 1988, and August 4, 1988; respectively, indicate that during low feedwater flow, thermal stratification can result in significant differences in thermal fatigue cycles. These differences have resulted in failures of the feedwater piping pressure boundary in PWR designs similar to the US-APWR. In **RAI 296-2254, Question 03.09.01-1, Item 5**, the staff requested the applicant if this issue had been addressed in the design and operation of the feedwater systems. The staff also requested the applicant to provide additional information on the basis for the number of cycles assumed for the main and EFWS. In its response to **RAI 296-2254, Question 03.09.01-1, Item 5**, dated May 14, 2009, the applicant stated that since the US-APWR SG design uses an elevated feedwater ring, thermal stratification is assumed to occur at the level of the feedwater ring. Therefore, the feedwater nozzle and piping pressure boundary, which are lower than the feed water ring, are not expected to experience thermal stratification. The applicant also clarified that MFW cycling is assumed to occur at hot standby or no-load conditions. The basis for the number of MFW cycles is the sum of the required MFW injections for the following events.

Transient Events	Number of MFW injections
Plant heat-up and cool down	480
Hot stand-by	1350
Hot functional test	200
Sum	2030

The number of MFW cycles is then 2100, including margin. The applicant stated that these numbers are based on data from operating PWRs and that the number of occurrences is conservative.

For EFW the applicant clarified that EFW cycling is assumed to occur after RT events. The basis for the number of EFW cycles is the sum of the following events that require EFW injection.

Transient Events	Number of emergency feed water injections
Loss of load	60
Loss of offsite power	360
Reactor Trip from full power	
• With no inadvertent cool down	60
• With cool down and no safety injection	30
• With cool down and safety injection	10
Control rod drop	30
Inadvertent safeguards actuation	30

Partial loss of reactor coolant flow	30
Inadvertent RCS depressurization, umbrella case	30
Sum	640

The number of emergency feed water cycles is 700, including 60 cycles of margin. The applicant stated that these numbers are based on data from operating PWRs, and the number of occurrences is conservative. The staff finds that this response is acceptable as it addresses the issue of preventing thermal stratification in the feed water injection piping and also provides the basis for the main and EFW cycles for a 60 year design life and is also based on current PWR operating data. Accordingly, **RAI 296-2254, Question 03.09.01-1, Item 5 is resolved**.

DCD Tier 2, Subsection 3.9.1.1.1.10, "Core Lifetime Extension," addresses the core lifetime extension transient. SRP 3.9.1, Subsection III.1 states that any deviation from previous accepted practice should be justified. The use of a decreased RCS average temperature with turbine inlet valve adjustments to extend the life of the core is a new transient that has not previously been approved. In **RAI 296-2254, Question 03.09.01-1, Item 6**, the staff requested the applicant to provide additional information and justification for this transient including impacts on core design and performance. In its response to **RAI 296-2254, Question 03.09.01-1, Item 6**, dated May 14, 2009, the applicant clarified that, for this transient, it assumes a duration of two weeks as a maximum core lifetime extension. The applicant also stated that the required temperature decrease to achieve criticality is conservatively determined by the analysis. The applicant also clarified that the US-APWR is not performing core lifetime extension evaluations at this time. The applicant indicated that such evaluations will be performed in the future by the respective licensees as part of the license renewal process in accordance with 10 CFR Part 54. The applicant clarified that this event is included among the US-APWR design transients to confirm that the stress evaluation is acceptable when core lifetime extension evaluations are conducted in the future.

The staff found that the applicant's response to **RAI 296-2254, Question 03.09.01-1, Item 6**, is incomplete and a portion of the applicant's response did not appear to be consistent with the description in DCD Tier 2, Subsection 3.9.1.1.1.10. The DCD indicates that the Core Lifetime Extension transient is assumed to occur at the end of an operating cycle and is assumed to occur 60 times during the plant design lifetime, not at some future time.

The staff closed as unresolved **RAI 296-2254, Question 03.09.01-1, Item 6 and** in follow-up **RAI 802-5931, Question 03.09.01-7**, the staff requested the applicant to provide the following additional information:

(1) Provide clarification on how a future effort to extend the life of the plant under the license renewal process is related to this Core Lifetime Extension transient.

(2) If there is no direct connection, explain why the inclusion of this transient is required to provide acceptable stress for the future core lifetime extension.

(3) Confirm whether the core-reload analyses for each cycle address a core lifetime extension transient as described in the DCD.

(4) As a follow-up to **RAI 296-2254, Question 03.09.01-1, Item 6**, provide a detailed discussion regarding the impacts of this transient on core design and performance.

(5) Provide clarification on any safety concerns or operational concerns with operating outside of the normal programmed RCS average temperature band for two weeks during this transient.

In its response to **RAI 802-5931, Question 03.09.01-7**, dated November 1, 2011, the applicant indicated that:

(1) The US-APWR is not performing core lifetime extension evaluations at this time. However, the core lifetime extension is included in the normal operating condition design transients in case a future US-APWR plant conducts core lifetime extension evaluations in support of such a license amendment. The COL licensee will decide if and when to process such a license amendment for the core lifetime extension operation. The duration of core lifetime extension operation and the band of the RCS average temperature decrease will be specified by the licensee based on the desirable operating duration and the ability of 100 percent power operation.

(2) Since the core lifetime extension evaluation is not performed, this core lifetime extension transient has no direct relation to a future effort to extend the life of plant under the license renewal process. This core lifetime extension transient is included in the design transients considering the conservatism for the equipment stress analysis.

(3) The core lifetime extension evaluation is not performed at this time, as described in item 2 above. Therefore, the transient conditions are not considered in the actual reload core analysis. If the core lifetime extension with the decrease of RCS average temperature is implemented in future, there could be impacts on the core power distributions and reactivity coefficients due to increased burn-up compared to the case without the core lifetime extension. However the impacts would be incorporated in the reload design by evaluating the reload checklist values at the extended EOC [End of Cycle].

(4) In the core lifetime extension operation, secondary side coolant temperature/ and pressure decreases according to the decrease in reactor coolant temperature. The decrease of reactor coolant temperature causes an increase in safety margin for the thermal transient. On the other hand, the decrease in MS pressure results in the increase of the delta between operational pressure and set point pressure of the MS safety valves opening. Therefore, when a transient or an accident occurs in the core lifetime extension operation, it will take a longer time period to initiate the opening of the MS safety valves, compared to the amount of time taken during normal operation. However, this impact to the safety margin will be small. Therefore, plant operability is hardly impacted by core lifetime extension. Stable plant operation will be achieved by the adjustment of reactor control system set points associated with the programmed RCS average temperature. Such analysis and transient concerns will need to be addressed in core lifetime extension evaluations in support of a license amendment, if a licensee decides to pursue approval of such operation.

The staff considers the applicant's response reasonable and adequate because the core extension transients will be evaluated during a future application for license renewal. There is no impact on the US-APWR design. The applicant also indicated that DCD Tier 2, Subsection

3.9.1.1.1.10 will be revised and provided a DCD markup. Since the applicant identified DCD changes, **RAI 802-5931, Question 03.09.01-7 is being tracked as a Confirmatory Item**.

DCD Tier 2, Subsection 3.9.1.1.1.13, “Primary-Side Leakage Test,” covers primary-side leakage tests and addresses performance of a primary side leak test with system pressure raised to 2500 psi (17.24 MPa). The ASME B&PV Code no longer requires increasing pressure above the normal operating pressure to perform these leakage tests. In **RAI 296-2254, Question 03.09.01-1, Item 7**, the staff requested the applicant to provide additional information and justification for raising the pressure to perform these leakage tests.

In its response to **RAI 296-2254, Question 03.09.01-1, Item 7**, dated May 14, 2009, the applicant stated that they recognize that it is not necessary to raise RCS pressure up to the design pressure of 2500 psia (17.24 MPa) during a primary-side leakage test in accordance with the ASME B&PV Code Section XI IWB-5221. However, the applicant clarified that they use the design pressure for the primary-side leakage test transient to ensure that a conservative stress evaluation is performed. The applicant also clarified that the testing is conducted below safety valve settings. Finally, the applicant stated that exposure to these conditions will not damage the RCS. The staff finds that the applicant’s response is acceptable as it provides the justification for continuing to conduct the primary-side leakage test using the design pressure. The applicant has stated that the test is conservative in regard to the stress analysis, and that the testing is performed below the safety valve settings and no damage to the RCS will occur. Accordingly, **RAI 296-2254, Question 03.09.01-1, Item 7, is resolved**.

DCD Tier 2, Subsection 3.9.1.1.1.14, “Secondary-Side Leakage Test,” addresses performance of a secondary side leak test with secondary system pressure raised to design pressure. Previously accepted practice for a typical four-loop PWR includes definition of pressures and temperatures for both the secondary side and the primary side to prevent damage to the steam generators and the RCS. Since the applicant did not specify how it determined the pressure and temperature for the leak test, in **RAI 296-2254, Question 03.09.01-1, Item 8**, the staff requested the applicant to provide additional information and justify the pressures and temperatures for both the secondary and primary sides.

In its response to **RAI 296-2254, Question 03.09.01-1, Item 8**, dated May 14, 2009, the applicant stated that they use atmospheric pressure for the primary side with the secondary side pressure just below the design pressure so as to not actuate the MS safety valves in the secondary side leakage test. The applicant clarified that the actual RCS temperature used during the test is atmospheric temperature, which is greater than the nil ductility temperature. The applicant stated that the conditions were selected to ensure that the stress evaluations are conservative. The applicant also stated that these conditions will not damage either the steam generators or the RCS. The staff finds that the applicant’s response is acceptable as it provides the requested information and justification for the pressures and temperatures used during the secondary side leak testing. The applicant also clarified that the test conditions are conservative in regard to the stress analysis, and no damage to the SGs or RCS will occur. Accordingly, **RAI 296-2254, Question 03.09.01-1, Item 8 is resolved**.

DCD Tier 2, Section 3.9.1.1.2.3, “RT from Full Power,” addresses RTs from full power. Previously accepted practice for a standard four-loop PWR uses a greater number of occurrences, typically 400 occurrences. In **RAI 296-2254, Question 03.09.01-1, Item 9**, the staff requested the applicant to provide additional information and justification for assuming 100 RTs.

In its response to **RAI 296-2254, Question 03.09.01-1, Item 9**, dated May 14, 2009, the applicant stated that the US-APWR defines 3 cases of RT and that each case assumes a different frequency based on the severity of the postulated event. The applicant defined these cases as follows:

Transient Case RT From Full Power	Frequency	Number of Occurrences	Notes
With no inadvertent cool down	1/year	60	Assumed as the same frequency of electrical disturbances.
With cool down and no safety injection	0.5/year	30	Assumed as the same frequency as a single failure.
With cool down and safety injection	10/plant life	10	Assumed as the same frequency as events more severe than single failure or operating error.

The applicant stated that these values are based upon PWR operating experience, and that the numbers of occurrences assumed by the applicant is conservative. The staff finds that the applicant's response is acceptable as it addresses the staff's concern regarding the justification for assuming 100 RTs and also provides the basis for a 60 year design life and is also based on current PWR operating experience. Accordingly, **RAI 296-2254, Question 03.09.01-1, Item 9 is resolved**.

The staff noted service level B conditions described in the following DCD Subsections:

- DCD Tier 2, Subsection 3.9.1.1.2.4, "Control Rod Drop": Assumes 30 times.
- DCD Tier 2, Subsection 3.9.1.1.2.6, "Inadvertent Safeguards Actuation": Assumes 30 times.
- DCD Tier 2, Subsection 3.9.1.1.2.7, "Partial Loss of Reactor Coolant Flow": Assumes 30 times

The occurrence numbers on a yearly basis are less than previously accepted practice. The reduction may be justified based on service experience; however, the amount of margin appeared to be less than typical for four-loop PWRs currently operating. In **RAI 296-2254, Question 03.09.01-1, Item 10**, for each service level B event listed, the staff requested the applicant to provide additional justification and information for the reduced number of occurrences assumed in the design-basis.

In its response to **RAI 296-2254, Question 03.09.01-1, Item 10**, dated May 14, 2009, the applicant stated that the service level B conditions Control Rod Drop, Inadvertent Safeguards Actuation, and Partial loss of Reactor Coolant Flow are assumed to occur at the same frequency as a single failure. The applicant stated that the frequencies of these postulated events are conservative relative to ANSI/ANS 51.1-1983. In DCD Tier 2, Subsection 3.9.1.1.2.7, the applicant assumes that the loss of two pumps occurs. This assumption is consistent with those postulated in the DCD Tier 2, Section 15.3.1.1, "Partial Loss of Forced Reactor Coolant Flow," safety analysis. The partial loss of flow analysis conservatively assumes that two RCPs trip at the same time. This is an extremely conservative assumption since the US-APWR is configured such that each RCP has its own source of electrical power. The staff finds that the applicant's response is acceptable as it addresses the staff's concerns regarding the justification for the number of occurrences for these selected service level B conditions and also provides a conservative basis for the number of occurrences assumed. Accordingly, **RAI 296-2254, Question 03.09.01-1, Item 10 is resolved**.

DCD Tier 2, Section 3.9.1.1.3, “Level C Service Conditions (Emergency Conditions),” addresses ASME Code, Section III Service Level C Conditions (emergency conditions). However, the applicant did not address the level C transients that are considered in the design-basis. In **RAI 296-2254, Question 03.09.01-1, Item 11**, the staff requested the applicant to confirm compliance with GDC 14 and 15 requirements for the following events:

a. DCD Tier 2, Subsection 3.9.1.1.3.1, “Small LOCA”: The applicant was asked to provide additional information on break size definitions and clarify description of the event used in the design-basis.

b. DCD Tier 2, Subsection 3.9.1.1.3.2, “Small Steam Line Break”: The applicant was asked to provide additional information on break size definitions and clarify description of the event used in the design-basis.

c. DCD Tier 2, Subsection 3.9.1.1.3.4, “Small Feedwater Line Break”: For previous four loop PWRs no distinction was made in previous designs between a small or large feedwater line break (Level D). The Level C small feed water line break description states, “No main feedwater is supplied to either of SG [steam generator] and then all of SG water level decrease. For receipt on low SG water level signal, the reactor is tripped and emergency feed water pumps are actuated automatically.” The applicant was asked to provide additional information and clarify as there does not appear to be a significant distinction between the Level C description and the Level D description.

d. DCD Tier 2, Subsection 3.9.1.1.3.5, “SG Tube Rupture”: Previous plant designs assumed steam generator tube ruptures (SGTRs) to be Level D faulted events not Level C emergency events. The applicant was asked to provide additional information and justification for why the SGTR is included as a Level C event verses a Level D event.

To address each of the above bullets in its response to **RAI 296-2254, Question 03.09.01-1, Item 11**, dated May 14, 2009, the applicant provided the following:

a. The applicant clarified that a small break LOCA is considered to be a break of a 1 in. (25 mm) inner diameter (ID) branch pipe of a reactor coolant pipe (a break with an ID smaller than 0.375 in. (9.53 mm) can be handled by the normal makeup system, which has sufficient capacity to compensate for the coolant loss with this break size, and produces no significant transient). For a break of a 1 in. (25 mm) ID branch pipe, the high head injection system is actuated to inject water at ambient temperature into the RCS. The staff finds that the applicant’s response is acceptable as it addresses the staff’s concerns regarding the definition of the small break LOCA break sizes and clarifies the description used in the design-basis. Accordingly, **RAI 296-2254, Question 03.09.01-1, Item 11a is resolved**.

b. The applicant clarified that a small steam line break is considered to be a break equivalent to a 10 in. (25 cm) ID pipe of a MS relief valve. The increase in steam generation rate caused by the postulated break removes heat from the RCS, which in turn, lowers the temperature and pressure of the RCS. The staff finds that the applicant's response is acceptable as it addresses the staff's concerns regarding the definition of the small steam line break and clarifies the description used in the design-basis. Accordingly, **RAI 296-2254, Question 03.09.01-1, Item 11b is resolved**.

c. The applicant provided additional information and clarified that a small feedwater line break is considered to be a break of the MFW cleanup line. Since MFW is not being supplied to any of the SGs under this scenario the water level decreases in all SGs. Upon receipt of a low SG water level signal, the reactor is tripped and the EFW pumps are actuated automatically. The applicant stated that this transient is basically the same as a large feed water line break (Level D) except for the break size. The staff finds that the applicant's response is acceptable as it addresses the staff's concerns regarding the definition of the small feed water line break and clarifies the difference between the Level C small and Level D large feed water line break. Accordingly, **RAI 296-2254, Question 03.09.01-1, Item 11c is resolved**.

d. The applicant clarified that in ANSI/ANS-51.1, a SGTR is classified as a plant condition 3 event which is equivalent to severity of level B or C events in the US-APWR design transient. The applicant also stated that they classified this event as level C, based on its severity. The staff finds that the applicant's response is acceptable as it addresses the staff's concerns regarding the classification of the SGTR event and provides justification for classifying this event as a level C event. Accordingly, **RAI 296-2254, Question 03.09.01-1, Item 11d is resolved**.

As **RAI 296-2254, Question 03.09.01-1, Items 11a through 11d are resolved, RAI 296-2254, Question 03.09.01-1, Item 11 is resolved.**

In **RAI 296-2254, Question 03.09.01-1, Item 12**, the staff requested the applicant to provide the basis to justify the exclusion of earthquakes as dynamic events at the rated operating power condition in DCD Tier 2, Table 3.9.1-1. In its response to **RAI 296-2254, Question 03.09.01-1, Item 12**, dated May 14, 2009, the applicant stated that since seismic events are evaluated in the load calculations, they are not considered in the design transient section. The applicant also clarified that this distinction is described in DCD Tier 2, Subsection 3.9.3.1.1, "Seismic Load Combinations." DCD Tier 2, Subsection 3.9.3.1.1 addresses seismic load combinations analyzed in accordance with the ASME Code and describes how earthquake events are combined with other plant conditions. The staff finds that the applicant's response is acceptable as it addresses the staff's concerns regarding categorizing the earthquake dynamic events and ensuring that these events have been included in the design-basis. Accordingly, **RAI 296-2254, Question 03.09.01-1, Item 12 is resolved**.

NRC Bulletin 79-13, "Cracking in Feedwater System Piping," issued June 25, 1979, addressed fatigue loading due to thermal stratification and high-cycle thermal striping during low-flow EFW injection. NRC Bulletin 88-08 and its supplements indicate that during low feedwater flow, stratification flow conditions can result in significant differences in thermal fatigue cycles. These differences have resulted in failures of the feedwater piping pressure boundary in PWRs. NRC Bulletin 88-11, "Pressurizer Surge Line Thermal Stratification," issued December 20, 1988, requires consideration of the effects of thermal stratification on the pressurizer surge line dynamic loads. In **RAI 296-2254, Question 03.09.01-1, Item 13**, the staff requested the applicant to discuss the basis for not considering the thermal stratification in DCD Tier 2, Section 3.9.1.1, even though it is an important design transient used in the design of piping.

In its response to **RAI 296-2254, Question 03.09.01-1, Item 13**, dated May 14, 2009, the applicant regarding feedwater piping stated that since the US-APWR employs an elevated feedwater ring, thermal stratification is not assumed to occur at the feedwater piping pressure boundary. The staff noted that DCD Tier 2, Section 3.12.5.10, “Thermal Stratification,” states that provisions of the thermal stratification of the feed water nozzle are described in DCD Tier 2, Subsection 5.4.2.1.2.12, “Thermal Stratification at Feedwater nozzle.” DCD Tier 2, Subsection 5.4.2.1.2.12 clarifies that this configuration forces the flow to fill the feedwater nozzle and thermal sleeve before flowing into the feedwater ring thus preventing thermal stratification within the nozzle or nearby feedwater piping. The staff finds that the applicant’s response with regard to feed water thermal fatigue is acceptable as it clarifies that feed water thermal stratification is not assumed to occur because of the elevated feed water ring configuration used in the US-APWR. Accordingly, regarding feedwater piping **RAI 296-2254, Question 03.09.01-1, Item 13 is resolved**.

Regarding the pressurizer surge line, the applicant clarified that thermal stratification is assumed to occur in the horizontal part of the pressurizer surge line and the load is considered in the stress evaluation. The applicant clarified that this analysis is described in DCD Tier 2, Section 3.12.5.10. DCD Tier 2, Section 3.12.5.10 addresses thermal stratification of the pressurizer surge line and addresses actions taken to ensure structural integrity of the pressurizer surge line. The staff finds that the applicant’s response with regard to thermal stratification in the pressurizer surge line is acceptable as this condition is assumed to occur and the loads are included in the stress evaluation. Accordingly, regarding the pressurizer surge line **RAI 296-2254, Question 03.09.01-1, Item 13 is resolved**. Based on the applicant addressing both the feedwater piping and the pressurizer surge line, **RAI 296-2254, Question 03.09.01-1, Item 13 is resolved**.

In **RAI 296-2254, Question 03.09.01-1, Item 14**, the staff requested the applicant about recent industry experience with the vibration effects on components and piping due to acoustic resonance. The staff pointed out that SRP Section 3.9.5, “Reactor Pressure Vessel Internals,” Revision 3, issued March 2007, and RG 1.20, “Comprehensive Vibration Assessment Program for Reactor Internals During Preoperational and Initial Startup Testing,” Revision 3, issued March 2007, were revised to include acoustic loads. The applicant was requested to provide the basis for not including this acoustic cyclic loading in the design transients. In its response to **RAI 296-2254, Question 03.09.01-1, Item 14**, dated May 14, 2009, the applicant stated that the design of the US-APWR RCS components, piping and the flow conditions are similar to that of the existing and currently operating PWRs in the U.S. and around the world. The applicant also stated that based on an extensive record of vibration-free operation, the applicant concludes that acoustic loadings are small enough and that it is not necessary to consider this loading in the stress evaluation. The staff finds that the applicant’s response is acceptable as it is based on an extensive record of vibration-free operation with a statement that the acoustic loadings are small enough so as not to be a concern in the stress evaluations. Accordingly, **RAI 296-2254, Question 03.09.01-1, Item 14 is resolved**.

In summary regarding **RAI 296-2254, Question 03.09.01-1, Items 1 to 5 and 7 to 14 are resolved. RAI 296-2254, Question 03.09.01-1, Item 6, is unresolved,** resulting in **RAI 802-5931, Question 03.09.01-7, which is being tracked as a Confirmatory Item**.

On the basis of this evaluation and the evaluation of the responses to **RAI 296-2254, Questions 03.09.01-1, Items 1 through 14** and **RAI 802-5931, Question 03.09.01-7**, the staff concludes that the use of operating plant experience, adjusted for a 60-year plant life, plus additional cycles to account for seismic events, provides an acceptable basis for estimating the total number of cycles for each transient. Therefore, the information relative to the US-APWR design transients in DCD Tier 2, Section 3.9.1.1, is consistent with the applicable guidelines in SRP Section 3.9.1 and is, therefore, acceptable, pending satisfactory resolution of **RAI 802-5931, Question 03.09.01-7, which is being tracked as a Confirmatory Item**.

3.9.1.4.2 Computer Programs Used in Analyses

This section evaluates computer programs that are used for analysis and design of seismic Category I structures, components and equipment to ensure these computer codes meeting the requirements of 10 CFR Part 50, Appendix B, and GDC 1 in accordance with SRP Section 3.9.1, Subsection II, SRP Acceptance Criteria 2.

DCD Tier 2, Subsection 3.9.1.2.1, "List of Programs," provides a list of computer programs used for analysis. An additional list of computer programs is provided in DCD Tier 2, Subsection 3.12.4.1.1, "List of Programs." The review procedures of SRP Section 3.9.1 Subsection III.2.B state that the submitted computer test problem solutions recommended in SRP Section 3.9.1 Subsection II.2.C are reviewed and compared to the reference solutions. If the solutions produced by the computer program agree within five percent with the reference solutions, the quality and adequacy of the computer programs are determined to be appropriate for the functions for which they were designed. There were no computer test problem solutions or summary tables provided in the DCD documentation. In addition, the applicant's Technical Reports MUAP-09001-P, "Summary of Design Transient," Revision 0, issued January 2009, and MUAP-09002-P, "Summary of Seismic and Accident Load Conditions for Primary Components and Piping," Revision 0, issued January 2009, provide a list of several computer codes (MARVEL-M, M-RELAP-5, WCOBRA/RTRAC, TWINKLE-M, VIPRE-01M and GOTHIC) that are not included in DCD Tier 2, Subsections 3.9.1.2.1 or 3.12.4.1.1.

In **RAI 296-2254, Question 03.09.01-2, Item 1**, the staff requested the applicant to provide additional information and details on the computer test problem methods, solution sets, and results. 10 CFR Part 50, Appendix B, requires applicants to ensure that appropriate standards are specified and included in design documents, including design methods and computer programs for the design and analysis of seismic Category I, ASME Code Class 1, 2, 3, and core support structures and non-Code structures. In **RAI 296-2254, Question 03.09.01-2, Item 2**, the staff requested the applicant to confirm that the computer programs used for US-APWR design and listed in DCD Tier 2, Subsections 3.9.1.2.1 and 3.12.4.1.1, including the pre- and post-processors used for the analyses, are in compliance with requirements of Appendix B to 10 CFR Part 50 and ASME NQA-1. The staff also requested the applicant to confirm that the documentation of these computer programs is available for staff review.

In its response to **RAI 296-2254, Questions 3.9.1-2, Items 1 and 2**, dated May 14, 2009, the applicant stated that they had verified the computer programs listed in DCD Tier 2, Subsection 3.9.1.2.1 in accordance with the methods described in SRP Section 3.9.1, SRP Acceptance Criteria 2.C. The applicant prepared a verification report of computer programs that described the computer test method, assumptions, analysis model, and solution set. The applicant also prepared computer code documents that described the author, source code, dated version, user's manual, and a theoretical description. The applicant stated that these documents have been prepared in accordance with 10 CFR Part 50, Appendix B, and ASME Code, NQA-1 requirements, and will be available for review during the NRC design audit. The staff confirmed during the site audit at the US-APWR offices in Rosslyn, Virginia, August 22 – 30, 2011, that the requirements and commitments listed above for design methods and computer programs have been completed. A summary of the staff's audit is available in, "Report of the August 22 - 30, 2011, Audit Regarding the United States - Advanced Pressurized Water Reactor Computer Programs and Piping Described in Design Control Document Section 3.9.1 and Section 3.12," dated November 15, 2012. Accordingly, **RAI 296-2254, Questions 03.09.01-2, Items 1 and 2 are resolved**.

In **RAI 296-2254, Question 03.09.01-2, Item 3** the staff requested the applicant to verify that all computer programs used for calculating stresses and cumulative usage factors for Class 1, 2, and 3 components include environmental effects in the fatigue curves. The staff requested the applicant to identify the computer programs used to perform the fatigue analysis. The staff also requested the applicant to confirm that the analyses for ASME Code, Section III Class 1 components and piping for the fatigue evaluation include environmental effects in accordance with the guidance in RG 1.207, "Guidelines for Evaluating Fatigue Analyses Incorporating the Life Reduction of Metal Components Due to the Effects of the Light-Water Reactor Environment for New Reactors," issued March 2007.

In its response to **RAI 296-2254, Question 03.09.01-2, Item 3**, dated May 14, 2009, the applicant indicated that the computer programs used to perform the stress analysis and cumulative fatigue usage factor are verified as noted in the above answers to **RAI 296-2254, Questions 03.09.01-2, Items 1 and 2**. The applicant further stated that fatigue analyses for ASME Code, Section III, Class 1 components and piping will include environmental effects in accordance with RG 1.207. The staff confirmed during the August 22 – 30, 2011, audit that the computer programs used to perform the stress analysis and cumulative fatigue usage factor have been properly verified and the analyses for ASME Code, Section III, Class I components and piping will include environmental effects in accordance with RG 1.207. Accordingly, **RAI 296-2254, Question 03.09.01-2, Item 3 is resolved**.

In **RAI 296-2254, Question 03.09.01-2, Item 4**, the staff requested the applicant to verify that all computer programs used for calculating stresses for Class 1, 2, and 3 piping include staff-endorsed methods when performing response spectrum analysis. The staff requested the applicant to verify that all computer programs employed in the US-APWR piping design that use the independent support motion (ISM) response spectrum analysis method comply with the staff position for combining mode, group (absolute sum), and direction responses, as stated in NUREG-1061, Volume 4, "Report of the U.S. Nuclear Regulatory Commission Piping Review Committee," issued December 1984.

In its response to **RAI 296-2254, Question 03.09.01-2, Item 4**, dated May 14, 2009, the applicant referred to its response to **RAI 260-2023, Question 03.12-5**, which noted that DCD Tier 2, Section 3.12.3.3, "Response Spectra Method (or Independent Support Motion Method)," will be revised to incorporate NUREG-1061, Volume 4, Section 2, so that group responses are combined by the absolute summation method, and inter-modal and inter-spatial responses are combined by the square root of sum of square (SRSS) method. The applicant also indicated that DCD Tier 2, Subsection 3.12.3.2.6, "Seismic Anchor Motions," on seismic anchor motions (SAMs) will also be revised, as noted in the answer to **RAI 260-2023, Question 03.12-6**, to address the analysis of SAM associated with the ISM method. The applicant stated that all computer programs used for US-APWR design of piping that use the ISM response spectrum analysis method comply with the staff position for combining mode, group (absolute sum), and direction responses, as stated in NUREG-1061, Volume 4. The staff finds that the applicant's response is acceptable as the applicant stated that the computer programs that use the ISM method comply with the staff position regarding the response spectrum analysis methods in NUREG-1061, Volume 4. The staff confirmed that the proposed DCD changes were incorporated into DCD Revision 3. Accordingly, **RAI 296-2254, Question 03.09.01-2, Item 4 is resolved**. Based on the applicant resolving all four items in the RAI, **RAI 296-2254, Question 03.09.01-2 is resolved.**

The staff reviewed the applicant's Technical Reports MUAP-09001-P, Revision 0, and MUAP-09002-P, Revision 0, which provide a list of new computer codes (MARVEL-M, M-RELAP-5, WCOBRA/RTRAC, TWINKLE-M, VIPRE-01 M and GOTHIC) that are not included in DCD Tier 2, Subsections 3.9.1.2.1 or 3.12.4.1.1. In **RAI 296-2254, Question 03.09.01-5**, the staff requested the applicant to provide additional information regarding how the computer codes were used, and discuss computer test problem methods, solution sets, and summary of the results in compliance with requirements of Appendix B to 10 CFR Part 50 and ASME NQA-1. The staff also requested the applicant to confirm that the documentation of these computer programs is available for staff review.

In its response to **RAI 296-2254, Question 03.09.01-5**, dated May 14, 2009, the applicant stated that they use several computer programs for both design transients for primary components and safety analyses. The specific computer codes mentioned in this RAI (including MARVEL-M, M-RELAP-5, WCOBRA/TRAC, TWINKLE-M, VIPRE-01M) are all examples of safety analysis codes that are also used for design transients. The applicant indicated that the information regarding these codes was submitted in support of DCD Tier 2, Chapter 15, "Transient and Accident Analyses." This information was also presented in the RAI response in Table 03.09.01-5.1, "Summary of Chapter 15 Code Documentation Submittals." The asymmetric pressurization analysis for accident load evaluation is performed by the GOTHIC code, which is also used for containment pressure and temperature analysis in DCD Tier 2, Chapter 6, "Engineered Safety Features," in support of containment functional design. Information on the GOTHIC code, including the input deck, was submitted in support of the review of Topical Report MUAP-07012-P, "LOCA Mass and Energy Release Analysis Code Applicability Report for US-APWR," Revision 2, issued May 2008.

The staff finds that the applicant's response is acceptable as the information for these computer programs was previously submitted to NRC in support of Chapters 6 and 15. The evaluation of these computer codes is included in Section 15.0.2.4 and Section 6.2.4 of this report. Accordingly, **RAI 296-2254, Question 03.09.01-5 is resolved.**

In its review of the applicant's technical reports providing summary stress analysis results (MUAP-09004, "Summary of Stress Analysis Results for Core Support Structures," Revision 1, issued January 2011, through MUAP-9013, "Summary of Stress Analysis Results for Accumulator," Revision 1, issued January 2011, and MUAP-11003," "Summary of Stress Analysis Results for Pressurizer Surge Line," Revision 1, issued March 2011), the staff found that certain computer codes used for the design of seismic Category I piping, components, and supports are not listed and discussed in DCD Tier 2, Sections 3.9.1.2, 3.12.4.1, "Computer Codes," or Appendix 3C, "Reactor Coolant Loop Analysis Methods." Additional computer codes are also listed in the applicant's submittal, "Revised Design Completion Plan for US-APWR Piping Systems and Components," the applicant dated May 12, 2011. In **RAI 770-5793, Question 03.09.01-6**, the staff requested the applicant to include in DCD Tier 2, Section 3.9.1.2, all computer programs used in the analysis and design of US-APWR safety-related piping, components and supports, in conformance with SRP Sections 3.9.1 and 3.12, "ASME Code Class 1, 2, and 3 Piping Systems, Piping Components and their Associated Supports," issued March 2007. The information should include program name, dated version and brief description of the program application. Since these programs were used for design of US-APWR seismic Category I components, the applicant was also requested to provide information on staff review of the codes or additional information on verification and validation of the codes in accordance with Appendix B to 10 CFR 50.55 or the ASME NQA-1 Code.

In its response to **RAI 770-5793, Question 03.09.01-6**, dated July 26, 2011, the applicant stated that it will provide additional information on computer codes, such as code verification and validation documents used for design of ASME Class 1, 2 and 3 components and piping in accordance with Appendix B to 10 CFR Part 50 and ASME Code NQA-1, for audit by the NRC staff. The applicant committed to prepare the computer code information listed in Table 1 of the applicant's May 12, 2011, submittal discussed above. The applicant stated that the detailed information for RELAP-5 had already been provided in support of the review of Topical Report MUAP-07013P, "Small Break LOCA Methodology for US-APWR," Revision 2, issued October 2010. This information was audited by the staff on October 19, 20, and 25, 2010. A summary of the staff's audit is available in, "The U.S. Nuclear Regulatory Commission Report for the Audit Performed on October 19, 20 And 25, 2010, Regarding the United States – Advanced Pressurized Water Reactor Software Quality Assurance Review of M-RELAP-5 and VIPRE-01M Codes," dated October 27, 2011. In its amended response to **RAI 770-5793, Question 03.09.01-6**, dated December 2, 2011, the applicant provided proposed revisions to DCD Tier 2, Sections 3.9.1.2, and 3.12.4.1 adding computer codes, including program name, dated version and brief description of the program application. The staff finds the amended response acceptable since the applicant included computer codes and associated information in accordance with SRP Section 3.9.1, Subsection II, SRP Acceptance Criterion 2. Since the applicant identified DCD changes, **RAI 770-5793, Question 03.09.01-6 is being tracked as a Confirmatory Item**.

Based on the review of the information in this section and the responses to the RAIs, the staff concludes that the computer code qualification methods described in this section meet the acceptance criteria provided in SRP Section 3.9.1 and are therefore acceptable.

3.9.1.4.3 Experimental Stress Analysis

In the Nuclear Steam Supply System, there exist gaps and non-linear material properties in the design of certain components, such as snubbers and pipe restraints, for which experimental stress analysis is frequently performed in conjunction with analytical evaluation. The experimental stress analysis methods should be used in compliance with the provisions of Appendix II to ASME Code, Section III, Division 1. In **RAI 296-2254, Question 03.09.01-3**, the staff asked the applicant about DCD Tier 2, Section 3.9.1.3, which indicates that experimental stress analysis is not used for the US-APWR to evaluate stresses for seismic Category I components and supports. The staff requested the applicant to discuss the stress analysis methods used to verify the design adequacy of specific US-APWR components: piping seismic snubbers, pipe whip restraints, and the prototype fine motion control rod drive.

In its response to **RAI 296-2254, Question 03.09.01-3**, dated May 14, 2009, the applicant addressed the requested components.

- CRDM: The applicant stated that experimental stress analysis methods are not used for the design of the CRDM. The applicant stated that the structural integrity of the CRDM pressure housing, as a RCS pressure boundary, is confirmed by stress analysis in accordance with ASME Code, Section III, Subsection NB. The staff finds that the applicant's response is acceptable as they stated that experimental stress analysis methods are not used for the CRDMs. Also, the structural integrity of the CRDMs is confirmed by stress analysis in accordance with ASME Code, Section III, Subsection NB. Accordingly, regarding CRDMs, **RAI 296-2254, Question 03.09.01-3 is resolved**.

- Pipe Whip Restraints: The applicant stated that experimental stress analysis methods are not used for the design of pipe whip restraints for the US-APWR. The applicant clarified that the analytical methods used in the design of pipe whip restraints are discussed in DCD Tier 2, Subsection 3.6.2.4.4.1, "Pipe Whip Restraints." The staff finds that the applicant's response is acceptable as they stated that experimental stress analysis methods are not used for the pipe whip restraints. The applicant also clarified the analytical methods used for pipe whip restraints in DCD Tier 2, Subsection 3.6.2.4.4.1. Accordingly, regarding pipe whip restraints, **RAI 296-2254, Question 03.09.01-3 is resolved**.

- Piping Seismic Snubbers: The applicant stated that there is no plan to use experimental stress analysis methods to verify the design adequacy of snubbers used for the US-APWR piping. The applicant clarified that snubbers used as shock arrestors and seismic restraints for piping are design verified using loads from a computer dynamic piping stress analysis. The snubber manufacturer determines the conditions and the limits of use for the snubber, and employs, among others, tests as required to establish those limits. These design limits consist of the four loading conditions as established by the applicable ASME Code, Section III, Subsection NF (i.e., normal, upset, emergency and faulted, maximum environmental temperature, maximum travel range, maximum allowable angularity, envelope space and any other applicable limitations). The staff finds that the applicants' response is acceptable as they stated that experimental stress analysis methods are not used to verify the design adequacy of snubbers and the applicant described an acceptable method for verifying that the snubbers used meet the design requirements including the applicable requirements of Section III of the ASME Code. Accordingly, regarding piping seismic snubbers, **RAI 296-2254, Question 03.09.01-3 is resolved**.

The staff finds that, consistent with the applicant's response to **RAI 296-2254, Question 03.09.01-3**, experimental stress analysis methods are not used in the design of US-APWR components including CRDMs, pipe whip restraints, and piping seismic snubbers. Therefore, the guidance in SRP Section 3.9.1, Subsection III.3 on experimental stress analysis methods is not applicable to the US-APWR. Accordingly, **RAI 296-2254, Question 03.09.01-3, is resolved**.

3.9.1.4.4 Considerations for the Evaluation of Faulted Conditions

The staff reviewed applicable portions of DCD Tier 2. Sections 3.9.3, "ASME Code Class 1, 2, and 3 Components, Component Supports, and Core Support Structures," and 3.12, "Piping Design Review," to ensure that when Service Level D limits are specified for Code Class 1 and core support components, as well as for supports, reactor internals, and other non-code items, the method of analysis to calculate the stresses and deformations conforms to the methods in Appendix F to ASME Code, Section III, Division 1.

DCD Tier 2, Section 3.9.3.4.5, "Special Engineered Pipe Supports" addresses the application of Appendix F of ASME Code, Section III, Division 1 to ASME Code, Section III piping supports and those supports or components not built to ASME Code, Section III. To meet the requirements of GDCs 1, 14, and 15, when Service Level D limits are specified, the methods of analysis should conform to the methods outlined in Appendix F to ASME Code, Section III, Division I. The second bullet of DCD Tier 2, Section 3.9.3.4.5 indicates that, when the effects of Level D service conditions are evaluated for supports or components not built to ASME Code, Section III, the allowable stress levels are based on tests or accepted industry standards "comparable" to those in Appendix F of ASME Code, Section III. In **RAI 296-2254, Question 03.09.01-4, Item 1**, the staff requested the applicant to provide additional information and details on the methods and allowable stress levels that will be applied for these Level D analyses to allow the staff to confirm that the methods satisfy Appendix F requirements.

In its response to **RAI 296-2254, Question 03.09.01-4, Item 1**, dated May 14, 2009, the applicant indicated that special engineered pipe supports will no longer be used in the US-APWR design, and the applicant planned to revise DCD Tier 2, Subsection 3.9.3.4.5 to delete the related information. The staff finds that the applicant's response is acceptable as the applicant stated that special engineered pipe supports will not be used. The staff confirmed that DCD Revision 2 reflects the change. Accordingly, **RAI 296-2254, Question 03.09.01-4, Item 1 is resolved**.

In DCD Tier 2, Section 3.9.3, the applicant states that all seismic Category I equipment are evaluated for the faulted (ASME Code, Section III Service Level D) loading conditions identified in Table 3.9-1. In **RAI 296-2254, Question 03.09.01-4, Item 2**, the staff requested that, for each of the components, supports, core support structures, and RV listed in DCD Tier 2, Section 3.9.3, the applicant identify the computer programs that were used to evaluate the stresses for determining that the ASME Code, Section III, Appendix F limits were met.

In its response to **RAI 296-2254, Question 03.09.01-4, Item 2**, dated May 14, 2009, the applicant failed to address the question in that no computer programs were identified. The applicant clarified that the seismic Category I equipment are listed in DCD Tier 2, Section 3.2, Table 3.2-2, "Classification of Mechanical and Fluid Systems, Components, and Equipment." DCD Tier 2, Table 3.2-2 provides a listing of detailed information on the classification of mechanical and fluid systems, components, and equipment. The table also includes a reference to the applicable codes and standards applied to each of the systems and components. The applicant also stated that a computer program is used for elastic analysis for stress evaluation of the components, supports, and core support structures, and a stress limit is applied from ASME Code, Section III, Appendix F requirements on Service Level D.

Following the August 22 – 30, 2011, audit, the applicant amended its response to **RAI 296-2254, Question 03.09.01-4, Item 2,** dated December 13, 2011. In the amended response, the applicant indicated that the computer programs that were used when determining ASME Code, Section III, Appendix F, limits were met were identified during the August 22 – 30, 2011, audit. The applicant indicated the requested computer programs were listed in the revised DCD mark-up of the amended response to **RAI 770-5793, Question 03.09.01-6**. The staff found the amended RAI response acceptable since the list of computer programs for **RAI 770-5793, Question 03.09.01-6**, addressed the concerns for **RAI 296-2254, Question 03.09.01-4, Item 2.** Accordingly, **RAI 296-2254, Question 03.09.01-4, Item 2 is resolved**. Based on the applicant resolving both items in the RAI, **RAI 296-2254, Question 03.09.01-4 is resolved.**

Based on its review of the information in the DCD and the applicant's responses to RAIs, the staff finds that the application of elastic and inelastic stress analyses are in conformance with ASME Code, Section III Appendix F, and therefore are acceptable, pending satisfactory resolution of **RAI 770-5793, Question 03.09.01-6, which is being tracked as Confirmatory Item**.

3.9.1.5 Combined License Information Items

There are no COL items in Table 1.8-2 of the DCD for Section 3.9.1.

3.9.1.6 Conclusions

On the basis of the evaluations in Section 3.9.1.4, the staff concludes that the design transients and resulting loads and load combinations with appropriate specified design and service limits for mechanical components are acceptable and meet the relevant requirements of GDC 1, 2, 14, 15; 10 CFR Part 50, Appendix B; and 10 CFR Part 50, Appendix S, as well as the guidelines in Section 3.9.1 of the SRP. The acceptance is pending on the satisfactory resolution of the confirmatory items shown above in the staff's technical evaluation.

3.9.2 Dynamic Testing and Analysis of Systems, Components, and Equipment

3.9.2.1 Introduction

This section of the DCD provides the criteria, testing procedures, and dynamic analyses employed by the applicant to ensure the structural and functional integrity of piping systems, mechanical equipment, reactor internals, and their supports under vibratory loadings, including those due to fluid flow and postulated seismic events.

This section addresses six main areas of review:

1. Piping Vibration, Thermal Expansion, and Dynamic Effects Testing.

2. Seismic Analysis and Qualification of Seismic Category I Mechanical Equipment.

3. Dynamic Response Analysis of Reactor Internals under Operational Flow Transients and Steady-State Conditions.

4. Preoperational Flow-Induced Vibration Testing of Reactor Internals.

5. Dynamic System Analysis of the Reactor Internals under Faulted Conditions.

6. Correlations of Reactor Internals Vibration Tests with the Analytical Results.

3.9.2.2 Summary of Application

DCD Tier 1/ITAAC: The Tier 1 information associated with this section is found in DCD Tier 1, Section 2.4.1, "Reactor System," related to reactor internals flow induced vibration (FIV) tests, and DCD Tier 1, Section 2.14, "Initial Test Program," which provides a non-system based description of the US-APWR ITP.

DCD Tier 2: The applicant has provided a DCD Tier 2 system description in Section 3.9.2, "Dynamic Testing and Analysis of Systems, Components, and Equipment," summarized here in part, as follows:

> This section provides the criteria, testing procedures, and dynamic analyses employed to assure that equipment maintains its structural and functional integrity for normal, off normal and postulated loads/events.

ITAAC: The ITAAC associated with DCD Tier 2, Section 3.9.2 are given in DCD Tier 1, Sections 2.4.1 and 2.14.

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

Technical Reports: The technical reports associated with DCD Tier 2, Section 3.9.2 are:

1. MUAP-07023-P, "APWR Reactor Internals 1/5 Scale Model Flow Test Report," Revision 0, issued December 2007.
2. MUAP-07023-P, "APWR Reactor Internals 1/5 Scale Model Flow Test Report," Revision 1, issued May 2009.
3. MUAP-07023-P, "APWR Reactor Internals 1/5 Scale Model Flow Test Report," Revision 2, issued August 2011.
4. MUAP-07027-P, "Comprehensive Vibration Assessment Program for US-APWR Reactor Internals," Revision 0, issued December 2007.
5. MUAP-07027-P, "Comprehensive Vibration Assessment Program for US-APWR Reactor Internals," Revision 1, issued May 2009.
6. MUAP-07027-P "Comprehensive Vibration Assessment Program for US-APWR Reactor Internals," Revision 2, issued August 2011.
7. MUAP-07027-P "Comprehensive Vibration Assessment Program for US-APWR Reactor Internals," Revision 3, issued November 2012.
8. MUAP-08007-P, "Evaluation Results of US-APWR Fuel System Structural Response to Seismic and LOCA Loads," Revision 1, issued August 2010.
9. MUAP-08009, "US-APWR Test Program Description," Revision 1, issued August 2010.
10. MUAP-09002, "Summary of Seismic and Accident Load Conditions for Primary Components and Piping," Revision 0, issued January 2009.
11. MUAP-09004-P, "Summary of Stress Analysis Results for the US-APWR Core Support Structures," Revision 0, issued March 2009.
12. MUAP-10006-P, "Soil-Structure Interaction Analyses and Results for US-APWR Standard Plants," Revision 3, issued November 2012.

Cross-cutting Requirements (TMI, USI/ GSI, Op Ex): There are no cross-cutting requirements for this area of review.

US-APWR Interface Issues Identified in the DCD: See DCD Tier 2, Table 1.8-2, "Compilation of All Combined License Applicant Items for Chapters 1-19."

Site Interface Issues Identified in the DCD: See DCD Tier 2, Table 1.8-1, "Significant Site Specific Interfaces with the Standard US-APWR Design."

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

3.9.2.3 Regulatory Basis

The relevant requirements of the Commission's regulations for this area of review, and the associated acceptance criteria are given in Section 3.9.2, "Dynamic Testing and Analysis of Systems, Components, and Equipment," Revision 3, issued March 2007, of NUREG-0800 and are summarized below. Review interfaces with other SRP sections can be found in Section 3.9.2 of NUREG-0800.

1. 10 CFR 50.55a and GDC 1, as they relate to the design, fabrication, erection, and testing of SSCs in accordance with quality standards commensurate with the importance of the safety function to be performed.

2. GDC 2, and 10 CFR Part 50, Appendix S, as they relate to the ability of SSCs without loss of capability to perform their safety function, to withstand the effects of natural phenomena, such as earthquakes, tornadoes, floods, and the appropriate combination of all loads.

3. GDC 4, as it relates to the protection of SSCs against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that may result from equipment failures and from events and conditions outside the nuclear power unit.

4. GDC 14, as it relates to designing SSCs of the RCPB to have an extremely low probability of rapidly propagating failure and of gross rupture.

5. GDC 15, as it relates to designing the RCS with sufficient margin to assure that the RCPB is not exceeded during normal operating conditions, including AOOs.

6. 10 CFR Part 50, Appendix B, as it relates to the QA criteria for nuclear power plants.

7. 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 has been constructed 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 1.20, “Comprehensive Vibration Assessment Program for Reactor Internals During Preoperational and Initial Startup Testing,” Revision 3, issued March 2007, as it related to a vibration assessment program for which the COL applicant/licensee will be responsible.

2. RG 1.199, “Anchoring Components and Structural Supports in Concrete,” issued November 2003, as it relates to the design of component supports that are anchored in concrete.

3. American Concrete Institute (ACI) Standard ACI-349, Appendix B, as it relates to the design of component supports that are anchored in concrete.

4. NRC Bulletin 79-02, “Pipe Support Base Plate Designs Using Concrete Expansion Anchor Bolts,” Revision 2, issued November 8, 1979, as it relates to the design of component supports that are anchored in concrete.

3.9.2.4 Technical Evaluation

As discussed in Section 3.9.2.1 of this report, there are six main areas of review in Section 3.9.2. Because of their distinctive character, for each area of review, the technical evaluation provides additional details on the summary of the application and identifies the regulatory basis particular to that area of review. The DCD Tier 1 information associated with this section regarding reactor internals FIV tests is found in DCD Tier 1, Section 2.4.1, which is evaluated in Section 14.4 of this report. The DCD Tier 1 information associated with this section regarding the ITP is found in DCD Tier 1, Section 2.14, which is evaluated in Section 14.2 of this report.

3.9.2.4.1 Piping Vibration, Thermal Expansion, and Dynamic Effects Testing

3.9.2.4.1.1 Summary of Application for Piping Vibration, Thermal Expansion, and Dynamic Effects Testing

DCD Tier 1/ITAAC: The Tier 1 information associated with this section is found in DCD Tier 1 Section 2.14, “Initial Test Program,” which provides a non-system based description of the US-APWR ITP.

DCD Tier 2: The applicant has provided a DCD Tier 2 system description in Section 3.9.2.1, “Piping Vibration, Thermal Expansion, and Dynamic Effects,” summarized here in part, as follows. DCD Tier 2, Section 3.9.2.1 describes the piping vibration, thermal expansion, and dynamic effects testing program to verify that the piping and piping restraints will remain within acceptable limits when subjected to piping vibrations and dynamic transients such as those caused by in-line component trip. The systems that are monitored during the program include all Class 1, 2, and 3 piping systems required by the ASME Code, Section III, to undergo a preoperational test program.

DCD Tier 2, Section 3.9.2.1 describes the requirements for system vibration and dynamic effects tests to validate that the piping, components, restraints, and supports of high-energy and moderate-energy systems have been designed to withstand the dynamic effects of transient and steady-state flow-induced vibration and AOOs. Thermal motion is monitored to assure that the predicted thermal movements meet the required design considerations within appropriate support and whip restraint gaps. The TS for the testing program are developed in accordance with the general guidance specified in RG 1.68, “Initial Test Programs for Water-Cooled Nuclear

Power Plants," Revision 3, issued March 2007, for vibration and dynamic effects testing, and specific vibration testing requirements identified in ASME Standards and Guides for Operation and Maintenance of Nuclear Power Plants (OM-S/G) Parts 3 and 7. These specifications address issues such as prerequisites, testing conditions, precautions, measurement techniques, monitoring requirements, test hold points, and acceptance criteria.

3.9.2.4.1.2 Regulatory Basis for Piping Vibration, Thermal Expansion, and Dynamic Effects Testing

The relevant requirements of the Commission's regulations for this area of review, and the associated acceptance criteria are given in SRP Section 3.9.2, SRP Acceptance Criterion 1 and are summarized below.

1. GDC 1, as it relates to structures and components being designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with the importance of the safety function to be performed.

2. GDC 2, as it relates to the ability of SSCs without loss of capability to perform their safety function, to withstand the effects of natural phenomena, such as earthquakes, tornadoes, floods, and the appropriate combination of all loads.

3. GDC 4, as it relates to the protection of SSCs against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that may result from equipment failures and from events and conditions outside the nuclear power unit.

4. GDC 14, as it relates to systems and components of the RCPB being designed, fabricated, erected, and tested so as to have an extremely low probability of rapidly propagating failure or gross rupture.

5. GDC 15, as it relates to the reactor coolant system being designed with sufficient margin to ensure that the RCPB will not be breached during normal operating conditions, including AOOs.

Acceptance criteria adequate to meet the above requirements include:

1. Relevant requirements of GDCs 1, 2, 4, 14, and 15 are met if vibration, thermal expansion, and dynamic effects testing are conducted during startup functional testing for specified high- and moderate-energy piping and their supports and restraints. The purposes of these tests are to confirm that the piping, components, restraints, and supports have been designed to withstand the dynamic loadings and operational transient conditions encountered during service as required by the code and to confirm that no unacceptable restraint of normal thermal motion occurs.

2. RG 1.68, "Initial Test Programs for Water-Cooled Nuclear Power Plants," Revision 3, issued March 2007.

3.9.2.4.1.3 Technical Evaluation of Piping Vibration, Thermal Expansion, and Dynamic Effects Testing

The specific areas of review for testing included the systems that were being monitored, test program details, acceptance criteria, and possible corrective actions when excessive vibration or indications of thermal motion restraint occur. SRP Section 3.9.2 states that piping vibration, safety relief valve vibration, thermal expansion, and dynamic effects testing should be conducted during startup functional testing for specific high- and moderate-energy piping and their supports and restraints to meet the relevant requirements of GDC 1, 14, and 15. The systems to be monitored should include: (i) all ASME Code Class 1, 2, and 3 piping systems, (ii) high-energy piping systems inside seismic Category I structures, (iii) high-energy portions of systems whose failure could reduce the functioning of seismic Category I plant features to an unacceptable safety level, and (iv) seismic Category I portions of moderate-energy piping systems located outside the containment. The purpose of these tests is to confirm that these piping systems, restraints, components, and supports have been adequately designed to withstand the dynamic loadings and operational transient conditions encountered during service as required by the Code, and to confirm that normal thermal motion is not restrained.

In DCD Tier 2, Revision 0, Section 3.9.2.1, the applicant stated that the testing of piping vibration, thermal expansion, and dynamic effects occurs during three phases, which make up the ITP as discussed in DCD Tier 2, Section 14.2.1.2, “Major Phases of Test Program.” The ITP is implemented to verify that the piping and piping restraints will remain within acceptable limits when subjected to piping vibrations and dynamic transients such as those caused by an in-line component trip. The applicant further stated that the SSCs for which this testing is performed are identified in DCD Tier 2, Section 14.2.1, “Summary of Test Program and Objectives,” and the systems listed are those discussed above.

In addition, the applicant stated that when applicable, the test program will include instrumentation lines to the first support in each of the three orthogonal directions from the process pipe or equipment connection point. The staff reviewed DCD Tier 2, Section 3.9.2.1 and DCD Tier 2, Section 14.2.1 but did not find where the applicant identified which specific systems are included in the testing program, and whether, as stated in SRP Section 3.9.2, testing is conducted on all ASME Code, Class 1, 2, and 3 piping systems. The staff requested the applicant in **RAI 204-1569, Question 03.09.02-1** to (a) provide a listing of the high- and moderate-energy piping systems which are covered by the vibration, thermal expansion, and dynamic effects testing program, and (b) verify that the systems to be monitored include all ASME Code Class 1, 2, and 3 piping systems.

In its response to **RAI 204-1569, Question 03.09.02-1**, dated March 25, 2009, the applicant stated that DCD Tier 2, Section 3.9.2.1 will be clarified by identifying the piping systems for which the vibration, thermal expansion, and dynamic effects testing are conducted as part of the ITP. The applicant further stated that in DCD Tier 2, Revision 2, Section 3.9, “Mechanical Systems and Components,” the first paragraph in DCD Tier 2, Revision 0, Section 3.9.2.1, will be changed to specifically site the types of systems to be monitored rather than simply referring to DCD Tier 2, Section 14.2.1.

The staff finds the applicant’s response acceptable because the applicant has clearly identified the piping systems for which the vibration, thermal expansion, and dynamic effects testing will be conducted as part of the initial test program, and the requested information has been incorporated in DCD Tier 2, Revision 2. Accordingly, **RAI 204-1569, Question 03.09.02-1 is resolved.**

DCD Tier 2, Revision 0, Subsection 3.9.2.1, also describes the three phases of ITP (i.e., construction, preoperational, and startup testing). During the construction test phase, the piping, pipe supports, and equipment supports are checked for proper assembly and design setting. The cold settings and cold gaps for pipe supports, whip restraints, equipment, and equipment supports are recorded for major piping systems, including the reactor coolant, RHR, and MS and feedwater systems. The preoperational tests are conducted to assure that the ASME Code, Section III, Class 1, 2, and 3, and other high-energy or seismic Category I piping systems meet the functional requirements, and that piping vibration is within acceptable levels. The specific systems and the associated transients considered in the preoperational test program are outlined in DCD Tier 2, Revision 0, Section 14.2, "Initial Plant Test Program." The startup testing is conducted to verify the performance characteristics of critical pumps, valves, controls, and auxiliary equipment; the requirements of the startup tests are outlined in DCD Tier 2, Subsection 14.2.1.2.

The staff reviewed DCD Tier 2, Section 14.2, and it is not clear whether the vibration, thermal expansion, and dynamic effects testing program simulates actual operating modes. SRP Section 3.9.2 states that an acceptable test program to confirm the adequacy of the design should include a list of the flow modes of operation and transients such as pump trips, valve closures, etc., to which the components will be subjected during the test. For example, the transients of the reactor coolant system heat-up tests should at least include start and trip of the RCP, operation of pressure relieving valves, and closure of turbine stop valve. Therefore, in **RAI 204-1569, Question 03.09.02-2**, the staff requested the applicant to provide a listing of the different flow modes to which the systems will be subjected during the vibration, thermal expansion, and dynamic effects testing program to confirm that the piping systems, restraints, components, and supports have been adequately designed to withstand flow-induced dynamic loadings under the steady-state and operational transient conditions anticipated during service.

In its response to **RAI 204-1569, Question 03.09.02-2**, dated March 25, 2009, the applicant stated that SRP Acceptance Criterion 1 of SRP Section 3.9.2 indicates flow modes of operation and transients like pump trips, valve closures, etc. are applicable during startup functional testing opposed to preoperational testing. The last paragraph in DCD Tier 2, Section 3.9.2.1 discusses hot functional (startup) testing, and references DCD Tier 2, Section 14.2.1.2 for the requirements of startup testing provided in DCD Tier 2, Subsection 14.2.1.2.3, "Startup Tests." To clarify the elements of an acceptable startup test program are incorporated as identified by SRP Section 3.9.2, the applicant agreed to add the different flow modes of operation and transients to which the systems will be subjected during startup functional testing of the specified piping systems in DCD Tier 2, Section 3.9.2.1. The applicant further agreed to revise DCD Tier 2, Section 3.9.2 to provide further detail on the flow modes and transients performed.

The staff finds the applicant's response acceptable because the applicant has provided sufficient information regarding the different flow modes to which the systems will be subjected during the testing program, and the requested information has been incorporated in DCD Tier 2, Revision 2. Accordingly, **RAI 204-1569, Question 03.09.02-2, is resolved**.

DCD Tier 2, Revision 0, Subsection 3.9.2.1.1, "System Vibration and System Dynamic Effects Tests," discusses the requirements for system vibration and dynamic effects tests to validate that the piping, components, restraints, and supports of high-energy and moderate-energy systems have been designed to withstand the dynamic effects of transient and steady-state flow-induced vibration and AOOs. The applicant stated in DCD Tier 2, Subsection 3.9.2.1.1 that the detailed technical specifications for the piping vibration, thermal expansion, and dynamic effects testing program are written in accordance with the general guidance specified in RG

1.68 and the specific vibration testing requirements of ASME OM-S/G Part 3, and that they address such issues as prerequisites, test conditions, precautions, measurement techniques, monitoring requirements, test hold points, and acceptance criteria. However, the applicant did not provide any details regarding specific locations in the systems where visual inspections and other measurements would be performed during the testing program. SRP Section 3.9.2 states that to confirm the adequacy of the design, an acceptable piping vibration, thermal expansion, and dynamic effects program should include a list of selected locations in the piping system at which visual inspections and, as needed, measurements will be performed during the testing.

In **RAI 204-1569, Question 03.09.02-3**, the staff requested the applicant to provide (a) a list of locations in the piping system selected for visual inspection and other measurements during the vibration, thermal expansion, and dynamic effects testing program and (b) assurance that for each of these selected locations, the deflection, pressure, or other appropriate criteria should be obtained during the tests to confirm that the stress and fatigue limits are within the design levels.

In its response to **RAI 204-1569, Question 03.09.02-3**, dated March 25, 2009, the applicant stated that the ITP plan will include a list of locations in the specific piping systems that are selected for visual inspection and other measurements during the vibration, thermal expansion, and dynamic effects testing program. In addition, the ITP plan will include acceptance criteria for the deflection, pressure, and/or other appropriate criteria to be obtained during the tests to determine if the stress and fatigue limits are within design levels.

In DCD Tier 2, Revision 2, Section 3.9.2, the applicant made a commitment that the ITP will be updated to include the list of locations for visual inspection and other measurements as well as acceptance criteria to confirm that the stress and fatigue limits are within the design levels. The applicant also revised appropriate portions of DCD Tier 2, Section 14.2, which contains a description of the ITP. The processes and controls employed for development of the ITP details such as test specifications, test procedures, test procedure performance, test results report development, test results acceptance, and test closeout, are described in the applicant's Technical Report MUAP-08009, "US-APWR Test Program Description," Revision 1, issued August 2010. Furthermore, in DCD Tier 2, Section 14.2.3, "Test Procedures," the applicant stated that approved test procedures will be provided to the NRC at least 60 days prior to their use. Therefore, based on the applicant's DCD revision to include within the scope of the ITP the requested information, **RAI 204-1569, Question 03.09.02-3, is resolved**.

In DCD Tier 2, Revision 0, Section 3.9.2.1.1, the applicant also stated that the detailed test specifications of the testing program are written in accordance with the general guidance of RG 1.68 and the specific vibration testing requirements of ASME OM Part 3 and address such issues as prerequisites, testing conditions, precautions, measurement techniques, monitoring requirements, test hold points, and acceptance criteria. The staff reviewed DCD Tier 2, Section 3.9.2.1, but did not find where the applicant had provided the details regarding any one of these issues. SRP Section 3.9.2 states that to confirm the adequacy of the design, an acceptable piping vibration and dynamic effects program should include a list of locations selected for visual inspections and other measurements, the acceptance criteria, and possible corrective actions if excessive vibrations are observed. Therefore, in **RAI 204-1569, Question 03.09.02-4**, the staff requested the applicant to provide a detailed description of the vibration and dynamic effects testing program to address issues such as measurement techniques, monitoring guidelines, acceptance criteria, and corrective action if necessary, and discuss in detail how compliance with the acceptance criteria is verified for the different measurement techniques. The discussion should include the vibration measurement and analysis methodology, monitoring requirements including considerations for selection of monitoring locations, test evaluation and acceptance criteria, and possible corrective action when the criteria are violated.

In its response to **RAI 204-1569, Question 03.09.02-4**, dated March 25, 2009, the applicant stated that the ITP plan will include the detailed test specifications for the testing program, which are written in accordance with the general requirements of RG 1.68 and the specific vibration testing requirements of ASME OM. The program will address such issues as prerequisites, test conditions, precautions, measurement techniques, monitoring requirements, test hold points, and acceptance criteria outlined in applicable sections of RG 1.68 and ASME OM. If the vibration is noted beyond the acceptable levels, corrective restraints are to be designed, incorporated in the piping system analysis, and installed.

The applicant revised DCD Tier 2, Revision 2, Section 3.9.2.1 to state that the requested information will be included in the ITP according to ASME OM Code and RG 1.68. Therefore, based on the applicant's commitment to revise the ITP to include the requested information, **RAI 204-1569, Question 03.09.02-4, is resolved**.

In the review of DCD Tier 2, Revision 0, Subsections 3.9.2.1, the staff found that the applicant did not identify which specific measurement technique would be used for a particular system. The choice of measurement technique depends on factors such as the safety significance of the system, expected mode and magnitude of the vibrations, and accessibility of the system during testing. SRP Section 3.9.2 states that an acceptable piping vibration and dynamic effects program to confirm the adequacy of the design should include a list of locations in the piping system selected for visual inspections and, as needed, other measurements during the testing. In **RAI 204-1569, Question 03.09.02-5**, the staff requested the applicant to identify which measurement technique (e.g., visual observation, remote monitoring, or local measurements) would be used for each of the piping systems covered by the vibration and dynamic effects testing program, and to describe methods used to anticipate piping movements and deflections.

In its response to **RAI 204-1569, Question 03.09.02-5**, dated March 25, 2009, the applicant stated that the ITP plan will identify which measurement technique (e.g., visual observation, remote monitoring, or local measurements) is to be used for each of the piping systems covered by the vibration and dynamic effects testing program. In addition, methods used to anticipate piping movements and deflections will be described. The applicant revised DCD Tier 2, Revision 2, Section 3.9.2.1 to state that the requested information will be included in the ITP. Therefore, based on the applicant's DCD revision to include within the scope of the ITP the requested information, **RAI 204-1569, Question 03.09.02-5, is resolved**.

DCD Tier 2, Subsection 3.9.2.1.1, states that the time-dependent dynamic analyses are performed on the system by combining transient-induced stresses with other operating stresses in accordance with the criteria identified in DCD Tier 2, Section 3.9.3, "ASME Code Class 1, 2, and 3 Components, Component Supports, and Core Support Structures," and Section 3.12, "Piping Design Review." The detailed test procedures include the criteria for the evaluation of data from the pipe movement monitors. The load, load combinations, load definitions, and required stress limits for ASME Code, Section III, Class 1, 2, and 3 components and structures are presented in DCD Tier 2, Tables 3.9-3, "Design Loading Combinations for ASME Code, Section III, Class 1, 2, and 3 CS Systems and Components," to 3.9-6, "Stress Criteria for ASME Code, Section III, Class 1, Components and Supports and Class CS Core Supports," and for piping and piping support in DCD Tier 2, Tables 3.12-1, ASME Code, Section III, Class 1, 2, 3, CS and Support Load Symbols and Definitions," to 3.12-4, "Loading Combinations for Piping Supports." As described in DCD Tier 2, Table 3.9-6, the ASME Code, Section III, Class 1 stress analyses for components and core structures consider sustained load (including dead load, pressure, and thermal expansion), system operational transient loads (thermal and fluid pressure transients), seismic loads, and pipe rupture loads [design pipe breaks, unless modified by LBB evaluations, and LOCA]. The applicant stated in DCD Tier 2, Table 3.9-6 that the ASME Code, Section III, Class 1 pressure boundary components are subject to fatigue usage evaluations over the 60-year design life, and that the environmental effects on fatigue of these components follow the guidance delineated in RG 1.207, "Guidelines for Evaluating Fatigue Analyses Incorporating the Life Reduction of Metal Components Due to the Effects of the Light-Water Reactor Environment for New Reactors," March 2007. Through this RG, the staff endorsed a new stainless steel air design curve given in Appendix A, Figure A.3 of NUREG/CR-6909, "Effect of LWR Coolant Environments on the Fatigue Life of Reactor Materials," issued February 2007. Recent evaluations of stainless steel test data indicate the ASME fatigue design curve is inconsistent with the appropriate test material and conduct of the fatigue test. Earlier the staff reviewed the non-conservatism of the current ASME Code design curve with respect to the existing fatigue data for austenitic stainless steels and endorsed use of the new stainless steel air curve for fatigue evaluations of ASME Code, Class 1 components.

The staff reviewed relevant sections of the DCD, and it is not clear whether the guidance of RG 1.207 regarding the new stainless steel air design curve will be followed for fatigue evaluations of ASME Code, Class 1 components of US-APWR. SRP Section 3.9.2 states that an acceptable piping vibration and dynamic effects program to confirm the adequacy of the design should include a list of locations in the piping system selected for measurements during the testing, and that for each of these locations, the deflection, pressure, or other appropriate criteria should be obtained during the tests to confirm that the stress and fatigue limits are within the design levels. In **RAI 204-1569, Question 03.09.02-6**, the staff requested the applicant to verify that the fatigue evaluations of ASME Code, Class 1 components will be performed in accordance with the guidelines of RG 1.207. Alternatively, the applicant was requested to provide a technical basis for using ASME Code, Section III, Appendix I, Figures I-9.2.1 and I-9.2.2 for the fatigue evaluations.

In its response to **RAI 204-1569, Question 03.09.02-6**, dated March 25, 2009, the applicant provided the following information:

> The allowable stresses using the evaluation of the pipe vibration required by SRP Section 3.9.2, SRP Acceptance Criteria 7, are clearly determined based on ASME OM-S/G-1990, Part 3, "Requirements for Preoperational and Initial Start-up Vibration Testing of Nuclear Power Plant Piping System." ASME OM Standard Part 3 specifies the usage of the design fatigue curve of Figures I-9.2.1 and I-9.2.2, ASME Code, Section III Appendix I. Therefore, the applicant carries out the fatigue evaluation following ASME as described in the SRP. Figure 1.3 of NUREG/CR-6909, Appendix A, is the fatigue design curve for the evaluation of the environmental fatigue; it is not used for the evaluation of the piping vibration.

Based on the information provided by the applicant regarding the basis for using Figures I-9.2.1 and I-9.2.2 for the fatigue evaluations, the staff finds that the applicant has provided adequate information to resolve the staff's concerns. Accordingly, **RAI 204-1569, Question 03.09.02-6, is resolved**.

In DCD Tier 2, Revision 0, Subsection 3.9.2.1.1, the applicant stated that for steady-state vibrations, the alternating stress amplitudes, S_{alt}, are determined from the maximum amplitudes measured during the initial operation and stress intensity levels based on the guidance of ASME

OM. DCD Tier 2, Subsections 3.9.2.1.1.1, “ASME Class Piping Systems,” and 3.9.2.1.1.2 “ASME Class 2 and 3 or ANSI B31.1 Piping,” give the ASME Code, Section III acceptable limit for maximum alternating stress for ASME Class 1 piping systems and ASME Class 2 and 3 [or ANSI B31.1] piping, respectively. These vibration criteria are based on ASME OM Standard, paragraph 3.2.1.2; for austenitic stainless steels, the stress limits are obtained from Figures I-9.2.1 and I-9.2.2 of the Mandatory Appendix I to Section III of the ASME BPV Code. As discussed in **RAI 204-1569, Question 03.09.02-6**, the applicant has provided a technical basis for using the ASME Code fatigue design curves for stainless steels. In addition, the DCD states that the results of vibration analysis are used to determine if support or system modifications are necessary. If modifications are performed, the system is retested until acceptable results are achieved. The staff finds this acceptable because piping vibration and dynamic effects analyses ensure that the stress and fatigue limits are within the design levels.

In DCD Tier 2, Subsection 3.9.2.1.2, “System Thermal Expansion Program,” the applicant stated that the thermal expansion tests are developed in accordance with the guidance of ASME OM, Part 7. However, the applicant did not provide a detailed description of the program in the DCD. SRP Section 3.9.2 states that an acceptable thermal expansion program to confirm the adequacy of the design should include a description of the thermal-motion monitoring program. In **RAI 204-1569, Question 03.09.02-7**, the staff requested the applicant to provide a detailed description of the thermal motion monitoring program for verification of snubber movement, adequate clearances and gaps, the acceptance criteria, and the method regarding how motion will be measured.

In its response to **RAI 204-1569, Question 03.09.02-7**, dated March 25, 2009, the applicant stated that DCD Tier 2, Subsection 3.9.2.1.2 will be changed to clarify that the provision of a detailed description of the thermal motion monitoring program, as identified by SRP Section 3.9.2, Acceptance Criterion 1, Item E, will be included as part of the ITP plan. The thermal motion monitoring program will include verification of snubber movement, adequate clearances and gaps, the acceptance criteria, and how the motion is to be measured.

The applicant revised DCD Tier 2, Section 3.9.2.1 to state that the requested information will be included in the ITP. Therefore, based on the applicant’s DCD revision to include within the scope of the ITP the requested information, the staff’s concerns are resolved. Accordingly, **RAI 204-1569, Question 03.09.02-7, is resolved**.

In DCD Tier 2, Subsection 3.9.2.1.2, the applicant stated that the specifications of manufactured standard supports, such as spring hangers, snubbers, and struts, are reviewed during the construction phase of the ITP to verify that thermal expansion is accommodated within acceptable limits during various operational modes. The applicant further stated in DCD Tier 2, Section 3.9.2.1, that operability of snubbers is checked during the preoperational testing by comparing their hot and cold positions with calculated values. The staff reviewed the DCD but did not find where the applicant had identified the snubbers that experience sufficient thermal movement, or described the acceptance criteria to ensure that the snubbers are operable. SRP Section 3.9.2 states that an acceptable piping vibration, thermal expansion, and dynamic effects program should include, in addition to measurements of snubber travel from cold to hot position, a description of the corrective action to be taken to ensure that the snubber is operable when no snubber piston movement is measured at these stations. In **RAI 204-1569, Question 03.09.02-8**, the staff requested the applicant to provide a list of snubbers on systems which experience sufficient thermal movement to measure snubber travel from cold to hot position during the vibration, thermal expansion, and dynamic effects testing program, and to describe the procedure used to verify snubber operability when no snubber piston movement is noted.

In its response to **RAI 204-1569, Question 03.09.02-8**, dated March 25, 2009, the applicant stated that SRP Section 3.9.2, SRP Acceptance Criterion 1, indicates the list of snubbers and measure of travel from cold to hot position are applicable during startup functional testing as opposed to preoperational testing. DCD Tier 2, Section 3.9.2.1 will be changed to indicate the list of snubbers on systems, which experience sufficient thermal movement to measure snubber travel from cold to hot position. In addition, the ITP plan will include the procedure necessary to verify snubber operability when no snubber piston movement is noted.

The applicant revised DCD Tier 2, Revision 2, Section 3.9.2.1 to state that the request information will be included in the ITP. Therefore, based on the applicant's inclusion of the requested information within the scope of the ITP, **RAI 204-1569, Question 03.09.02-8, is resolved**.

The applicant stated in DCD Tier 2, Subsection 3.9.2.1.2 that system thermal expansion tests are developed in accordance with the guidance of ASME OM Part 7. The applicant stated that thermal monitoring of the systems was performed to assure that predicted thermal movements meet the required design considerations within appropriate support and whip restraint gaps. Excessive thermal deflections are noted and checked against as-analyzed piping results. The applicant does not provide possible corrective actions when excessive indications of thermal motion are recorded. The staff finds this information provided in the DCD inadequate to ensure that no unacceptable restraint of normal thermal motion occurs. SRP Section 3.9.2 states that the piping vibration, thermal expansion, and dynamic effects testing program should include possible corrective actions if excessive indications of thermal motion restraint occur. In **RAI 204-1569, Question 03.09.02-9**, the staff requested the applicant to describe possible corrective actions taken when excessive thermal motion is noted during system thermal expansion testing.

In its response to **RAI 204-1569, Question 03.09.02-9**, dated March 25, 2009, the applicant stated that DCD Tier 2, Subsection 3.9.2.1.2 will be changed to clarify that if piping system restraints are determined during the test to be inadequate or are damaged, corrective restraints are to be installed and another test performed to determine whether the thermal motion has been reduced to an acceptable level.

Based on the information provided by the applicant regarding the corrective actions taken when restraints are determined to be inadequate or are damaged during system thermal expansion testing, the staff finds that the applicant has provided adequate information, and the requested information has been incorporated in the DCD. Accordingly, **RAI 204-1569, Question 03.09.02-9, is resolved**.

3.9.2.4.1.4 Conclusions for Piping Vibration, Thermal Expansion, and Dynamic Effects Testing

The staff concludes that by having an acceptable vibration, thermal expansion, and dynamic effects test program that will be conducted during preoperational and startup on specified high- and moderate-energy piping, and all associated systems, restraints, and supports, the applicant shall meet the relevant requirements of GDCs 1, 2, 4, 14, and 15. These tests provide adequate assurance that the piping, components, restraints, and supports have been designed to withstand the dynamic loadings and operational transient conditions encountered during service, and that adequate clearances and free movement of snubbers exist for unrestrained thermal movement of piping and supports during normal system heat-up and cool-down operations.

3.9.2.4.2 Seismic Analysis and Qualification of Seismic Category I Mechanical Equipment

3.9.2.4.2.1 Summary of Application for Seismic Analysis and Qualification of Seismic Category I Mechanical Equipment

DCD Tier 1/ITAAC: There is no Tier 1 information associated with this section. DCD Tier 1, lists system based ITAAC for seismic Category 1 components.

DCD Tier 2: The applicant has provided a DCD Tier 2 system description in Section 3.9.2.2, “Seismic Analysis and Qualification of Seismic Category I Mechanical Equipment,” summarized here in part, as follows. DCD Tier 2, Section 3.9.2.2, describes the seismic analysis methods and seismic qualification testing of safety-related mechanical equipment and associated supports. Seismic qualification of the equipment is performed by analysis, testing, or a combination of the two. The seismic qualification testing and analysis methods are described in DCD Tier 2, Sections 3.10.2, “Methods and Procedures for Qualifying Mechanical and Electrical Equipment and Instrumentation,” and 3.10.3, “Methods and Procedures of Analysis or Testing of Supports of Mechanical and Electrical Equipment and Instrumentation.” The seismic analysis methods, including response spectrum analysis, time history analysis, and equivalent static load analysis, are discussed in DCD Tier 2, Sections 3.7.2, “Seismic System Analysis,” and 3.7.3, “Seismic Subsystem Analysis,” and the method for piping and supports is described in DCD Tier 2, Section 3.12. The methods for qualification of mechanical equipment and supports are based on the guidelines of Institute of Electrical and Electronic Engineers (IEEE) Std. 344-1987, “IEEE Recommended Practice for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations,” ASME QME-1-2007, “Qualification of Active Mechanical Equipment Used in Nuclear Power Plant” and DCD Tier 2, Sections 3.10.2 and 3.10.3.

The areas of review included seismic analysis methods, determination of number of earthquake cycles, basis for selection of frequencies, three components of earthquake motion, combination of modal responses, analytical procedures for piping systems, multiply-supported equipment and components with distinct inputs, use of constant vertical static factors, torsional effects of eccentric masses, seismic Category I buried piping systems, interaction of other piping with seismic Category I piping, and criteria used for damping.

3.9.2.4.2.2 Regulatory Basis for Seismic Analysis and Qualification of Seismic Category I Mechanical Equipment

The relevant requirements of the Commission’s regulations for this area of review, and the associated acceptance criteria are given in SRP Section 3.9.2, SRP Acceptance Criterion 2 and are summarized below.

1. 10 CFR 50.55a, as it relates to codes and standards.

2. GDC 2, as it relates to systems, components, and equipment important to safety being designed to withstand appropriate combinations of the effects of normal and accident conditions with the effects of expected natural phenomena without loss of capability to perform their safety functions.

3. 10 CFR Part 50, Appendix B, as it relates to QA.

4. 10 CFR Part 50, Appendix S, as it relates to the suitability of the plant design bases for mechanical components established in consideration of site seismic characteristics.

Acceptance criteria adequate to meet the above requirements include:

1. RG 1.61, Revision 1, "Damping Values for Seismic Design of Nuclear Power Plants."

2. RG 1.92, Revision 2, "Combining Modal Responses and Spatial Components in Seismic Response Analysis."

3.9.2.4.2.3 Technical Evaluation of Seismic Analysis and Qualification of Seismic Category I Mechanical Equipment

The review of the seismic analysis and qualification of seismic Category I mechanical equipment was performed in accordance with SRP Section 3.9.2 to ensure conformance with GDC 2. The review consisted of an evaluation of DCD Subsection 3.9.2.2; portions of DCD Tier 2, Sections 3.7.2, 3.7.3, 3.10, "Seismic and Dynamic Qualification of Mechanical and Electrical Equipment," and 3.12; DCD Tier 2, Appendices 3C "Reactor Coolant Loop Analysis Methods," 3F, "Design of Conduit and Conduit Support," and 3G "Seismic Qualification of Cable Trays and Support," and the applicant's Technical Report MUAP-10006-P, "Soil-Structure Interaction Analyses and Results for US-APWR Standard Plants," Revision 3, issued November 2012.

DCD Tier 2, Subsection 3.9.2.2.1, "Seismic Qualification Testing," states that the seismic Category I mechanical equipment and supports are designed to withstand the combined effects of postulated earthquakes and normal and accident conditions without loss of their intended safety-related function. Seismic qualification of the equipment is performed by analysis, testing, or a combination of both testing and analysis. The seismic qualification of safety-related mechanical equipment by testing and analysis is performed in accordance with the recommendations of ANSI/IEEE Std. 344-1987 and ASME QME-1-2007, as endorsed by NRC, RG 1.100, Revision 2, "Seismic Qualification of Electric and Mechanical Equipment for Nuclear Power Plants," issued June 1998. The seismic qualification testing and analysis methods are described in DCD Tier 2, Sections 3.10.2 and 3.10.3. The review of seismic qualification of seismic Category I mechanical equipment and supports by analysis includes the review areas identified under SRP Section 3.9.2, SRP Acceptance Criterion 2.

3.9.2.4.2.3.1 Seismic System Analysis Methods

DCD Tier 2, Subsection 3.9.2.2.2, "Seismic System Analysis Methods," states that the seismic system and subsystem analysis methods, including response spectrum analysis, time history analysis, and equivalent static load analysis, are discussed in DCD Tier 2, Sections 3.7.2 and 3.7.3, and the method for piping and supports is described in DCD Tier 2, Section 3.12.

The applicant stated in DCD Tier 2, Subsection 3.9.2.2.2, that the stiffness of the seismic subsystem anchorage must be determined, and the assumptions made in the seismic analysis must be verified as accurately reflecting the mounting condition. In **RAI 214-1920, Question 03.09.02-34**, the staff requested the applicant to address the following:

1. Discuss how the dynamic characteristics of the support anchorages, including base plate and anchor bolts or through bolts, connecting to the building structure are determined. Discuss how the equipment seismic analysis accounts for the dynamic characteristics of the support anchorage, especially for heavy equipment.

2. Provide the plant-specific compensatory measures or quality control/assurance programs used to alleviate the effects of anchor bolt torque relaxation. Anchor bolt torque relaxation may occur after years of operation and cause reduction in the natural frequency of the equipment and support assembly, and increase in its seismic response.

3. Clarify whether, and explain why, expansion anchor bolts will or will not be used for safety-related systems and components.

In its response to **RAI 214-1920, Question 03.09.02-34**, dated April 30, 2009, the applicant provided the following information regarding the characteristics of the support anchorages:

> The flexibility of the equipment support anchorages, which includes the base plate and anchor bolts or through bolts, may affect the natural frequency of the equipment and support assembly and therefore may affect the equipment seismic analysis results. To account for the support anchorage flexibility effects, the stiffness of the anchorage system is considered in the equipment seismic analysis when significant. The stiffness of the anchorage can be determined either by a hand calculation method for a simple anchorage (anchor bolt flexibility only) or by a finite element analysis method for a complicated anchorage arrangement (base plate and anchor bolt flexibility).

The applicant provided details for determining whether stiffness is significant. The applicant stated that when the in-structure response spectra (ISRS) peak acceleration is used for the seismic design load, stiffness is ignored, and if the ISRS peak is not used, then anchorage stiffness is considered. The applicant also discussed analytical techniques and computer programs for determining the stiffness of the anchorage system. The applicant stated that an example of a computer program designed to perform analysis of the pipe support structure, including the base plate flexibility per NRC Bulletin 79-02, "Pipe Support Base Plate Designs Using Concrete Expansion Anchor Bolts," issued March 8, 1979, is E/PD STRUDL, which is listed in DCD Tier 2, Subsection 3.12.4.1.1, "List of Programs."

Regarding the support for heavy equipment the applicant stated that they are modeled as spring elements in the dynamic analysis of the RCL- R/B- PCCV- CIS coupled model that is presented in MUAP-10006-P, Revision 3.

Regarding torque relaxation of expansion type anchor bolts, the applicant stated that the US-APWR design is in accordance with the requirements of NRC Bulletin 79-02, Revision 2, issued November 8, 1979, which should help alleviate the problems associated with bolt torque relaxation.

Regarding the use of expansion anchor bolts the applicant stated that embedded plates or cast-in-place anchor rods are the preferred choices for the US-APWR anchorage for safety-related systems and components, and expansion anchors conforming to the requirements and provisions of ACI 349 Appendix B, RG 1.199, "Anchoring Components and Structural Supports in Concrete," issued November 2003, and NRC Bulletin 79-02 are used sparingly, and only when necessary. The applicant further stated that post-installed expansion anchor bolts such as wedge-type or sleeve-type anchors, which rely on friction, will not be used for the systems or components subject to the vibratory motion in the normal operation conditions, since the anchor bolt torque relaxation may occur under this condition. Also, wedge-type or sleeve-type expansion anchors will also not be used for heavily loaded supports or equipment where limited by their allowable load capacities.

Based on a review of the information provided by the applicant regarding the seismic analysis of subsystem anchorage, the staff finds the response acceptable because the minimal use of expansion bolts should keep the potential problem of bolt torque relaxation under control. Accordingly, **RAI 214-1920, Question 03.09.02-34, is resolved**.

The staff further noted that the DCD does not provide sufficient information to allow the review of the seismic subsystem analysis. The anchorage stiffness can affect natural frequencies of the subsystem significantly. Therefore, in **RAI 205-1584, Question 03.09.02-10**, the staff requested the applicant to provide a list of anchorage type, the method for determining their stiffness, the related assumptions, and the procedure for verification of the assumptions.

In its response to **RAI 205-1584, Question 03.09.02-10**, dated April 30, 2009, the applicant presented a description of the types of anchorages considered for use on the US-APWR standard plant and described the methods used for seismic analyses of the subsystems (i.e., anchorages). The applicant also referred to the response to **RAI 205-1584, Question 03.09.02-34**, discussed above, for further information. The applicant stated that the equivalent static load method of analysis is the preferred method for use in seismic analysis of subsystems such as equipment and piping anchorages. The applicant further stated that when necessary, dynamic analysis methods are employed and the stiffness of the anchorage assembly is determined by a hand calculation or by a FE analysis (used for more complicated anchorages). In its response the applicant also included a list of anchorage types. The applicant also committed to include a list which would summarize the method for determining the stiffness, the related assumptions, and the procedure for verification of the assumptions for all of the anchorage types considered for use on the US-APWR.

The staff reviewed the response and noted that the equivalent static load method of analysis is the preferred method for use in seismic analysis of subsystems such as equipment and piping anchorages. The staff also noted that SRP Section 3.9.2, Acceptance Criterion 2.A.(ii) states, "An equivalent static load method is acceptable if: (1) there is a justification that the system can be realistically represented by a simple model and the method produces conservative results in responses, (2) the design and simplified analysis account for the relative motion between all points of supports, and (3) to obtain an equivalent static load of equipment or components which can be represented by a simple model, a factor of 1.5 is applied to the peak acceleration of the applicable floor response spectrum. A factor of less than 1.5 may be used with adequate justification." The applicant did not provide detailed technical information to demonstrate how these three criteria were being satisfied. Therefore, the staff closed as unresolved **RAI 205-1584, Question 03.09.02-10** and in follow-up **RAI 498-3782, Question 03.09.02-59,** the staff requested the applicant in to provide detailed technical information to demonstrate how the acceptance criteria are satisfied, and in accordance with the commitment in the response to **RAI 205-1584, Question 03.09.02-10**, to include in DCD Revision 2 a list which would summarize the method for determining the stiffness, the related assumptions, and the procedure for verification of the assumptions for all of the anchorage types considered for use on the US-APWR.

In its response to **RAI 498-3782, Question 03.09.02-59**, dated January 15, 2010, the applicant referred to its responses to several staff questions closely related to this **RAI 498-3782, Question 03.09.02-59**. The applicant reaffirmed its commitment to comply with the SRP acceptance criteria. The applicant further stated that details of the various anchorage types, the method for determining their stiffness, the related assumptions, and the verification of the assumptions can only be prepared after all anchorage systems are finalized.

The staff noted that the main thrust of **RAI 498-3782, Question 03.09.02-59,** was on conformance with the three SRP Section 3.9.2 acceptance criteria, which must be met in order to use equivalent static load methods in performing seismic analyses of subsystems. The staff reviewed the previous responses and documents referred in the applicant's response, and found that the following information demonstrated compliance with the three criteria: (a) the applicant's response to **RAI 213-1951, Question 03.07.03-1, Subquestion 3.7.3-15**, April 24, 2009, explains that the applicant will show that the equipment and piping systems that use the equivalent static method for anchorage design have a dominant single mode; (b) DCD Tier 2, Section 3.7.3.1, "Seismic Analysis Methods," indicates that the independent support motion (ISM) method given in NUREG-1061, "Evaluation of Other Dynamic Loads and Load Combinations," Volume 4, issued December 1984, will be used to account for the relative motion between points of supports; and (c) the applicant's response to **RAI 213-1951, Question 03.07.03-1, Subquestion 3.7.3-02**, dated March 27, 2009, provided a satisfactory explanation that shows how the factor of 1.5 is used to obtain the seismic demand on the SSCs. Therefore, the staff finds the applicant's response to **RAI 498-3782, Question 03.09.02-59,** to be acceptable. The applicant revised DCD Tier 2, Revision 2, Section 3.9.3.4, to include the details of anchor design, type, and base plate arrangement. Accordingly, **RAI 498-3782, Question 03.09.02-59, is resolved**.

The design of conduit and conduit support is discussed in DCD Tier 2, Appendix 3F. The applicant stated that seismic Category II conduit systems, the electrical conduit containing non-1E cable in seismic Category I buildings, are not essential for safe shutdown of the plant and need not remain functional during, and after, a SSE. However, such conduit systems must not fall or displace excessively where they could damage any seismic Category I SSCs. In DCD Tier 2, Section 3F.1.2, "Seismic Category II Conduit Systems," for seismic Category II conduit systems, the applicant stated that these conduit systems, including support anchorages, are analyzed and designed by the COL applicant for the site SSE using the same methods and stress limits specified for seismic Category I structures and subsystems, except structural steel in-plane stress limits are permitted to reach 1.0 F_y. In **RAI 204-1584, Question 03.09.02-40**, the staff requested the applicant to clarify where in the DCD this COL information item is described.

In its response to **RAI 214-1920, Question 03.09.02-40**, dated April 30, 2009, the applicant stated the following:

> Site-specific seismic Category II (and I) structures, systems, and components (SSCs) are designed by the COL applicant. These include subsystems such as conduits, cable trays, and ducts whose design is dependent on the in-structure response spectra developed for the building structure to which the subsystems are mounted. This process is described collectively by COL items 3.7(3), 3.7(4), 3.7(21), 3.7(26), 3.8(15), and 3.8(19).

> The site-specific SSE is developed by the COL applicant from the site-specific ground motion response spectra (GMRS) and foundation input response spectra (FIRS). The site-specific SSE is increased if necessary to envelope the minimum response spectra required by 10 CFR Part 50, Appendix S. This process is described collectively by COL items 3.7(5), 3.7(6), 3.7(22), 3.7(24), 3.7(26), and 3.7(30). Note that for conduit, cable tray, and duct subsystems located in standard plant buildings, the ISRS used will be based on the certified seismic design response spectra (CSDRS). If located in site-specific buildings and structures, the conduit, cable tray, and duct subsystem seismic design will be based on the site-specific SSE.

The staff finds the applicant's response acceptable because designing both seismic Category I and II SSCs, such as conduits, cable trays, and ducts based on the ISRS to which the subsystems are mounted meets the SRP guidance. Accordingly, **RAI 214-1920, Question 03.09.02-40 is resolved**.

In DCD Tier 2, Section 3.7.2.9, "Effects of Parameter Variations on Floor Response Spectra," the applicant stated that to account for variations in the structural frequencies due to the uncertainty in parameters, such as material and mass properties of the structures, damping values, and soil properties, SSI analysis techniques, and the seismic modeling methods, the computed ISRS are smoothed by filling valleys between peaks. The staff reviewed MUAP-10006-P, Revision 3, and found that the computed ISRS are properly smoothed and valleys are filled in the Figures 3-B.1.0-1, "Design-Basis ISRS at Top of Reactor Cavity – 3% Damped Response in NS Direction (X)," through 3-B.1.0-51, "Design-Basis ISRS at Top of PCCV – 5% Damped Response in Vertical Direction (z) at Model Elevation 232'-0"," for the use of subsystem seismic analysis.

3.9.2.4.2.3.2 Determination of Number of Earthquake Cycles

As discussed in DCD Tier 2, Section 3.7.1.1, "Design Ground Motion," the OBE is chosen as 1/3 of the SSE for the US-APWR; therefore, an explicit design or analysis is not required for the OBE. For fatigue evaluation, the number of earthquake cycles is based on the guidance of SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs," issued April 2, 1993, and the associated SRM, issued July 21, 1983, as discussed in DCD Tier 2, Section 3.7.1.1 and DCD Tier 2, Section 3.10.2. The number of earthquake cycles to consider is two SSE events with 10 maximum stress cycles per event. Alternatively, a number of fractional peak cycles equivalent to the maximum peak cycles for five ½ SSE events when followed by one full SSE may be used in accordance with Appendix D of IEEE Std. 344-1987. For piping analysis, the number of earthquake cycles is defined in DCD Tier 2, Section 3.12, Table 3.12-2, "Loading Combinations for ASME Code, Section III, Class 1 Piping." Note 3.

3.9.2.4.2.3.3 Basis for Selection of Frequencies

To avoid resonance, all components and equipment are designed such that their fundamental frequencies are either less than one-half or more than twice the dominant frequencies of the forcing frequencies of the support structure.

3.9.2.4.2.3.4 Three Components of Earthquake Motion

For seismic Category I systems and subsystems, the combination of three components of earthquake motion is described in DCD Tier 2, Sections 3.7.2.6, “Three Components of Earthquake Motion,” and 3.7.3.4, “Three Components of Earthquake Motion.” Two horizontal components and one vertical component of seismic response spectra, associated with the SSE, are employed as input to a modal response spectra analysis. In its response spectra and equivalent static analyses, the effects of the three components of earthquake motion are combined in accordance with RG 1.92, Revision 2, using one of the following methods:

- The peak responses due to the three earthquake components from the response spectra analyses are combined using the square root sum of the squares (SRSS) method.

- The peak responses due to the three earthquake components are combined directly, using the assumption that when the peak response from one component occurs, the responses from the other two components are 40 percent of the peak (i.e., the Newmark 100 percent-40 percent-40 percent method). Combinations of seismic responses from the three earthquake components, together with variations in sign (plus or minus), are considered.

The applicant stated that for piping analysis, the three sets of mutually orthogonal components of earthquake motion are combined in accordance with RG 1.92, Revision 1, by the SRSS method; details are discussed in DCD Tier 2, Section 3.12.3.2, “Modal Response Spectrum Method.” DCD Tier 2, Subsection 3.12.3.2.6, “Seismic Anchor Motions,” states that where supports are located within different structures, the seismic motions at these locations are assumed to move 180 degree out-of-phase (i.e., the most unfavorable condition) in the analysis. Where supports are located within a single structure, the seismic motions are considered to be in-phase, and the relative displacement between the support locations is considered in the analysis.

3.9.2.4.2.3.5 Combination of Modal Responses

In DCD Tier 2, Sections 3.7.2.7, “Combination of Modal Responses,” and 3.7.3.5, “Combination of Modal Responses,” and Subsection 3.9.2.2.6, “Combination of Modal Responses,” the applicant stated that when the response spectrum method of analysis is used, because the phase relationship between various modes is lost, and only the maximum responses for each mode are determined, the modal responses are combined by one of the RG 1.92, Revision 2, methods or by the 10 percent grouping method. The applicant further stated that in some applications, the more conservative modal combination methods contained in Revision 1 of RG 1.92 are also used, as permitted in Revision 2 of RG 1.92. The staff found this method acceptable because it meets the RG guidance.

The applicant also stated that for piping analysis, as discussed in DCD Tier 2, Section 3.12.3.2, the individual modal results are combined in accordance with RG 1.92, Revision 1. The response spectra method consists of either the uniform support motion (USM) or independent support motion (ISM) technique. To account for uncertainty in the seismic response spectrum, either the peak broadening method or peak shifting method is used for the design and analysis of the piping systems.

In addition, the design of conduit and conduit support is discussed in DCD Tier 2, Appendix 3F, and seismic qualification of cable trays and supports is discussed in DCD Tier 2, Appendix 3G.

In DCD Tier 2, Sections 3F.4.2, “Response Spectrum Modal Analysis,” and 3G.4.2, “Modal Response Spectrum Analysis,” respectively, for response spectrum modal analysis of conduit systems and cable tray systems, the applicant states that the conduit systems and cable tray systems can be analyzed by using the envelope-broadened response spectra methods, considering USM, or ISM method. The staff found the methods used to analyze conduits and cable tray systems acceptable because they meet the guidance in SRP Section 3.9.2 SRP Acceptance Criterion 2.G, RG 1.92 and NUREG-1061, Vol. 4.

3.9.2.4.2.3.6 Analytical Procedures for Piping

The seismic analysis methods for seismic Category I and II piping are in accordance with SRP Section 3.7.3, “Seismic Subsystem Analysis,” Revision 3, issued March 2007, and discussed in DCD Tier 2, Section 3.12.3, ”Piping Analysis Methods.” These methods include the response spectrum method, time history method, or where applicable, the equivalent static load method. For modeling supports in the piping analysis, either the decoupled support model is used, whereby the support is modeled as a restraint with specified support stiffness, or an integrated support model is used, whereby the actual structural model of the support is included in the analysis. In the former model, requirements to maintain minimum support stiffness and limit deflection under dynamic load to a preset limit have to be addressed separately (as described in DCD Tier 2, Section 3.12.6.7, “Pipe Support Stiffness”), whereas in the latter model, these parameters are considered, and no separate check is required. The detailed evaluation of the piping analysis is discussed in Section 3.12 of this report.

3.9.2.4.2.3.7 Multiple-Supported Equipment Components with Distinct Inputs

For equipment and components supported at several points by either a single structure at different elevations, or two separate structures, the methods used to account for the different input motions are described in DCD Tier 2, Subsection 3.7.3.1.7, “Multiple Support Response Spectra Input Methods.” The USM method and the ISM method are used to account for the phasing and interdependence characteristics of the various support points. These methods use multiple-input response spectra and are based on the guidelines provided by the Pressure Vessel Research Council) Technical Committee on Piping Systems (ISM Method of Modal Spectra Seismic Analysis). Although these methods have been most often applied to plant piping subsystems, they are also applicable to other subsystems with multiple support points.

The USM method is used for analyzing plant equipment and components supported at multiple locations within a single structure. A uniform response spectra is defined that envelopes all of the individual response spectra at the various support locations. The uniform response spectra are applied at all support locations to calculate the maximum inertial responses of the equipment. For this method, modal combinations, including missing mass computations, are performed in accordance with RG 1.92, Revision 2. When there is more than one supporting structure, the ISM method for seismic response spectra may be used. Each support group is considered to be in a random-phase relationship to the other support groups. The responses caused by each support group are combined by the SRSS method. The details for these methods are given in DCD Tier 2, Subsections 3.7.3.1.7.1, “Uniform Support Motion Method,” and 3.7.3.1.7.2, “Independent Support Motion Method,” respectively. For some equipment (e.g., RCS components), the coupled model with supported structures is used. In **RAI 214-1920, Question 03.09.02-38**, the staff requested the applicant to clarify if the USM method of analysis used for equipment and components is in accordance with the guidance of SRP Section 3.9.2, SRP Acceptance Criterion 2.G.

In its response to **RAI 214-1920, Question 03.09.02-38**, dated April 30, 2009, the applicant stated that SRP Section 3.9.2, SRP Acceptance Criterion 2.G addresses the response spectrum envelope method and time history approach to multiple-supported systems. It describes the application methods of response spectra and maximum relative support displacement. As stated in DCD Tier 2, Subsection 3.9.2.2.8, "Multiple-Supported Equipment Components with Distinct Inputs," equipment supported at two or more locations with distinct seismic input, uses the upper bound of the envelope of all individual response spectra for these locations. As discussed in DCD Tier 2, Subsection 3.12.3.2.6, the analysis of seismic anchor motions (i.e., maximum relative support displacement), is performed as a static analysis with all dynamic supports active and the results of this analysis are combined with the piping system seismic inertia analysis results by absolute summation. This approach for piping systems is the same for equipment and components.

The staff finds the applicant's response acceptable because the applicant clarified that the USM method of analysis used for equipment and components is in accordance with the guidance of SRP Section 3.9.2, SRP Acceptance Criterion 2.G. The staff also noted that DCD Tier 2, Subsections 3.9.2.2.8 and 3.7.3.1.7 have been revised in Revision 2, to provide clarification with respect to the application of the time history approach used for the RCS components. Accordingly, **RAI 214-1920, Question 03.09.02-38, is resolved**.

For piping systems that are supported by multiple support structures or at multiple levels within a structure, the method used is described in DCD Tier 2, Section 3.12.3. In DCD Tier 2, Subsection 3.12.3.2.3, "Uniform Support Motion," the applicant stated that piping systems supported by structures located at multiple elevations within one or more buildings may be analyzed by USM. This analysis method applies a single set of spectra at all support locations, which envelops all of the individual response spectra for these locations. The enveloped response spectrum is developed and applied in the two mutually perpendicular horizontal directions and the vertical direction. Floor response spectrum curves used for USM may be generated from the damping values identified in Table 3 or the frequency-dependent damping values of Figure 1, "Frequency-Dependent Damping," from RG 1.61, Revision 1. The applicant stated in DCD Tier 2, Section 3.12.5.1, "Seismic Input Envelope vs. Site-Specific Spectra," per COL Information Item 3.12(2), that if any piping is laid out in the yard, the COL applicant is to generate site-specific seismic response spectra, which can be used for the design of these piping systems or portions of piping system. However, the DCD does not provide sufficient details regarding the seismic analysis methods. Also, the exact definition of "portions of the piping" is not clear. In **RAI 204-1584, Question 03.09.02-18**, the staff requested the applicant to provide the seismic analysis methods (including support displacements) for a piping that is partly laid out in the yard and partly supported by a building, equipment, or components.

In its response to RAI 204-1584**, Question 03.09.02-18**, dated April 30, 2009, the applicant stated that any safety-related seismic Category I underground piping for the US-APWR standard plant, as well as for site-specific applications, will be enclosed in and supported by a pipe tunnel, trench, or similar structure. The yard piping will not be in direct contact with the soil, so the seismic analysis method (including support displacements) used for the yard piping is the same as that used for piping supported by buildings, equipment, or components.

The staff reviewed the RAI response and relevant sections of DCD Revision 2 and noted that DCD Tier 2, Subsection 3.7.3.1.7 of the revised DCD addressed the issues of combining seismic responses for piping with multiple supports and independent inputs. This information is correctly included in DCD Revision 2. In addition, the applicant revised DCD Tier 2, Subsection 3.9.2.2.8 in Revision 2 to describe the analysis of seismic anchor motions.

The staff noted that the combination of results of static analysis of support motions with the seismic inertia analysis using the absolute sum method is consistent with the acceptance criteria of multiple-supported equipment and components with distinct input in SRP Section 3.7.3. The revised paragraph adequately addresses the staff's concerns about the relative displacements of support points in the seismic analysis. In addition, the applicant revised the DCD to clarify that non-exceedances would need to be considered if the CSDRS were used to design site-specific pipe tunnels or similar buried SSCs and that the COL applicant is to assure that the design or location of any site-specific seismic Category I SSCs will not expose those SSCs to possible impact or damage. The staff noted that the changes were incorporated and that the applicant has provided the seismic analysis method for underground piping, including support displacements. The absolute sum of the response due to seismic inertia effects and support displacement is conservative and consistent with SRP Section 3.7.3. For piping or portion of the piping that is enclosed in site-specific pipe tunnel, site-specific seismic response spectra will be generated for design of these piping systems. Accordingly, **RAI 204-1584, Question 03.09.02-18, is resolved**.

3.9.2.4.2.3.8 Use of Constant Vertical Static Factors

The constant vertical static factors method is not used to determine the vertical response loads in seismic analysis for the US-APWR. The vertical component of the seismic motion is obtained using one of the analysis methods described in DCD Tier 2, Section 3.7.2.1, "Seismic Analysis Methods." The vertical component is combined with the horizontal components of the seismic motion as described Section 3.9.2.4.2.3.4 of this report above.

3.9.2.4.2.3.9 Torsional Effects of Eccentric Masses

DCD Tier 2, Subsection 3.9.2.2.10, "Torsional Effects of Eccentric Masses," states that most of the equipment is designed such that torsional effects of eccentric masses do not occur. For some components (e.g., RCS components), the torsional effects are considered in the modeling and the analysis methods. The methods used to account for the torsional effects of valves and other eccentric masses such as valve operators in the seismic subsystem analyses are to be as follows:

- When valves and other eccentric masses are considered rigid, the masses of the operator and valve body or other eccentric mass are located at the center of gravity. The eccentric components (that is, yoke and valve body) are modeled as rigid members.

- When valves and other eccentric masses are not considered rigid, the dynamic models are simulated by the lumped masses in discrete locations (that is, center of gravity of valve body and valve operator), coupled by elastic members with properties of the eccentric components.

DCD Tier 2, Section 3.12.4.2, "Dynamic Piping Model," states that torsional effects of eccentric masses affecting the piping design are included in the dynamic analysis. The staff concludes that the modeling considers the torsional effects of eccentric masses acceptable because the analysis methods considered the torsional effects.

3.9.2.4.2.3.10 Buried Seismic Category I Piping, Conduits, and Tunnels

Buried seismic Category I piping, conduits, or tunnels are not used in the US-APWR standard plant design. Physical space is reserved and planned to provide an ESWPT, designed as a site-specific seismic Category I structure by the COL Applicant referencing the US-APWR. The methodology to be used in developing the site-specific design to be completed by the COL applicant is described in DCD Tier 2, Section 3.7.3.7, "Buried Seismic Category I Piping, Conduits, and Tunnels."

3.9.2.4.2.3.11 Interaction of Other Piping with Seismic Category I Piping

In the US-APWR design, the primary method of protection for seismic Category I piping is isolation from piping that is not required to be designed to seismic Category I requirements. In cases where this is not possible or practical, adjacent non-seismic piping is classified as seismic Category II and analyzed and supported such that an SSE event would not cause an unacceptable interaction with the seismic Category I piping. The displacements due to the seismic loading on seismic Category II piping are reviewed for interaction with seismic Category I piping and components, and interacting supports between seismic Category I and seismic Category II piping are designed for SSE loadings. If necessary, the seismic Category II portion of the piping is analyzed to the same design criteria as the seismic Category I piping.

For piping systems that include an interface between seismic Category I and non-seismic Category I portions, the seismic Category I dynamic analysis includes the first anchor point in the non-seismic system. Anchor points are defined as the extremities of piping runs that connect to structures, components (e.g., vessels and pumps), or pipe anchors that act as rigid constraints relative to piping motion and thermal expansions. The staff concludes that this is acceptable per SRP Section 3.9.2 acceptance criteria.

3.9.2.4.2.3.12 Analysis Procedure for Damping

The damping values used for seismic analysis are consistent with RG 1.61, Revision 1; the analysis procedure for damping is described in DCD Tier 2, Section 3.7.3.3, "Analysis Procedure for Damping." The damping values used for seismic analysis of piping systems are addressed in DCD Tier 2, Section 3.12.5.4, "Damping Values." In **RAI 204-1584, Question 03.09.02-39**, the staff requested the applicant to provide the basis of assigning a 5 percent damping value for CRDMs, as shown in DCD Tier 2, Table 3.7.3-1(a), "SSE Damping Values."

In its response to **RAI 214-1920, Question 03.09.02-39**, dated April 30, 2009, the applicant stated that the SSE analysis for the CRDM used a damping value of 4 percent, not five percent.

The staff finds the applicant's response acceptable because the applicant stated that SSE analysis for the CRDM used a damping value of 4 percent, not five percent. However, the applicant did not mention in its response that in the DCD Tier 2, Table 3.7.3-1(a) the damping value for the CRDM will be changed. The applicant was requested in follow-up **RAI 498-3782, Question 03.09.02-83**, to revise the CRDM damping value in DCD Tier 2, Table 3.7.3-1(a) and submit the revised DCD for staff review.

In its response to **RAI 498-3782, Question 03.09.02-83**, dated January 15, 2010, the applicant stated that in Table 3.7.3-1(a) of the DCD, 4 percent damping ratio for CRDM is not directly specified but assumed for one of the "welded and friction bolted steel structures and equipment" in that table. Therefore, there is no need to revise the CRDM damping in DCD Tier 2, Table 3.7.3-1.

The staff noted that in DCD Tier 2, Revision 2, Tables 3.7.3-1(a) and 3.7.3-1(b), “OBE Damping Values,” have been revised, and the SSE and OBE damping values listed in the tables are consistent with the acceptable values specified in RG 1.61, Revision 1. The staff further noted that although the damping values specified for CRDM have been deleted in Tables 3.7.3-1(a) and 3.7.3-1(b) of DCD Revision 2, the applicant stated that the damping values specified for welded and friction-bolted steel structures and equipment are also used for CRDM. The staff finds this acceptable because it is consistent with the acceptable damping values in RG 1.61, Revision 1, for seismic analysis of nuclear power plant SSCs. Accordingly, **RAI 498-3782, Question 03.09.02-83, is resolved**.

3.9.2.4.2.4 Conclusions for Seismic Analysis and Qualification of Seismic Category I Mechanical Equipment

The staff concludes that the US-APWR plant design meets the relevant requirements of 1GDC 2, with respect to demonstrating the design adequacy of all seismic Category I mechanical equipment and piping and their supports to withstand the combined effects of postulated earthquakes and normal and accident conditions. The staff also concludes that by providing acceptable seismic systems analysis procedures and seismic qualification criteria, the applicant has met the guidelines of SRP Section 3.9.2, including the applicable regulatory positions of RGs 1.61 and 1.92.

3.9.2.4.3 Dynamic Response Analysis of Reactor Internals under Operational Flow Transients and Steady-State Conditions

3.9.2.4.3.1 Summary of Application for Dynamic Response Analysis of Reactor Internals under Operational Flow Transients and Steady-State Conditions

DCD Tier 1/ITAAC: The Tier 1 information associated with this section is found in DCD Tier 1, Section 2.4.1, which lists ITAAC for reactor internals FIV tests.

DCD Tier 2: The applicant has provided a DCD Tier 2 system description in Section 3.9.2.3, “Dynamic Response Analysis of Reactor Internals under Operational Flow Transients and Steady-State Conditions,” summarized here in part, as follows. The reactor internals of the US-APWR represent a unique, first-of-a-kind design because it differs from the current 4-loop PWR. In accordance with the recommendations of RG 1.20, “Comprehensive Vibration Assessment Program for Reactor Internals during Preoperational and Initial Startup Testing,” Revision 3, issued March 2007, the applicant classifies the first US-APWR as a “Prototype Reactor” and commits to performing a comprehensive vibration analysis for the first US-APWR to assure that the vibration responses of the reactor internals during normal operations are acceptable before the pre-operational, hot functional test.

Since the reactor internals of the US-APWR are evolved from the design of the well-proven 4-loop reactor design, which is currently in operation in the U.S. and Japan, the applicant first provides a comparative analysis between the current 4-loop reactor and the US-APWR. A more detailed description of the reactor internals is also given in DCD Tier 2, Section 3.9.5, “Reactor Pressure Vessel Internals.” In comparison to the 4-loop design, the US-APWR has a larger diameter to accommodate a larger number of fuel elements (257 instead of 193). The RV and core barrel diameters of the US-APWR are, therefore, about 20 percent larger than the 4-loop design, the flow rate is higher by 30 percent, and the size of secondary core support assembly is also larger. In addition, the design of the neutron reflector and the arrangements of the rod cluster control assembly (RCCA) tubes and upper support columns have been changed from the 4-loop design.

In the comparative analysis, the applicant compares the ratio between the flow excitation force and structural stiffness for various internal components of the US-APWR with those of the 4-loop design and concludes that sufficient margins of safety are maintained regarding FIV.

A more quantitative FIV analysis of the US-APWR reactor internals is discussed in DCD Tier 2, Subsection 3.9.2.3.3, “Quantitative FIV Analysis of the US-APWR.” The analysis methodology is based on the ASME Code, Section III, Appendix N-1300. The applicant also performed small-scale model tests to validate the structural modeling and the forcing functions. Additional details of the scale-model tests and the FIV analysis are given in the following two technical reports:

1. MUAP-07023-P: “APWR Reactor Internals 1/5 Scale Model Flow Test Report,” Revision 2, issued August 2011

2. MUAP-07027-P: “Comprehensive Vibration Assessment Program for US-APWR Reactor Internals,” Revision 3, issued November 2012

Each of these two reports was revised at least twice to incorporate the additional information requested by the staff. In order to maintain the chronological development of the review process, unless otherwise stated, all RAIs refer to the original revision of the reports. The revised revisions of the reports have been also reviewed to ensure that the staff’s concerns are satisfactorily addressed and the relevant RAIs are included.

The vibration analysis of the reactor internals deals with various flow excitation mechanisms such as flow turbulence, vortex shedding, and fluid-elastic instability. Acoustic excitations generated by the RCP are also considered in the analysis. From the results of the quantitative vibration analysis, the applicant concludes the following:

- Alternating stress levels of reactor internals due to FIV are acceptably low in comparison with the limit for HCF.

- The vibration responses of the reactor internals without the core are the same or slightly larger than those with the core.

Based on the latter conclusion, the applicant proposes to conduct the preoperational and start-up vibration testing before core loading only because the vibration levels after loading the core will be bounded by those measured without the core.

3.9.2.4.3.2 Regulatory Basis for Dynamic Response Analysis of Reactor Internals under Operational Flow Transients and Steady-State Conditions

The relevant requirements of the Commission’s regulations for this area of review, and the associated acceptance criteria are given in SRP Section 3.9.2, SRP Acceptance Criterion 3 and are summarized below.

1. GDC 1, as it relates to structures and components being designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with the importance of the safety function to be performed.

2. GDC 4, as it relates to systems and components important to safety being appropriately protected against the dynamic effects.

Acceptance criteria adequate to meet the above requirements include:

1. To meet the requirements of GDCs 1 and 4, the following guidelines, in addition to RG 1.20 "Comprehensive Vibration Assessment Program for Reactor Internals During Preoperational and Initial Startup Testing", apply to the analytical solutions to predict vibrations of reactor internals for prototype plants. Generally, this analysis is required only for prototype designs and power uprate of existing plants; however, it is not required for non-prototypes except that segments of an analysis (in particular, assessments of any potential adverse flow effects) may be necessary if there are deviations from the prototype internals design or operating conditions or if the non-prototype is based on a conditional prototype which has experienced problems from adverse flow effects. If the reactor internal structures are a non-prototype design, the applicant should refer to the results of tests and analyses for the prototype reactor and give a brief summary of the results. A more detailed summary of results of assessment of the potential of any adverse flow effects also should be given.

2. RG 1.20, "Comprehensive Vibration Assessment Program for Reactor Internals during Preoperational and Initial Startup Testing," Revision 3, issued March 2007.

3.9.2.4.3.3 Technical Evaluation of Dynamic Response Analysis of Reactor Internals under Operational Flow Transients and Steady-State Conditions

The review of the dynamic response analysis of reactor internals under operational flow transients and steady-state conditions was performed in accordance with SRP Section 3.9.2 and RG 1.20. The review objective was to ensure the structural and functional integrity of the reactor and SG internals under vibratory loadings and thereby assure conformance with GDC 1 and 4. Although DCD Tier 2, Section 3.9.2.3 includes a commitment to meet these expectations, not all the information and documents needed to complete the review are provided. Most notable is the absence of any dynamic analysis of the SG internals, the steam separators, and the safety relief valves.

The applicant states that the evaluation of the SG is described in DCD Tier 2, Section 5.4.2.1, "Steam Generator Materials." In **RAI 272-1585, Question 03.09.02-42** (identified as RAI 3.9.2-19 in applicant's response), the staff requested the applicant to provide appropriate analysis of dynamic responses of structural components within the SG caused by steady and operational transient flow conditions, including an assessment of potential adverse flow effects, such as flow-induced vibrations and acoustic resonances. The analysis should be supplemented with the expected bias error and uncertainties.

In its response to **RAI 272-1585, Question 03.09.02-42**, dated April 9, 2009, the applicant stated that the topic of SG upper internals vibration is addressed in DCD Tier 2, Subsection 5.4.2.1.2.10, "Flow Induced Vibration of Secondary Side Internals" (via a cross reference from

DCD Tier 2, Subsection 3.9.2.4.1, "Background"). The applicant will revise DCD Tier 2, Subsection 5.4.2.1.2.10 to more fully address the structural adequacy of the SG internals.

The staff agrees with the applicant's response regarding the upper internals of the US-APWR SG, namely, that since they have been in use in existing plants for more than 20 years in sizes and flow rates which bound those of the US-APWR without any vibration problems, there is no need to perform detailed vibration analysis of the upper internals of the SG. The applicant, however, does not address the other (lower) internals of the SG, such as the tube bundle and the U-tube vibration due to cross and axial flows. Therefore, **RAI 272-1585, Question 03.09.02-42** was closed as unresolved and in follow-up **RAI 498-3782, Question 03.09.02-63**, the staff requested the applicant to provide appropriate vibration analysis for the SG lower internals, including the tube bundles and the U-tubes which are exposed to cross and axial flows. If the design of the SG lower internals is not prototypical, it suffices to refer to in-service SGs with similar design and flow conditions.

In its response to the **RAI 498-3782, Question 03.09.02-63**, dated January 15, 2010, the applicant referred to DCD Tier 2, Subsection 5.4.2.1.2.6, "Flow Induced Vibration of Tube Bundle," stating that it includes the requested information about the SG tube bundle. The staff reviewed DCD Tier 2, Revision 2, Subsections 5.4.2.1.2.6 and 5.4.2.1.2.10, and concluded that the FIV analysis of the SG internals, including the tube bundle and the U-tubes, is adequately addressed in these sections. Accordingly, **RAI 498-3782, Question 03.09.02-63, is resolved**. Later in the review, however, operating experience in the nuclear industry necessitated further evaluation of the US-APWR steam generator FIV analysis. This evaluation is documented later in this section, with reference to **RAI 930-6494, Question 05.04.02-13**, and **RAI 1013-7031, Questions 03.09.02-103 and 03.09.02-104**.

Because of recent concerns in the nuclear industry regarding the excitation of acoustic resonances in the standpipes of safety relief valves, the new revisions of SRP Section 3.9.2 and RG 1.20 recommend that potential adverse flow effects be evaluated for the steam delivery system, including safety relief valves. Since the DCD does not address this issue at all, in **RAI 272-1585, Question 03.09.02-43** (identified as RAI 3.9.2-20 in applicant's response), the staff requested the applicant to analyze the potential of flow-excited acoustic resonance occurring in the standpipes of the safety relief valves (or in any other blind side-branches), which are mounted on the MS lines exiting the SGs. If any acoustic resonance is anticipated, the applicant is expected to explain the countermeasure(s) that will be implemented to avoid or mitigate the resonance.

In its response to the **RAI 272-1585, Question 03.09.02-43**, dated April 9, 2009, the applicant stated that the design of the US-APWR steam delivery system (including the safety relief valves and the steam separator) and the flow conditions they experience are similar to the existing and currently operating steam delivery systems in the U.S. and around the world. The applicant designed the US-APWR steam delivery system using the structural design rules based on years of empirical experience with similar equipment. The configuration employed in the US-APWR steam delivery system has been operating in the U.S. for more than 20 years with sizes and flow rates that bound those of the US-APWR steam delivery system. Based on an extensive record of vibration-free operation, the applicant concludes that the structural and vibration design bases are proven. This nonsafety-related steam delivery system will not experience excessive vibration; therefore, the analysis of the flow excited acoustic resonance occurring in the standpipes of the safety relief valves (or in any other blind standpipes) is not expected.

The staff agrees with its response that since the steam separator and the safety relief valves of the US-APWR have been in use in existing plants for more than 20 years in sizes and flow rates that bound those of the US-APWR without any vibration problems, there is no need to perform detailed vibration analysis of these parts. However, since this information was not included in the DCD, the staff closed as unresolved **RAI 272-1585, Question 03.09.02-43,** and in follow-up **RAI 498-3782, Question 03.09.02-64**, the staff requested the applicant to revise the DCD and refer to the information provided in the response to **RAI 272-1585, Question 03.09.02-43**. The staff also requested the applicant to provide specific references from existing operating reactors.

In its response **RAI 498-3782, Question 03.09.02-44**, dated February 3, 2010, the applicant provided the relevant data of a plant operating in the U.S. since 1987 with a power uprate in 1993. The provided information indicates that the steam flow velocity in the MS lines of the US-APWR (145 to 153 ft/s) (44.2 to 46.6 m/s) is lower than that in similar plants operating in the U.S. for many years (161 ft/s) (49.1 m/s). Also, the standpipe diameter of the safety relief valves in the US-APWR is similar to that of operating plants.

Based on the provided data, the staff agrees that acoustic resonances in the standpipes of the safety relief valves are not expected to occur in the steam delivery system of the US-APWR. The staff finds the applicant's response to **RAI 498-3782, Question 03.09.02-64** acceptable because the provided data of operating plants bound the US-APWR operating conditions, and reference to this information is provided in the DCD. Accordingly, **RAI 498-3782, Question 03.09.02-64, is resolved**.

The methodology used to analyze the dynamic responses of the reactor internals consists of performing 1/5-scale model tests (MUAP-07023-P, Revision 0) to characterize the dynamic fluid forces and validate the computational tools used to simulate the dynamic responses. In addition, the SYSNOISE code is used to simulate the acoustic environment inside the RV and the reactor coolant piping and to formulate the acoustic forcing functions acting on various structural components within the RV. The details of the vibration analysis of the reactor internals are given in MUAP-07027-P, Revision 0. In the following, the scale model tests, the simulation tools, and the methodology are evaluated.

MUAP-07023-P, Revision 0, states that the scale model is similar to the US-APWR, except that the core model is shorter; its length corresponds to the J-APWR (in Japan), which is 12 ft (3.7 m), and not to the US-APWR, which is 14 ft (4.3 m). However, comparison of Figure 3-2 in MUAP-07023-P, Revision 0, with Figure 2.1-1 in MUAP-07027-P, Revision 0, suggests that the small-scale model reflects the geometry of the 4-loop reactor rather than that of the US-APWR, with the exception of including the neutron reflector. Therefore, in **RAI 272-1585, Question 03.09.02-44** (identified as RAI 3.9.2-21 in applicant's response), the staff requested the applicant to provide the details of any deviations in the 1/5-scale model geometry from the US-APWR geometry and to explain the effects of these differences on the validity of the model test results and their applicability to the US-APWR. In addition, it is not clear how the effects of geometry deviations, such as use of a shorter core in the model, are accounted for in assessing potential adverse flow effects, in defining the fluid forcing functions, and in scaling the forcing functions from the model to the full-size reactor. Since the scale model tests are used by the applicant to validate the forcing function definitions and the structural models for dynamic analysis, the details of the model geometry and test conditions, in comparison to the US-APWR, need to be submitted for review. The applicant was also requested to revise the 1/5-scale model test report to include the requested information.

In its response to **RAI 272-1585, Question 03.09.02-44**, dated April 9, 2009, the applicant referred to the response to **RAI 206-1576, Question 03.09.02-22** (identified as RAI 3.9.2-43 in applicant's response), which gives the differences between the 4-loop reactor, the J-APWR, and the US-APWR. Regarding the analysis procedure, including the modeling and forcing functions, the applicant stated that the FIV analysis program consists of two separate tasks. One was to estimate the responses of the model used in the J-APWR scale model test (SMT) and the other was the prototypical analysis of the US-APWR. The modeling and forcing functions were customized for each task. In summary, the applicant used the 1/5-scale model structural and flow properties in the first task to validate the analysis by means of the model test results. In the second task, the analysis of the US-APWR was performed using the full-scale structural properties. In addition, the flow-induced forcing functions were scaled from the J-APWR SMT to the full-scale and operating conditions of the US-APWR. To explain this scaling procedure, the applicant stated that for the down-comer turbulence and the cross flow loads in the upper plenum, the force time history data generated for the 1/5 SMT in the J-APWR analysis were directly converted to those of US-APWR prototype in the following manner. Conversion ratios C1, C2 and C3 were factors for the force area, the dynamic pressure and time (or frequency). The constants of proportionality are assumed to be the same because the structural configurations and flow conditions are the same in these regions in both plants. The applicant also gives the values of the conversion ratios (C1, C2, and C3) for the turbulence excitation in the downcomer and for the cross flow loads in the upper and lower plenums. Additionally, the empirical formula used to model the turbulence excitation in the downcomer is compared with field measurements and with the results of the 1/5-scale model of the J-APWR. The applicant contends that these comparisons validate the forcing function model. The structural modeling procedure was validated by comparing the computed natural frequencies of the J-APWR SMT with the measured frequencies. The applicant also revised DCD Tier 2, Revision 2, Subsection 3.9.2.3.3, to add a pointer on where to find further information in the technical report.

While the applicant's response is informative regarding the general approach of the FIV analysis program, it falls short on many important details of the procedure. For example, the response does not address why the scale model appears to replicate the 4-loop reactor instead of the US-APWR. Additionally, validation of the structural models by comparing the resonance frequency only does not seem to be an adequate validation approach. Finally, the forcing function upper bound, which is shown in Figure 2, "J-APWR 1/5 SMT D/C Normalized PSD with Au Yang's Empirical Equation," of the response, seems to underestimate the turbulence excitation in the upper portion of the downcomer. Therefore, the staff closed as unresolved **RAI 272-1585, Question 03.09.02-44** and in three follow-up questions, **RAI 498-3782, Questions 03.09.02-65, 03.09.02-66, and 03.09.02-67**, the staff identified its concerns.

In its response to **RAI 272-1585, Question 03.09.02-44** (identified as RAI 3.9.2-21 in applicant's response), the applicant also addressed the global differences between the US-APWR and the 1/5 SMT, and did not clarify other differences that may appear small but can have important effect on the test results. For example, MUAP-07023-P, Revision 0, Figure 3-2, "Overview of the Test Model," indicates that the scale model has a lower core plate and a lower support plate, whereas MUAP-07027-P, Revision 0, Figure 2.1-1, "Typical Flow Velocities: Comparison between the US-APWR and the Current 4-loops Plant," shows that the US-APWR has one plate only (lower core support plate). Therefore in follow-up **RAI 498-3782, Question 03.09.02-65,** the staff requested the applicant to explain the reasons for these differences (and others which may not be apparent in the above-noted figures), and to clarify the effect of these differences on the test results, structural modeling, and forcing functions.

In its response to the **RAI 498-3782, Question 03.09.02-65**, dated February 3, 2010, the applicant provided tables and figures explaining the differences between the J-APWR, which is in operation in Japan, and the US-APWR. The effects of these differences on the FIV of the reactor internals were also clarified. A summary is included below.

Because the core of the U.S. reactor is longer than the Japanese reactor (14 ft (4.3m) instead of 12 ft (3.7m)), the fuel assemblies and neutron reflector are also longer. The lower core plate is integrated with the lower core support plate in the U.S. reactor, which necessitated redesigning the flow holes of the lower core support plate. Also, the in-core instrumentation system (ICIS) of the U.S. reactor is inserted from the top of the vessel instead of from the bottom of the vessel as in the Japanese reactor. The ICIS is guided and protected from cross flow loads by guide tubes and upper support columns in the upper plenum. The applicant also stated that the information included in the response to this RAI will be added in a revised revision of MUAP-07027-P, Revision 0.

The staff finds the applicant's response to **RAI 498-3782, Question 03.09.02-65,** acceptable because it clarifies the differences between the US-APWR and the J-APWR (which is similar to the scale model). It also explains the effects of these differences on the natural frequencies and flow excitations of various components of the reactor internals. It is also clear that these differences are accounted for in the finite element model of the US-APWR. The only remaining concern related to this RAI is the effect of increasing the weight of the fuel assembly and the neutron reflector on the stresses at the junction between the core barrel and the lower core support plate and also on the vibration of the fuel assemblies. In the new follow-up **RAI 614-4853, Question 03.09.02-88,** the staff requested the applicant to confirm that the effect of this increase in weight is considered in the analysis of the fuel assembly and the core barrel.

In its response to **RAI 614-4853, Question 03.09.02-88**, dated September 29, 2010, the applicant stated that the weight of the neutron reflector and the fuel assembly are directly simulated in the mass vibration analysis model, and the results of the vibration analysis are used as input to the stress analysis. The applicant also stated that weights are used for the dead weight loads. Therefore, the stress analysis results take into account the increase in weight of the neutron reflector and the fuel assembly both in vibration loads and dead weights. The applicant added that the results (i.e., the primary stress on the core barrel) have been reported in Table 10-1 of the applicant's Technical Report MUAP-09004-P, Revision 0. The stress at the core barrel/lower core support plate junction is less than half the allowable limit in design conditions and all operating conditions.

The applicant further stated that the weight increase is also considered in the vibration analysis model of the fuel assembly; further information regarding the analysis model and its results are presented in the applicant's technical report MUAP-08007-P, Revision 1. In addition, significant FIV was not observed in the hydraulic test with a full-scaled mockup fuel assembly. The applicant added that there is no structural integrity concern regarding the weight increase of the neutron reflector and the fuel assembly. The staff finds the applicant's response acceptable because the weight of fuel assembly and neutron reflector is directly simulated in the vibration analysis model, and is also considered in the vibration analysis model of the fuel assemblies. Therefore, **RAI 498-3782, Question 03.09.02-65, and RAI 614-4853, Question 03.09.02-88, are resolved**.

In its response to **RAI 272-1585, Question 03.09.02-44** (identified as RAI 3.9.2-21 in the applicant's response), the applicant also stated that the validation of the method of structure modeling was conducted by comparing the computed natural frequencies of the J-APWR SMT with the measured data, as discussed in MUAP-07027-P, Revision 0, Section 3.2.1. The staff finds this validation procedure inadequate because it does not take into account the frequency response functions (FRFs), which express the relationship between the structural response and the forcing functions. In addition, the structural modeling of the US-APWR should be validated from measurements on other full-size installations. SRP Section 3.9.2 recommends that uncertainties and bias errors in FE simulations be estimated from comparisons with measurements made on structures similar in construction to the reactor internals being modeled. The staff appreciates that the validated model will not be that of the US-APWR. However, the procedure of modeling boundary conditions, structural tolerances, damping, welds, etc., and the resulting bias and uncertainty errors can be validated. In follow-up **RAI 498-3782, Question 03.09.02-66**, the staff requested the applicant to provide additional information to assure the staff that:

a) the structural modeling approach has been adequately validated, and

b) the bias error and uncertainties have been adequately assessed and incorporated in the dynamic analysis of the reactor internals. In addressing the bias error and uncertainties, the applicant was requested to address how the systematic bias and the random uncertainties are separately estimated.

In its response to **RAI 498-3782, Question 03.09.02-66**, dated February 3, 2010, the applicant stated that because measurement results from existing plants are not available, the J-APWR 1/5-scale model test data are used for validation of the analysis procedure. The applicant added that the US-APWR reactor internals are “more similar” to those of the J-APWR than those of the existing plants and pointed out that the adequacy of the analysis will be verified with the measured data in the preoperational test of the US-APWR.

The applicant also discussed the acceptance criteria for the analysis uncertainty, stating that based on past experience, the applicant assumed a factor of three for the uncertainty in FIV responses and a factor of two for the uncertainty in the flow-induced loads. Therefore the uncertainty of the analysis model itself is expected to be no larger than a factor of 1.5 in the response. To achieve this, the uncertainty in the fundamental modal frequency was limited to be less than 10 percent, which corresponds to 20 percent in the response considering the frequency transfer function.

Regarding the second item of the RAI dealing with bias error and uncertainties, the applicant provided tables of sources of uncertainties and bias errors, including dimension tolerances, material properties at elevated temperature, effect of thermal expansion, support conditions at the mating surfaces, and close fittings. Because the impact of these uncertainties depends on the magnitude of the displacement or the applied load, the applicant developed two models: one is used for FIV with small responses, and the other is used for seismic/LOCA analysis with larger responses.

The staff finds the applicant’s response reasonable, as a factor of three for uncertainties in the FIV responses is conservatively high considering the validation procedure applied to load definition and modal analysis. Accordingly, **RAI 498-3782, Question 03.09.02-66, is resolved**.

In its response to **RAI 272-1585, Question 03.09.02-44** (identified as RAI 3.9.2-21 in applicant’s response), the applicant provided a comparison between the empirical normalized forcing function (e.g., power spectral density) in the downcomer and that of the 1/5 SMT. In this comparison, the turbulence power spectral density (PSD) in the upper portion of the downcomer is about an order of magnitude higher than the upper bound of the empirical PSD. In follow-up **RAI 498-3782, Question 03.09.02-67**, the staff requested the applicant to explain how this large difference is accounted for in estimating the forcing function of the US-APWR. In particular, the applicant was requested to elaborate on the axial and circumferential distributions of the forcing function.

In its response to the **RAI 498-3782, Question 03.09.02-67**, dated January 15, 2010, the applicant provided additional explanations and figures to clarify the axial and circumferential distributions of the forcing function and explain how these distributions were implemented in the dynamic analysis of the core barrel. Regarding the upper bound of the PSD, the applicant stated that both the empirical functions and the field data are based on a measured point in the middle of the downcomer. "Upper bound" does not mean the upper bound of the entire downcomer but the envelope of spectral peaks at that single measuring point. The inlet nozzles are located in the upper portion of the downcomer, where the magnitude of pressure PSD is three-four times larger than that at the mid-section of the downcomer. This difference is reasonable because the local flow velocity in the inlet nozzle is about twice the average velocity in the downcomer, resulting in four times larger in the dynamic pressure level.

Regarding the axial and circumferential distributions of the forcing function for the US-APWR, the applicant provided figures showing examples of the power spectral density of the pressure fluctuations measured at four locations (A to D) in the downcomer of the US-APWR 1/7 Scale Model. To explain the distribution of the pressure fluctuations, the applicant added that location "A" is nearest to the inlet nozzle so the PSD is much higher than the others for reasons discussed above. The mapping of the pressure forcing functions was also shown in a figure attached to the RAI response. The entire surface of the core barrel facing the downcomer was divided into 16 segments with two elevations and eight segments around the circumference, which correspond to the nozzle layouts. For example, the forcing functions for the upper segments facing the inlet nozzles were generated from the same PSD "A" by the inverse Fourier Transform Method. But the time histories for the four segments with this PSD were statistically independent of one another and with random phase. In the same manner, a total of 16 time histories were defined from the four PSDs.

The staff finds the applicant's response acceptable because it provides a reasonable distribution of the forcing function as measured from the scale model tests. The response also clarified how this distribution is implemented in the dynamic analysis of the core barrel. Accordingly, **RAI 498-3782, Question 03.09.02-67, is resolved**.

Apart from the requirement of geometric similarity, when FIV mechanisms are investigated by means of model tests, the model and prototype must also be dynamically similar. Since the applicant appears to be using the 1/5-scale model to confirm the reactor structural integrity against flow-induced vibration, it is necessary that the model design and test conditions be dynamically similar to those of the prototype. Neither the DCD nor the scale model test report (MUAP-07023-P, Revision 0) discusses the dynamic similarity of the scale model tests despite its importance to the reproduction of flow-induced vibration mechanisms. Therefore, in **RAI 272-1585, Question 03.09.02-45,** (identified as RAI 3.9.2-22 in the applicant's response), the staff requested the applicant to compare the relevant dimensionless parameters for the 1/5-scale model tests and the full-size reactor at normal operation conditions to demonstrate dynamic similarity between the small-scale model and the prototype. In its request, the staff noted that examples of dimensionless parameters include, but are not limited to, the fluid-elastic parameter, Strouhal number, reduced velocity, and the ratio of excitation to resonance frequencies. Deviations of the model parameters from dynamic similarity should be shown to be conservative; otherwise, flow excitation mechanisms that occur under the prototype test conditions may not be reproduced in the model tests.

In its response to the **RAI 272-1585, Question 03.09.02-45**, dated April 9, 2009, the applicant stated that detailed comparisons of dimensionless parameters between the J-APWR SMT and the US-APWR plant are listed in the response to **RAI 206-1576, Question 03.09.02-22,** (identified as RAI 3.9.2-43 in applicant's response) under Item (b).

The staff's review of the dimensionless parameters provided by the applicant is included in the evaluation of **RAI 206-1576, Question 03.09.02-22**. The staff also received a revised MUAP-07027-P, Revision 1, and the review of this report is included in the evaluation of **RAI 272-1585, Question 03.09.02-56** (identified as RAI 3.9.2-33 in applicant's response). Accordingly, **RAI 272-1585, Question 03.09.02-45 is resolved**, and the staff's concerns are addressed further in **RAI 272-1686, Question 03.09.02-56** and **RAI 206-1576, Question 03.09.02-22**.

In MUAP-07027, Revision 0, the resonance frequencies of the small-scale model were used to validate the FE dynamic simulation of the small-scale model. However, for the US-APWR, it was not clear whether the FE simulation has already been validated (for example, by comparison with the results of the 4-loop reactors). SRP Section 3.9.2 recommends that uncertainties and bias errors in FE simulations be estimated from comparisons with measurements made on structures similar in construction to the reactor internals being modeled. In **RAI 272-1585, Question 03.09.02-46** (identified as RAI 3.9.2-23 in applicant's response), the staff requested the applicant to explain the methodology used to validate the structural models of the prototype reactor internals and to provide typical results of the validation tests together with the uncertainties and bias errors which are expected in the results of FE structural modelling. The staff also requested the applicant to briefly describe the measurements performed to determine the structural resonance frequencies, the mode shapes, and the FRFs.

In its response to the **RAI 272-1585, Question 03.09.02-46**, dated April 9, 2009, the applicant stated that the validation of the structure modeling was conducted by the comparison of the computed natural frequencies of the J-APWR SMT with the measured data, as discussed in MUAP-07027-P, Revision 0, Section 3.2.1. A typical value of the uncertainties and bias errors of the structural resonance frequencies is 7.5 percent, which is obtained from averaging the errors in a total of 12 pairs of natural frequencies as shown in MUAP-07027P, Table 3.2-1. The effect of the 7.5 percent error in the structural resonance frequencies on the vibration response is estimated by considering the mode shapes and the FRF. It leads to 15 percent error in the random vibration response, with a conservative assumption that the 7.5 percent error in the natural frequency is totally caused by the uncertainty in the stiffness of structures. However the error in the mass of the model has little effect on the vibration response.

The applicant also stated that comparison with measured data from a currently operating plant is not a better method of validating the analysis methods used in the US-APWR due to the following reasons.

(1) The vibration characteristics of the US-APWR reactor internals are close to those of the J-APWR rather than to the current plant.

(2) This method cannot be applied to first-of-a-kind design with significantly different dimensions or configurations, such as the neutron reflector or the core barrel.

Upon reviewing the applicant's response, the staff still had some concerns regarding both the validation of the structural models and the assessment of bias errors and uncertainties. Therefore, the staff closed as unresolved **RAI 272-1585, Question 03.09.02-46** and in follow-up **RAI 498-3782, Question 03.09.02-66**, the staff raised these and other concerns. As discussed above, the applicant's response to follow-up **RAI 498-3782, Question 03.09.02-66** is acceptable.

To evaluate the effect of the acoustic pulsations, which are generated by the RCP, on the dynamic response of reactor internals, the applicant uses the acoustic code SYSNOISE to model the acoustic environment within the RVand the reactor coolant piping. In MUAP-07027-P, Revision 0, the resonance frequencies of two simple flow configurations (an annulus or a cylinder) are used to validate the acoustic model formulated by SYSNOISE. This "validation" method is inadequate because the geometry of the reactor and cooling system is much more complex than that of an annulus or a cylinder. According to SRP Section 3.9.2 and RG 1.20, the applicant is expected to validate the analytical tools by measurements made on structures similar in construction to the reactor internals being modelled. In **RAI 272-1585, Question 03.09.02-47** (identified as RAI 3.9.2-24 in applicant's response), the staff requested the applicant to describe the method used to validate the acoustic model developed by SYSNOISE and to discuss the bias errors and uncertainties associated with the model predictions. On another issue related to the validation of the SYSNOISE simulation, in **RAI 272-1585, Question 03.09.02-48** (identified as RAI 3.9.2-25 in applicant's response) the staff requested the applicant to discuss and substantiate the sound attenuation coefficient values, which are used in the acoustic simulation of the small-scale model and the US-APWR.

In its response to **RAI 272-1585, Question 03.09.02-47**, dated April 9, 2009, the applicant stated that using test data from the 1/5 scale model of the APWR to validate the acoustic analysis procedure is not adequate because they do not simulate the acoustic pressure pulsations induced by the RCPs. The applicant instead selected comparison with theoretical results as the validation procedure due to lack of measured data from the existing 4-loop reactors. The down comer and the upper plenum were selected for the validation of the SYSNOISE acoustic analysis. These regions have high possibilities of acoustic resonance induced by the RCP. The down comer was analyzed as an annulus while the upper plenum was analyzed as a cylinder to compare with the theoretical results. The results are reported in MUAP-07027-P, Revision 0, Table 3.3-5, "SYSNOISE Code Verification Analysis." In the design analysis model, the lower plenum and the reactor core connecting the above region were added. The head plenum was excluded from the model because it is an acoustically-isolated closed space. The applicant also reported that a sensitivity analysis indicated that the uncertainty and bias errors of the calculated resonance frequencies are within 10 percent, even with uncertainties in the sound speed.

The staff found the applicant's response to **RAI 272-1585, Question 03.09.02-47,** inadequate because the geometries used to verify the acoustic modeling are simple and do not reflect the complexity of the reactor internals and its piping, especially the boundary conditions. Therefore, the staff closed, as unresolved, **RAI 272-1585, Question 03.09.02-47,** and in follow-up **RAI 498-3782, Question 03.09.02-68**, the staff requested the applicant to respond to the original RAI and comply with SRP Section 3.9.2 and RG 1.20, which expect the analytical tools, such as the SYSNOISE model, to be validated by means of measurements made on structures similar in construction to the reactor internals being modeled. In addition, the applicant was requested to discuss the bias and uncertainty errors that may be included in the predictions of the SYSNOISE model.

In its response to **RAI 498-3782, Question 03.09.02-68**, dated February 3, 2010, the applicant stated that the verification of SYSNOISE was performed on simple models with dimensions selected to represent the downcomer or the upper plenum of the actual reactor. The applicant explained that the upper plenum of an actual reactor is more complicated because of the many components inside, such as the RCCA guide tubes. The applicant assumed that these internal components may act as an acoustic resistance (damper of pressure) but do not have significant effects on the basic acoustic modes, because the diameters and their pitches (0.1-0.2 m (0.3-0.7 ft)) are much smaller than the wave lengths of RCP pulsation (7.1 m (23 ft) for 140 Hz and 3.6 m (12 ft) for 280 Hz), although it is difficult to verify this estimation. The 1/5 SMT test data also provided no information on this issue because the RCP characteristics were not simulated in this test. Therefore the applicant used the simple models without internal structures for the benchmark problem, where theoretical values of acoustic resonance modes can be calculated. To ensure that the analysis results of the SYSNOISE code are sufficiently conservative, the applicant neglected the acoustic damping effects due to the structural flexibility. A sensitivity study performed by the applicant showed that the pulsation amplitude is almost one order of magnitude smaller if the structural flexibility is taken into account.

In the absence of in-plant measurements of the RCP pulsation, and based of the sensitivity analysis performed by the applicant, the staff finds the margin of 10 for the RCP pulsation loads to be sufficiently conservative. In addition, the neglect of sound absorption in water while using the SYSNOISE code will also add conservative bias in the analysis results. Therefore, **RAI 498-3782, Question 03.09.02-68, is resolved**.

Regarding the values of the sound attenuation coefficient, in its response to **RAI 272-1585, Question 03.09.02-48**, dated April 9, 2009, the applicant stated:

(a) Test data from the 1/5-scale model of the J-APWR is inadequate to validate the sound attenuation in the SYSNOISE model because they do not simulate the RCP pulsation.

(b) The sound attenuation included in the SYSNOISE model is discussed in the analysis of acoustic loading in the reactor internals of the US-APWR.

The applicant also included two tables explaining various mechanisms of sound attenuation in the reactor internals and the conservative values assumed in the analysis of the US-APWR. To retain conservatism in the SYSNOISE model, the applicant ignored all acoustic attenuation mechanisms except the perforated plate damping mechanism, which is associated with the flow in the spray nozzles and through the lower and upper core plates. The acoustic attenuation values used in the SYSNOISE model are based on a paper by A.W. Guess ("Calculation of Perforated Plate Liner Parameters from Specified Acoustic Resistance and Reactance," *Journal of Sound and Vibration* 40(1), pp 119-137, 1975).

The acoustic attenuation values provided by the applicant seem to be reasonable and sufficiently conservative. However, regarding the statement in item (a) above, it is not clear why the acoustic characteristics of the system cannot be determined with any means of acoustic pulsations. Because the staff raised this latter issue in RAI 03.09.02-47, and the applicant adequately addressed it, **RAI 272-1585, Question 03.09.02-48, is resolved**.

As mentioned earlier, the applicant has committed to performing a comprehensive vibration analysis program for the first US-APWR in COL Information Item 3.9(2). The DCD and the vibration analysis report address various excitation mechanisms, including vortex shedding, flow turbulence, and fluid-elastic instability excitations. The strength of these excitation mechanisms is obviously dependent on the value of the cross-flow velocity component. The DCD and the vibration analysis report discuss the velocity distributions only qualitatively, and therefore, the methodology used to determine the cross-flow velocity component is not clear. Therefore, in **RAI 272-1585, Question 03.09.02-50** (identified as RAI 3.9.2-27 in applicant's response), the staff requested the applicant to explain the methodology used to determine the cross-flow velocity over various components of the reactor internals, such as the cross-flow velocity near the exit nozzles and over the core supporting structures (e.g., upper and lower support columns, guide tubes, and instrumentation support structures).

In its response to **RAI 272-1585, Question 03.09.02-50**, dated April 9, 2009, the applicant stated that the cross flow velocity around the structure in the lower and upper plenum of the RV was evaluated in the following manner:

(1) Upper plenum

a. The cross flow velocities in the upper plenum were calculated based on the potential flow theory without structures in the plenum.

b. The cross flow velocity distribution between the structures were determined based on the equation of continuity and the pitch-to-diameter ratio of the structures

c. When the cross flow is not uniform along the axis of the structure in the upper plenum, the maximum cross flow was used for vortex shedding and fluid elastic instability evaluation.

(2) Lower plenum

a. The cross flow velocity in the lower plenum was assumed to be equal to the down comer average velocity.

b. The cross flow velocity distribution between the diffuser plate support columns was determined based on the equation of continuity and the pitch-to-diameter ratio of the support columns.

c. When the cross flow is not uniform along the axis, the maximum cross flow was used in vortex shedding and fluid elastic instability evaluations.

The staff finds this response acceptable because it clarifies the procedure for calculating the cross flow velocity in various parts of the reactor. Since this procedure is conservative, and the requested information has been incorporated in DCD Tier 2, Revision 2, **RAI 272-1585, Question 03.09.02-50, is resolved**.

However, in the revised MUAP-07027-P, Revision 2, the applicant used cross-flow velocity which varied along the length of the structures in the upper plenum, whereas this velocity was assumed to be constant in the original version of the report. Therefore, in **RAI 916-6343, Questions 03.09.02-101 and 03.09.02-102**, the staff requested the applicant to clarify how the local velocity is determined along the structures in the upper plenum and to explain the phase condition which is imposed on the forcing function along the length of the analyzed structures.

In its response to the **RAI 916-6343, Questions 03.09.02-101 and 03.09.02-102**, dated April 26, 2012, the applicant explained that the flow velocity is calculated based on the potential flow theory without taking into account the presence of the internal structures (guide tube, upper support column, or top slotted column). The effect of these structures on the flow velocity is taken into account by multiplying the flow velocity by the blockage factor for conversion into the local flow velocity (which is equivalent to the gap velocity in the tube bundle FIV analysis). Regarding the phasing of the forcing function, the applicant responded that the computed load is separately input into each axial segment. Therefore, the forces are un-correlated between different axial segments in the upper plenum. Within each segment, the coherence of the forcing function is governed by the input correlation length. To reduce the error due to ignoring the coherence of the forcing function across different segments, the axial segment is set to be longer than the correction length.

The staff finds these assumptions for the forcing functions along the structures in the upper plenum reasonable because the correlation length of the flow excitation is accounted for and also because the excitation frequency changes along the length of structures in the upper plenum, which weakens and un-correlates the cross-flow excitation. These assumptions were also validated by comparing the response of the upper plenum structures with measurements of the scale model response. Therefore, the staff's concern regarding the forcing functions of the structures in the upper plenum is resolved. Accordingly, **RAI 916-6343, Questions 03.09.02-101 and 03.09.02-102, are resolved**.

The reactor coolant pumps of the US-APWR are larger than those used in current 4-loop reactors. They provide 27-percent higher flow rate at 10-percent higher head. Neither the DCD nor the vibration analysis report explains how this increase in the pump capacity is accounted for in the acoustic forcing function, or in the acoustic source representing the pump excitation. Section 3.3.4 of MUAP-07027-P, Revision 0, states that: "The RCP pulsation amplitudes at each rotation speed are assumed as shown in Table 3.3-3." In **RAI 272-1585, Question 03.09.02-51** (identified as RAI 3.9.2-28 in applicant's response), the staff requested the applicant to substantiate the values of pressure pulsations assumed to be generated by the RCP. In addition, the staff requested the applicant to explain the effect of using RCPs with higher flow rates and delivery heads on the acoustic excitation generated by the pumps.

In its response to **RAI 272-1585, Question 03.09.02-51**, dated April 9, 2009, the applicant stated that the effect of increasing the capacity of the RCP from the current 4-loop capacity was encountered on the evaluation of the US-APWR RCP pulsation load as follows:

(1) The RCP forcing function was determined based on a model test of the RCP for the J-APWR, the flow rate and the head of which are higher than the current 4-loop plant. The design flow rate for the J-APWR is the same with the US-APWR as discussed in the response to **RAI 206-1576, Question 03.09.02-22**.

(2) The head of the US-APWR RCP is also considered to the forcing function.

In the revised MUAP-07027-P, Revision 2, the applicant added that the model tests were performed on ½ scale model pump which is similar to the pump planned for the US-APWR.

The staff finds this response acceptable because the applicant has fully substantiated the value of the RCP pulsation amplitude, which is used in the acoustic simulation, and the requested information has been incorporated in DCD Tier 2, Revision 2. Accordingly, **RAI 272-1585, Question 03.09.02-51, is resolved**.

In MUAP-07027-P, Revision 0, Section 3.3.1, the flow turbulence excitation in the downcomer is addressed. It is stated that "The methodology of the turbulence force generation proposed by Au-Yang (Reference 4 of MUAP-07027-P, Revision 0) is applied for the down comer forcing function with some modifications." However, no information is given in the report as to what these modifications are. In **RAI 272-1585, Question 03.09.02-52** (identified as RAI 3.9.2-29 in applicant's response), the staff requested the applicant to describe the modifications made in the methodology suggested by Au-Yang to define the turbulence excitation forces and to clarify the reason(s) for introducing these modifications.

In its response to **RAI 272-1585, Question 03.09.02-52**, dated April 9, 2009, the applicant pointed out that the original definition of the joint acceptance integral proposed by Au-Yang has been simplified in the following manner:

1. Assumed constant mode shape functions inside the acceptance integral.

2. Assume the down comer flow is purely axial so that the convection term in the pressure coherence function in the circumferential direction could be eliminated.

Because the joint acceptance involves integration over the entire mode shape, assumption 1 has only a secondary effect on the joint acceptance. An example of this is shown in Figure 8.5 in Au-Yang's book (Flow-Induced Vibration of Power and Process Plant Components). Assumption 2 is generally valid over most of the down comer flow surface. Since neither assumption involves the modal frequencies, the modal transfer functions are not affected. Therefore, the above two assumptions have no significant impact on the validity of the original method.

The staff finds this response acceptable because the applicant has fully clarified the modifications incorporated into the original formula proposed by Au-Yang. Since these modifications do not significantly impact the resulting forcing functions, and the requested information has been incorporated in MUAP-07027, Revision 1 and the DCD Tier 2, Revision 2, the staff concerns are resolved. Accordingly, **RAI 272-1585, Question 03.09.02-52, is resolved**.

The alternating stress Sa_{FIV} resulting from flow-induced forces is discussed in MUAP-07027-P, Revision 0, Section 3.4.2. A stress index (factor K) is used to account for structural discontinuity, but the used value for this factor is not given or discussed in the report. In **RAI 272-1585, Question 03.09.02-53** (identified as RAI 3.9.2-30 in applicant's response), the staff requested the applicant to provide and substantiate the assumed values of the stress index (factor K) for typical structural discontinuities, such as welds and joints between components with different thicknesses.

In its response to **RAI 272-1585, Question 03.09.02-53**, dated April 9, 2009, the applicant stated that a constant value 5.0 was used as the stress index (stress concentration factor K) for structural discontinuities in the equation of alternating stresses (Sa_{FIV}) in accordance with ASME B&PV Code, Section III. However, in the revised MUAP-07027-P, Revision 2, the applicant revised this statement. In the evaluation procedure described on page 87 (item 2(a) of Section 3.3.3.2.), it is stated that to simplify the analysis procedure, a general stress concentration factor of 5 was used for all low stress locations. However, for the higher stress locations, such as the cross-shaped legs in the fixed parts of the column structures, a stress concentration factor of two was used to calculate the peak stresses.

The staff finds the values recommended by the applicant for the stress concentration factor (K) to be sufficiently high to account for structural discontinuities such as partial and full penetration welds and joints between components of different thickness. The applicant has also included the value of K in the revised vibration analysis report MUAP-07027-P, Revision 2. Therefore, the staff finds the response satisfactory. Accordingly, **RAI 272-1585, Question 03.09.02-53, is resolved**.

With regard to the flow-induced vertical force acting on the reactor core, a PSD function as that used in the downcomer is assumed to act on the core support plates (see page 29 of MUAP-07027-P, Revision 0). It is not clear why this approach is used rather than use of the PSD of orifice flow to calculate the flow-induced vertical forces generated by the flow through the holes in the core plates. In **RAI 272-1585, Question 03.09.02-54** (identified as RAI 3.9.2-31 in the applicant's response), the staff requested the applicant to explain the reasons for using this approach and to explain the differences that may result in the flow-induced vertical forces if the PSD of a jet flow issuing from an orifice is used (instead of that of the downcomer). The staff also requested the applicant to explain the physical mechanism that causes the atypical discontinuities depicted in the vertical force PSD given in Figure 3.3-8, "RCP Pulsation Loads on Core Support Plates," of MUAP-07027-P, Revision 0.

In its response to **RAI 272-1585, Question 03.09.02-54**, dated April 9, 2009, the applicant addressed the similarity between the jet flow turbulence spectra existing in the reference literature and those measured in the downcomer just downstream of the inlet nozzles. In particular, the applicant applied the same measured pressure data in the down comer to the lower core support and upper core plate flow holes. The justification for this premise is based on the assumption that the pressure fluctuation close to the RPV inlet nozzle is caused by jet flow turbulence exiting from the inlet nozzle and, therefore is assumed to be similar to the jet flow turbulence through the lower core support plate and upper core plate flow holes.

Furthermore, the applicant provided a comparison between the PSD measured in the downcomer, which is used to estimate the vertical forcing function, and the PSD of a jet flow taken from the reference literature. These PSDs seem sufficiently similar such that they are expected to generate similar vertical forcing functions. In addition, the applicant does not use joint acceptance or correlation length coefficients to estimate the force. Instead, the total force is calculated from the SRSS of all flow holes on the plate. This approach will give a conservative estimate of the vertical forcing function. For these reasons, the staff finds the applicant's response to this RAI reasonable and concludes that the method used to estimate the vertical force is conservative. Accordingly, **RAI 272-1585, Question 03.09.02-54, is resolved**.

As mentioned earlier, SRP Section 3.9.2 and RG 1.20 recommend the validation of all numerical models and forcing functions that are used in the design process. Obviously, the reliability of these tools depends on the acceptance criteria adopted to compare the experimental results with the results of numerical models. The staff was concerned about the rigor of the acceptance criteria used by the applicant to validate the computational models and forcing functions. This concern can be explained by referring to MUAP-07027-P, Revision 0, Table 3.3-5 and Figure 3.4-2. In the table, the validation is judged acceptable, although the reactor design is much more complex than the simple annulus/cylinder geometry used to validate the SYSNOISE model, and in Figure 3.4-2, the analysis predictions of the core barrel and neutron reflector displacements are judged acceptable, although their predicted PSDs deviate substantially from those measured. The acceptance criteria for the validation of structural models (FE programs), acoustic excitations (SYSNOISE), and forcing function definitions need to be clarified to ensure that the reactor internal structures are designed to quality standards commensurate with the importance of their safety functions. In **RAI 272-1585, Question 03.09.02-55** (identified as RAI 3.9.2-32 in applicant's response), the staff requested the applicant to address this concern about the acceptance criteria.

In its response to **RAI 272-1585, Question 03.09.02-55**, dated April 9, 2009, the applicant referred to its response to **RAI 272-1585, Question 03.09.02-47** (identified as 3.9.2-24 in applicant's response) and **RAI 208-1574, Question 03.09.02-33** (identified as 3.9.2-70 in applicant's response), which deal with the acceptance criteria for SYSNOISE modeling and the start-up tests, respectively. On the validation of the simulation analysis of the 1/5 SMT, the applicant applied Category 2 acceptance criteria, namely, that:

a. Natural frequency for fundamental beam mode and lowest shell mode: within 10 percent.

b. Random response (displacement, stress etc.): factor of 3.

Although the applicant provided the information requested in the RAI, the basis and the consequences of the acceptance criterion of a factor of 3 in the random response and stresses are not clear. The staff questioned whether this meant that the actual stresses of the reactor internals can be up to a factor of three higher than the computed stresses. And if this is the case, how this factor is accounted for in the bias error and random uncertainties? Regarding the acceptance criterion of 10 percent in the resonance frequency, the applicant has not explained how the analysis accounts for an unanticipated coincidence between a resonance frequency and an excitation frequency. Therefore the staff closed, as unresolved, **RAI 272-1585, Question 03.09.02-55** and in follow-up **RAI 498-3782, Question 03.09.02-69**, the staff requested the applicant to address these concerns.

In its response to **RAI 498-3782, Question 03.09.02-69**, dated February 3, 2010, the applicant referred to the acceptance criteria for the measured responses in the pre-operational test, and stated that two categories (of acceptance criteria) were defined. Category 1 is the criteria on the structural integrity such as the alternating stress amplitude for the HCF evaluation. Category 2 criteria are on the comparison between the analysis and measured results, which are defined to check the prediction analysis reliability.

The applicant also noted that since Category 1 criteria has a higher priority, the high margin assumed in Category 2 to ensure conservative prediction of the actual stresses may be reduced in order to meet Category 1. The applicant also referred to the response to **RAI 272-1585, Question 03.09.02-44** and **RAI 272-1585, Question 03.09.02-56**, which include further discussions of the uncertainties in frequency and stress predictions. The applicant's clarification of the acceptance criteria, as well as the additional discussion of uncertainties included in the response to **RAI 272-1585, Question 03.09.02-44** and **RAI 272-1585, Question 03.09.02-56**, which are evaluated elsewhere in this report, have resolved the staff's concerns. Accordingly, **RAI 498-3782, Question 03.09.02-69** is resolved.

In **RAI 916-6343, Question 03.09.02-93**, the staff requested the applicant to explain why the difference between the estimated and measured natural frequencies of top slotted column exceeded the frequency acceptance criterion of 10 percent as discussed in MUAP-07027-P, Revision 2. In its response to **RAI 916-6343, Question 03.09.02-93**, dated April 26, 2012, the applicant explained that the calculated frequency was higher than the measured one because of the assumption of fixed conditions at both ends and the difficulties in accurately simulating the effect of slot openings with the beam model analysis. These features were precisely reproduced in the 1/5 scale model and therefore the measured frequency was considered more accurate. The applicant also added that this frequency difference is accounted for by adjusting the column stiffness in the analysis of the US-APWR. The staff finds this explanation reasonable because the difference in the frequency is clarified and accounted for in the analysis. Accordingly, **RAI 916-6343, Question 03.09.02-93, is resolved**.

In general, the staff found that the summaries provided in MUAP-07023-P, Revision 0, and MUAP-07027-P, Revision 0, inadequate. For example, the 1/5-scale model flow test report had an inadequate level of detail. It is therefore difficult to evaluate the data and results included in these important reports. In **RAI 272-1585, Question 03.09.02-56** (identified as RAI 3.9.2-33 in applicant's response), the staff requested the applicant to provide revised versions of the MUAP-07023-P and MUAP-07027-P with expanded detail to provide sufficient explanation of the included tests and results. On each table and figure included in these reports, the applicant was requested to give the relevant information, such as considered geometry (e.g., SMT, prototype of US-APWR, J-APWR, or 4-loop PWR) and source of data (e.g., measurements, FE simulation, or scaling up from SMT results).

In its response to **RAI 272-1585, Question 03.09.02-56**, the applicant submitted revised versions of the above-mentioned reports. In MUAP-07023-P, Revision 1, the applicant made revisions in order to:

1. Clarify the position of this technical report on the vibration assessment of the US-APWR reactor internals.

2. Add explanation of the test results and evaluation.

3. Identify the database for tables and figures, such as the direct measured data, measured data scaled to J-APWR dimensions, and analysis results.

In the revised MUAP-07027-P, Revision 1, substantial changes were made in order to accommodate the following requirements:

1. Reflect responses to RAIs.

2. Change the composition of Chapter 3 to correspond to the procedure in the validation of the analysis methodology, and add description to respond to the above RAIs.

3. Replace the analysis results of the J-APWR SMT and US-APWR Prototype based on re-evaluated forcing functions.

While the additional information provided in the revised reports clarified many aspects, several issues regarding the SMT and the validation of the dynamic analysis procedure still remain unclear. For example, the method used to validate the dynamic analysis of the reactor internals by means of the SMT is still unclear; the scaling laws used to convert the SMT results to the J-APWR are not substantiated; the approach used to validate the full-scale numerical codes seems inadequate; and the margin of safety in the HCF analysis does not seem conservative when compared with the level of uncertainties included in the acceptance criterion. Therefore, the staff closed as unresolved **RAI 272-1585, Question 03.09.02-56** and the staff's concerns are addressed in the following: first, for the SMT report, MUAP-07023-P, Revision 1, in follow-up **RAI 498-3782, Questions 03.09.02-70 to 03.09.02-74**, and then for the vibration assessment report, MUAP-07027-P, Revision 1, in follow-up **RAI 498-3782, Questions 03.09.02-75 to 03.09.02-77**.

In Section 3 of the revised MUAP-07023-P, Revision 1, the applicant indicates that the designs of the fuel assembly, the radial support of the core barrel, and the holes in the neutron reflector were modified in the scale model for the sake of simplicity. However, the details of these modifications and their possible effects on the test results are not addressed. In **RAI 498-3782, Question 03.09.02-70**, the staff requested the applicant to explain these modifications as well as any possible effects on the test results.

In its response to **RAI 498-3782, Question 03.09.02-70**, dated February 3, 2010, the applicant provided a table detailing the modifications used in the scale model. Concerning the fuel assemblies, the applicant explained that because the vibration of the fuel assembly was verified with a full scale mock-up test, fuel assembly in the 1/5 scale model was simplified. The numbers of rods and grids were reduced although the scaled mass and pressure drop were still simulated. The natural frequency of the fuel was not simulated. But its impact on the vibration responses of the reactor internals was small because their natural frequencies are well separated. The applicant also clarified the modification of the neutron reflector, stating that for the neutron reflector, the numbers and diameter of flow-holes were modified so that the total section area of flow holes was properly scaled (1/25 of that in the actual plant). This modification had no impact on the shell mode stiffness and natural frequencies as confirmed by FE analysis. With regard to the radial support of the core barrel, the applicant stated that the modification of the radial key was not for simplification but to control the test conditions. The shapes and locations of the radial key were modeled to simulate the flow around the radial key. Because the support condition of the core barrel bottom was controlled by additional push bolts, the clearance of the radial key was extend to assure the no contact condition.

The staff concurs that these modifications in the scale model are acceptable because their effect on the dynamic response of tested reactor internals is small. Accordingly, **RAI 498-3782, Question 03.09.02-70, is resolved**.

Section 6.1 of the revised MUAP-07023-P, Revision 1 suggests that the SMT results were scaled up to the J-APWR, and the dynamic analysis was performed on the J-APWR. However, in its response to the **RAI 272-1585, Question 03.09.02-44**, and in the revised version of MUAP-07027-P, Revision 1, the applicant explained that in the FIV analysis program, the measured responses of the J-APWR scale model tests were compared with those estimated by the dynamic analysis applied to the SMT size and test conditions. Also, in Figures 3.2.1-3 to 12 of MUAP-07027-P, Revision 1, the figure captions refer to “actual dimensions” without indicating whether these dimensions are those for the SMT or the full-scale reactor. In **RAI 498-3782, Question 03.09.02-71**, the staff requested the applicant to confirm how the dynamic analysis of the reactor internals was benchmarked by means of the SMT in order to resolve the apparent contradictions mentioned above. In particular, the staff requested the applicant to explain whether the dynamic analysis was performed on the size and flow conditions of the small scale model or the full-scale J-APWR.

In its response to **RAI 498-3782, Question 03.09.02-71**, dated February 3, 2010, the applicant stated that the benchmark analysis model was developed with the 1/5 scale model dimensions. The material properties of structures and coolant are defined at room temperature as the test conditions. Comparisons of the analysis response and measured values were performed after scaling up to the plant because the measured results had been scaled in the test report. The scaling factor from test dimensions to actual ones is shown in Table 3.1, "Outline of the Test Facilities," of MUAP-07023-P, Revision 1. In addition to the scaling factors, the effect of the temperature difference such as fluid mass density and Young moduli has been discussed in the response to **RAI 498-3782, Question 03.09.02-72**.

The applicant's response explains the source of confusion: although the analysis model and the measurements were performed on the 1/5-scale model, the comparison of the results was made after the results were scaled to the full-size reactor. The staff found the response acceptable because the applicant clarified the source of the confusion. The applicant also included a discussion of this response in MUAP-07023-P, Revision 2. Accordingly, **RAI 498-3782, Question 03.09.02-71, is resolved**.

Table 3.1 of the revised MUAP-07023-P, Revision 1 indicates that no scaling is needed to convert the strain and stress from the SMT measurements to the J-APWR. This conclusion is not appropriate since the SMT and J-APWR are not identical in size or flow conditions. In Tables 6.8, "Fatigue Evaluation in the Case of 77 GTs (100% Flow)," to 6.14, "Fatigue Evaluation of the Bolts in the Case of 85 GTs (100% Flow)," of the same report, the method of strain and stress conversion is not clear, and in Tables 6.2, "Converted Vibration Response of Displacement (RMS) Corresponding to Actual Reactor Internals (In the Case of 77 GTs)," and 6.3, "Converted Vibration Response of Displacement (RMS) Corresponding to Actual Reactor Internals (In the Case of 85 GTs)," the conversion of measured displacement to the J-APWR is not explained. In addition, the source of the stress equation for HCF, which is cited on page 4, is not given. Therefore, in **RAI 498-3782, Question 03.09.02-72**, the staff requested the applicant to substantiate the methods used to convert or scale the displacement, as well as the strain and the stress, from the SMT data to the full-scale J-APWR.

In its response to **RAI 498-3782, Question 03.09.02-72**, dated January 15, 2010, the applicant explained the scaling rules used to convert the SMT results to the full size plan t. In its response, the applicant stated that in general, in addition to geometric scaling, adjustments due to differences in the fluid mass densities and Young moduli are needed to convert flow-induced dynamic responses from a scale model test at room temperature to those in the full-size reactor under plant operating conditions.

In addition, the applicant provided the formula used to calculate the displacement, as well as the strain and the stress, of the J-APWR from the SMT measurements. The staff finds these formulae appropriate because they take into account the differences in the size and test conditions between the scale mode and the J-APWR. In fact, some of these differences are intentionally ignored in order to maintain additional conservatism in the scaling procedure.

The staff finds the applicant's response reasonable because it indicates that appropriate scaling rules have been used in the dynamic analysis. However, these scaling rules are valid only if the scale model geometry is an accurate replica of the full size plant. The staff examined the effects of the geometrical differences between the SMT and the US-APWR on the scaling procedure and concluded that these effects are addressed satisfactorily in the responses to **RAI 498-3782, Question 03.09.02-65**, discussed above, and **RAI 498-3782, Question 03.09.02-84**, discussed below. Accordingly, **RAI 498-3782, Question 03.09.02-72, is resolved**.

In Table 6.4, "Evaluation Results of Flow Load (120 % Flow) (77 GTs)," of MUAP-07023-P, Revision 1, the staff found some parameters undefined and conversion (or scaling) methods unexplained. For example, it is not clear how the moment was converted from the SMT to the

J-APWR. Also, the applicant does not explain what is meant by the term "design load," especially when this "design load" is lower than that measured from the SMT. In **RAI 498-3782, Question 03.09.02-73**, the staff requested the applicant to explain the procedure used to estimate the design load and to scale it to the full-size reactor. The staff also requested the applicant to explain when and how the design load will be determined for the bottom-mounted instrumentation nozzle.

In its response to **RAI 498-3782, Question 03.09.02-73**, dated January 15, 2010, the applicant responded to three issues: 1) the procedure of evaluating the bending moment, 2) the relevance of the design load, 3) and the determination of the design load for the bottom-mounted instrumentation nozzle.

Regarding the first issue, the applicant stated that the relationship between the loading moment and the measured strain was obtained by a series of unit loading tests for the column structures, which have been performed as part of the flow test. The moment at the end of the column was derived from the measured strain and correlation factor determined by the unit loading test.

Regarding the second issue with respect to the relevance of the design load, the applicant added that the design load had been determined by analysis or hand calculation in the process of sizing the structural components. In case a design load is lower than that measured in the SMT, the need for revising the design load should be evaluated. For example the moment of the RV water level instrumentation support tube is lower than the measured value as shown in Table 6-4 of MUAP-07023-P, Revision 1, but the design load is not revised because this component is not located in a high cross flow region in the upper plenum and the corresponding measured stress is much lower than the allowable limit as shown in Table 6-7, "Stress Evaluation in the Case of 77 GTs (120% Flow)," of MUAP-07023-P, Revision 1.

Regarding the third issue, the applicant stated that the determination of the design load of the J-APWR bottom mounted instrumentation nozzle was not performed because of its very limited length exposed to the cross flow. The corresponding measured stress is much lower than the allowable limit as shown in Table 6-7 of MUAP-07023-P, Revision 1.

The staff finds the applicant's response to the three concerns raised in **RAI 498-3782, Question 03.09.02-73** reasonable, except for the issue of scaling the bending moment. In this regard, the applicant clarified how the bending moment was measured during the SMT, but did not explain how the moment measured from the SMT was scaled up to the full size US-APWR. Since the scaling procedure and the effects of geometrical differences between the scale model and the US-APWR are addressed satisfactorily in **RAI498-3782, Questions 03.09.02-65, 03.09.02-72, and 03.09.02-84**, the staff considers **RAI 498-3782, Question 03.09.02-73, is resolved**.

In Section 6.1 of the revised MUAP-07023-P, Revision 1, the applicant states the following:

> These natural frequencies, after scaling up to the J-APWR reactor internals in water were shown in Table 6-1, then test results were compared with the J-APWR pre-analysis results to confirm the adequacy of the J-APWR 1/5 test models.

The staff believes that one of the objectives of the SMT is to validate the dynamic analysis, and not to use the dynamic analysis to confirm the adequacy of the small-scale test models. Therefore, in **RAI 498-3782, Question 03.09.02-74**, the staff requested the applicant to explain what is meant by the above-cited statement.

In its response to **RAI 498-3782, Question 03.09.02-74**, dated January 15, 2010, the applicant clarified the test objective of the J-APWR scale model tests and explained the use of the results to design the US-APWR.

Regarding the test objective for the J-APWR design, the applicant stated:

> First, we must clarify the background of the J-APWR 1/5 SMT and the historic evolution of the validation process. The J-APWR 1/5 scale model was performed in 1996 as the design confirmation of the J-APWR. MUAP-07023-P, Revision 1 is the English translation version of the original Japanese report written in 1996. The pre-analysis was performed before the test with actual plant dimensions and with a FEM code developed in Japan. In the test procedure, it was true that the scale model test results were cross-checked by comparison with the predicted natural frequencies from the FE analysis as described in Section 6.1 of MUAP-07023-P, Revision 1.

Concerning the use of J-APWR SMT results for the US-APWR design, the applicant added:

> In the design process of US-APWR, the J-APWR SMT results are used for the verification of dynamic analysis method through the benchmark analysis described in Chapter 3 of MUAP-07027-P, Revision 1. The benchmark analysis was performed in 2006. For this purpose, a 1/5 scale dimensions model was re-constructed from the original J-APWR actual plant model described above. Any fine tuning with the test results was not included. The FEM code was changed to ANSYS which was also used for the US-APWR analysis.

The staff reviewed the applicant response and accepts the applicant clarification of the statement. Since the applicant response confirms that the dynamic analysis of the US-APWR is verified by means of the SMT, and not the other way around, the staff's concerns regarding this statement are resolved. Accordingly, **RAI 498-3782, Question 03.09.02-74, is resolved**.

With respect to the revised MUAP-07027-P, Revision 1, the review indicated that substantial uncertainties exist in the dynamic analysis. For example, in the revised SYSNOISE analysis, the RCP pulsation amplitude is reduced by a factor of five, and the response of the reactor internals to this RCP pulsation increases by a factor of five when the simulation time step is refined. Moreover, when comparing the SMT random response with the response obtained from the dynamic analysis, a ratio of three between the measured and predicted values is considered acceptable. Despite these substantial uncertainties indicated above, the applicant considers a margin of safety of 30 percent acceptable for the HCF analysis, as indicated in Table 3.3.3-4, "High Cycle Fatigue Evaluation Based on Analysis Responses (US-APWR Analysis Results)," of the above-mentioned report. In **RAI 498-3782, Question 03.09.02-75**, the staff requested the applicant to explain why this margin of safety (30 percent) is considered conservative despite the existing high degree of uncertainty.

In its response to **RAI 498-3782, Question 03.09.02-75**, dated February 3, 2010, the applicant stated that the margin of safety of 0.3 takes into account the excitations caused by cross flow and RCP pulsation. The applicant added also that this margin of safety was verified to be acceptable based on the following considerations:

1. The alternating stresses due to the RCP pulsation have large uncertainty (factor of 5) but the absolute values are lower than those due to the cross flow by one order of magnitude.

2. The RCP pulsation loads include a conservative bias by neglecting the acoustic damping due to structural flexibility as discussed in the response to **RAI 498-3782, Question 03.09.02-68**. Because this effect is also the main part of the uncertainty in the acoustic resonance analysis, the magnitude of bias error is approximately the same as the uncertainty (factor of five). Therefore, the analysis results due to RCP pulsation may be 10 times larger than the actual values, but not smaller.

3. The cross flow loads on the upper and lower plenum structures are determined with peak cross flow velocity along the entire length of structures. The bias due to neglecting the cross flow distribution is estimated to be around a factor of two, which is comparable to the assumed uncertainty in the flow-induced loads.

4. From the above discussions, the minimum margin of safety of 0.3 for the upper plenum structures due to cross flow loads includes a conservative bias of around two due to non-uniform cross flow distribution. Because this bias is comparable to the assumed uncertainty in the flow induced loads (factor of two), the margin of safety 0.3 was considered acceptable.

The staff reviewed the applicant's response and finds it acceptable because the margin of safety already covers conservatively estimated bias errors and uncertainties associated with determining the loading functions due to cross flow and RCP pulsation. The applicant included this discussion in MUAP-07027-P, Revision 2. Accordingly, **RAI 498-3782, Question 03.09.02-75, is resolved**.

Figure 3.1.1-1, "US-APWR Reactor Internals FIV Response Analysis Procedure," of MUAP-07027-P, Revision 1 shows a revised flow chart of the dynamic analysis procedure of the US-APWR reactor internals. This flow chart suggests that if the calculated response of J-APWR does not agree with the scale model test measured response, a correction factor of the forcing function is determined. Since this correction factor was not included in the original version of the report, in **RAI 916-6343, Question 03.09.02-94**, the staff requested the applicant to explain what is meant by this correction factor and if any correction factors are used in the validation procedure to formulate the FIV forces acting on the internal structures of the US-APWR. The staff also requested the applicant to explain why a correction factor in the forcing function is sought rather than in the FE modeling, damping values, or any other input parameters of the analysis procedure.

In its response to **RAI 916-6343, Question 03.09.02**, dated April 26, 2012, the applicant stated that a correction factor would be used only when the FE results agree with the measured frequencies but the estimated response does not agree with the measured response. However, no correction factors were applied in the analysis of the J-APWR SMT and the US-APWR because the difference between calculated and measured response was within the acceptance criterion. The staff finds this response acceptable because no correction factors were applied to the forcing functions. Accordingly, **RAI 916-6343, Question 03.09.02-94, is resolved**.

Figure 3.2.2-3, "Normalized Pressure PSD vs. Reduced Frequency (Semi-log Scales)," of MUAP-07027-P, Revision 1 shows the PSD of the turbulence excitation in the downcomer. This

PSD is clearly stronger near the inlet nozzles and becomes weaker as the flow progresses along the downcomer. In **RAI 498-3782, Question 03.09.02-76**, the staff requested the applicant to explain the axial and circumferential distributions of the turbulence excitation PSD, which are used in the dynamic analysis of the reactor internals.

In its response to **RAI 498-3782, Question 03.09.02-76**, dated January 15, 2010, the applicant referred to the response to **RAI 498-3782, Question 03.09.02-67**, which includes the requested distributions of the turbulence excitation PSD. As stated in the evaluation of **RAI 498-3782, Question 03.09.02-67**, the applicant used appropriate PSD distributions to estimate the dynamic response of the core barrel. Therefore, the staff's concerns about the PSD distributions are resolved. Accordingly, **RAI 498-3782, Question 03.09.02-76, is resolved**.

In MUAP-07027-P, Revision 1, the applicant attached a comparison (Figure 3.2.3-1) between the measured and predicted amplitudes of the relative displacement between the core barrel and the RV. The staff noticed the predicted amplitude of the peak at 30 Hz to be substantially smaller than that measured from the SMT. Therefore, in **RAI 916-6343, Question 03.09.02-95**, the staff requested the applicant to clarify this difference. In its response to **RAI 916-6343, Question 03.09.02-95**, dated April 26, 2012, the applicant stated that the difference between the analysis and measured amplitude at 30 Hz is caused by the assumed spatial distribution of the downcomer forcing function. The applicant provided a detailed analysis illustrating the effect of this spatial distribution on the relative motion and concluded that the overall root mean square (rms) response, which is used in the stress analysis, is very weakly affected by the assumed distribution. In fact, the assumptions made in the analysis provided a conservative estimate of the overall rms response in comparison to other plausible distributions of the forcing function. The staff reviewed the analysis provided by the applicant and concurs that the estimated rms response of the core barrel is conservative. Accordingly, **RAI 916-6343, Question 03.09.02-95, is resolved**.

In the introduction section of MUAP-07027-P, Revision 1, as well as in several other sections of the same report, the applicant states:

> …measured data in the J-APWR scale model test was used for the forcing functions due to the downcomer flow turbulence. After the completion of Revision 0 of this report at the end of 2007, new data pertinent to the US-APWR configuration was obtained in the US-APWR Reactor Vessel Lower Plenum 1/7 Scale Model Flow Test.

Therefore, in **RAI 498-3782, Question 03.09.02-77**, the staff requested the applicant to explain why the SMT of the J-APWR in the revised reports was not entirely replaced by the available SMT of the US-APWR.

In its response to **RAI 498-3782, Question 03.09.02-77**, dated January 15, 2010, the applicant stated that only data pertaining to the lower plenum from the US-APWR 1/7 scale lower plenum test was included in the vibration assessment report because of the following reasons:

1. The test model was set up-side down to give easy accessibility to the lower plenum structures and flow visualization. In addition, the fuel assemblies, the radial reflector and the upper core internals were not included in this model. Thus FIV of these structures were not within the scope of this test, which was designed specifically for the lower plenum structures.

2. The flow paths from the vessel inlet nozzle to the downcomer and the lower plenum were simulated in the Lower Plenum 1/7-Scale Model Test, and the layout and dimensions of these configurations were not changed from J-APWR. Therefore, the pressure fluctuation data of the downcomer measured in US-APWR lower plenum test can be applied both to the simulation analysis of the J-APWR SMT and the US-APWR prediction analysis.

The staff reviewed the applicant's response and the report describing the 1/7-scale lower plenum model tests of the US-APWR, and agrees that only the data pertaining to the pressure pulsations in the downcomer are applicable to the current application because the internals of the core barrel and the upper plenum were not included in the tested 1/7-scale model. The staff's concerns about this issue are resolved. Accordingly, **RAI 498-3782, Question 03.09.02-77, is resolved**.

The 1/5-scale model tests reported in MUAP-07023, Revision 0, were performed with either 77 or 85 guide tubes. The report does not explain why these numbers of guide tubes were chosen for the model tests. Therefore, in **RAI 272-1585, Question 03.09.02-57** (identified as RAI 3.9.2-34 in applicant's response), the staff requested the applicant to explain the reason(s) for choosing these numbers of guide tubes for the model tests and to elaborate on the significance of these numbers of guide tubes, in comparison to the total numbers of guide tubes that will be used during normal operation of the US-APWR.

In its response to **RAI 272-1585, Question 03.09.02-57**, dated April 9, 2009, the applicant stated that both the 77-guide tube and 85-guide tube configurations selected for the J-APWR scale model test were based on requests from Japanese customers. In general, increasing the number of guide tubes leads to an increase in the blockage factor and cross flow loads. Therefore, the cross flow loads on the guide tubes in the US-APWR, with its 69-guide tube design, are bounded by the test results of J-PWR SMT.

The staff finds the tested number of guide tubes in the J-APWR scale model acceptable because these tests result in higher cross-flow velocities than in the US-APWR design, and the measured forcing functions in these scale model tests will be conservative for the US-APWR. Accordingly, **RAI 272-1585, Question 03.09.02-57, is resolved**.

MUAP-07027, Revision 0, notes that the flow velocity in the vessel exit nozzle of the US-APWR will be increased, compared with the current 4-loop reactors, from 54 ft/s (16 m/s) to 61 ft/s (19 m/s), and the cross flow velocity in the upper core plenum will also be increased from 26.6 ft/s (8.11 m/s) to 31.2 ft/s (9.51 m/s). In **RAI 272-1585, Question 03.09.02-58** (identified as RAI 3.9.2-35 in applicant's response), the staff requested the applicant to discuss the analysis performed and the tests planned to demonstrate that adverse flow effects will not cause unanticipated excessive FIV or structural damage to the reactor piping systems and the internal structures in the upper core plenum near the exit nozzles, which will be subjected to flow velocities higher than those in the 4-loop reactor.

In its response to **RAI 272-1585, Question 03.09.02-58**, dated April 9, 2009, the applicant stated that the structural components have large structural and HCF margins of safety to accept higher FIV loadings and cross-flow loads loadings from the higher flow velocities.

Furthermore, the applicant concludes that the structures in the upper plenum have sufficient margins of safety for the adverse flow effect due to the cross flow such as the vortex shedding lock-in or fluid elastic instability, as discussed in Subsection 3.2.3 of MUAP-07027-P, Revision 0. Finally, the applicant refers to similar statements in the DCD Tier 2, Subsection 3.9.2.3.2, “Comparative Analysis of the US-APWR and the Current Plant,” which states that the reduced velocity U/f_nD for the upper plenum structures is slightly higher than that of the current 4-loop plant, but sufficient margins of safety are maintained for cross-flow induced vibrations, such as fluid elastic instability and turbulence-induced vibration. In addition, since the vortex shedding frequencies are lower than 70 percent of the structural fundamental frequencies, lock-in vortex-induced vibration is avoided per ASME Code, Section III, Appendix N-1324.

The above response dealing with the adverse flow effects on the upper reactor internals seems reasonable, especially in light of the fact that the cross flow velocity has been estimated rather conservatively, as described in the applicant’s response to **RAI 272-1585, Question 03.09.02-50**. However, the applicant did not comment on the adverse flow effects on the reactor piping due to the increase in flow velocity. Therefore, the staff closed as unresolved **RAI 272-1585, Question 03.09.02-58** and in follow-up **RAI 498-3782, Question 03.09.02-78**, the staff requested the applicant to address this issue.

In its response to **RAI 498-3782, Question 03.09.02-78**, dated January 15, 2010, the applicant stated that MUAP-07027-P, Revision 1 describes that the flow velocity in the vessel exit nozzle of the US-APWR will be increased in comparison with the current 4-loop reactors. The effects of higher flow velocity on the piping system have been confirmed based on the analysis of the current 4-loop reactors. The result shows the structures in the piping have sufficient margins of safety for the vortex shedding lock-in and fluid elastic instability. Therefore, it is concluded that the flow velocity increase has no impact on the instability of vibration of the piping system.

The staff finds the response to this RAI unsatisfactory. The applicant was requested to discuss the analysis performed to assess adverse flow effects on the reactor piping system due to increasing the flow velocity above that of the 4-loop reactors. In its response, the applicant does not give, or reference, any details of the analysis and just mentions that based on the analysis of the 4-loop reactor, there is sufficient margin of safety against flow-induced vibration. Therefore, the staff closed as unresolved **RAI 498-3782, Question 03.09.02-78** and in a new follow-up **RAI 614-4853, Question 03.09.02-91**, the staff requested the applicant to discuss the analysis performed to assess adverse flow effects on the reactor piping system due to the increased flow velocity at the vessel outlet nozzle as identified in Table 2.1-1, “Comparison of Typical Flow Velocities between the US-APWR and the Current 4-loop Plant,” of MUAP-07027-P, Revision 1.

In its amended response to **RAI 614-4853, Question 03.09.02-91**, dated December 2, 2011, the applicant stated that it had planned to adopt its thermowell design of the current 4-loop reactor for US-APWR reactor coolant loop piping. This thermowell design was confirmed to have sufficient margin against FIV under the flow condition of US-APWR reactor. The applicant added that it was also confirmed that reduced velocity (V/f_1D) is less than one in accordance with ASME Code, Section III, Appendix N-1324, and that generated stress is less than the lower bound of design fatigue curve. The applicant further stated that this plan, however, has been changed and the applicant has determined to procure thermowell from a thermocouple supplier. The applicant also revised the reactor coolant loop piping specification to confirm that FIV assessment will be performed in accordance with ASME Code, Section III, Appendix N-1300.

The staff finds the applicant’s response acceptable because the applicant has committed to use thermowells which meet the ASME Code requirements to avoid FIV under the flow conditions of the US-APWR. Accordingly, **RAI 614-4853, Question 03.09.02-91, is resolved**.

Based on the evaluation of the applicant's response to the RAIs on DCD Tier 2, Revision 0, Section 3.9.2.4, "Preoperational Flow-Induced Vibration Testing of Reactor Internals," the staff was still concerned about the differences between the scale model geometry and the US-APWR. The applicant had already addressed some of these differences, but others seem to exist in the submitted drawings that had not been addressed by the applicant (e.g., follow-up **RAI 498-3782, Question 03.09.02-65**). The staff was also concerned that additional differences could be seen in the scale model drawings. There were also several open issues concerning the validation of the simulation models. Therefore, in **RAI 498-3782, Question 03.09.02-84**, the staff requested the applicant to provide a list of all the differences between the US-APWR and the geometry of the scale model that is used in the vibration testing. The staff also requested the applicant to demonstrate that the effect of each of these differences on the estimated vibration response of the US-APWR is conservative.

In its response to **RAI 498-3782, Question 03.09.02-84**, dated February 3, 2010, the applicant referred to the response to **RAI 498-3782, Question 03.09.02-65**, which provided a list of all the differences between the US-APWR and the geometry of the scale model, and demonstrated that the effect of each of these differences on the estimated vibration response of the US-APWR is conservative. Additional simplifications of the SMT geometry were explained and rationalized in the response to **RAI 498-3782, Question 03.09.02-70**. The staff found the response to these two RAIs acceptable and addressed the concerns in **RAI 498-3782, Question 03.09.02-84**. Accordingly, **RAI 498-3782, Question 03.09.02-84 is resolved**.

Based on the SG tube degradation event at San Onofre Nuclear Generating Station (SONGS) Unit 3 in 2012, in **RAI 930-6494, Question 05.04.02.01-13**, the staff requested the applicant to demonstrate that the US-APWR SG tube bundle design will prevent such degradation.

In its response to **RAI 930-6494, Question 05.04.02.01-13**, dated December 12, 2012, the applicant attributes the tube-to-tube wear and tube-to-support [anti vibration bar (AVB) and tube support plate (TSP)] to the high steam quality and small contact force with AVBs. For the SGs of the US-APWR, the applicant states that the steam quality will be lower and the contact force will be more effective than that of the SONGS replacement SGs (RSGs), without discussing the effect of these parameters on the critical velocity of fluid elastic instability. The staff found the response incomplete and in **RAI 1013-7031, Question 03.09.02-103**, the staff requested the applicant to:

(a) Provide a comparison of technical data of both SGs (SONGS and US-APWR) to illustrate that the critical velocity has been exceeded in the SONGS case but will not be exceeded in the US-APWR case.

(b) Explain any design differences between the SG of the US-APWR and the Fort Calhoun RSGs and clarify the effect of these differences (if any) on the flow-induced vibration response of the tube bundle.

(c) Explain how the contact force of the AVBs will be checked to ensure it is sufficiently high to prevent in-plane tube instability.

(d) Explain why the wear at the TSP is considered to be caused by turbulence excitation and not by in-plane tube instability.

RAI 1013-7031, Question 03.09.02-103 is being tracked as an Open Item.

In **RAI 1013-7031, Question 03.09.02-104**, the staff requested the applicant to provide the preliminary design of the SG tube bundle and the design criteria for the SG tubes and retainer bars against flow-induced excitations, including random turbulence, fluid elastic instability (out-of-plane and in-plane), and vortex shedding. **RAI 1013-7031, Question 03.09.02-104 is being tracked as an Open Item**.

3.9.2.4.3.4 Combined License Information Items for Dynamic Response Analysis of Reactor Internals under Operational Flow Transients and Steady-State Conditions

The following is a list of pertinent COL items and descriptions from DCD Tier 2, Table 1.8-2, "Compilation of All Combined License Applicant Items for Chapters 1-19," applicable to dynamic response analysis of reactor internals under operational flow transients and steady-state conditions:

Table 3.9.2-1
US-APWR Combined License Information Items

Item No.	Description	Section
COL 3.9(2)	The first COL Applicant is to complete the vibration assessment program, including the vibration test results, consistent with guidance of RG 1.20. Subsequent COL applicants need only provide information in accordance with the applicable portion of Position C.3 of RG 1.20 for non-prototype internals.	3.9.2.3

The finds COL Information Item 3.9(2) acceptable since it implements the guidance of RG 1.20.

3.9.2.4.3.5 Conclusion for Dynamic Response Analysis of Reactor Internals under Operational Flow Transients and Steady-State Conditions

As a result of the open items for **RAI 1013-7031, Question 03.09.02-103** and **RAI 1013-7031, Question 03.09.02-104**, the staff is unable to finalize its conclusions on section 3.9.2.4.3 related to the dynamic response analysis of reactor internals under operational flow transients and steady-state conditions, in accordance with NRC regulations.

3.9.2.4.4 Preoperational Flow-Induced Vibration Testing of Reactor Internals

3.9.2.4.4.1 Summary of Application for Preoperational Flow-Induced Vibration Testing of Reactor Internals

DCD Tier 1/ITAAC: The Tier 1 information associated with this section is found in DCD Tier 1, Section 2.4.1, which lists ITAAC for reactor internals FIV tests.

DCD Tier 2: The applicant has provided a DCD Tier 2 system description in Section 3.9.2.4, summarized here in part, as follows. Following the recommendation of RG 1.20, the design of the US-APWR is classified as a "Prototype Design." Therefore, the applicant is planning to perform a preoperational vibration measurement program to confirm that unexpected, abnormal vibrations do not occur, and to ensure that the vibration responses of the reactor internals are sufficiently small compared to an acceptance criterion based on the design fatigue curves in the ASME Code, Section III. The test results will also be used to confirm the vibration analysis of the reactor internals (see Section 3.9.2.4.6 of this report).

The measurement program described in DCD Tier 2, Section 3.9.2.4 includes installing strain gages, accelerometers, pressure sensors, and displacement transducers at specific locations on the core barrel, lower core support plate, neutron reflector, secondary core support assembly, upper core support, RCCA guide tubes, and upper support columns. The applicant also provided an inspection program and acceptance criteria for the preoperational testing program of reactor internals. DCD Tier 2, Section 3.9.2.4 includes only a brief summary of the preoperational testing program, but additional details are provided in MUAP-07027-P, Revision 3.

3.9.2.4.4.2 Regulatory Basis for Preoperational Flow-Induced Vibration Testing of Reactor Internals

The relevant requirements of the Commission's regulations for this area of review, and the associated acceptance criteria are given in SRP Section 3.9.2, SRP Acceptance Criterion 4 and are summarized below.

1. GDC 1, as it relates to structures and components being designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with the importance of the safety function to be performed.

2. GDC 4, as it relates to designing reactor internals to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents.

Acceptance criteria adequate to meet the above requirements include:

1. For requirements of GDCs 1 and 4, the preoperational vibration and stress test program for the internals of a prototype reactor, for existing reactors under consideration for power uprate, and for non-prototype reactors whose valid or conditional prototypes have experienced structural failures due to adverse flow effects in any plant (e.g., steam dryer cracking and valve failures) should conform to the requirements for a prototype test as specified in RG 1.20, including vibration prediction, vibration monitoring, adverse flow effects (flow-induced acoustic and structural resonances, data reduction, bias errors and uncertainty analysis, and walkdown and surface inspections.

2. RG 1.20, "Comprehensive Vibration Assessment Program for Reactor Internals during Preoperational and Initial Startup Testing," Revision 3, issued March 2007.

3.9.2.4.4.3 Technical Evaluation of Preoperational Flow-Induced Vibration Testing of Reactor Internals

The review of the preoperational vibration testing of reactor internals was performed in accordance with SRP Section 3.9.2 and RG 1.20 to ensure the structural and functional integrity of the reactor and SG internals under vibratory loadings and thereby assure conformance with GDC 1 and 4. Although DCD Tier 2, Section 3.9.2.4 includes a commitment to performing preoperational vibration testing consistent with COL Information Item 3.9(2) and provides details of the sensors to be used, the DCD did not meet some of the guidance recommended in RG

1.20 and SRP Section 3.9.2. Most notable is the absence of preoperational vibration testing of the steam delivery system, including the SG, safety relief valves, and the MS lines.

According to SRP Section 3.9.2, the SG preoperational test program is expected to be described in DCD Tier 2, Section 3.9.2. Therefore, in **RAI 206-1576, Question 03.09.02-19** (identified as 3.9.2-40 in applicant's response), the staff requested the applicant to provide appropriate information about the preoperational vibration testing of the SG. As stated in the RAI, the nature of the requested information depends on whether the SG is classified as a prototype or not. Regarding the rest of the steam delivery system, which is not addressed in the DCD, in **RAI 206-1576, Question 03.09.02-20** (identified as RAI 3.9.2-41 in applicant's response), the staff requested the applicant to provide additional details about the FIV measuring and monitoring program for the preoperational and start-up tests of the steam separator, the safety relief valves, and the MS lines. For the two RAIs mentioned above, the requested additional information should include the measurement locations with diagrams, test conditions, hold points to allow data acquisition and analysis, and inspection and monitoring program.

In its response to **RAI 206-1576, Question 03.09.02-19**, dated March 27, 2009, the applicant stated that the topic of SG upper internals vibration is addressed in DCD Subsection 5.4.2.1.2.10 (via a cross reference from Subsection 3.9.2.4.1). The applicant will revise DCD Tier 2, Subsection 5.4.2.1.2.10 to more fully address the structural adequacy of the SG internals.

The staff finds this response acceptable and agrees that since similar SGs have been in use in existing plants for many years without any vibration problems, there is no need to perform startup testing of the SGs. However, contrary to the statement of the applicant, the staff could not find any reference in DCD Subsection 3.9.2.4.1 to Subsection 5.4.2.1.2.10 that addresses the SG dynamic response. Therefore, the staff closed as unresolved **RAI 206-1576, Question 03.09.02-19** and in follow-up **RAI 498-3782, Question 03.09.02-79**, the staff requested the applicant to add the cross reference to Subsection 3.9.2.4.1 mentioned above.

The staff also requested the applicant to include a reference to the statement that the design of the US-APWR steam delivery system, including the SG upper internals, the safety relief valves, and the steam lines, has been employed in the U.S. for more than 20 years in sizes and flow rates that bound those of the US-APWR. These additions are requested so that the DCD document meets the guidance of RG 1.20 and SRP Section 3.9.2.

In its response to **RAI 498-3782, Question 03.09.02-79**, dated January 15, 2010, the applicant stated that the cross reference from DCD Tier 2, Subsection 3.9.2.4.1 to DCD Tier 2, Subsection 5.4.2.1.2.10 is described in the last paragraph of DCD Tier 2, Revision 2, Subsection 3.9.2.4.1.

Regarding the steam delivery system, the applicant added:

> The similar design of the US-APWR steam delivery system has been operating almost 20 years as shown in FSARs (e.g., Comanche Peak and Alvin Vogtle). Flow velocity is an important factor relative to vibration, and the flow velocity of the US-APWR steam delivery system is:
>
> - MS piping: approximately 150 ft/s (45.7 m/s).

- MS safety valve inlet piping: approximately 500 ft/s (152 m/s).

MS piping and MS safety valve inlet piping have almost the same flow velocity as current existing, 4-loop plants in U.S.

The staff finds the applicant's response acceptable because, as requested in the RAI, the applicant added a cross reference to DCD Tier 2, Subsection 5.4.2.1.2.10, which deals with the dynamic analysis of the SG internals. In addition, the applicant provided adequate information about the steam delivery system. Accordingly, **RAI 498-3782, Question 03.09.02-79, is resolved**.

With respect to **RAI 206-1576, Question 03.09.02-20**, regarding the rest of the steam delivery system, the applicant informed the staff that similar systems have also been in use in the U.S. for many years without any vibration problems. In particular, the applicant stated that the design of the US-APWR steam delivery system (including the steam separator, safety relief valves, and steam lines) and the flow conditions they experience are similar to the existing and currently operating steam delivery systems in the U.S. and around the world. The applicant designed the US-APWR steam delivery system using the structural design rules based on years of empirical experience with similar equipment. The configuration employed in the US-APWR steam delivery system has been operating in the U.S. for more than 20 years in the steam delivery system of sizes and flow rates that bound those of the US-APWR steam delivery system. Based on an extensive record of vibration-free operation, the applicant concludes that the structural and vibration design bases are proven. This nonsafety-related steam delivery system will not experience excessive vibration; therefore, no startup testing is planned for the steam delivery system.

The staff finds this response acceptable and agrees that since similar steam delivery systems have been in use in existing plants for many years without any vibration problems, they do not need to be included in the startup testing. In addition, as discussed earlier in the evaluation of the response to **RAI 498-3782, Question 03.09.02-64**, the applicant added this information in the revised DCD. Accordingly, **RAI 206-1576, Question 03.09.02-20, is resolved**.

MUAP-07027-P, Revision 0, provides details about the types and locations of the transducers that will be used in the preoperational vibration test program, the test conditions, and inspection program. In total, the reactor internals will be instrumented with 50 transducers consisting of 33 strain gages, 11 accelerometers, four pressure sensors, and two displacement transducers. The report also includes drawings of the locations of these transducers as well as estimates of the vibration responses at the transducers locations. These response estimates are obtained from the vibration analysis (see Table 3.4-8, "Estimated Transducers Responses in US-APWR Reactor Internal Vibration Measurement in Hot Functional Testing," of MUAP-07027-P, Revision 0). Although the overall concept of this test program seems reasonable, it is not clear what provisions are made to ensure that adequate data will be obtained even if several sensors fail during the preoperational test. Therefore, in **RAI 206-1576, Question 03.09.02-21** (identified as RAI 3.9.2-42 in applicant's response), the staff requested the applicant to elaborate on the provisions made in the vibration test program to ensure sufficient redundancy in the instrumentation such that adequate information is obtained from the preoperational and start-up vibration test program.

In its response to **RAI 206-1576, Question 03.09.02-21**, dated March 27, 2009, the applicant provided additional information regarding the provisions needed to ensure sufficient redundancy in the instrumentation. The response addressed various components of the reactor internals, including:

a. Core Barrel/Lower Core Support Plate: To measure the beam mode response of the lower internals, theoretically three sensors are needed: one for each of the two-horizontal directions and a third for the vertical direction. However, a total of four strain gages will be installed to measure the strains due to beam mode vibration of the lower internals. The fourth strain gage is a redundant sensor serving as backup in case one of them fails prematurely. In addition, two additional strain gages will be installed on the inner surface of the core barrel flange to measure the strains caused by local bending.

b. Neutron Reflector/Tie Rod: Since the neutron reflector is a first-of-a-kind design, more data on this component will be acquired to ensure that no unexpected vibrations occur. Taking into consideration the symmetry of this component, the applicant believes that a total of four accelerometers with axes in the radial direction will provide sufficient redundancy to define the vibration characteristics of this component, even if one fails prematurely. In addition, an additional accelerometer will be installed to measure the vertical motion.

c. Upper Plenum Structures: The top slotted column is another first-of-a-kind design in US-APWR. For this component, three strain gages will be installed to measure the vibration responses in the two horizontal directions with the third sensor serving as the redundant back up sensor.

d. Roc Cluster Control (RCC) Guide Tube: Because the RCC guide tube has a safety-related function and is one of the most important subassembly of the reactor internals, three strain gages will be installed on one of the RCC guide tubes to measure the responses in three directions and two strain gages will be installed in the upper guide tube of this same RCC guide tube. For redundancy, two strain gages will be installed on the lower guide tube of another RCC guide tube.

To maintain sufficient redundancy during the measurements of the lower plenum structures, the applicant is planning to measure the strain on two support columns for each of the upper and lower diffuser plate assemblies.

The staff finds the provisions planned by the applicant for the vibration test program to be acceptable. These provisions will ensure sufficient redundancy in the instrumentation such that adequate information is obtained from the preoperational and start-up vibration test program. The staff also reviewed the revised version of MUAP-07027-P, Revision 1 and found that sufficient redundancy in the instrumentation is included in the test program. Accordingly, **RAI 206-1576, Question 03.09.02-21, is resolved**.

A major conclusion based on the results of the vibration assessment program (MUAP-07027-P, Revision 0) is that the vibration responses of the reactor internals without the core are the same or slightly larger than those with the core. Therefore, the applicant proposes to conduct the preoperational and start-up vibration testing (cold hydraulic test and hot functional test) only before loading the core. It is argued that the vibration levels after loading the core will be bounded by those measured without the core. The staff is concerned about the validity of this conclusion. In **RAI 206-1576, Question 03.09.02-22** (identified as RAI 3.9.2-43 in applicant's response), the staff requested the applicant to substantiate the validity of the argument that the dynamic response of the reactor internals under normal and operational flow transient conditions with fuel assemblies in the core is the same or slightly smaller than that under hot functional test conditions without the core. The staff requested the applicant to reassess this conclusion in light of the issues raised in this RAI. In its request the staff noted that the applicant should discuss the effect of existing differences between the scale model geometry (including dynamic similarity) and the prototype on the reactor core vibration with or without the core. In addition, the effect of loading the fuel assembly on flow distribution, pressure drop, and pump characteristics should also be clarified. Finally, preoperational vibration testing of the fuel assemblies would not be possible without loading the reactor core.

In its response to **RAI 206-1576, Question 03.09.02-22**, dated March 27, 2009, the applicant responded to the six issues (a to f) which are raised by the RAI. The applicant's response to each issue is evaluated separately.

(a) Comparison of the Reactor - Current- 4-loop/J-APWR/US-APWR:

The applicant presented a table comparing the key (or global) specifications of the reactor for the current 4-loop, J-APWR, and US-APWR. Key specifications include the RV and core barrel dimensions, number of fuel assemblies, and flow rate. The applicant then concludes that the FIV response characteristics of the US-APWR reactor internals are equivalent to those of the J-APWR.

The staff was still concerned about the effect of other differences in the design details as expressed in **RAI 498-3782, Question 03.09.02-65.** As explained in the evaluation of the response to **RAI 498-3782, Question 03.09.02-65**, the applicant's response illustrated that the effects of these differences are conservative, and therefore, the staff's concerns are resolved.

(b) Comparison of dimensionless parameters between J-APWR SMT and US-APWR plant:

The applicant provided a comparison of the dimensionless parameters related to FIVs, such as the Reynolds number, the reduced velocity, and the Strouhal number for the J-APWR SMT, J-APWR plant, and US-APWR plant.

The staff reviewed these parameters and agrees that they are reasonable and acceptably close to each other for the three designs.

(c) Effect of fuel assemblies on the flow conditions inside the reactor, including the lower and upper plenums. In response to this issue, the applicant stated that for the following reasons, the fuel assembly has little effect on RV flow conditions, including cross flow velocities in the lower and upper plenums.

i. The maximum cross-flow distribution in the upper plenum depends on the outlet nozzle flow velocity and geometries of structures near the outlet. It does not depend on the core outlet flow distribution into the upper plenum. And because of a little bit increase of total flow rate with lower pressure loss in the core, the maximum cross flow velocity in the upper plenum in the hot functional test without core will be higher than the normal operating condition.

ii. The maximum cross flow distribution in the lower plenum depends on the flow velocity in the downcomer and geometries of structures in the peripheral region of the lower plenum. It does not depend on the core inlet flow distribution in the downstream side. And because of the increase of total flow rate with lower pressure loss in the core, the maximum cross flow velocity in the lower plenum during the hot functional test without the core will be higher than the normal operating condition.

The applicant therefore concluded that the mechanical integrity of structures subjected to cross-flow in the lower and upper plenums can be verified without fuel assemblies.

The staff agrees with the conclusion that the cross-flow velocity components in the lower and upper plenums are similar or higher without the core than those with the core in place.

(d) Excitation by leakage flow in the nozzle gap between the core barrel and RV: In response to this issue, the applicant stated that the bypass flow rate from the outlet nozzle gap between the core barrel/RV during normal operation is not larger than that in the pre-operational testing because the gap clearance is designed to be minimum in the normal operating condition considering the core barrel thermal expansion. In any case, the bypass flow between the outlet nozzle gap has little effect on the core barrel vibration because both the flow rate and the surface area contact to the flow are much smaller than those of the downcomer flow.

The above response indicates that the gap will be smaller because of thermal expansion, but the pressure drop with the core will be higher. Therefore, the staff's concerns regarding the potential of leakage flow vibrations have not been resolved. These concerns have been also expressed in **RAI 374-2446, Question 03.09.05-25,** discussed in Section 3.9.5 of this report. In DCD Tier 2, Subsection 3.9.5.3.2, "Thermal-Hydraulics Design-Basis," the applicant stated that some percentage of the main coolant flow is bypass flow, which is either for cooling metal or leakage between gaps. However, the applicant did not assess the liability of the core barrel flange to leakage FIV. Therefore, in **RAI 374-2446, Question 03.09.05-25**, the staff requested the applicant to discuss the liability of the core barrel flange to FIV caused by the leakage (or bypass) flow between the outlet nozzle of the core barrel flange and the RV exit nozzle. Since the diameter of the core barrel flange is larger than that of the 4-loop reactors, its shell modes have lower frequencies. In addition, the leakage flow rate is higher in the US-APWR than in the 4-loop reactors. The applicant was requested to provide evidence showing that the leakage flow between the outlet nozzle of the core barrel flange and the RV exit nozzle will not cause excessive vibration of the core barrel flange.

In its response to **RAI 374-2446, Question 03.09.05-25**, dated June 19, 2009, the applicant stated that there has been no reported evidence of nozzle gap by-pass flow being a major contributor to the core barrel vibration response through the experience of previous plants operation or testing. The bypass flow from the outlet nozzle gap between the core barrel and the RV has little effect on the core barrel vibration because the flow rate and the flow contact area of the gap are much smaller than those of the downcomer as discussed in the response to **RAI 206-1576, Question 03.09.02-22** and Appendix-A of MUAP-07027-P, Revision 1.

In its response to **RAI 374-2446, Question 03.09.05-25**, the applicant argued that the US-APWR is not expected to experience leakage FIV because such vibration has not been experienced by other in-service reactors. However, the in-service reactors are smaller in size, operating at lower flow rates, and experience smaller pressure drops than the US-APWR. This argument is therefore unacceptable. Therefore, the staff closed as unresolved **RAI 206-1576, Question 03.09.02-22**, and in follow-up **RAI 646-5065, Question 03.09.02-92**, the staff requested the applicant to provide evidence or a basis for stating that leakage flow vibration is not a concern in the US-APWR.

In its response to **RAI 646-5065, Question 03.09.02-92**, dated October 14, 2011, the applicant provided a more detailed response assessing possible excitation mechanisms of the core barrel by leakage flow at the exit nozzles. Both turbulence excitation and leakage flow instability are discussed. Regarding the former, the applicant assessed the turbulent fluid force caused by leakage flow and estimated it to be less than 3.5 percent of the turbulent force generated by flow impingement from the inlet nozzles on the core barrel. Thus, this excitation source does not raise any concern. The applicant also addressed the potential of leakage flow instability by means of two different approaches. In the first, a theoretical model is sought and the pressure pulsations resulting from the core barrel vibrations are found to be stabilizing. In the second approach, the design parameters of the US-APWR are compared with those of existing 4-loop PWR. Although the core barrel of the US-APWR is larger, its wall is thicker and therefore its resonance frequency is higher than that of the 4-loop PWR. As a result, the reduced velocity of the leakage flow in the US-APWR, which is a key parameter for FIV, remains comparable to (or slightly lower than) that of the 4-loop PWR, which did not experience any leakage flow vibration.

The staff finds the applicant response acceptable because it shows by analytical means and comparisons with existing plants data that leakage flow vibrations of the core barrel is not likely to occur. Accordingly, **RAI 646-5065, Question 03.09.02-92, is resolved**.

(e) RCP operating conditions: In response to this issue, the applicant stated that the difference of operating point on the RCP Q-H curve (pump flow rate versus pump head curve) has been considered in the estimation of test flow rate from that for normal operating conditions with the fuel loaded.

The staff finds this response reasonable because it clarifies how the changes in the RCP operating conditions are accounted for.

(f) The need to measure the vibrations of fuel assemblies: In response to this issue, the applicant stated that there is little need for fuel assembly vibration measurement in pre-operational or start-up testing because of following reasons:

i. FIV response of the fuel assembly will be confirmed in a full size mock-up testing.

ii. Vibration of the fuel assemblies in the core can be checked by the Fast Fourier Transform analysis of the ex-core nuclear instrumentation signals in the startup testing, if needed.

The staff's concern about this issue is resolved because the fuel assembly dynamic response will be confirmed by means of a full-size mock-up testing.

In summary, the staff found the responses to issues (a), (b), (c), (e), and (f) acceptable. The response to issue (d) resulted in follow-up **RAI 646-5065, Question 03.09.02-92** and consequently **RAI 206-1576, Question 03.09.02-22** was closed as unresolved.

3.9.2.4.4.4 Conclusions for Preoperational Flow-Induced Vibration Testing of Reactor Internals

The staff concludes that the applicant shall meet the relevant requirements of GDC 1 and 4 with regard to the internals of a prototype reactor being tested to quality standards commensurate with the importance of the safety functions being performed and being appropriately protected against dynamic effects. The staff also concludes that by meeting the guidance of a prototype test as specified in RG 1.20 and by having a preoperational vibration measurement program planned to confirm that unexpected, abnormal vibrations do not occur, the pre-operational FIV testing program ensures that the vibration responses of the reactor internals are sufficiently small compared to an acceptance criterion based on the design fatigue curves in the ASME Code, Section III. The combination of preoperational testing program, analysis of test results, and post-test inspection program and acceptance criteria provides adequate assurance that the reactor internals will, during their service life, withstand the FIVs of the reactor without loss of structural integrity.

3.9.2.4.5 Dynamic System Analysis of the Reactor Internals under Faulted Conditions

3.9.2.4.5.1 Summary of Application for Dynamic System Analysis of the Reactor Internals under Faulted Conditions

DCD Tier 1/ITAAC: There is no Tier 1 information associated with this section.

DCD Tier 2: The applicant has provided a DCD Tier 2 system description in Section 3.9.2.5, "Dynamic System Analysis of the Reactor Internals under Faulted Conditions," summarized here in part, as follows. DCD Tier 2, Section 3.9.2.5, describes the dynamic system analyses performed to confirm the structural design adequacy of the reactor internals and unbroken loops of the reactor coolant piping under faulted conditions. The pipe rupture analysis methodology is similar to the seismic analysis methodology (i.e., a dynamic computer model is used to determine the maximum accelerations, displacements, and loadings); and the static computer models for the reactor internals are used to determine the component stresses and displacements. The details regarding the mathematical models for seismic and pipe rupture analyses, the methods of analyses, description of the forcing functions, and the acceptance criteria are presented.

The results of the pipe rupture analysis are time-history accelerations, displacement (absolute and relative), and loadings (forces and moments), which are input into the static models for reactor internals to determine the component maximum stress intensities and displacements. To assure structural and functional integrity of the reactor internals under faulted conditions, the analytical results are required to meet the stress limits of the ASME Code, Section III Subsection NG for core support structures and the functional requirements listed in DCD Tier 2, Subsection 3.9.2.5.3, "Structural Design Adequacy Criteria for Level D Combined Loadings."

3.9.2.4.5.2 Regulatory Basis for Dynamic System Analysis of the Reactor Internals under Faulted Conditions

The relevant requirements of the Commission's regulations for this area of review, and the associated acceptance criteria are given in SRP Section 3.9.2, SRP Acceptance Criterion 5 and are summarized below.

1. GDC 2, as it relates to systems, components, and equipment important to safety being designed to withstand appropriate combinations of the effects of normal and accident conditions with the effects of earthquake without loss of capability to perform their safety functions.

2. GDC 4, as it relates to systems and components important to safety being appropriately protected against the dynamic effects of LOCAs.

Acceptance criteria adequate to meet the above requirements include:

1. For requirements of GDCs 2, 4, 14, and 15 dynamic system analyses should confirm the structural design adequacy of the reactor internals and the reactor coolant piping (unbroken loops) to withstand the dynamic loadings of the most severe LOCA in combination with the SSE. Where a substantial separation between the forcing frequencies of the LOCA (or SSE) loading and the natural frequencies of the internal structures can be demonstrated, the analysis may treat the loadings statically.

3.9.2.4.5.3 Technical Evaluation of Dynamic System Analysis of the Reactor Internals under Faulted Conditions

The dynamic system analysis was reviewed to determine if the applicant has provided adequate information to satisfy the guidance of SRP Section 3.9.2 and relevant requirements of GDC 2 and 4, as well as the applicable portions of the other Regulatory Basis stated in Subsection 3.9.2.4.5.2 above. SRP Section 3.9.2 states that dynamic system analyses should confirm the structural design adequacy and ability, with no loss of function, of the reactor internals and unbroken loops of the reactor coolant piping to withstand the loads from a LOCA in combination with a SSE. The staff's review covered the methods of analysis, the considerations in defining the mathematical models, the descriptions of the forcing functions, the calculation scheme, the acceptance criteria, and the interpretation of analytical results.

DCD Tier 2, Revision 0, Section 3.9.2.5, presents a detailed discussion of the dynamic system analyses of reactor internals subjected to combined seismic and postulated pipe rupture events under ASME Code, Section III Level D (faulted) service conditions. To assure structural and functional integrity of the reactor internals under faulted conditions, the analytical results are required to meet the stress limits of the ASME Code, Section III, Subsection NG, for CSSs and the functional requirements of the reactor internals design specification.

DCD Tier 2, Subsection 3.9.2.5.1, “Seismic Analysis Methodology and Acceptance Criteria,” describes the seismic analysis methodology and acceptance criteria. Two mathematical models are used for the seismic analysis: a three-dimensional non-linear dynamic FE computer model of the core internals and the support system to determine the maximum accelerations, displacements, and loadings, and a three-dimensional static FE computer model that uses the results from the first model to determine the maximum seismic stress intensities and displacements. The applicant stated in DCD Tier 2, Subsection 3.9.2.5.1, that the mathematical model for dynamic system analysis includes representation of the RV support system, inlet and outlet piping nozzles, CRDM system, integrated head support system, in-core instrumentation support system, and fuel assembly nozzles and grids. The applicant further stated that the fluid-structure interaction effects were accounted for by matrices developed for that purpose. The staff reviewed DCD Tier 2, Section 3.9.2.5 and found that the applicant did not provide sufficient details. SRP Section 3.9.2 states that mathematical model used for dynamic system analysis for LOCA in combination with SSE effects should include fluid-structure effects when applicable. Also, typical diagrams and modeling basis should be described. In **RAI 207-1577, Question 03.09.02-23** (identified as RAI 3.9.2-50 in applicant’s response), the staff requested the applicant to provide the details to explain how the fluid-structure effects are accounted for in the modeling of the reactor internals and dynamically related piping, pipe supports, and components.

In its response to **RAI 207-1577, Question 03.09.02-23**, dated March 27, 2009, the applicant stated that both seismic and LOCA dynamic analysis models are three dimensional, non-linear FE models representing the RV and its internals in six degrees of freedom. The modeling is described in the applicant’s Technical Report MUAP-09002, “Summary of Seismic and Accident Load Conditions for Primary Components and Piping,” Revision 0, issued January 2009.

For the seismic dynamic analysis model, the hydrodynamic mass matrices for following three locations are included:

(1) Between the RV and the core barrel in two horizontal directions.

(2) Between the core barrel and the neutron reflector in two horizontal directions.

(3) Between the upper core support and the RV head in vertical direction.

The hydrodynamic mass matrices calculated for locations (1) and (2) were determined from the three dimensional solid-fluid FE analyses that are described in MUAP-07027-P, Revision 0. And the matrix for location (3) was derived from a hand calculation. For the-LOCA dynamic analysis model, the hydrodynamic mass matrices for location (1) are excluded because the hydrodynamic mass effect in this location is accounted for in the blowdown analysis with MULTIFLEX.

The staff finds the applicant’s response acceptable because the applicant has provided sufficient information, including the information in MUAP-09002, Revision 0, describing how the fluid-structure interaction effects are accounted for by using the hydrodynamic mass matrices in the modeling of the reactor internals. Also, the staff confirmed that the details regarding the fluid-structure effects have been incorporated in DCD Tier 2, Revision 2. Accordingly, **RAI 207-1577, Question 03.09.02-23, is resolved**.

DCD Tier 2, Subsection 3.9.2.5.1 states that the pipe rupture analysis methodology is similar to the seismic analysis methodology. The reactor internals are represented in the model by beam elements; and the connectivity of the reactor internals and interfacing structures is represented by mass inertia effect, stiffness and hydrodynamic matrices, springs, and/or impact elements, including gap and damping. Dominant frequencies are identified by comparing the frequency response of the reactor internals with the response based on experience and measurements. Based on its review of DCD Tier 2, Section 3.9.2.5, the staff indicated that the applicant did not provide any discussion regarding system structural partitioning and directional decoupling employed in the model. SRP Section 3.9.2 states that the mathematical model used for dynamic system analysis of reactor internals should include a justification regarding any system structural partitioning and directional decoupling employed in the model. Therefore, in **RAI 207-1577, Question 03.09.02-24** (identified as RAI 3.9.2-51 in applicant's response), the staff requested the applicant to provide this information.

In its response to **RAI 207-1577, Question 03.09.02-24**, dated March 27, 2009, the applicant stated that the nodal point degrees of freedom, and damping coefficients of the reactor internals and surrounding structures are selected such that most dominant frequencies are represented in the seismic-LOCA response. This forms the basis for establishing any directional decoupling and system structural partitioning in the seismic-LOCA model.

Both the seismic and LOCA dynamic analyses models consist of beam elements, linear and nonlinear springs, gaps, hydrodynamic mass matrices, and stiffness matrices. The only shell element modeled is for the diffuser plates. The main structures are modeled by the beam elements. The structural interfaces are modeled by the spring, gap, and impact elements.

The RCS loops are not explicitly modeled in the seismic and LOCA dynamic models; however, they are simulated by stiffness matrices connected to the RV nozzle center location. The fluid-structural effects are accounted for between the RV and core barrel, core barrel and neutron reflector, upper core support and RV by hydrodynamic mass matrices. The effects of friction between the core barrel flange and RV, the upper core support flange and RV are accounted for by the friction elements. The shell elements are used for modeling the diffuser plates. The core region is divided to five regions, one center region and four outer regions. The beam elements are used for representing the upper core support, upper core plate and lower core support plate vertical vibration and out of plane stiffness.

The time history seismic accelerations developed from the RCL seismic analysis are applied to the RV supports and integrated head package locations. The LOCA time history forcing functions from the MULTIFLEX analyses are applied to the RV and core barrel in the horizontal direction and the vertical forces are applied to the RV, upper core support, core barrel, lower core support plate, upper core plate and fuel assemblies.

Based on the information provided by the applicant regarding the seismic and LOCA dynamic models, the staff finds that the applicant has provided adequate details and justification for establishing any directional decoupling and system structural partitioning in the dynamic system modeling of the reactor internals and the RPV. In addition, the staff confirmed that as stated in applicant's response discussed above, these details have been incorporated in DCD Tier 2, Revision 2. Accordingly, **RAI 207-1577, Question 03.09.02-24, is resolved**.

The reactor internals seismic input for the models can be from either in-structure response spectra or in-structure time-history accelerations obtained from the seismic system analysis described in detail in DCD Tier 2, Section 3.7.2 and the results presented in DCD Tier 2, Appendix 3H, "Model Properties and Seismic Analysis Results for Lump Mass Stick Models of R/B-PCCV-Containment Interior Structures on a Common Basemat, and PS/Bs on Individual Basemat." The present analysis employs the design response spectra based on modified input from RG 1.60, "Design Response Spectra for Seismic Design of Nuclear Power Plants," Revision 1, issued December 1973, to determine the effects of vibratory motion for SSE and 1/3 SSE seismic conditions. Additional input to the seismic analysis includes vertical pressure loadings converted to nodal point external loads, and the vertical weights of the reactor internals and interfacing components, which is used as input to density on the beams with spring effects or mass nodal points. The static computer models for reactor internals are used to determine their component stresses and displacements.

DCD Tier 2, Revision 0, Subsection 3.9.2.5.2, "Pipe Rupture Analysis Methodology and Acceptance Criteria," states that the mathematical model for dynamic systems analysis includes such structural characteristics as the flexibility, mass inertia effect, geometric configuration, and damping (including possible coexistence of viscous and Coulomb damping). However, the staff found that the applicant did not include any justification that the model is representative of the system structural characteristics, or provide a reference document where such information is available. SRP Section 3.9.2 states that the mathematical model used for dynamic system analysis of reactor internals under faulted conditions should typify such system structural characteristics as flexibility, mass inertia effect, geometric configuration, and damping. In **RAI 207-1577, Question 03.09.02-25** (identified as RAI 3.9.2-52 in applicant's response), the staff requested the applicant to provide this information.

In its response to **RAI 207-1577, Question 03.09.02-25**, dated March 27, 2009, the applicant stated that the dynamic system structural characteristics are discussed in MUAP-09002, Revision 0. Validation that the reactor internals dynamic models are representative is made by the comparison of a simulation analysis of 1/5 scale model test and the test results as discussed in MUAP-07027-P, Revision 0.

The staff finds the applicant's response acceptable because the applicant used a scale model test to justify that the dynamic reactor internal models are representative of system structural characteristics and the applicant has included in DCD Tier 2, Revision 2, the two reference documents where this information is available. Accordingly, **RAI 207-1577, Question 03.09.02-25, is resolved**.

In DCD Tier 2, Subsection 3.9.2.5.2, the applicant also stated that the effects of flow upon the mass and flexibility properties of the system are accounted for in the mathematical model for dynamic system analysis. The staff found that the applicant did not provide sufficient details regarding the mathematical model. SRP Section 3.9.2 states that the mathematical model used for dynamic system analysis for LOCA in combination with SSE effects should address the effects of flow upon the lumped-mass and flexibility properties of the system. Therefore, in **RAI 207-1577, Question 03.09.02-26** (identified as RAI 3.9.2-53 in applicant's response) the staff requested the applicant to provide this information.

In its response to **RAI 207-1577, Question 03.09.02-26**, dated March 27, 2009, the applicant stated that the effects of flow upon both the lumped mass and flexibility properties in the LOCA dynamic system model are accounted for in the model because the MULTIFLEX results used as input to the LOCA dynamic system model included fluid-structural interaction. The applicant referred response to **RAI 207-1577, Questions 03.09.02-23 and 03.09.02-24** for additional discussion of the modeling. The hydraulic forces acting on the reactor internal structures during normal operation are accounted for in the seismic dynamic analysis by applying the loads as steady-state uniform loadings.

The staff finds response related to the effects of flow upon the lumped-mass and flexibility properties of the system acceptable because it meets the SRP criteria. Also, the requested information has been incorporated in DCD Tier 2, Revision 2. Accordingly, **RAI 207-1577, Question 03.09.02-26, is resolved**.

On the basis of its review of DCD Tier 2, Subsection 3.9.2.5.2, the staff found that the applicant did not include a discussion to assure that there is no significant dynamic amplification of the load on reactor internals as a result of the oscillatory nature of the blow-down forces during a postulated LOCA. SRP Section 3.9.2 states that evaluation of the dynamic effects on reactor internals associated with postulated pipe rupture should include a description of the methods and procedures for dynamic system analyses, including the governing equations of motion and the computational scheme for deriving results. Therefore, in **RAI 207-1577, Question 03.09.02-29** (identified as RAI 3.9.2-56 in applicant's response), the staff requested the applicant to provide the analytical results to demonstrate that there is no significant amplification of the loads on reactor internals and core support structures as a result of postulated pipe rupture.

In its response to **RAI 207-1577, Question 03.09.02-29**, dated March 27, 2009, the applicant stated that the methods and procedures used for the seismic and LOCA dynamic system analyses including the governing equations of motion and computational scheme for deriving results are provided in MUAP-09002, Revision 0. The dynamic system analysis results for LOCA in combination with SSE are also provided in MUAP-09002, Revision 0. These dynamic output parameters are used into the detailed structures component static FE model and the maximum stress intensities are calculated. The results are compared to the ASME Code, Section III Level D service limits.

The staff finds the applicant's response acceptable because the applicant has provided sufficient information, including the information in Section 6 of MUAP-09002, Revision 0, describing how the dynamic amplification of blow-down forces on reactor internals is accounted for by using the time history analysis with the direct integration method in the LOCA response analysis. Furthermore, in DCD Tier 2, Revision 2, Subsection 3.9.2.5.2, the applicant added a statement that Section 6 of MUAP-09002, Revision 0 provides the details regarding the methods of LOCA dynamic system analysis. Accordingly, **RAI 207-1577, Question 03.09.02-29, is resolved**.

DCD Tier 2, Subsections 3.9.2.5.2 and 3.9.2.5.3 state that the outputs of the LOCA response analysis are time-history accelerations, displacement (absolute and relative), and loadings (forces and moments). The maximum loadings and displacements are input into the reactor internals static models to determine the component maximum stress intensities and displacements. The combined effect of seismic and postulated pipe rupture events is determined by combining the maximum stresses and displacements for each condition with the SRSS rule. The applicant stated that the maximum stress intensities and displacements are compared with the ASME Code, Section III, Subsection NG, stress limits, and the allowable interface load and displacement limits given in DCD Tier 2, Table 3.9-2, "Reactor Internals Interface Load and Displacement Limits." The applicant further stated that the LOCA dynamic system analyses results confirm that the structural design adequacy of the reactor internals can withstand the dynamic loadings of the most severe LOCA in combination with the SSE. Based on its review of the DCD the staff found that the applicant had not provided any details regarding the dynamic systems analyses. SRP Section 3.9.2 states that the dynamic system analyses should confirm the design adequacy of the reactor internals, i.e., their ability to withstand the dynamic loadings of the most severe LOCA in combination with SSE. In **RAI 207-1577, Question 03.09.02-30** (identified as RAI 3.9.2-57 in applicant's response) the staff requested the applicant to identify the locations in the reactor internals where the stress deformation and fatigue are determined to be the highest. Also, the staff requested the applicant was asked to identify the corresponding loading combination and to revise the DCD to include the requested information.

In its response to **RAI 207-1577, Question 03.09.02-30**, dated March 27, 2009, the applicant stated that locations that are likely to be sources of high stresses during a ASME Code, Section III Level D event are structures that have to transmit high loads and have limiting minimum thicknesses. Critical core support structure locations identified in MUAP-09002-P, Revision 0, Figures 8-12 and 8-13 include:

- Core Barrel Flange Discontinuity.
- Upper/Lower Core Barrel Discontinuity.
- Lower Core Barrel/Lower Core Support Plate Discontinuity.
- Radial Support Key.
- Lower Core Support Plate.
- Upper Core Support.
- Upper Core Support Flange/Skirt Discontinuity.
- Top Slotted Column Extension.
- Top Slotted Column.
- Top Slotted Column Fastener.
- Upper Core Support Column Extension.
- Upper Core Support Column.
- Upper Core Support Column Fastener.

The critical location of the deformation is identified in DCD Tier 2, Table 3.9-2.

Based on a review of MUAP-09002, Revision 0, and the information provided by the applicant regarding locations in the reactor internals where the stress deformation and fatigue are determined to be the highest, the staff finds that the applicant has provided adequate information to resolve the staff's concerns. Also, in DCD Tier 2, Revision 2, Subsection 3.9.2.5.3, the applicant provided the reference which identifies the locations in the reactor internals where the stress deformation and fatigue are determined to be the highest. Accordingly, **RAI 207-1577, Question 03.09.02-30 is resolved**.

DCD Tier 2, Subsections 3.9.2.5.2 and 3.9.2.5.3 states that the maximum stress intensities and displacements obtained from the LOCA dynamic system analyses are compared with the ASME Code, Section III, stress limits and with the allowable interface load and displacement limits given in DCD Tier 2, Table 3.9-2. In addition, the functional requirements that need to be met include the following:

(a) Allowable horizontal load of the guide tube should not impede insertion of the control rod after a LOCA.

(b) Displacement of the upper core barrel is not to impede the downcomer emergency core cooling (ECC) flow after a LOCA.

(c) Reaction loads at the RV connections are not to exceed allowable values of the interface load.

(d) Maximum vertical displacement of the upper core plate relative to the upper support plate should preclude buckling of the guide tube.

(e) Permanent displacement of the upper core barrel should not prevent loss of function of the control rod assembly by radial inward deformation of the upper guide tube.

The staff reviewed the DCD and found that the applicant did not address the stability of the core barrel in compression. SRP Section 3.9.2 states that the dynamic system analyses of the reactor internals under pipe rupture loadings should be employed to investigate the stability of the elements in compression, such as the core barrel and control rod guide tubes. In **RAI 207-1577, Question 03.09.02-32** (identified as RAI 3.9.2-58 in applicant's response), the staff requested the applicant to describe how the stability of the core barrel in compression was investigated under pipe rupture loadings.

In its response to **RAI 207-1577, Question 03.09.02-32**, dated March 27, 2009, the applicant stated that the stability of the core barrel and control rod guide tubes during a pipe rupture are evaluated with the pressure loadings from the blow-down analysis using MULTIFLEX code. The design limit for the evaluation is determined in according to ASME Code, Section III, Division 1, Nonmandatory Appendix F, Article F-1000, "Rules for Evaluation of Service Loadings with Level D Service Limits," F-1331.5, "Requirements for Compressive Loads."

The staff finds the response acceptable because based on a review of the information provided by the applicant adequately valuated the stability of the elements in compression, such as the core barrel and control rod guide tubes, during a pipe rupture event using the ASME Code Section II, Division 1, Appendix F. Also, stability of core barrel and guide tube during a pipe break was added to the list of functional requirements in DCD Tier 2, Revision 2, Subsection 3.9.2.5.3. Accordingly, **RAI 207-1577, Question 03.09.02-32, is resolved**.

The applicant stated in DCD Tier 2, Revision 0, Subsection 3.9.2.5.3, that the pipe break sizes of current 4-loop plants were based on the largest LOCA loads that resulted from either a 0.093 m^2 (1.0 ft^2) single-ended cold leg break or a double-ended hot leg break, whereas the LBB criteria are applied to determine the break condition for the US-APWR design input. The magnitude of blow-down hydraulic loads applying LBB is smaller than either the loads for the large cold leg or hot leg breaks. Thus, maximum stresses and displacements of the reactor internals under faulted conditions meet the ASME Code, Section III, Subsection NG, stress and deflection limits. On the basis of its review, the staff found that the DCD did not provide sufficient details. In **RAI 207-1577, Question 03.09.02-31** (identified as RAI 3.9.2-59 in applicant's response), the staff requested the applicant to:

(a) Confirm that to eliminate the dynamic effects of pipe rupture from the design-basis, LBB evaluation was performed in accordance with SRP Section 3.6.3, "Leak-Before-Break Evaluation Procedures," Revision 1, issued March 2007, to demonstrate that the probability of pipe rupture is extremely low for the applied loading resulting from normal conditions, anticipated transients, and postulated SSE.

(b) Identify the piping systems that were included in the evaluations.

(c) State what were the nominal pipe diameter and postulated pipe break flow area for the limiting design-basis pipe size used to determine the pipe rupture dynamic response.

In its response to **RAI 207-1577, Question 03.09.02-31**, dated March 27, 2009, the applicant identified the postulated pipe ruptures and provided the results of the dynamic analysis in MUAP-09002-P, Revision 0. MUAP-09002-P, Revision 0, Section 4.1 provides the cases of postulated pipe ruptures as discussed below.

The applicant applied LBB criteria for main coolant piping and MS Line. As a result of the LBB evaluation, the following postulated pipe break events have been selected for the RCS components.

1. Hot leg branch line break at the 10 in. (25 cm), Schedule 160, RHR/SI line nozzle, nominal pipe diameter: NPS 10 in. (25 cm),

2. Cold leg branch line break at the 14 in. (36 cm), Schedule 160 accumulator line nozzle, nominal pipe diameter: NPS 14 in. (36 cm),

3. Feedwater line break at the SG feedwater nozzle, nominal pipe diameter: NPS 16 in. (41 cm),

4. MS line break at the SG MS nozzle, nominal pipe diameter: NPS 32 in. (81 cm).

Based on a review of the information provided by the applicant, including the information in MUAP-09002, Revision 0, Section 4.1, the staff finds the applicant has provided sufficient information regarding the LBB evaluation. The staff's evaluation of LBB analyses is presented in Section 3.6.3 of this report. Accordingly, **RAI 207-1577, Question 03.09.02-31,** is resolved.

3.9.2.4.5.4 Conclusions for Dynamic System Analysis of the Reactor Internals under Faulted Conditions

Based on the staff's evaluation above, the staff concludes that the dynamic system analyses have been performed to confirm that the structural design of the reactor internals is able to withstand the dynamic loadings of the most severe LOCA in combination with the SSE, with no loss of function. The staff also concludes that the methods and procedures for dynamic systems analyses, the considerations in defining the mathematical models, the descriptions of the forcing functions and the acceptance criteria, and the interpretation of the analytical results are in conformance with the relevant requirements of GDC 2 and 4.

3.9.2.4.6 Correlations of Reactor Internals Vibration Tests with the Analytical Results

3.9.2.4.6.1 Summary of Application for Correlations of Reactor Internals Vibration Tests with the Analytical Results

DCD Tier 1/ITAAC: The Tier 1 information associated with this section is found in DCD Tier 1, Section 2.4.1, which lists ITAAC for reactor internals FIV test.

DCD Tier 2: The applicant has provided a DCD Tier 2 system description in Section 3.9.2.6, "Correlations of Reactor Internals Vibration Tests with the Analytical Results," summarized here in part, as follows. To confirm the computational methodology used to assess the dynamic responses of the reactor internals, the applicant states that a simulation of the 1/5-scale model has been performed, and the computed vibration response of the core barrel, with the best estimate of damping coefficient, agreed with the measured response from the scale model. The applicant also commits to performing a comparison between the results of the preoperational vibration test of the first US-APWR with the predictions of the vibration analysis. It is also stated that any discrepancies between the predicted and measured values will be accounted for and fully explained, if necessary, by adjusting the input parameters, such as the forcing functions or damping coefficients.

3.9.2.4.6.2 Regulatory Basis for Correlations of Reactor Internals Vibration Tests with the Analytical Results

The relevant requirements of the Commission's regulations for this area of review, and the associated acceptance criteria are given in SRP Section 3.9.2, SRP Acceptance Criterion 6 and are summarized below.

1. GDC 1, as it relates to structures and components being designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with the importance of the safety function to be performed.

3.9.2.4.6.3 Technical Evaluation Correlations of Reactor Internals Vibration Tests with the Analytical Results

In conformance with the recommendations of RG 1.20, the applicant is committed to performing correlations between the vibration tests and the analytical results and also to resolving any discrepancies that may arise. However, several issues were identified, some of which have been already discussed in the evaluation of DCD Tier 2, Section 3.9.2.3 (Section 3.9.2.4.3 of this report, and others are addressed in the Section 3.9.5 of this report. For example, the staff concerns about differences between the scale model and the prototype and about the validation of the structural, acoustic, and forcing functions models have been addressed in Section 3.9.2.4.3 of this report and in the discussion of **RAI 272-1585, Questions 03.09.02-44 to 03.09.02-48** in that section. In addition, Section 3.9.5 of this report discusses other concerns dealing with assumed values of damping coefficients and the bias errors and uncertainties associated with the vibration analysis predictions.

Since the preoperational vibration tests will employ strain gages, pressure sensors, displacement transducers, and accelerometers, more rigorous acceptance criteria are needed than those applied when comparing the model test results with the analysis predictions (see discussion of **RAI 272-1585, Question 03.09.02-55** above). In this regard, in **RAI 208-1574, Question 03.09.02-33**, the staff requested the applicant to describe the acceptance criteria that will be used when comparing the results of the preoperational vibration tests with the analytical results. These may include acceptance criteria for load definitions (e.g., pressure measurement results), dynamic characteristics of the system (e.g., resonance frequencies, mode shapes, and vibration response), stress analysis calculations (e.g., strain gages results), and acoustic response of the reactor and the SG environment.

In its response to **RAI 208-1574, Question 03.09.02-33**, dated March 27, 2009, the applicant described two categories of acceptance criteria to be used during the preoperational tests. Category 1 criteria are related to the integrity of the components, while Category 2 criteria are related to the adequacy of the analysis technique. The applicant also explained contingency plans in case these criteria are not met. In summary, Category 1 applies to the measured stresses and the occurrence of FIV. It states that the measured stresses must be less than the fatigue limit, and fluid-elastic instability and vortex shedding lock-in are not allowed to occur during the pre-operational tests. On the other hand, Category 2 applies to comparisons between the results of vibration analysis and the pre-operational test results. Category 2 covers modal frequencies, damping ratios, forcing functions, and the adequacy of the dynamic system models. The applicant also commits to the following contingency plans in case the acceptance criteria are not met:

> In case any Category 1 criterion is not met, the overall impact on the design of the reactor internals will be evaluated. The reactor will not be put into operation until it is sure that the design can accommodate the larger than expected vibration responses.
>
> In case any Category 2 criterion is not met, the difference will be resolved by the post-preoperational test analysis.

Additionally the applicant states that the information provided in this response will be included in the revised version of MUAP-07027-P, Section 3.5, "Test Acceptance Criteria." Also, a cross reference to the revised acceptance criteria and contingency plans in Technical Report MUAP-07027-P will be added to DCD Tier 2, Section 3.9.2.

The staff finds the proposed acceptance criteria and contingency plans acceptable, except the Category 2 criterion which deals with the damping ratios. This criterion states that "The measured damping ratios, as determined from the half-power point method, must be within a factor of 2.0 of what are used in the prediction analysis." The staff finds that a measured damping ratio of 50 percent lower than that used in the prediction to be excessively non-conservative and recommends a ratio larger than unity be adapted to maintain sufficient conservatism. Therefore, the staff closed as unresolved **RAI 208-1574, Question 03.09.02-33** and in follow-up **RAI 498-3782, Question 03.09.02-80**, the staff requested the applicant to use damping ratios that are equal or smaller than those determined from measurements to maintain sufficient conservatism in the analysis.

In its response to the **RAI 498-3782, Question 03.09.02-80**, dated February 3, 2010, the applicant explained that for the acceptance criterion, the value for the measured damping ratio must always be higher than the damping ratio used in the analysis. Thus, the ratio of two mentioned in the acceptance criterion means that the measured damping ratio is expected to be at least similar to, but not greater than twice the value used in the dynamic analysis.

The staff finds this response acceptable because the acceptance criteria proposed by the applicant is conservative. Accordingly, **RAI 498-3782, Question 03.09.02-80 is resolved**.

3.9.2.4.6.4 Conclusions for Correlations of Reactor Internals Vibration Tests with the Analytical Results

Since the applicant has adequately responded to all RAI discussed in Subsection 3.9.2.6.3, all the concerns raised by the staff are resolved. Based on the staff's evaluation above, it is concluded that the applicant meets the relevant requirements of GDC 1 with regard to the internals of a prototype reactor being tested to quality standards commensurate with the importance of the safety functions being performed by the proposed program to correlate the test measurements with the analysis results. The staff also concludes that the program provides an acceptable basis for demonstrating the compatibility of the results from tests and analyses, the consistency among mathematical models used for different loadings, and the validity of the interpretation of the test and analysis results.

3.9.2.5 Combined License Information Items

The following is a list of COL item numbers and descriptions from Table 1.8-2, "Compilation of All Combined License Applicant Items for Chapters 1-19," of the DCD applicable to the dynamic testing and analysis of systems, components, and equipment:

Table 3.9.2-2
US-APWR Combined License Information Items

Item No.	Description	Section
COL 3.9(2)	The first COL Applicant is to complete the vibration assessment program, including the vibration test results, consistent with guidance of RG 1.20.Subsequent COL applicants need only provide information in accordance with the applicable portion of Position C.3 of RG 1.20 for non-prototype internals.	3.9.2.3

The staff finds COL Information Item 3.9(2) acceptable since it implements the guidance of RG 1.20.

3.9.2.6 Conclusions

As a result of the open items for **RAI 1013-7031, Question 03.09.02-103** and **RAI 1013-7031, Question 03.09.02-104**, the staff is unable to finalize its conclusions on Section 3.9.2 related to the dynamic testing and analysis of systems, components, and equipment, in accordance with NRC regulations.

3.9.3 ASME Code Class 1, 2 and 3 Components, Component Supports, and Core Support Structures

3.9.3.1 Introduction

This section discusses the requirements for maintaining the structural and pressure boundary integrity of safety-related, pressure-retaining components, core support structures, and component supports, which are designed to the criteria specified in the ASME B&PV Code.

3.9.3.2 Summary of Application

DCD Tier 1/ITAAC: The Tier 1 information associated with this section is found in DCD Tier 1, Section 2.0, "Design Descriptions and ITAAC."

DCD Tier 2: The applicant has provided a DCD Tier 2 system description in Section 3.9.3, "ASME Code Class 1, 2, and 3 Components, Component Supports, and Core Support Structures," summarized here in part, as follows: This section addresses several areas of review including: loading combinations, system operating transients, and stress limits for component design; the design and installation of pressure-relief devices; pump and valve operability assurance; and the design of component supports.

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

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

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

Technical Reports: Technical reports associated with DCD Tier 2, Section 3.9.3 are as follows:

1. MUAP-08005, "Dynamic Analysis of the Coupled RCL-R/B-PCCV-CIS Lumped Mass Stick Model," Revision 0, issued April 2008.

2. MUAP-09002, "Summary of Seismic and Accident Load Conditions for Primary Components and Piping," Revision 0, issued January 2009.

3. MUAP-10006, "Soil-Structure Interaction Analyses and Results for the US-APWR Standard Plant," Revision 3, issued November 2012.

Cross-cutting Requirements (TMI, USI/ GSI, Op Ex): There are no cross-cutting requirements for this area of review.

US-APWR Interface Issues Identified in the DCD: See DCD Tier 2, Table 1.8-2, "Compilation of All Combined License Applicant Items for Chapters 1-19."

Site Interface Issues Identified in the DCD: See DCD Tier 2, Table 1.8-1, "Significant Site Specific Interfaces with the Standard US-APWR Design."

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

3.9.3.3 Regulatory Basis

The relevant Commission regulations for this area of review and the associated acceptance criteria are given in Section 3.9.3, "ASME Code Class 1, 2, and 3 Components, Component Supports, and Core Support Structures," Revision 2, issued March 2007, of NUREG-0800 and are summarized below. Review interfaces with other SRP sections can be found in Section 3.9.3 of NUREG-0800.

1. 10 CFR 50.55a and GDC 1, as they relate to structures and components being designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety function to be performed.

2. GDC 2, and 10 CFR Part 50, Appendix S, as they relate to structures and components important to safety being designed to withstand the effects of earthquakes, without loss of capability to perform their safety function.

3. GDC 4, as it relates to structures and components important to safety being designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant accidents.

4. GDC 14, as it relates to the RCPB being designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure and of gross rupture.

5. GDC 15, as it relates to the RCS and associated auxiliary, control, and protection systems being designed with sufficient margin to assure that the design conditions of the RCPB are not exceeded during normal operating conditions, including anticipated operational occurrences (AOOs).

6. 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 has been constructed 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 requirement include:

1. RG 1.26, "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants," Revision 4, issued March 2007.

2. RG 1.124, "Service Limits and Loading Combinations for Class 1 Linear-Type Supports," Revision 2, issued February 2007.

3. RG 1.130, "Service Limits and Loading Combinations for Class 1 Plate-and-Shell-Type Component Supports," Revision 2, issued March 2007.

4. ASME Code, Section III, Subarticle NCA-3250, which requires that a design specification be prepared for Class 1, 2, and 3 components such as pumps, valves, and piping systems.

5. Subarticle NCA-3260 of ASME Code, which also requires a design report for ASME Code, Class 1, 2, and 3 piping and components that demonstrate that the as-built components satisfy the requirements of the respective ASME design specification for each component and the applicable Subsection of the ASME Code.

3.9.3.4 Technical Evaluation

ASME Code, Section III, Subarticle NCA-3250 requires that a design specification be prepared for Class 1, 2, and 3 components such as pumps, valves, and piping systems. The design specification is intended to become a principal document governing the design and construction of these components and should specify loadings and their combinations; design, service and test limits; and other design data inputs. Subarticle NCA-3260 of the Code also requires a design report for ASME Code, Class 1, 2, and 3 piping and components that demonstrates that the as-built components satisfy the requirements of the respective ASME design specification for each component and the applicable Subsection of the ASME Code.

In DCD Tier 2, Section 3.9.3, the applicant states that the licensee is responsible for developing design specifications and design reports in accordance with the responsibilities outlined under the ASME Code, Section III rules. The staff noted that this requirement is not included as one of the COL information items. Therefore, in **RAI 209-1803, Question 03.09.03-1** the staff requested that the applicant discuss how the above ASME Code, Section III requirements concerning design specifications and design reports for mechanical components, specifically for risk-significant components, will be met.

In its response to **RAI 209-1803, Question 03.09.03-1**, dated March 4, 2011, the applicant stated that the applicant provided the information on the ASME Code, Section III requirements, design specifications, and stress reports for the risk-significant mechanical components in the applicant's letter dated July 21, 2010, "Updated Design Completion Plan for US-APWR Piping Systems and Components." The applicant stated that design completeness will be verified during the reconciliation of the "as-built" plant against pertinent design documents as committed in the system specific ITAAC.

By letter dated May 12, 2011, "Revised Design Completion Plan for US-APWR Piping Systems and Components," the applicant submitted a revised design completion plan for US-APWR PSCs. This letter states that design specifications of all risk-significant ASME Class 1, 2, and 3 components, except valves and orifice, will be modified to incorporate the seismic responses resulting from the FEM and will be available in March 2012. The letter provides general lists of PSCs and schedules for the availability of design specifications for various PSC groups. The letter does not provide specific lists of components requested by the staff.

The staff reviewed the lists of PSCs in DCD Tier 2. DCD Tier 2, Table 17.4-1, "Risk-significant SSCs," lists risk-significant components but does not identify which are categorized as ASME Code, Section III components. The staff reviewed DCD Tier 2, Table 3.2-2, "Classification of Mechanical and Fluid Systems, Components, and Equipment," that identifies which PSC are ASME Code, Section III, but Tables 3.2-2 and 17.4-1 are not directly comparable. These tables are organized differently since they address different review criteria. Therefore, the staff cannot readily identify a specific list of risk-significant mechanical components from the DCD.

Accordingly, the staff closed as unresolved **RAI 209-1803, Question 03.09.03-1**, and in follow-up **RAI 822-5861, Question 03.09.03-26**, the staff requested the applicant to provide a specific list of risk-significant ASME Code, Section III components. In its response to **RAI 822-5861, Question 03.09.03-26**, dated January 26, 2012, the applicant stated that seismic and structural analysis is changing. As a result, the applicant committed to revise and resubmit the PSCs design completion plan. In the revised PSCs design completion plan, the applicant will include a table providing a specific list of risk-significant ASME Division 1, Section III PSCs. The table will be consistent with DCD Tier 2, Table 3.2-2 and Table 17.4-1. The staff finds the response acceptable since the applicant has agreed to provide a list of risk-significant ASME Code, Section III PSCs consistent with DCD Tier 2, Table 3.2-2 and Table 17.4-1. By letter dated December 7, 2012, "Revised Design Completion Plan for US-APWR Piping Systems and Components," the applicant submitted a revised design completion plan for US-APWR PSCs. This letter states that design specifications of all risk-significant ASME Class 1, 2, and 3 components, except valves and orifices, will be modified to incorporate changes from the resolution of seismic issues and will be made available. Subsequently, by letter dated March 1, 2013, "List of Risk-Significant ASME Code, Section III Piping Systems and Components Associated with Revised Design Completion Plan for US-APWR Piping Systems and Components," the applicant provided the completed list of risk-significant ASME Class 1, 2 and 3 components in the PSCs design completion plan that is consistent with DCD Tier 2, Table 3.2-2 and Table 17.4-1. As the staff has confirmed the applicant has completed the actions described in response, **RAI 822-5861, Question 03.09.03-26, is resolved**. To track the need for the applicant make the documents available for audit all risk-significant ASME Code, Section III Class 1, 2, and PSCs, except valves and orifices, consistent with the December 7, 2012, Design Completion Plan, the staff issued **RAI 1015-7054, Question 03.09.03-31.** Pending the staff completing and documenting the audit, **RAI 1015-7054, Question 03.09.03-31 is being tracked as an Open Item**.

3.9.3.4.1 Loading Combinations, System Operating Transients, and Stress Limits

In DCD Tier 2 Section 3.9.3.1, "Loading Combinations, System Operating Transients, and Stress Limits," the applicant establishes the criteria for the selection and definition of design limits and loading combinations associated with normal operation, postulated accident, and specified seismic and other transient events for the design of other safety-related ASME Code, Section III components. In **RAI 209-1803, Question 03.09.03-2**, the staff requested the applicant to clarify what other safety-related components are referenced in the above statement and also to identify the codes and standards to be used for Quality Group (QG) D components (per RG 1.26, "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants," Revision 4, issued March 2007).

In its response to **RAI 209-1803, Question 03.09.03-2**, dated March 4, 2011, the applicant removed the word 'other' from the phrase "for the design of other safety-related ASME Code, Section III components in the first paragraph, first sentence of DCD Tier 2, Section 3.9.3.1. The applicant stated that for QG D components, the applicant will use industry code and standard requirements and will address these requirements in the component's design specification. However, the applicant neither acknowledged the use of ASME Code B31.1 per RG 1.26 nor identified any other industry code. The staff finds the changes to DCD Tier 2, Section 3.9.3.1 acceptable since they clarified that the subsection pertains to all safety-related components; however, the applicant's response did not identify in the DCD the applicable code and standard requirements for QG D components (per RG 1.26 for systems not part of the RCPB but which may contain radioactive materials). Subsequently, the staff reviewed DCD Tier 2, Section 3.2.2.4, "Equipment Class 4," and concluded that the applicable codes and standards to QG D components are tabulated in DCD Tier 2, Section 3.2.2.4 and are consistent with RG 1.26. The staff found that the design information in DCD Tier 2, Section 3.2.2.4 addressed **RAI 209-1803, Question 03.09.03-2**. Accordingly, **RAI 209-1803, Question 03.09.03-2 is resolved.**

For the US-APWR, the safety-related components are designed in accordance with the requirements of the ASME Code, Section III, Subsection NB for Class 1, Subsection NC/ND for Class 2/3, Subsection NF for Class 1, 2, and 3 component supports, and Subsection NG for core support structures depending on their component and service level classifications. The applicant has also defined, in DCD Tier 2, Table 3.9-1, "RCS Design Transients," the ASME service level RCS system operating transients and their expected frequencies over a plant life of 60 years for service Level A (normal), Level B (abnormal), Level C (emergency), and Level D (faulted). The effects of seismic events are also included in the evaluation of cyclic fatigue by defining a 1/3 SSE seismic event as a Level B service condition in DCD Tier 2, Table 3.9-3, "Design Loading Combinations for ASME Code, Section III, Class 1, 2, and 3 CS Systems and Components," which will require fatigue evaluation of both thermal as well as seismic effects. The number of cycles of earthquake loads considered is based on the equivalent of a usage factor where 300 cycles at 1/3 of the peak SSE load equals the same usage factor as 20 cycles at the peak SSE load, consistent with Appendix D of IEEE Standard 344-2004, "IEEE Recommended Practice for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations." The staff noted that the applicant has referenced IEEE Standard 344-

2004 to 10 CFR Part 50, Appendix S (Reference 3.9-33 instead of Reference 3.9-34), which provides seismic design criteria for a single earthquake design at SSE level for nuclear power plants. Also, Note 3 in DCD Tier 2, Section 3.9.3.1.1, “Seismic Load Combinations,” states that in certain cases for non-standard SSCs, the 1/3 SSE may be adjusted higher for plant specific site as justified in a COLA per SECY 93-087, “Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs,” dated April 2, 1993. Therefore, in **RAI 209-1803, Question 03.09.03-3**, the staff requested the applicant to clarify the reference (Reference 3.9-33) for the equivalent SSE fatigue cycles for calculating the usage factor or seismic qualification testing of safety-related Class 1 components and the statement regarding non-standard SSCs in Note 3 in DCD Tier 2, Subsection 3.9.3.1.1.

In its response to **RAI 209-1803, Question 03.09.03-3**, dated April 30, 2009, the applicant corrected the DCD reference for IEEE Standard 344-2004 and noted typographical errors in Note 3 on DCD Tier 2, Subsection 3.9.3.1.1. In addition, the applicant revised Note 3 by removing the reference to SECY 93-087, which does not specifically address adjusting the OBE based on site-specific information. Further, the applicant stated that operating basis earthquake (OBE) values that will be used will be chosen by the COL applicant and may be higher than 1/3 SSE based on the site-specific seismic information for non-standard plant. For the standard US-APWR design, the OBE is 1/3 of the SSE and, therefore, per 10 CFR 50, Appendix S.IV(a)(2)(i), no explicit analysis or design is required. The staff determined that the corrections made addressed the concern and that the clarification on the OBE values defines the seismic requirements for standard plant and non-standard plant (site-specific) designs. Therefore, the staff finds the response acceptable. The staff confirmed that the responses were incorporated into DCD Tier 2, Revision 2. Accordingly, **RAI 209-1803, Question 03.09.03-3, is resolved**.

In DCD Tier 2, Section 3.9.3.1.1, the applicant states that due to the low probability of occurrence of a SSE during operating modes occurring less than 10 percent of plant operation time, the SSE is analyzed in combination with only those operating modes that occur greater than 10 percent of plant operation time. One of the conditions for combining SSE with other transient loads states that it is assumed that a simultaneous LOOP and a single failure of a safety-related system occur as a result of an SSE event. In **RAI 209-1803, Question 03.09.03-4**, the staff requested the applicant to clarify certain aspects of these criteria including:

1. Providing a technical basis for combining SSE with only those operating modes that occur greater than 10 percent of the plant operation time; how the SSE during operational modes occurring less than 10 percent of the plant operation time that correlate to the system operating mode that occurs greater than 10 percent of the time;

2. The meaning of a single failure of a safety-related system in light of the RG 1.53, “Application of the Single-Failure Criterion to Nuclear Power Plant Protection Systems” Revision 2, issued November 2003, definition of a single failure; how the SSE duration is established for the design; and to discuss if there are other loads that will be included in the design.

In its response to **RAI 209-1803, Question 03.09.03-4**, dated April 30, 2009, the applicant stated that the load combinations were in accordance with SRP Section 3.9.3. In addition, the applicant deleted the criteria associated with the 10 percent plant operation time and suggested changes to the DCD Tier 2, Subsection 3.9.3.1.1 accordingly. DCD Tier 2, Subsection 3.9.3.1.1 was changed to be more consistent with RG 1.53. The applicant explained that the SSE duration of 22 seconds satisfies the 20 second minimum total duration per SRP Section 3.7.1, “Seismic Design Parameters.” Finally, the applicant stated that there are no other additional loads other than those identified in the DCD based on past precedence and regulatory guidelines that will be included in the US-APWR design. Since these responses clarified the staff’s concern on the criteria associated with the 10 percent of the plant operation time and the 22 second SSE duration that satisfies the SRP’s minimum 20 second requirement, the staff finds them acceptable. The staff confirmed that the responses were incorporated into DCD Tier 2, Revision 2. Accordingly, **RAI 209-1803, Question 03.09.03-4, is resolved**.

In DCD Tier 2, Table 3.9-3, “Design Loading Combinations for ASME Code, Section III, Class 1, 2, and 3 CS Systems and Components,” the applicant provides the minimum design loading combinations for ASME Code, Section III, Class 1, 2, and 3 and core support SSCs, and in DCD Tier 2, Table 3.9-4, “Design Loading Combinations for Supports for ASME Code, Section III, Class 1, 2, and 3 Components,” the applicant provides the same for ASME Code, Section III, Class 1, 2, and 3 component supports. In **RAI 209-1803, Question 03.09.03-5**, the staff requested the applicant to clarify several statements included in notes below these tables on the load combination criteria.

In its response to **RAI 209-1803, Question 03.09.03-5**, dated April 30, 2009, the applicant provided appropriate revisions to DCD Tier 2 Tables 3.9-3 and 3.9-4 and clarified staff’s concerns relating to defining the loads and their combinations. These revisions included adding hydrostatic load combinations, deleting of Notes 5 and 12 from DCD Tier 2, Table 3.9-3, which did not apply, and correcting the terms in Table 3.9-4. The staff found that the suggested changes to these tables acceptable since they follow the recommendations in SRP Section 3.9.3 and methodology given in NUREG-0484, "Methodology for Combining Dynamic Responses," Revision 1, issued May 1980. The staff confirmed that the responses were incorporated into DCD Tier 2, Revision 3. Accordingly, **RAI 209-1803, Question 03.09.03-5, is resolved**.

In DCD Tier 2, Revision 3, Section 3.9.3, Table 3.9-3, and Table 3.9-4, the applicant has outlined the load combinations for components and component supports associated with ASME service level A, B, C and D. The load combinations of dynamic loads in Level D Service have not clearly demonstrated the methodology used in NUREG-0484. In follow-up **RAI 847-6064, Question 03.09.03-27**, the staff requested the applicant to provide additional clarification and justification of using square root sum of the squares (SRSS) for the third and fourth load combination lines of Level D Service in DCD Tier 2, Tables 3.9-3 and 3.9-4.

In its response to **RAI 847-6064, Question 03.09.03-27**, dated January 24, 2012, the applicant provided the corrections in markups of DCD Tier 2, Table 3.9.-3, Table 3.9-4, Table 3.9-5, “ASME Code, Section III, Class 1, 2, 3, CS, and Support Load Symbols and Definitions,” and Table 3.12-4 to address the staff’s concerns. The staff reviewed the draft revision changes of DCD Tier 2, Revision 3, Table 3.9.-3, Table 3.9-4, Table 3.9-5, and Table 3.12-4 and found the changes to be acceptable. Until the changes are incorporated into the next DCD Tier 2 revision, accordingly, **RAI 847-6064, Question 03.09.03-27 is being tracked as a Confirmatory Item**.

In DCD Tier 2, Section 3.9.3.1.2, “Loads for ASME Code, Section III, Class Components, Core Support, and Component Supports,” the applicant discusses loads for ASME Code, Section III, Class 1, 2, and 3 components, core support structures, and component supports. In **RAI 209-1803, Question 03.09.03-6**, the staff requested that the applicant clarify the descriptions of certain loads including: how the asymmetric blowdown load is characterized and included; how the LBB criteria are applied to all lines; what types of pipe break loads are applied to the RCL piping at the branch connections and how these loads are determined; describe and technically justify the dynamic load factor to be used; and explain statements made on the reactor coolant pump locked rotor and heavy lift loads.

In its response to **RAI 209-1803, Question 03.09.03-6**, dated April 30, 2009, the applicant clarified the description and application of asymmetric blowdown loads consistent with the guidelines given in NUREG-0609, “Asymmetric Blowdown Loads on PWR Primary Systems,” issued January 1981. The applicant also clarified that since the pressurizer relief valves are installed on the piping to the pressurizer and are modeled as such in the US-APWR Class 1 piping analysis, the resulting transient load on the pressurizer relief valve is a result of the piping analysis and includes the relief valve opening loads. Therefore, a dynamic load factor for analyzing the relief valve open system transient as a static load is not needed. Finally, with regard to the RCP locked rotor analysis, the applicant clarified that the RCP locked rotor is postulated to occur in a single loop and therefore, only that loop is evaluated in accordance with ASME Code, Section III Level D design criteria for the accident conditions. The other three non-accident loops are evaluated for integrity using the ASME Code, Section III Level B design criteria which appropriately reflects the loads for the non-accident loops. The LBB criteria discussed in DCD Tier 2, Section 3.6.3, “LBB Evaluation Procedures,” has been applied to all of the lines listed in DCD Tier 2, Appendix 3B, “Bounding Analysis Curve Development for Leak-Before-Break Evaluation of High-Energy Piping for US-APWR,” Table 3B-2, “List of BACs for LBB Evaluation.” The applicant has applied the LBB criteria to RCS Class 1 piping. As a result of the LBB evaluations, the main reactor coolant piping break and surge line break dynamic evaluations were eliminated. The applicant provided a mark-up of DCD Tier 2, Subsection 3.9.3.1.2 that states that the specific RCL ASME Code, Section III, Class 1 branch lines, and MS lines listed in DCD Tier 2, Appendix 3B that can be exempted from required pipe rupture considerations by meeting LBB criteria. The applicant’s markup also stated that components and piping are evaluated for the dynamic response to transient loads. The staff confirmed that the changes were incorporated into DCD Tier 2, Revision 2, Subsection 3.9.3.1.2. Additional details of LBB criteria are discussed in section 3.6.3 of this report. The staff found the RAI response acceptable since the applicant’s approaches are consistent with the staff guidance in NUREG-0609. Accordingly, **RAI 209-1803, Question 03.09.03-6, is resolved**.

In DCD Tier 2, Subsection 3.9.3.1.3, “ASME Code, Section III, Class Components and Supports and Class CS Core Support Loading Combinations and Stress Limits.” the applicant provides the loading combinations and stress limits criteria for ASME Code, Section III, Class 1 components and supports and ASME Class CS core support structures. DCD Tier 2, Table 3.9-6, “Stress Criteria for ASME Code, Section III, Class 1, Components and Supports and Class CS Core Supports,” summarizes stress criteria per ASME Code Subarticles applicable to these Class 1 and Class CS components and their supports. In **RAI 209-1803, Question 03.09.03-7**, the staff requested the applicant to clarify certain ASME Code Sections applicable to the vessel and other components as identified in DCD Tier 2, Table 3.9-6.

In its response to **RAI 209-1803, Question 03.09.03-7**, dated April 30, 2009, the applicant stated that although ASME Code, Section III, NB-3300 and 3400, describe the scope of the vessel and pump design, both sections refer to ASME Code, Section III, NB-3200 for stress criteria. ASME Code Section NB-3527 states that if the design specifications specify any loadings for which level D limits are designated, the guidelines of ASME Code, Section III Appendix F may be used in evaluating those loadings independently of other loadings. Rather than referencing NB-3527 in Table 3.9-6, the applicant elected to use the notation afforded by that section to refer to ASME Code, Section III Appendix F for pressure boundary integrity requirements. Based on this, the staff determined that the intent of those referenced Subarticles in DCD Tier 2 Table 3.9-6 on ASME Code, Section III is consistent with the required design criteria for Class 1 components such as vessel, pumps, and valves. Therefore, the staff finds the response acceptable. Accordingly, **RAI 209-1803, Question 03.09.03-7, is resolved**.

In DCD Tier 2, Section 3.9.3.1.4, "RCL Piping Model," the applicant describes the modeling and analysis of the RCL piping along with all major components within the containment. This model also includes the attachments to the R/B, the containment and the internal structure. DCD Tier 2, Appendix 3C, "Reactor Coolant Loop Analysis Methods," provides a detailed description of the RCL piping, components, and support system model. The modeling of primary components and supports consist of the RCL hot, cold, and cross-over loop piping, SG, RCP, RV, and component supports, as applicable, for each of four RCL loops. The combined system models include both the translational and rotational stiffness, mass characteristics of the RCL piping and components, and the stiffness of supports. The stiffness and mass effects of associated line piping are also considered when they affect the system. The staff found that the modeling of components and supports are acceptable. Additional evaluation of four-loop RCL piping models is detailed in Section 3.12 of this report

In DCD Tier 2, Section 3.9.3.1.5, "ASME Code, Section III, Class 2 and 3 Components," the applicant provides the loading combinations and stress limits criteria for ASME Code, Section III, Class 2 and Class 3 components and supports. DCD Tier 2, Table 3.9-8, "Stress Criteria for ASME Code, Section III Class and Components and Supports," summarizes the stress criteria per ASME Code Subarticles applicable to Class 2 and Class 3 components and their supports. In **RAI 209-1803, Question 03.09.03-8** the staff requested the applicant to clarify several items included in DCD Tier 2, Table 3.9-8 including:

a) Discuss the criteria used in the design of vessels in accordance with ASME Code Subsection NC-3217 for Service Level A and why these criteria are not applicable to the other service level conditions for the vessel design; and

b) Explain why the verification of whether environmental fatigue on ASME Class 2 and 3 components will follow guidelines established by the NRC would not be a COL information item.

In its response to **RAI 209-1803, Question 03.09.03-8**, dated March 4, 2011, the applicant suggested the deletion of the reference to ASME Code Subsection NC-3217 in DCD Tier 2, Table 3.9-8, since this section of the Code provides the design criteria instead of stress criteria. The staff verified that the applicant's suggestions for revising the DCD Tier 2, Subsection 3.9.3.1.5 and Table 3.9-8 satisfy the appropriate Code Subarticles stated in the table. Therefore, the staff finds this portion of the response acceptable. In response to concerns that some ASME Class 2 and 3 components are exposed to high radiation or high thermal cycles and therefore their materials are susceptible to environmental fatigue, the applicant stated that ASME Class 2 and 3 components are out of scope of the environmental fatigue evaluation in accordance with RG 1.207, "Guidelines for Evaluating Fatigue Analyses Incorporating the Life Reduction of Metal Components Due to the Effects of the Light-Water Reactor Environment for New Reactors," March 2007." The applicant further stated that it will perform screening evaluations of the cases where temperature fluctuations occur in the junction between cold and hot water, and assess the environmental impact on fatigue. The staff agrees that ASME Class 2 and 3 components are out of scope for environmental fatigue evaluation per RG 1.207. The staff confirmed that the applicant revised DCD Tier 2, Subsection 3.9.3.1.5 to incorporate the proposed changes. However, the description in the DCD of the assessment of environmental fatigue of ASME Class 2 and 3 components was unclear and did not match the RAI response.

Therefore, the staff closed **RAI 209-1803, Question 03.09.03-8,** as unresolved and the staff asked follow-up **RAI 851-6065, Question 03.09.03-28** as follows.

In DCD Tier 2, Section 3.9.3.1.5, the last sentence of second paragraph stated, "The environmental impact on fatigue of Class 2 and 3 components will follow guidelines established by the NRC at the time of the actual analysis." Currently, there is no existing RG for the environmental impact on fatigue of ASME Class 2 and 3 components; therefore, this statement does not apply to US-APWR DC. In **RAI 851-6065, Question 03.09.03-28**, the staff requested the applicant to modify the statement in DCD Tier 2, Section 3.9.3.1.5 reflect the actual approach to be followed or to delete the statement from the DCD. In its response to **RAI 851-6065, Question 03.09.03-28**, dated December 20, 2011, the applicant agreed to remove the statement from the DCD. The staff finds the response to delete the statement acceptable since Class 2 and 3 components are outside the scope of NRC guidelines. The staff will confirm that the DCD change is incorporated in the next revision. Accordingly, **RAI 851-6065, Question 03.09.03-28 is being tracked as a Confirmatory Item**.

3.9.3.4.2 Design and Installation of Pressure-Relief Devices

In DCD Tier 2 Section 3.9.3.2, "Design and Installation of Pressure-Relief Devices," the applicant provides the criteria for the design and installation of pressure-relief devices that complies with the requirements of ASME Code, Section III, Appendix O, "Rules for the Design of Safety Valve Installations." The applicant also describes the ASME Code, Section III, Class 1 and 2 pressure relief devices. The only ASME Code, Section III, Class 1 pressure relief valves in the US-APWR standard plant are the safety valves connected to the pressurizer upper head. The ASME Code, Section III, Class 2 pressure relieving devices include the safety valves and power operated relief valves on the steam line and the relief valve on the containment isolation portion of the normal RHR system.

The design of pressure-relieving devices can be generally grouped in two categories: open discharge to the atmosphere or a vent stack open to the atmosphere and closed discharge by piping between the valve and a tank or some other terminal end. The applicant describes the analytical methods and assumptions to be used in the design and installation of these devices consistent with the SRP Section 3.9.3. The staff found that the design of pressure-relieving devices for the US-APWR piping and components will provide adequate protection against any high pressure failures and therefore, is acceptable.

3.9.3.4.3 Pump and Valve Operability Assurance

The staff reviewed DCD Tier 2 Section 3.9.3.3, "Pump and Valve Operability Assurance," which discusses criteria for operability assurance of safety-related pumps and valves being capable of performing their intended functions during the life of the plant under various postulated transient conditions. Active pumps and valves are also required to function under faulted conditions. The applicant states that DCD Tier 2, Section 3.10, "Seismic and Dynamic Qualification of Mechanical and Electrical Equipment," provides the equipment specifications to assess the functional capability of the required components. These criteria and considerations include collapse and deflection limits associated with these components. The staff requested in **RAI 209-1803, Question 03.09.03-9**, the applicant to clarify these criteria and considerations including the collapse and deflection limits.

In its response to **RAI 209-1803, Question 03.09.03-9**, dated April 30, 2009, the applicant stated that the stress evaluation applied by ASME Code, Section III Service Level D requirements is intended to assure that the pressure retaining integrity is maintained, but is not intended to assure operability of components. Pump and valve operability is assured by tests and analysis in accordance with SRP Section 3.10, "Seismic and Dynamic Qualification of Mechanical and Electrical Equipment," Revision 3, issued March 2007, for seismic/dynamic qualification and SRP Section 3.11, “Environmental Qualification of Mechanical and Electrical Equipment,” Revision 3, issued March 2007, for EQ of active pumps and valves during their design and installation phases. In addition, DCD Tier 2, Section 3.9.6, “Functional Design, Qualification, and Inservice Testing Programs for Pumps, Valves, and Dynamic Restraints,” provides details for the IST Program to assure pump and valve operability during their service lives. The applicant’s description of the stress evaluation is consistent with the current staff position in SRP 3.9.3. Further evaluation of seismic and dynamic qualification, environmental qualification, and the IST program are provided in Sections 3.10, 3.11, and 3.9.6, respectively. Accordingly, **RAI 209-1803, Question 03.09.03-9, is resolved**.

Regarding, pump operability assurance, in DCD Tier 2, Section 3.9.3.3.1, “Pump Operability,” the applicant provides definitions of active and inactive pumps. Active pumps are those whose operability is relied upon to perform a safety-related function during transients or events in the respective operating condition categories. The only criterion included in this section is that the design of active pumps is in accordance with ASME Code, Section III requirements as outlined in DCD Tier 2, Table 3.9-6 for Class 1 pumps and DCD Tier 2, Table 3.9-8 for Class 2 and 3 pumps. In **RAI 209-1803, Question 03.09.03-10**, the staff requested that the applicant clarify how the operability of safety-related pumps is ensured by designing them in accordance with the ASME Code requirements.

In its response to **RAI 209-1803, Question 03.09.03-10**, dated April 30, 2009, the applicant stated that the operability of safety-related pumps is assured through various means. Pump operability is initially assured through factory tests conducted before installation. These factory tests include a hydrostatic test for pressure retaining parts, pump seal leakage tests at the hydrostatic test pressure, and pump head versus flow tests. The ability of safety-related pumps to operate as a result of adverse seismic and environmental conditions (i.e., equipment qualification), as described in DCD Tier 2, Sections 3.10, “Seismic and Dynamic Qualification of Mechanical and Electrical Equipment,” and 3.11, “Environmental Qualification of Mechanical and Electrical Equipment,” respectively, is also required to be demonstrated prior to plant operation. Other, post-installation pump testing includes system tests, integrated system tests, cold hydrostatic tests and hot functional tests. Plant operational tests to assure pump operability include the IST program during the in-service life of the pump as described in DCD Tier 2, Section 3.9.6. The applicant’s description of the qualification methods for Class 1 pump operability assurance is consistent with the current staff position in SRP 3.9.3. Further evaluation of seismic and dynamic qualification, environmental qualification, and the IST program are provided in Sections 3.10, 3.11, and 3.9.6, respectively. Accordingly, **RAI 209-1803, Question 03.09.03-10, is resolved**.

With regards to valve operability assurance, in DCD Tier 2 Section 3.9.3.3.2, “Valve Operability,” the applicant also provides definitions of active and inactive valves. Active valves are those whose operability are relied upon to perform safety-related functions during transients or events in the respective operating condition categories. The criteria included in this section is that the design of active valves is in accordance with ASME Code, Section III requirements as outlined in DCD Tier 2, Table 3.9-6 for Class 1 valves and DCD Tier 2, Table 3.9-8 for Class 2 and 3 valves and a series of tests and inspections prior to installation and service as well as during the plant life. In **RAI 209-1803, Question 03.09.03-11**, the staff requested the applicant to elaborate on how the operability of safety-related valves is ensured by the stated criteria in the DCD.

In its response to **RAI 209-1803, Question 03.09.03-11**, dated April 30, 2009, the applicant addressed valve motor operator operability qualification, testing and analyses for ASME Code, Section III Level D service conditions, and the use of IEEE 344-2004. Regarding valve motor operator operability qualification, the applicant stated that all valve operators listed in DCD Tier 2, Appendix 3D, Table 3D-2, "US-APWR Environmental Qualification Equipment List," will be qualified to the environmental conditions that will be experienced for the particular application (i.e., the equipment will be included in the environmental qualification program whose implementation is verified by ITAAC). Qualification verification is assured through actuator testing, type-testing, or analysis prior to plant operation. Other vendor tests (such as flow tests, cycle tests, hydrostatic tests) will also be performed if required by the component specifications, which also assure that the actuators are capable of performing their intended design functions. Operability tests in accordance with the IST program are conducted after installation to verify the component function.

Regarding testing and analyses for ASME Code, Section III Level D service conditions, the applicant indicated that the valve manufacturer determines the representative number of valves to verify seismic adequacy based on valve attributes such as size, type, model, flow characteristics, etc. The dynamic load, other than an SSE, considered by the applicant is the jet impingement load. Verification that the jet impingement load will not adversely affect an intact train is demonstrated through stress analysis or physical separation.

Regarding the use of IEEE 344-2004, the applicant clarified that it is adopting only the updated Appendix D from IEEE 344-2004 and otherwise continue to adopt IEEE 344-1987. The applicant committed to revise DCD Tier 2, Subsection 3.9.3.3.2 to ensure that the reference is only to Appendix D of IEEE 344-2004. The applicant stated that the DCD has been reviewed and other DCD subsections which reference IEEE 344-2004 will be changed to reference only Appendix D of IEEE 344-2004 or reference IEEE 344-1987.

The staff finds the applicant's response acceptable since the described qualification methods for the valve operability assurance are consistent with the guidance in SRP Section 3.9.3, and will ensure that safety-related valves will perform their intended functions during their service lives. The staff confirmed that the suggested changes to DCD Tier 2, Subsection 3.9.3.3.2 were incorporated into DCD Tier 2, Revision 3. Accordingly, **RAI 209-1803, Question 03.09.03-11, is resolved**.

On the basis of the above evaluations, the staff determined that the criteria for operability assurance of pumps and valves are acceptable. The testing and analysis methods to be used in the qualification of these components are commonly used in the industry and thus provide an adequate margin of safety to withstand the loadings as a result of normal operating, transient, emergency and accident service condition levels. Therefore, the staff concludes that the applicant satisfies the requirements of 10 CFR 50.55a, 10 CFR 52.47(b)(1), GDC 1, 2, 4, 14 and 15, and 10 CFR Part 50, Appendix S, by specifying appropriate analysis and testing methods for designing safety-related pumps and valves for the US-APWR plant.

3.9.3.4.4 Component Supports

The staff reviewed DCD Tier 2 Section 3.9.3.4, "Component Supports," that discusses criteria for the design of ASME Code, Section III, Class 1, 2, and 3 component supports and their attachments. The design of ASME Code, Section III, Class 1, 2, and 3 component supports and their attachments is in accordance with ASME Code, Subsection NF up to the interface of the building structure, with jurisdictional boundaries as defined by ASME Code, Subsection NF. The building structure component of the supports (connecting the ASME Subsection NF support boundary component to the existing building structure) is designed in accordance with ANSI/AISC N690-1994, "Nuclear Facilities-Steel Safety-Related Structures for Design, Fabrication and Erection," and AISC "Manual for Steel Construction," 9th Edition, 1989, for the design, fabrication, and erection of structural steel structures. The stress limits are in accordance with ASME Code, Subsection NF and Appendix F; RG 1.124, "Service Limits and Loading Combinations for Class 1 Linear-Type Supports," Revision 2, issued February 2007; and RG 1.130, "Service Limits and Loading Combinations for Class 1 Plate-and-Shell-Type Component Supports," Revision 2, issued March 2007. ITAAC associated with ASME Code, Section III, Class 1, 2, and 3 component supports and their attachments are discussed in Section 3.9.3.4.5 of this report.

The applicant states that the maximum calculated static and dynamic deflections of the component at support locations should not exceed the allowable limits specified in the component design specification. But the applicant did not discuss how the maximum static and dynamic deflections are combined from multiple loads under the four service level conditions and how the allowable limits are established for the component in its design specification. Therefore, in **RAI 209-1803, Question 03.09.03-12,** the staff requested the applicant to discuss details on calculating the component deflections from different load conditions and establishing the allowable limits.

In its response to **RAI 209-1803, Question 03.09.03-12**, dated April 30, 2009, the applicant stated that as described in MUAP-09002-P, the maximum displacement is calculated by dynamic or seismic response analysis using the coupling model of the four-loop RCL model and building structure model. The displacement for accident conditions is calculated by dynamic analysis using a four-loop RCL model. The maximum displacement of each component is evaluated. Displacement for dead weight, thermal expansion and internal pressure is calculated by static analysis using a four-loop RCL model. Component deflections from these load conditions are combined and compared with the allowable limits given in the design specification for the component. Based on this, the staff concluded that the use of the four-loop RCL model for calculating the component maximum displacement is acceptable. Additional evaluation of the four-loop RCL model is detailed in Section 3.12 of this report. Therefore, since the applicant explained how maximum displacements are determined, the staff finds the applicant's response acceptable. Accordingly**, RAI 209-1803, Question 03.09.03-12, is resolved**.

In DCD Tier 2, Section 3.9.3.4, the applicant classified all supports into two groups: manufactured standard supports and supplementary steel supports. The manufactured standard supports include spring hangers, snubbers, and struts that are commercially available in accordance with their rated loads. They are designed in accordance with the ASME Code, Section III or equivalent structural codes for all four service level conditions. The supplementary steel supports are fabricated from steel sections such as steel tubing, wide flange sections, steel plates, and steel angles that are welded in permanent configuration.

In DCD Tier 2, Subsection 3.9.3.4.1, "Spring Hangers," the applicant states that spring hangers are typically used for supporting components with dead weight loads because they allow specified movement of a supported component at different thermal conditions. They are evaluated for expected movements and field calibrated to assure adequate consideration of operational thermal conditions.

In DCD Tier 2, Subsection 3.9.3.4.3, "Struts," the applicant states that struts are pin-connected rigid passive axial supports with no active mechanism. Like all other types of supports, the forces on struts are obtained from an analysis and confirmed not to exceed the design loads for various service level conditions. They are typically used in all three spatial directions to support dead weight and other dynamic events including seismic and system operating transients.

In DCD Tier 2, Subsection 3.9.3.4.2, "Snubbers," the applicant states that two types of snubber supports (i.e., hydraulic and mechanical) are commercially available for supporting nuclear power plant piping and components for dynamic restraints. They allow free thermal expansion in the axial direction of the support while restraining sudden dynamic movement of the component they support. These dynamic loads include seismic, building hydrodynamic loads, safety relief valve discharge, and steam/water hammer due to sudden valve closures.

In DCD Tier 2, Subsection 3.9.3.4.2.1, "Assurance of Snubber Functionality," the applicant states that systems and components that use snubbers as shock absorbers are analyzed to determine the interaction of the snubbers with the systems and components to which they are attached. Cyclic fatigue is analyzed unless the load cycle is small or overall thermal movements are verified to be below limits.

The applicant provides these acceptable methods for spring hangers, struts, snubbers, and snubber functionality as discussed above and they are consistent with the guidelines in SRP Section 3.9.3 and therefore, the staff finds them acceptable.

The staff noted that DCD Tier 2, Subsection 3.9.3.4.2.4, "Snubber Mechanical and Structural Properties," provides insufficient information for potential snubber end fitting clearances, mismatch of activation and release rates, and lost motion. In **RAI 209-1803, Question 03.09.03-14**, the staff requested that the applicant discuss how it accounted for snubber end fitting clearances, mismatch of activation and release rates, and lost motion, and how they would affect the calculations of snubber reaction loads and stresses using a linear analysis methodology. In multiple snubber applications where mismatch of end fitting clearance and lost motion exists, the staff requested that the applicant discuss their potential impact on the synchronism of activation level or release rate and, consequently, on the assumption of the load sharing of multiple snubber supports.

In its response to **RAI 209-1803, Question 03.09.03-14**, dated March 4, 2011, the applicant stated that since the procurement specifications will require snubber vendors/manufacturers to provide the design parameters of tight fitting pins and spherical bearings that allow for off axis movement while minimizing lost motion at both ends of the connection (i.e., the pipe clamp and the end structural attachment), the end clearances are minimal. The design parameters of end fitting clearances, as well as release rates and lost motion, will be accounted for in the average dynamic spring rate provided to the designer by the manufacturer. Since these design parameters are accounted in the spring rate, these parameters will also affect snubber load resulting from the piping stress analysis.

The staff finds the supplied snubber design information acceptable because the applicant provided the staff's requested descriptions of how the design parameters are being applied to the snubber designs and equipment and will be incorporated into procurement specifications in accordance with ASME Code, Section III. Therefore, the staff found that the applicant's response is acceptable. Accordingly, **RAI 209-1803, Question 03.09.03-14, is resolved**.

In DCD Tier 2, Subsection 3.9.3.4.2.7, "Snubber Design and Testing," the applicant states that the support design specification requires snubbers to be designed in accordance with ASME Code, Section III, Subsection NF. The design requirement includes analysis for normal, upset, emergency, and faulted loads. The applicant also states that these calculated loads are then compared against the manufacturer's design and/or test capacities to ensure that the stresses are below the ASME Code's allowable limits. The staff, however, found no applicant-specific design requirements provided for snubbers. In **RAI 209-1803, Question 03.09.03-15**, the staff requested that the applicant provide a detailed discussion on the specific design rules of ASME Code Subsection NF that apply to snubbers. The staff also asked the applicant to provide a detailed discussion on how the load capacity for design, normal, upset, emergency, and faulted conditions is derived and compared against the vendor's allowables, for both mechanical and hydraulic snubbers.

In its response to **RAI 209-1803, Question 03.09.03-15**, dated April 30, 2009, the applicant stated that ASME Code Subsection NF imposes rules for the design of snubbers that pertain primarily to the snubber vendor who must incorporate all of the ASME Code parameters into the equipment design. For example, the parameters stated in ASME Code Subarticle NF-3412.4, such as design loadings, required force, time and displacement relationship, capability to operate in environmental conditions, and material characteristics, are all relevant to the manufacturer of snubbers. The attributes required for the piping design will be included in the snubber procurement specifications. Once these requirements have been incorporated into the design, the pipe support designer will select a snubber that is appropriate for the snubber design conditions.

Based on the above snubber design information on the use of ASME Code parameters in the snubber design, the staff finds the response acceptable. Accordingly, **RAI 209-1803, Question 03.09.03-15, is resolved**.

DCD Tier 2, Subsection 3.9.3.4.2.7, states that specific environmental design considerations, and that the snubber functionality is assured under harsh service conditions are important snubber design considerations. DCD Tier 2, Subsection 3.9.3.4.2.5, "Design Specifications," states, per COL Information Item 3.9(1), that the COL Applicant is to assure snubber functionality in harsh service conditions, including snubber materials (e.g., lubricants, hydraulic fluids, seals). These harsh service conditions require evaluation and life projection, including provisions for high level radiation areas and snubber materials, for example snubber seals and fluids,. Also, the applicant states that based on initial in-situ snubber dynamic lock-up testing and thermal motion testing, a comparison of test data with analytical data (force and/or displacement time histories due to earthquakes and/or dynamic transients) assures that the piping or component stress analysis model and as-built snubber configuration performs within the analytical boundaries. However, a detailed description of snubber qualification and manufacturing tests has not been provided for the snubber designs. In **RAI 209-1803, Question 03.09.03-16**, the staff requested the applicant to (1) discuss the procedure and scope of manufacturing, qualification, and installation test programs, separately, for both the mechanical and hydraulic snubbers of different sizes and manufacturers, (2) discuss how the criteria for each pertinent snubber functional parameter are met in the testing, and (3) provide the codes and standards used for the test programs.

In its response to **RAI 209-1803, Question 03.09.03-16**, dated April 30, 2009, the applicant stated that the snubber vendor/manufacturer builds the snubber in compliance with ASME Code Subsection NF-3412.4, which specifies that the snubber must be capable of operating in the environmental conditions to which it will be exposed; namely, temperature, irradiation, corrosive atmosphere, moisture and airborne particles. The qualification testing for mechanical snubbers normally consists of dynamic load tests, temperature tests, environmental tests (humidity, salt spray, and sand and dust), life (i.e., cycle) tests, and faulted load tests. Between each qualification test, an acceptance test will be performed consisting of an acceleration test, lost motion test, and a drag test. Manufactures will also perform functional testing of mechanical snubbers prior to shipment consisting of lost motion, acceleration (activation), and final drag testing. Functional testing for in-service conditions can consist of drag tests and acceleration tests. The applicant also described a snubber qualification process typically adopted by the industry.

The staff found that the applicant has adequately addressed the qualification process for the snubber design and installation since it is in accordance with ASME QME-1 requirements. This ensures that the snubber subject to these above tests and design criteria prior to service would perform its intended functions. Therefore, the staff finds the response acceptable. Accordingly, **RAI 209-1803, Question 03.09.03-16, is resolved**.

To ensure continual functional operation of snubbers associated with safety-related components during the life of the plant, the staff noted that an ASME Section XI in-service snubber-testing program in accordance with ASME Code OM, Subsection ISTD is required. Additional evaluation of operation and in-service testing for ASME components is detailed in Section 3.9.6 of this report.

In DCD Tier 2 Subsection 3.9.3.4.2.6, “Considerations for Inspection, Testing, Repair, and/or Replacement of Snubbers,” the applicant indicated that the guidance for testing, maintenance, and repair of snubbers are provided in the snubber instruction manual provided by the vendor. This manual specifies the required inspection locations and the periods of inspection. The staff notes the applicant utilizes this information in accordance with ASME QME-1-2007, “Qualification of Active Mechanical Equipment Used in Nuclear Power Plants,” as accepted in RG 1.100, “Seismic Qualification of Electric and Mechanical Equipment for Nuclear Power Plants,” Revision 3, issued September 2009. As described in DCD Tier 2, Section 3.9.6.1, “Functional Design and Qualification of Pumps, Valves, and Dynamic Restraints.” hydraulic snubbers for piping require that a fluid level indicator is equipped to ascertain the level of fluid in the snubber. Snubber thermal movement, clearance, and gaps are periodically verified, including motion measurements, and acceptance criteria assure compliance with ASME Code, Section III, Subsection NF. The staff noted that the applicant has indicated the hydraulic snubbers for piping are equipped with level indicators, but not those used for component supports. In **RAI 209-1803, Question 03.09.03-17**, the staff requested the applicant to clarify why the fluid level indicators in hydraulic snubbers are limited to piping supports only.

In its response to **RAI 209-1803, Question 03.09.03-17**, dated April 30, 2009, the applicant clarified that hydraulic snubbers are equipped with fluid level indicators, whether they are used for pipe supports or for equipment supports. Also, ASME QME-1 requires the applicant to apply an instruction manual for both hydraulic and mechanical snubbers. The contents of the instruction manual used by the applicant for mechanical snubbers will include information such as the mechanical shock arrestor service life extension program and address maintenance items such as snubber re-greasing and repair. The staff found the response acceptable because the applicant clarified the statements in the DCD on level indicators and snubber instruction manuals. The staff confirmed that changes to the DCD reflecting these clarifications were incorporated into DCD Tier 2 Revision 3. Accordingly, **RAI 209-1803, Question 03.09.03-17, is resolved**.

In DCD Tier 2 Subsection 3.9.3.4.2.5, the applicant lists the contents required in design specification for snubbers. This list is consistent with SRP Section 3.9.3 and the staff found the list acceptable.

In DCD Tier 2, Subsection 3.9.3.4.4, “Frame Type Pipe Supports,” the applicant states that frame type pipe supports within the supplementary steel support category are rigid supports and are typically used as guides. They consist of frames that are constructed of structural steel elements and allow axial and rotational movement of the pipe but serve as rigid restraints similar to struts in either one or two directions. They are used in conditions not suitable for manufactured supports. For insulated pipes, special pipe guides such as pipe saddles with one or two-way restraint are used in order to minimize the heat loss of piping systems. A limited total gap of 1/8th in. (3.2 mm) around the piping is allowed to avoid thermal binding due to radial thermal expansion of the pipe. For large pipes with higher temperatures, this gap is evaluated to assure that no thermal bending occurs.

In DCD Tier 2, Subsection 3.9.3.4.4, “Frame Type Pipe Supports,” the design of frame type supports considers an appropriate coefficient of friction to calculate friction loading on the support due to friction between the pipe and frame support that occurs as a result of sliding. DCD Tier 2, Section 3.12.6.10, “Consideration of Friction Forces,” provides a coefficient of friction of 0.3 for steel-to-steel contact and 0.1 for slide/bearing plates. This subsection also provides a criterion that the friction force should always be less than the force exerted by the pipe displacement in the direction of movement. The staff found that the described coefficients of friction and the criterion addressed the design of frame type pipe supports and is acceptable.

In DCD Tier 2, Subsection 3.9.3.4.5, “Component Support Baseplate and Anchor Bolt,” the applicant states that special engineered pipe supports, designed without the use of manufactured standard supports or supplementary steel supports, are used for pipe supports in the US-APWR design. They utilize non-standard specialized components and can have both mechanical and structural characteristics. These support types are used generally on systems that have high thermal expansion and require seismic or vibration support to minimize the use of snubbers. The staff noted that the applicant did not provide sufficient details regarding the design criteria and dynamic testing of these supports. In **RAI 209-1803, Question 03.09.03-18**, the staff requested the applicant to provide some of these design criteria.

In its response to **RAI 209-1803, Question 03.09.03-18**, dated April 30, 2009, the applicant determined that special engineered pipe supports will not be used in the US-APWR design and will replace the discussion with the text on baseplate design in DCD Tier 2, Subsection 3.9.3.4.5. The applicant stated that the design of components anchorage to concrete follows ACI-349, Appendix B, considering the limitations of RG 1.199, “Anchoring Components and Structural Supports in Concrete,” issued November 2003. All aspects of anchor bolt design, baseplate flexibility and factors of safety are utilized as identified in NRC Bulletin 79-02, Pipe Support Base Plate Designs Using Concrete Expansion Anchor Bolts, Revision 2, issued November 8, 1979. The staff found that the applicant decision not to use special engineering pipe supports in the US-APWR design acceptable because the applicant used the design of components anchorage to concrete in accordance with of ACI-349 Appendix B. Also, the design of the baseplate and anchor bolts is consistent with the current staff position in NRC

Bulletin 79-02, Revision 2. Therefore, the staff finds the RAI response acceptable. Accordingly, **RAI 209-1803, Question 03.09.03-18, is resolved**.

In DCD Tier 2 Subsection 3.9.3.4.6, “ASME Code, Section III, Class 1, 2, and 3 Component Supports,” the applicant provides design criteria for ASME Code, Section III, Class 1, 2, and 3 component supports in accordance with ASME Code Subsection NF. The design specification for each component and its support includes loadings, load combinations, and stress limits for establishing its structural integrity. RG 1.124 (for Class 1 linear type supports) and RG 1.130 (for Class 1 plate and shell type supports) provide acceptable levels of service limits and appropriate combinations of loading associated with normal operation, postulated accidents, and specified seismic events, since ASME Code does not specify loading combinations for these types of component supports.

The applicant also states that where the design and service stress limits specified in the code do not necessarily provide direction for the proper consideration of operability requirements for conditions which warrant consideration, Section II.3 and Appendix A to SRP Section 3.9.3, RG 1.124 and RG 1.130 are used for guidance. Where these stress limits apply, the treatment of functional capability, including collapse, deformation and deflection limits is evaluated and design information is developed for inclusion into the design specification. In **RAI 209-1803, Question 03.09.03-19**, the staff requested the applicant to discuss the operability requirements and functional capability of component supports.

In its response to **RAI 209-1803, Question 03.09.03-19**, dated April 30, 2009, the applicant stated that the operability requirements of component supports will be provided in detail in the design specifications. The staff finds the response acceptable since the use of design specifications is consistent with the commitment stated in DCD Tier 2, Subsection 3.9.3.4.6. Accordingly, **RAI 209-1803, Question 03.09.03-19, is resolved**.

In DCD Tier 2, Subsection 3.9.3.4.6, the applicant further stated that ASME Code, Section III component supports are designed, manufactured, installed, and tested in accordance with all applicable codes and standards. Supports include hangers, snubbers, struts, spring hangers, frames, energy absorbers, and limit stops. In **RAI 209-1803, Question 03.09.03-20**, the staff requested the applicant to provide additional information regarding the design of these types of component supports, specifically the energy absorbers and limit stops.

In its response to **RAI 209-1803, Question 03.09.03-20**, dated April 30, 2009, the applicant stated that the codes and standards to be used in the design of component supports will be provided in the design specification. In addition, the applicant stated that energy absorbers and limit stops will not be used in the US-APWR design and provided DCD changes. The staff finds the response acceptable since the applicant clarified design information on component supports. The staff confirmed the applicant has inserted a new DCD Tier 2 Subsection 3.9.3.4.7 indicating that energy absorbers and limit stops will not be used in the US-APWR design. Accordingly, **RAI 209-1803, Question 03.09.03-20, is resolved**.

In DCD Tier 2 Subsection 3.9.3.4.6.1, “ASME Code, Section III, Class Component Supports Models and Methods,” the applicant discusses the design methods for ASME Class 1 component supports and includes, in that discussion, supports for RV, SGs, RCPs, and the pressurizer. The structural analysis of these ASME Code, Section III, Class 1 component supports includes the loads, load combinations, and stress allowable limits in accordance with the ASME Code, Section III, Subsection NF and Appendix F. Externally applied loads for each system operating, transient, and accident condition that are generated from the RCL piping analysis are applied and are appropriately combined with component generated support loads. The combination of loadings considered for each component support uses the criteria in Appendix A of SRP Section 3.9.3, RG 1.124, and RG 1.130. Computerized finite element analysis programs are used to determine the support stresses and reaction loads. The seismic and design analysis method for RCL system components and supports are presented in MUAP-09002, Revision 2, and MUAP-10006, Revision 3. In **RAI 209-1803, Question 03.09.03-21**, the staff requested the applicant to clarify a number of concerns associated with these major component support designs.

In its response to **RAI 209-1803, Question 03.09.03-21**, dated April 30, 2009, the applicant stated that the methodology of modeling of component supports is described in the applicant's Technical Report MUAP-08005, Revision 0, Section 6, "Coupled Analytical Model." The applicant further stated that fatigue evaluation will be performed in accordance with ASME Code, Section III, Subarticle NF. A coupled RCL model includes the component supports attached to the RV, SG, and RCP and there are no supports attached to RCL piping. Subsequently, the applicant informed the staff that it was updating its seismic models and MUAP-08005 was no longer being used. In November 2012, the applicant issued Technical Report MUAP-10006, "Soil-Structure Interaction Analyses and Results for the US-APWR Standard Plant," Revision 3, which supersedes MUAP-08005. The staff will review the new information of MUAP-10006 and the related Technical Report MUAP-09002. Until the review of MUAP-10006 and MUAP-09002 is completed, **RAI 209-1803, Question 03.09.03-21, is being tracked as an Open Item**.

In DCD Tier 2, Subsection 3.9.3.4.6.2, "ASME Code, Section III, Class 2 and 3 Component Supports Models and Methods," the applicant discusses the design criteria (models and methods) for ASME Code Class 2 and 3 component supports. These component supports are generally of linear or plate and shell type; however, standard component supports may be used. The component support design complies with ASME Code, Section III, Subsection NF and Appendix F and includes ASME Code, Section III Level A, B, C, and D service load and load combination requirements. ASME Code, Section III, Class 2 and 3 piping supports are designed and analyzed as discussed in DCD Tier 2, Section 3.12, "Piping Design Review." In **RAI 209-1803, Question 03.09.03-22**, the staff requested the applicant to clarify certain details of Class 2 and 3 component support design requirements.

In its response to **RAI 209-1803, Question 03.09.03-22**, dated April 30, 2009, the applicant stated that:

- It does not use the allowable stress limits and the evaluation of linear supports for ASME Code, Section III Service Level D conditions. The signature test methods are used and defined in ASME Code, Section III, Appendix F for Class 2 and 3 component supports.

- The load combination for the component supports is determined in accordance with SRP Section 3.9.3, Appendix A, and provided in DCD Tier 2, Table 3.9-4. The design criteria conform to the requirements of Section C of RG 1.124 and Section C of RG 1.130. For linear type, plates, and shell type component supports, there is no difference between the load combination criteria presented in the DCD and that presented in the RGs.

The applicant also revised DCD Tier 2, Subsection 3.9.3.4.6.2 and incorporated the requirements of allowable stresses for plant emergency conditions as follows:

- DCD Tier 2, Subsection 3.9.3.4.6.2 to be changed to the "Emergency - For emergency conditions, the allowable stresses or load ratings are 33% higher than those specified for normal conditions. This is consistent with Subsection NF of ASME Code, Section III (Reference 3.9-1) in which (see NF-3250 and NF-3260) limits for emergency conditions are 33 percent greater than the normal condition limits."

- Delete the sixth paragraph in DCD Tier 2, Subsection 3.9.3.4.6.2 in its entirety.

- Delete the seventh paragraph in DCD Tier 2, Subsection 3.9.3.4.6.2 in its entirety.

The staff finds the RAI response acceptable since the applicant adequately addressed the design criteria for the Class 2 and 3 component supports. The staff confirmed that the changes to DCD Tier 2, Subsection 3.9.3.4.6.2 were incorporated into DCD Tier 2 Revision 2. Accordingly, **RAI 209-1803, Question 03.09.03-22, is resolved**.

In DCD Tier 2, Revision 1, Subsection 3.9.3.4.7, “Snubbers Used as Component Supports,” the applicant discusses snubbers used as component supports. Snubbers are generally hydraulic; however, there are mechanical snubbers available that lock-up at equivalent hydraulic velocities. Details of snubber design, testing, operation, maintenance, inspection, and other functional characteristics are presented in DCD Tier 2, Subsection 3.9.3.4.2. In **RAI 209-1803, Question 03.09.03-23**, the staff requested that the applicant clarify the details of the snubber design requirements for major components.

In its response to **RAI 209-1803, Question 03.09.03-23** dated March 4, 2011, the applicant stated that for hydraulic and mechanical snubbers, the movement is not equivalent so that the lock-up system is different for arresting a movement. The applicant provided a mark-up for Subsection 3.9.3.4.7 to indicate that snubbers are generally hydraulic; however, there are mechanical snubbers which have adequate functionality that is resistance to drift velocity change. Also, snubbers are used at the SG intermediate shell support and upper shell support in the RCL. The support design is performed using the necessary and sufficient number of snubbers to satisfy the SG seismic design. The results generated by using the calculated main coolant piping load to meet the LBB criteria minimize the use of snubbers. The staff determined that these explanations are satisfactory.

In its response to **RAI 209-1803, Question 03.09.03-23**, the applicant also provided changes to be incorporated to DCD Tier 2 Section 3.9.3, Revision 2, as follows:.

The first paragraph in Subsection 3.9.3.4.7 to be changed, to read as follows:

> Snubbers are considered manufactured standard support components. Snubber manufacturers provide various sizes of snubbers and rated loading consistent with ASME Level A, B, C, and D service conditions. Snubbers are generally hydraulic, however, there are mechanical snubbers, which have adequate functionality that is resistance to drift velocity change. Details of snubber design, testing, operation,

maintenance, inspection, and other functional characteristics are presented in this subsection.

The staff finds the proposed DCD changes acceptable since the changes reflect the RAI response. The staff confirmed that the changes in DCD Tier 2, Section 3.9.3, were incorporated into DCD Tier 2 Revision 2. Accordingly, **RAI 209-1803, Question 03.09.03-23, is resolved**.

The applicant submitted its Technical Report MUAP-08012, "US-APWR Sump Strainer Stress Report" dated March 2011, for staff review. MUAP-08012 summarizes the evaluation of the structural components of the ECC and CSS sump strainer. The strainer design is evaluated for compliance with ASME Code allowable stress in accordance to ASME Code, Section III, Division 1, Subsection NC requirements. The staff reviewed sump strainer designs, methodology and the sump strain stress reports and found that the MUAP-08012 is acceptable.

During the April 16 - 17, 2012, audit of the applicant's Recirculation Flow Path Design Change, the staff reviewed Design Specification 4CS-UAP-20120006, "Basic Design Requirement and Specifications for Debris Interceptors" Rev. 0, dated April 12, 2012, which depicts a transfer pipe debris interceptor. The transfer pipe opening into the containment is protected from large debris by a vertical debris interceptor. The audit report is available in, "The U.S. Nuclear Regulatory Commission Report for the Audit Performed Between April 16 - 17, 2012, Regarding the United States – Advanced Pressurized Water Reactor Refueling Water Storage Pit Recirculation Flow Path Design Change," dated October 2, 2012.

In **RAI 943-6526, Question 03.09.03-30**, the staff requested the applicant to describe the design-basis load combinations and the use of jet loads for the debris interceptor stress analysis. In addition, the staff requested the applicant to provide the basis for the methodology and assumptions used to analyze the debris interceptor. The staff also requested the applicant to revise the DCD to include a summary of this information.

In its response to **RAI 943-6526, Question 03.09.03-30**, the applicant stated that the load combinations shown in DCD Tier 2, Tables 3.9-3 and 3.9-4 will be used for the stress analysis of the debris interceptor. The postulated break locations and jet sources as well as the regions affected by jet impingement are identified and evaluated as described in DCD Tier 2 Section 3.6.2, "Determination of Rupture Locations and Dynamic Effects Associated with the Postulated Rupture of Piping." The stress analysis of the debris interceptor is performed using elastic methods for the defined loads, including SSE and jet impingement loads. The applicant also identified that DCD Tier 2, Section 6.2.2.3, "Design Evaluation," will be revised to include this information. Based on the applicant providing the information requested by the staff, the staff found the RAI response acceptable. The staff will confirm the changes in DCD Tier 2, Section 6.2.2.3. Therefore, **RAI 943-6526, Question 03.09.03-30 is being tracked as a Confirmatory Item**.

As a result of the open item for **RAI 209-1803, Question 03.09.03-21**, the staff is unable to finalize its conclusions on component support designs, in accordance with NRC regulations.

3.9.3.4.5 Inspections, Tests, Analyses, and Acceptance Criteria for ASME Code Class 1, 2, and 3 Components, Component Supports, and Core Support Structures

DCD Tier 1, Section 2.3.1, "Design Description," states that components, component supports, and core support structures are analyzed and designed to the requirements of the ASME Code, Section III, based on code classification and ASME Service Level. DCD Tier 1, Section 2.3.1, also identifies the ASME Code Subsections associated with Class 1, 2, and 3, components, component support and their attachments, and core support structures. DCD Tier 1, Table 2.3-3, “Systems with ASME Code, Section III, Class 1, 2, and 3 Piping Systems and Components,” lists systems that have ASME Code, Section III, Class 1, 2, and 3 components, component supports, and core support structures. The associated ITAAC are included in DCD Tier 1, Table 2.3-2, “Piping Systems and Components Inspections, Tests, Analyses, and Acceptance Criteria,” and are evaluated in Section 14.3.3 of this report.

ITAAC for the fabrication, installation, and inspection, and reconciliation with the design requirements of ASME Code Class 1, 2, and 3 components, component supports, and core support structures are provided with specific systems. These ITAAC are evaluated with their respective systems as described in 14.3 of this report. Accordingly, the staff finds that the applicant meets the requirements of 10 CFR 52.47(b)(1) with regards to ASME Code Class 1, 2, and 3 components, component supports, and core support structures.

3.9.3.5 Combined License Information Items

The following is a list of COL item numbers and descriptions from the DCD Tier 2, Table 1.8-2 related to ASME Code Class 1, 2, and 3 components, component supports, and core support structures:

Table 3.9.2-2
US-APWR Combined License Information Items

Item No.	Description	Section
COL 3.9(1)	The COL applicant is to assure snubber functionality in harsh service conditions, including snubber materials (e.g., lubricants, hydraulic fluids, seals).	3.9.3.4.2.5
COL 3.9(10)	The COL applicant is to identify the site-specific active pumps.	3.9.3.3.1

The staff evaluated the above COL Items and concluded that the applicant appropriately lists the information that will be provided by the COL applicant for the APWR plant. The staff found this list acceptable.

3.9.3.6 Conclusions

As a result of the open item for **RAI 209-1803, Question 03.09.03-21** and **RAI 1015-7054, Question 03.09.03-31**, the staff is unable to finalize its conclusions on ASME Code Class 1, 2, and 3 components, component supports, and core support structures, in accordance with NRC regulations.

3.9.4 Control Rod Drive System

3.9.4.1 Introduction

The control rod drive system (CRDS) consists of the control rods and the related mechanical components, which provide the means for mechanical movement. 10 CFR Part 50, Appendix A, GDC 26, requires that one of the independent reactivity control systems use control rods. The rods and the drive mechanism shall be capable of reliably controlling reactivity changes under conditions of normal operation, including AOOs, and under postulated accident conditions. A positive means for inserting the rods should be maintained to ensure appropriate margin for malfunction, such as stuck rods. The applicant's information regarding design criteria; testing programs; summary of method of operation of the CRDS; applicable design codes and standards; design loads and combinations; and operability assurance program is reviewed in this SE Section. This information pertains to the CRDS, which is considered to extend to the coupling interface with the reactivity control elements in the RPV. The review in this section is limited to the control rod drive mechanism (CRDM) portion of the CRDS.

3.9.4.2 Summary of Application

DCD Tier 1: The Tier 1 information associated with the CRDMs is found in Sections 2.4.1.1, "Design Description," and 2.7.5.3.1.3, "Control Rod Drive Mechanism (CRDM) Cooling System," Table 2.4.1-1, "Equipment Key Attributes," and Figure 2.4.1-1 "Reactor General Assembly."

DCD Tier 2: The applicant provided a DCD Tier 2 system description in Section 3.9.4, "Control Rod Drive Systems," summarized here in part, as follows: Section 3.9.4 presents the technical information supporting the design-basis for the CRDM. The primary functions of the CRDMs are to insert or withdraw the rod control cluster assemblies (RCCAs) from the reactor core to control average core temperature and to control changes in reactivity during reactor startup and shutdown. A standard Westinghouse design utilized in currently operating reactors provides the basis for the US-APWR CRDM. The US-APWR CRDM is a magnetically operated jack consisting of an arrangement of three electromagnets energized in a controlled sequence to insert or withdraw the rod control assemblies in the reactor core in discrete steps. The CRDM is designed to release the rod control assemblies during any part of the power cycle sequencing in the event that electrical power to the electromagnets is interrupted. When released from the CRDM, the rod control assemblies fall by gravity into a fully inserted position within the reactor core. The pressure housing subassembly of the CRDM forms a part of the RCPB. The CRDM pressure housing is constructed in conformance with the requirements of 10 CFR 50.55a, including design, analysis, materials, fabrication, and QA requirements for Class 1 components specified in Section III of the ASME Code as incorporated in 10 CFR 50.55a.

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

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

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

Technical Reports: Technical reports associated with DCD Tier 2, Section 3.9.4 are as follows:

1. MUAP-09002-P, "Summary of Seismic and Accident Load Conditions for Primary Components and Piping," Revision 2, issued December 2010.

2. MUAP-09009-P, "Summary of Stress Analysis Results for the US-APWR Control Rod Drive Mechanism," Revision 1, issued February 2011

Cross-cutting Requirements (TMI, USI/ GSI, Op Ex): There are no cross-cutting requirements for this area of review.

US-APWR Interface Issues Identified in the DCD: See DCD Tier 2, Table 1.8-2, "Compilation of All Combined License Applicant Items for Chapters 1-19."

Site Interface Issues Identified in the DCD: See DCD Tier 2, Table 1.8-1, "Significant Site Specific Interfaces with the Standard US-APWR Design."

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

3.9.4.3 Regulatory Basis

The relevant Commission regulations for this area of review, and the associated acceptance criteria, are given in Section 3.9.4, "Control Rod Drive Systems," Revision 3, issued March 2007, of NUREG-0800 (the SRP) and are summarized below. Review interfaces with other SRP sections can be found in Section 3.9.4 of NUREG-0800.

1. GDC 1 and 10 CFR 50.55a, as they relate to the CRDS, require that the CRDS be designed to quality standards commensurate with the importance of the safety functions to be performed.

2. GDC 2, as it relates to CRDS, requires that the CRDS be designed to withstand the effects of an earthquake without loss of capability to perform its safety functions.

3. GDC 14, as it relates to CRDS, requires that the RCPB portion of the CRDS be designed, constructed, and tested for the extremely low probability of abnormal leakage or gross rupture.

4. GDC 26, as it relates to CRDS, requires that the CRDS be one of the independent reactivity control systems that are designed with appropriate margin to assure its reactivity control function under conditions of normal operation, including anticipated operational occurrences (AOOs).

5. GDC 27, as it relates to CRDS, requires that the CRDS be designed with appropriate margin, and in conjunction with the emergency core cooling system, be capable of reliably controlling reactivity changes and cooling the core under postulated accident conditions.

6. GDC 29, as it relates to CRDS, requires that the CRDS, in conjunction with protection systems, be designed to assure an extremely high probability of accomplishing its safety functions in the event of AOOs.

Acceptance criteria adequate to meet the above requirements include:

1. The descriptive information is determined to be sufficient provided the minimum requirements for such information meet Section 3.9.4 of RG 1.29.

2. Construction (as defined in NCA-1110 of Section III of the ASME Code) should meet the following codes and standards utilized by the nuclear industry which have been reviewed and found acceptable:

   a. For pressurized portions of equipment classified as Quality Group (QG) A, B, C (RG 1.26): Section III of the ASME Code, Class 1, 2, or 3 as appropriate.

b. For pressurized portions of equipment classified as QG D as discussed in RG 1.26:

   i. Section VIII, Division 1, of the ASME Code for vessels and pump casings.
   ii. For piping systems (ANSI):

B16.5	Steel Pipe Flanges and Flanged Fittings
B16.9	Steel Butt Welding Fittings
B16.11	Steel Socket Welding Fittings
B16.25	Butt Welding Ends
B16.34	Steel Valves with Flanged and Butt Welding Ends
B31.1	Power Piping
MSS-SP-25	Marking for Valves, Fittings, Flanges, and Unions

c. For non-pressurized equipment (Non-ASME Code):

   Design margins presented for allowable stress, deformation, and fatigue should be equal to or greater than margins for other plants of similar design with successful operating experience. A justification of any decreases in design margins should be provided.

3. For the various design and service conditions defined in NB-3113 of Section III of the ASME Code, load combination sets are as given in SRP Section 3.9.3. The stress limits applicable to pressurized and non-pressurized portions of the CRDSs should be as given in SRP Section 3.9.3 for the response to each loading set. For boiling water reactors (BWRs), the CRDS design should adequately consider water hammer loads to assure that system safety functions can be achieved.

4. The operability assurance program will be acceptable provided the observed performance as to wear, functioning times, latching, and ability to overcome a stuck rod meets system design requirements provided in the FSAR.

3.9.4.4 Technical Evaluation

The staff reviewed the information in DCD Tier 2, Section 3.9.4, related to the criteria used to ensure the structural integrity of the CRDS during normal operation, under postulated accident conditions, and during seismic events. The staff reviewed the criteria for conformance to the acceptance criteria in SRP Section 3.9.4.

The guidance in SRP Section 3.9.4, Part I, Item 4, states that a review of the applicant's plans for the conduct of an operability assurance program or that references previous test programs or standard industry procedures for similar apparatus is performed. The staff was unable to locate a reference documenting CRDM qualification to operate in a reactor environment. Therefore, in **RAI 107-1293, Question 03.09.04-1, Subquestion 1293-01**, the staff requested one or more references that document CRDM qualification to operate in a reactor environment. The staff noted that this could include either details of an operability assurance program for the US-APWR CRDM that covers all the items contained in the guidance in SRP Section 3.9.4, Part I, Item 4, or reference to a previous testing program that has been approved by the NRC, including:

1. A description of any differences in the two designs and their effects on the applicability of the previous operability tests,

2. Identification of any differences in the operating conditions or loads, and

3. A comparison of the LOCA plus SSE loads, and a description of the basis for the 60-year lifetime for the US-APWR CRDM internals.

In its response to **RAI 107-1293, Question 03.09.04-1, Subquestion 1293-01**, dated August 11, 2011, the applicant stated that the US-APWR CRDM is based on the L-106A type CRDM which has been used in many operating plants in the U.S. and Japan. The response described the differences between the L-106A CRDM and the US-APWR CRDM, including: chrome carbide coating on the latch arms, cap and rod travel housing machined from one piece of material, latch housing and CRDM adapter machined from one piece of material, butt welding of the latch housing to the CRDM nozzle of the RV head, and extension of the drive rod assembly for 14 foot (4.3 m) long fuel. The chrome carbide coating is thin enough to not affect operability and was tested to show an improvement in wear resistance. The pressure housing improvements results in an extremely low probability of primary coolant system leakage and operability is not affected. The drive line weight is about 10 percent heavier than the current 4-loop drive line weight, but is less than the test weight used in the endurance test of the CRDM. The new pressure housing design will be constructed according to the ASME B&PV Code, and will undergo hydrostatic testing prior to plant startup. A sentence was added to DCD Tier 2, Revision 2, Subsection 3.9.4.1.1, "CRDM," that these design changes do not affect operability. The staff confirmed that DCD Revision 2 incorporated these changes.

Pressure of the primary coolant water of US-APWR is the same as current 4-loop plants. Temperature and core average core coolant velocity of US-APWR are slightly lower than current 4-loop plants. This change was also verified in the endurance test of the CRDM, which was conducted in high pressure and high temperature with no core flow to allow for conservatism.

The effects of LOCA and SSE loads are reported in the applicant's Technical Report MUAP-09009-P Revision 1. The estimated deflection of the CRDM pressure housing at ASME Code, Section III Level D condition is within the 1.18 in. (3.00 cm) allowable design limit. The basis for assuring the 60 year design life of the US-APWR CRDM was that the stress and fatigue strength of the pressure boundary is evaluated by application of design transients covering 60 years of expected plant life, and the stress analysis results of the pressure housing were found to be within the allowable limit. The results were reported in MUAP-09009-P. The integrity of the CRDM latch mechanism was confirmed by the endurance test. The coil stack assembly and drive rod assembly are not required to have a 60 year operating time. These assemblies can be replaced during the life of the plant.

The staff finds that the applicant's response addresses the operability of the design changes, difference in operating conditions, effects of LOCA and SSE loads, and the applicant's basis for a 60-year lifetime through analysis and testing. However, the applicant informed the staff that the response to **RAI 107-1293, Question 03.09.04-1, Subquestion 1293-01** is being revised to account for updated seismic information. As the staff is awaiting this revised response, **RAI 107-1293, Question 03.09.04-1, Subquestion 1293-01, is being tracked as an Open Item**.

The acceptance criteria in SRP Section 3.9.4, Part II, states that the operability assurance program will be acceptable provided that observed performance as to wear, functioning times, latching, and ability to overcome a stuck rod meet system design requirements. DCD Tier 2, Section 3.9.4.4, “CRDS Operability Assurance Program,” states that the capability of the CRDM functions, including withdrawal, insertion, and trip delay, are confirmed by both lead unit tests and production unit tests to demonstrate that the design specification requirements are met prior to shipment. System design requirements for cold stepping, hot and cold trip delay times, and hot stepping are given in DCD Tier 2, Section 3.9.4, and preoperational and operational tests are discussed in DCD Tier 2, Section 14.2, “Initial Plant Test Program,” but there is no discussion of wear and overcoming a stuck rod. In **RAI 107-1293, Question 03.09.04-1, Subquestion 1293-02**, the staff requested a discussion of how wear and overcoming a stuck rod are addressed in the operability assurance program.

In its response to **RAI 107-1293, Question 03.09.04-1, Subquestion 1293-02**, dated December 19, 2008, the applicant stated that wear had been demonstrated in endurance testing of the US-APWR CRDM, and past successful operation of the L-106A design, as well as evaluation of thermal expansion of the latch assembly, latch arm, and coil assembly in designing for proper clearance to avoid a stuck rod condition, address the ability to overcome a stuck rod. Additional text was added at the end of DCD Tier 2, Section 3.9.4.3, “Design Loads, Stress Limits, and Allowable Deformations,” in DCD Revision 2, to state this more clearly. Based on this past experience, endurance testing, and the controlled clearances in the latch assembly parts, the staff finds this response acceptable. The staff confirmed that the DCD changes were incorporated into DCD Revision 2. Accordingly, **RAI 107-1293, Question 03.09.04-1, Subquestion 1293-02, is resolved**.

SRP Section 3.9.4, Part II, SRP Acceptance Criteria 2.C, states that for non-pressurized equipment (non-ASME Code), design margins presented for allowable stress, deformation, and fatigue should be equal to or greater than margins for other plants of similar design with successful operating experience. A justification of any decreases in design margins should be provided. DCD Tier 2, Subsection 3.9.4.2.3, “Internal Component Requirements,” states that the non-pressurized portion of the US-APWR CRDM is non ASME Code, Section III limited; however, no description is provided on the criteria for structural analyses, design margins, or how design margins were obtained. In **RAI 107-1293, Question 03.09.04-1, Subquestion 1293-03**, the staff requested a description of results of stress, deflection, and fatigue analyses for the non-pressurized portion of the US-APWR CRDM, including design loads and loading combinations, values of material properties used and the justification for their basis, stress, deflection, and fatigue criteria used and the justification for their basis, and design margins and how they compare with previous designs.

In its response to **RAI 107-1293, Question 03.09.04-1, Subquestion 1293-03**, dated December 19, 2008, the applicant stated that the design of the nonsafety-related components was confirmed by test, rather than analysis, and that the latch mechanism has been tested for functionality to ten millions steps. The design endurance criterion of the latch mechanism is six million steps, accommodating a margin of 60 years. The first sentence of the second paragraph of DCD Tier 2, Subsection 3.9.4.2.3, and the last bullet of the first paragraph of DCD Tier 2, Subsection 3.9.4.2.1, “CRDM Functional Requirements,” of DCD Revision 2, were revised to include this information. The staff found the response acceptable since the endurance testing of the latch mechanism and design endurance criterion accommodating a margin of 60 years is an acceptable alternative to analysis of the latch mechanism. The staff confirmed the DCD changes were incorporated into DCD Revision 2. Accordingly, **RAI 107-1293, Question 03.09.04-1, Subquestion 1293-03, is resolved**.

The guidance in SRP Section 3.9.4, Part I, Item 1, states that the descriptive information, including design criteria and testing programs, is reviewed to permit an evaluation of the adequacy of the system to perform its mechanical function properly. In **RAI 107-1293, Question 03.09.04-1, Subquestion 1293-06**, the staff requested the applicant to provide the basis of the 1.18 in. (3.00 cm) allowable rod travel housing deflection during a seismic event as described in DCD Tier 2, Section 3.9.4.3, "Design Loads, Stress Limits, and Allowable Deformations," and to provide a description of how it has been verified that the rod control assemblies would be inserted into the core at this deflection. In its response to **RAI 107-1293, Question 03.09.04-1, Subquestion 1293-06**, dated May 29, 2009, the applicant stated that the 1.18 in. (3.00 cm) allowable deflection was obtained by testing and included a margin, and the estimated deflection was obtained by analysis and is less than the allowable value. These are documented in MUAP-09009-P and Attachment 1 to the response. The staff finds the response acceptable since the design allowable deflection is less than the tested deflection and includes a margin; therefore, **RAI 107-1293, Question 03.09.04-1 Subquestion 1293-06 is resolved**.

In **RAI 107-1293, Question 03.09.04-1, Subquestion 1293-07**, the staff requested that the criteria used for CRDM operational capability, including the margin, following exposure to the combined effects of a LOCA and an SSE be provided. In its response to **RAI 107-1293, Question 03.09.04-1, Subquestion 1293-07**, dated May 29, 2009, the applicant stated that the CRDM operational capability, including a margin, was documented in test and stress reports contained in Attachment 1 of the response and MUAP-09009-P, respectively. The estimated deflection of the CRDM pressure housing at an ASME Code, Section III Level D condition is less than the allowable limit of 1.18 in. (3.00 cm). Because the actual deflection value was conservative, the staff found the response acceptable. Accordingly, **RAI 107-1293, Question 03.09.04-1, Subquestion 1293-07 is resolved**.

On August 11, 2011, the applicant submitted an amended response to **RAI 107-1293, Question 03.09.04-1 Subquestions 1293-01, 1293-06, and 1293-07;** updating the maximum CRDM deflection due to the LOCA and seismic loads. This resulted in an increased maximum CRDM deflection which significantly reduced the margin between the deflection calculated by analysis and the design limit. In **RAI 848-6093, Question 03.09.04-14**, the staff requested the applicant to provide the justification for the increase in maximum CRDM deflection due to the LOCA and seismic loads.

In its response to **RAI 848-6093, Question 03.09.04-14**, dated November 18, 2011, the applicant stated that in considering the CRDM deflection due to LOCA and seismic loads, the latest results show that the deflection due to seismic loads is predominant over the deflection due to LOCA loads. The seismic response of the CRDM was calculated using acceleration time histories at the R/B complex, based on eight detailed soil conditions described in DCD Tier 2, Revision 3, Section 3.7.1, "Seismic Design Parameters." The CRDM dynamic response analysis was reported in the applicant Technical Report MUAP-09002-P, Revision 2. Subsequently, the applicant informed the staff that the responses to **RAI 107-1293, Question 03.09.04-1, Subquestions 1293-01, 1293-06, and 1293-07** and **RAI 848-6093, Question 03.09.04-14** are being revised to account for updated seismic information. As the staff is awaiting these revised responses, **RAI 107-1293, Question 03.09.04-1, Subquestions 1293-01, 1293-06, and 1293-07 and RAI 848-6093, Question 03.09.04-14 are being tracked as Open Items**.

GDC 26 and 27, as they relate to the CRDS, require that the CRDS be designed to withstand the effects of an earthquake, and be designed with appropriate margin to assure its functionality under conditions of postulated accident conditions. In **RAI 107-1293, Question 03.09.04-1,**

Subquestion 1293-08, the staff requested the applicant provide references in the DCD that show that the CRDM design conforms to its design criteria and limits; and if the design verification includes loading combination analysis in conjunction with testing, that a reference also be included. In its response to **RAI 107-1293, Question 03.09.04-1, Subquestion 1293-08**, dated December 19, 2008, the applicant added a reference for MUAP-09009-P in Section 3.9.4.4 of DCD Revision 2. The staff finds the response acceptable since the referenced stress report addresses the staff's concern. The staff confirmed that the DCD changes were incorporated into DCD Revision 2. Accordingly, **RAI 107-1293, Question 03.09.04-1, Subquestion 1293-08, is resolved**.

DCD Tier 2, Section 3.9.4.2.1 states that the "rod drop time…is evaluated by analysis," however, no analysis is included or referenced in the application. In **RAI 107-1293, Question 03.09.04-1, Subquestion 1293-09**, the staff requested evidence be provided that the insertion and withdrawal times in the stepping mode, and the drop times meet the design requirements, and that the design requirements for these functions, their bases, and margins be provided. In its response to **RAI 107-1293, Question 03.09.04-1, Subquestion 1293-09** dated December 19, 2008, the applicant stated that the design requirements from the safety analysis and the functional requirements are identical. The functionality of the US-APWR CRDM is assured through many years of operating experience in Japan. Production tests are performed on all CRDMs prior to shipment to demonstrate that the design specification requirements are met. The effect for the rod drop time is evaluated by the calculated deflection of the CRDM pressure housing and test results, which is the basis of the deflection criteria of the CRDM pressure housing. The staff found the production tests adequately demonstrate that the design specification required step and drop times will be met. The staff further found the calculated deflection value for the CRDM pressure housing to be conservative in comparison to the design allowable deflection value. Based on the testing discussed above, the staff found the RAI response to be acceptable with the exception that the applicant did not incorporate necessary information into the DCD. Therefore, the staff closed as unresolved **RAI 107-1293, Question 03.09.04-1, Subquestion 1293-09** and issued follow-up **RAI 835-6060, Question 03.09.04-13**, requesting the applicant to update the DCD with its response to **RAI 107-1293, Question 03.09.04-1, Subquestion 1293-09**. The applicant's response to **RAI 835-6060, Question 03.09.04-13**, dated November 2, 2011, proposed to update the DCD with its response to **RAI 107-1293, Question 03.09.04-1, Subquestion 1293-09**. The staff finds the response acceptable since the applicant proposed to incorporate the requested information into the DCD. **RAI 835-6060, Question 03.09.04-13 is being tracked as a Confirmatory Item** pending the applicant updating the DCD to incorporate the proposed changes.

DCD Tier 2, Section 4.6.3, "Testing and Verification of the CRDS," lists four stages of tests planned for verifying proper function of the CRDS, which are stated to be in DCD Tier 2, Section 3.9.4.4 and DCD Tier 2, Section 14.2. These sections give some information on pre-shipment and pre-operational testing, but none on periodic in-service or post-refueling startup tests. In **RAI 107-1293, Question 03.09.04-1, Subquestion 1293-10**, the staff requested clarification of whether all CRDMs go through the functional verification tests, and at what stage. In its response to **RAI 107-1293, Question 03.09.04-1, Subquestion 1293-10**, the applicant dated December 19, 2008, stated that all CRDMs undergo in-service and post-refueling functional testing. In addition, the applicant inserted a "Post-Refueling Startup Test" at the end of DCD Tier 2, Section 3.9.4.4 of DCD Revision 2. The proposed description for the newly inserted test, "The stepping and the drop tests are performed as in-service/post-refueling tests," was not clear as to exactly which tests are being performed and what the frequencies are. Therefore, the staff closed as unresolved **RAI 107-1293, Question 03.09.04-1, Subquestion 1293-10** and issued follow-up **RAI 570-4428, Question 03.09.04-4**, requesting that the applicant clarify the tests to be performed, their respective frequencies, and the criteria which apply to the post-refueling startup test.

In its response to **RAI 570-4428, Question 03.09.04-4**, dated May 19, 2010, the applicant stated that the stepping and rod drop tests are performed as post-refueling startup tests. Additionally, the stepping test is also performed periodically during plant operation. The frequencies and criteria are specified in Section 14.2, which refers to Chapter 16. The staff found the response acceptable since it clarified the tests to be performed and their frequencies. However, the applicant proposed DCD changes that were not consistent with the response. Therefore the staff closed as unresolved **RAI 570-4428, Question 03.09.04-4** and issued follow-up **RAI 604-4775, Question 03.09.04-7**, requesting the applicant address the difference between the response and the DCD markup. In its response to **RAI 604-4775, Question 03.09.04-7**, dated July 28, 2010, the applicant corrected the proposed DCD change, including the revision to the Post-Refueling Startup Test. The staff confirmed that, the applicant incorporated an appropriate change to DCD Revision 3. Accordingly, **RAI 604-4775, Question 03.09.04-7, is resolved**.

DCD Tier 2, Revision 2, Section 3.9.4.4, lists production tests which are performed on all CRDM units before shipment. After the units are installed and prior to fuel loading, CRDM preoperational system testing is performed as required by DCD Tier 2, Section 14.2. However, there was no information regarding testing of individual CRDM components to ensure their proper installation prior to preoperational system testing. In **RAI 570-4428, Question 03.09.04-6**, the staff requested the applicant to describe what testing or checks are performed to ensure the proper installation of the individual CRDM components prior to system testing. Also, the staff requested the applicant to describe what testing or checks are included in the two prerequisites for the preoperational tests. In its response to **RAI 570-4428, Question 03.09.04-6**, dated May 19, 2010, the applicant provided details on the on-site checks of the CRDMs prior to initial startup testing. The applicant also clarified the testing and checks included in the two prerequisites for the preoperational tests. However, the staff found that no changes were made to the DCD to address these details. Therefore, the staff closed as unresolved **RAI 570-4428, Question 03.09.04-6** and issued follow-up **RAI 604-4775, Question 03.09.04-8**, to request that the applicant update the DCD with on-site checks and testing and checks included in the two prerequisites. In its response to **RAI 604-4775, Question 03.09.04-8**, dated July 28, 2010, the applicant stated that DCD Tier 2, Section 3.9.4.4, will be revised to include several bullets under the heading "On-site checks." In addition, in addition the applicant stated that DCD Tier 2, Section 14.2 will be revised to include the details concerning the prerequisites for the preoperational tests in DCD Revision 3. The staff finds the response acceptable with regard to the changes to DCD Tier 2, Section 3.9.4.4 since it clarified the onsite checks. However, the staff determined that additional clarification was needed regarding preoperational tests and the associated prerequisites. Therefore, the staff closed as unresolved **RAI 570-604-4775, Question 03.09.04-8** and in **RAI 679-4985, Question 03.09.04-11** the staff requested the applicant to update DCD Tier 2, Section 3.9.4.4 with a summary of the preoperational tests performed and to revise the prerequisite markup in DCD Tier 2, Section 14.2 to be consistent. In its response to **RAI 679-4985, Question 03.09.04-11**, the applicant proposed updates to DCD Tier 2, Section 3.9.4.4 and DCD Tier 2, Section 14.2. The staff found the response acceptable since the applicant's proposed markup incorporates the requested information into the DCD. **RAI 679-4985, Question 03.09.04-11 is being tracked as a Confirmatory Item** pending the applicant updating the DCD to incorporate the proposed changes.

During its review of DCD Tier 2, Chapter 3, the staff noted that the seismic qualification of the CRDM for the US-APWR standard design may not be adequate. GDC 2 states that SSCs important to safety shall be designed to withstand the effects of natural phenomena such as earthquakes, combined with appropriate effects of normal and accident conditions, without loss of capability to perform their safety functions. DCD Tier 2, Subsection 3.9.4.2.2, "Pressure Housing Requirements," describes how the CRDM pressure boundary components, such as CRDM latch housing and CRDM rod travel housing are designed and qualified to meet the ASME Code requirements. However, as stated in DCD Tier 2, Sections 3.9.4.2.3 and 3.9.4.2.4, "Coil Stack Assembly Requirements," the CRDM non-pressure boundary components, including the latch assembly and coil stack assembly are not explicitly qualified for seismic events. The potential for these seismic components to not work properly such as in the jamming of the latch mechanism or the malfunction of the coils due to seismic events may prevent the control rods from dropping as designed in a seismic event. In **RAI 569-4433, Question 03.09.04-2**, the staff requested the applicant to explain why the latch mechanism and coil stack assembly does not need to be seismically qualified to comply with GDC 2, or to revise the seismic classifications of the CRDM components to ensure adequate seismic qualification for the safety functions of the CRDS. In its response to **RAI 569-4433, Question 03.09.04-2**, dated May 13, 2010, the applicant described the construction of the latch assembly, gave a justification for the claim that the latch assembly will not jam, and explained that the latch assembly is supported by the CRDM pressure housing, which is seismic Category I. The applicant stated that the latch assembly is classified as a nonsafety-related component and proposed deletions to DCD Tier 2, Subsection 3.9.4.2.3 in DCD Revision 3.

The staff found that the explanation provided was inadequate and closed as unresolved **RAI 569-4433, Question 03.09.04-2** and issued follow-up **RAI 604-4775, Question 03.09.04-9**, requesting the applicant to clarify how the latch assembly cannot be jammed. The staff also requested the applicant to explain why the latch mechanism does not need to be seismically qualified to comply with GDC 2, or to revise the seismic classifications of the CRDM components to ensure adequate seismic qualification for the safety functions of the CRDS. In addition, the staff requested the applicant to clarify the deletions to be incorporated into DCD Revision 3 of the second paragraph of DCD Tier 2, Subsection 3.9.4.2.3.

In its response to **RAI 604-4775, Question 03.09.04-9**, dated July 28, 2010, the applicant stated that the integrity of the latch assembly was confirmed by the endurance test described in Reference 1 of the December 19, 2008, response to **RAI 107-1293, Question 03.09.04-1, Subquestion 1293-01**. The latch assembly was continually operated in ten million steps during the test, and the motion of the latch assembly was not jammed. The applicant is not aware of a jamming event for type L-106A latch assemblies. However, a transitory behavior such as miss stepping or slipping occurs at a fairly low rate of occurrence. Several research programs for the control rod behavior during earthquakes had been carried out in Japan. One of those test results was already shown in the Attachment 1 of the May 29, 2009, response to **RAI 107-1293, Question 03.09.04-1, Subquestion 1293-06**. Another test result was published in the Nuclear Engineering International's April 1990, issue. Those test results show that the control rods were released into the core during earthquakes, and that the release function of the latch was maintained during earthquakes.

Therefore the staff closed as unresolved **RAI 604-4775, Question 03.09.04-9,** and issued follow-up **RAI 679-4985, Question 03.09.04-12**, requesting the applicant to provide the test result referenced in the response (Nuclear Engineering International, April 1990 issue). The control rod drop function test does not provide assurance that latches will function after an earthquake as the test only assessed dropped rods during earthquake (excitation). The staff requested that the applicant clarify their response in Section 3.2, Test Procedures of Attachment-1 of the May 29, 2009, response to **RAI 107-1293, Question 03.09.04-1,**

Subquestion 1293-06, which stated "When it…obtained the target amplitude, the insertion time was measured with a rod position indicator by dropping the rod cluster control assembly (RCCA) with the drive rod of CRDM from all withdrawal positions to all insert positions." The staff also requested clarification on the duration of the excitation applied during the rod drops, including the time scale in relation to the release of latches, and greater detail in general with regards to test procedures, graphs, and data collected during the test.

In its response to **RAI 679-4985, Question 03.09.04-12**, dated February 9, 2011, the applicant stated the two documents would be available for audit. Further, the referenced statement of Section 3.2, Test Procedures of Attachment-1 of the May 29, 2009, response to **RAI 107-1293, Question 03.09.04-1, Subquestion 1293-06,** concerning test procedure is clarified as follows. First, the equipment was excited by a sine wave corresponding to the natural frequency of the equipment at the fully withdrawn position of the drive rod. During the resonance condition of the target amplitude, the power to the CRDM was interrupted, and the drive rod with RCCA was dropped. Starting when power was interrupted, the rod drop time from the full out position to the fully inserted position was measured by the traces generated by the rod position indicator. The duration of the excitation applied during each of the rod drop tests, including the timescale in relation to the release of the latches, and greater detail in general with regard to test procedures, graphs, and data collected during test would be made available during the audit.

The staff conducted the audit on March 17, 2011, to review the raw data for “Summary of the Control Rod Drop Function Test Results in Japan,” which was the paper published in the Nuclear Engineering International April 1990 Issue, and to understand the design and qualification of latches for the function of CRDM under seismic excitation. In addition, the staff reviewed the “Joint Research on the Evaluation of the Functionality of Active Components in an Earthquake, Final Report (Supplementary Volume), Control Rod Drop Function Test,” issued March 1983, and noted that in the area concerning the latch housing, and subsequently the latches, there were no response spectra in the report. In the report, this area was designated as “A4.” The applicant could not locate the test response spectra for S2 excitation at “A4” location. In addition, the required response spectra (RRS) were based on Japanese Nuclear Safety Commission seismic requirements. The staff requested the applicant to provide the RRS for the US-APWR CRDM housing location and determine if it can be bounded by the test response spectra (TRS) for S2 excitation at the “A4” housing location. The staff also requested details of a test carried out by the Japanese Nuclear Energy Safety Organization (JNES) that was mentioned in Attachment-1 of the May 29, 2009, in the applicant’s response to **RAI 107-1293, Question 03.09.04-1, Subquestion 1293-06**. A summary of the staff’s audit is available in, “U.S. Nuclear Regulatory Commission Audit Report for the Audit Performed During March 17, 2011, Regarding Section 3.9.4, ‘Control Rod Drive Mechanism,’ of the United States – Advanced Pressurized Water Reactor Design Control Document,” dated March 23, 2012.

The applicant was unable to locate the TRS at the “A4” location during the audit, but in the amended response to **RAI 604-4775, Question 03.09.04-9**, dated August 11, 2011, the applicant provided details on the JNES test referenced in the Attachment-1 of the May 29, 2009, response to **RAI 107-1293, Question 03.09.04-1, Subquestion 1293-06**. In the JNES test, it was confirmed that the CRDM operates normally when releasing the control rod under the condition of 3.3 times the design-basis extreme earthquake S2, which is similar to an SSE. Although there was no measured data that corresponds to the latch housing location in the test, a comparison between the response spectrum generated for the latch housing location that corresponds to the antinodes in the analysis and the response spectrum generated for middle seismic support location that corresponds to a node in the test was determined to be a conservative evaluation. The TRS generated from the test results nearly exceeds the RRS generated from the analysis results of the US-APWR. Under severe conditions for the frequency range where the CRDM responds largely to the large input, it was confirmed that the TRS exceeds the RRS. Thus, staff determined that the CRDM functions, including the latch assembly functions, can be maintained at the assumed seismic conditions of the US-APWR. The staff finds the applicant response acceptable because the TRS exceeds the RRS under SSE conditions where the CRDM is most responsive to seismic input. Accordingly, **RAI 679-4985, Question 03.09.04-12, is resolved**.

In **RAI 570-4428, Question 03.09.04-5**, the staff requested the applicant to clarify the preoperational test discussed in DCD Tier 2, Revision 2, Section 3.9.4.4, since it actually described a startup test and not a preoperational test. In its response to **RAI 570-4428, Question 03.09.04-5**, dated May 19, 2009, the applicant proposed changes to DCD Tier 2, Section 3.9.4.4 and DCD Tier 2, Section 4.6.3 to rename the preoperational test to initial startup test. However, this change resulted in an inconsistency between DCD Tier 2, Section 3.9.4.4 and DCD Tier 2, Section 4.6.3. Therefore the staff closed as unresolved **RAI 570-4428, Question 03.09.04-5** and in **RAI 679-4985, Question 03.09.04-10**, the staff requested that DCD Tier 2, Section 4.6.3 be updated to include preoperational tests, inserted before initial startup tests.

In its response to **RAI 679-4985, Question 03.09.04-10, dated April 25, 2011**, the applicant stated that the bullet, "Preoperational tests," will remain, and the bullet, "Initial startup tests," will be inserted after "Preoperational tests." The staff finds that the applicant's response and proposed updates to the DCD address the staff's concern. Accordingly, **RAI 679-4985, Question 03.09.04-10 is being tracked as a Confirmatory Item** pending the applicant updating the DCD to incorporate the proposed changes**.**

3.9.4.5 Combined License Information Items

No applicable items were identified in the DCD for this area of review.

3.9.4.6 Conclusions

As a result of the open items for **RAI 107-1293, Question 03.09.04-1, Subquestions 1293-01, 1293-06, and 1293-07 and RAI 848-6093, Question 03.09.04-14**, the staff is unable to finalize its conclusions on Section 3.9.4 related to the CRDS, in accordance with NRC regulations.

3.9.5 Reactor Pressure Vessel Internals

3.9.5.1 Introduction

The reactor pressure vessel internals include the core support and internal structures and structural and mechanical elements inside the reactor vessel with the following exceptions:

- Reactor fuel elements and the reactivity control elements.
- Control rod drive elements.
- In-core and thermocouple instrumentation.

Support structures and guide tubes associated with the above items are included in this section.

3.9.5.2 Summary of Application

DCD Tier 1: The Tier 1 information associated with this section is found in DCD Tier 1, Section 2.4.1, “Reactor System.”

DCD Tier 2: The applicant has provided a DCD Tier 2 system description in Section 3.9.5, “Reactor Pressure Vessel Internals,” summarized here in part, as follows:

> With respect to the RPV internals, this section discusses reactor internals design arrangements, design loads, acceptance criteria, computational methods, confirmation of computational methods, interface requirements, and preservice and ISI plans.

ITAAC: The ITAAC associated with DCD Tier 2, Section 3.9.5 are given in DCD Tier 1, Section 2.4.1.2, “Inspections, Tests, Analyses, and Acceptance Criteria.”

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

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

Technical Reports: The Technical Reports associated with DCD Tier 2, Section 3.9.5 is:

1. MUAP-07022-P, “Reactor Vessel Lower Plenum 1/7 Scale Model Flow Test Report,” Revision 0, issued June 2008.

2. MUAP-07023-P, “APWR Reactor Internals 1/5 Scale Model Flow Test Report,” Revision 1, issued May 2009.

3. MUAP-07027-P, “Comprehensive Vibration Assessment Program for the US-APWR Reactor Internals,” Revision 1, issued May 2009.

4. MUAP-09002-P “Summary of Seismic and Accident Load Conditions for Primary Components and Piping” Revision 1, issued March 2009.

5. MUAP-09004-P, “Summary of Stress Analysis Results for Core Support Structures,” Revision 0, issued March 2009.

6. MUAP-11012-P, “US-APWR RCCA Insertion Limit Load Test Report,” Revision 0, issued March 2011.

Cross-cutting Requirements (TMI, USI/ GSI, Op Ex): There are no cross-cutting requirements for this area of review.

US-APWR Interface Issues Identified in the DCD: See DCD Tier 2, Table 1.8-2, “Compilation of All Combined License Applicant Items for Chapters 1-19.”

Site Interface Issues Identified in the DCD: See DCD Tier 2, Table 1.8-1, “Significant Site Specific Interfaces with the Standard US-APWR Design.”

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

3.9.5.3 Regulatory Basis

The relevant Commission regulations for this area of review, and the associated acceptance criteria are given in Section 3.9.5, “Reactor Pressure Vessel Internals,” Revision 3, issued March 2007, of NUREG-0800, the SRP, and are summarized below. Review interfaces with other SRP sections can be found in Section 3.9.5 of NUREG-0800.

1. GDC 1 and 10 CFR 50.55a, as they relate to the design, fabrication, erection, and testing of SSCs in accordance with quality standards commensurate with the importance of the safety function to be performed.

2. GDC 2, as it relates to the ability of SSCs without loss of capability to perform their safety function, to withstand the effects of natural phenomena, such as earthquakes, tornadoes, floods, and the appropriate combination of all loads.

3. GDC 4, as it relates to the protection of SSCs against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that may result from equipment failures and from events and conditions outside the nuclear power unit.

4. GDC 10, requires that reactor internals be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of AOOs.

5. 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 has been constructed and will operate in accordance with the DC, the provisions of the Atomic Energy Act, and the NRC’s regulations.

Acceptance criteria adequate to meet the above requirements include:

1. Requirements for loads, loading combinations, and limits applicable to those portions of reactor internals constructed to Subsection NG of the ASME Code, Section III are presented in SRP Section 3.9.3, “ASME Code Class 1, 2, and 3 Components, and Component Supports, and Core Support Structures.”

2. The design and construction of the core support structures should comply with the requirements of Subsection NG, "Core Support Structures," of the ASME Code, Section III and SRP Section 3.9.3.

3. The design criteria, loading conditions, and analyses that provide the bases for the design of reactor internals other than the core support structures should meet the guidelines of ASME Code, Section III, NG-3000 and be constructed not to affect the integrity of the core support structures adversely (ASME Code, Section III, NG-1122). If other guidelines (e.g., manufacturer standards or empirical methods based on field experience and testing) are the bases for the stress, deformation, and fatigue criteria, those guidelines should be identified and their use justified.

4. Deformation limits for reactor internals should be established by the applicant and presented in the safety analysis report. The basis for these limits should be included. The stresses of these displacements should not exceed the specified limits. The requirements and guidelines for dynamic analysis of these components are addressed in SRP Section 3.9.2, "Dynamic Testing and Analysis of Systems, Structures, and Components."

5. The reactor internals should be designed to accommodate asymmetric blowdown loads from postulated pipe ruptures. The applicant's evaluation of such loads should demonstrate that they do not exceed the limits imposed by the applicable codes and standards. Where double-ended guillotine break of reactor coolant piping is postulated, acceptable criteria for evaluating loading transients and structural components are specified in NUREG-0609, "Asymmetric Blowdown Loads on PWR Primary Systems: Resolution of Generic Task Action Plan A-2," issued January, 1981.

6. Potential adverse flow effects of flow-induced vibration and acoustic resonances on reactor internals (including the steam dryer in BWRs) should be adequately addressed in accordance with relevant criteria stated in SRP Section 3.9.5, Appendix A.

3.9.5.4 Technical Evaluation

3.9.5.4.1 Summary of Design Arrangement

The applicant discussed the physical or design arrangements of all reactor internals structures, components, assemblies, and systems, including the manner of positioning and securing of such items, providing for axial and lateral retention and support, and the accommodation of dimensional changes due to thermal and other effects. The basic design of the US-APWR evolved from existing 4-loop PWR plants. The reactor internals of the US-APWR are classified as core support structures, threaded structural fasteners, and internal structures. The materials to be used for the construction of the reactor internals are also briefly described.

The reactor internals of the US-APWR can be divided into two major assemblies, i.e., upper and lower. The applicant provided a detailed description of the major subassemblies and components of the upper and lower reactor internals assemblies including the functional requirements of each component. The loading conditions for the various reactor internals and the means by which these loads are transmitted from the reactor internal components to the RV are also discussed. The fuel assembly cavity of the US-APWR is defined horizontally by the neutron reflector inside surface, and vertically by the upper core plate that compresses the fuel assembly hold-down springs and a lower core support plate that supports the fuel assemblies.

3.9.5.4.1.1 Design Arrangement - General

DCD Tier 2, Section 3.9.5.1, "Design Arrangements," states that the US-APWR is a 4-loop PWR plant with 257 fuel assemblies having 17 by 17 fuel rod arrays, with ~4.27-m (14 ft) active fuel length. The basic design of the US-APWR reactor internals evolved from existing 4-loop plant technologies.

The reactor internals of the US-APWR are classified as core support structures, threaded structural fasteners, and internal structures. They include all structural and mechanical elements inside the RPV, with the exception of reactor fuel elements and reactivity control elements, control rod drive elements, and in-core instrumentation. However, only a few components are designed as core support structures and threaded structural fasteners, as summarized below:

- Upper Core Support Assembly
    - Upper core support plate, flange, and skirt cylinder
    - Upper core plate
    - Upper core plate fuel alignment pins
    - Upper core support columns, and threaded structural fasteners
    - Upper core plate clevis and threaded structural fasteners
- Lower Core Support Assembly
    - Core barrel flange
    - Upper and lower core barrel
    - Lower core support plate
    - Radial support key and clevis
    - Lower core support plate fuel alignment pins

The design bases requirements provided in DCD Tier 2 Section 3.9.5.3, "Design Bases" specifies that those reactor internals components classified as core support structure conform to the materials, design, fabrication, examination, and documentation requirements of the ASME B&PV Code, Section III, Subsection NG, 2001, Edition through the 2003, Addenda.

The staff's review of DCD Tier 2 Section 3.9.5.1, together with DCD Tier 2 Subsection 3.9.5.1.3, "Jurisdictional Boundaries of the Reactor Internals," revealed that the applicant did not clearly define the classification of the reactor internals hold-down spring, which is a load bearing component of the upper core support assembly. In **RAI 374-2446, Question 03.09.05-27,** the staff requested the applicant to: (a) provide clarification for the classification of the reactor internals hold-down spring; (b) provide technical justification for any classification which would not require use of the design, fabrication, examination, and documentation requirements of the ASME Code, Section III, Subsection NG for design of the hold-down spring; and (c) revise DCD Tier 2 Section 3.9.5.1 and Subsection 3.9.5.1.3, and DCD Tier 2, Table 3.2-2, "Classification of Mechanical and Fluid Systems, Components, and Equipment," including the requested information.

In its response to **RAI 374-2446, Question 03.09.05-27**, dated July 17, 2009, the applicant stated that the hold-down spring is classified as an internal structure because it is not required to directly support the core. The primary functions of the hold-down spring are to allow compliance for thermal expansion between the RV and the reactor internals (upper support flange and core barrel flange), and provide sufficient preloads to the flanges to prevent excessive vibration or sliding during operation.

The applicant further stated that even if the hold-down spring loses all its preload from stress relaxation, the shape of the hold-down spring will remain unchanged and the vertical loads from the core can still be transferred through the hold-down spring to the upper support and core barrel flanges and then to the vessel head and vessel flange.

In its review of the applicant's response, including the response to **RAI 374-2446 Question 03.09.05-2,** discussed below, the staff noted that the applicant had not provided adequate technical justification for classifying the hold-down spring as an internal structure and not as a core support structure. ASME Code, Section III, Article NG-1121 defines core support structures as structures or parts of structures, which provide direct support or restraint of the core within the RPV. DCD Tier 2, Subsection 3.9.5.1.1, "Upper Reactor Internals Assembly Design Arrangement," states that the horizontal loads on the upper core support assembly are transmitted from the upper core support flange to the RV head and hold-down spring by friction or direct contact with the RV flange. Furthermore, in its response the applicant stated that this extreme example of complete loss of preload is undesirable from a functional standpoint because of the potential adverse effects on vibration and sliding of the reactor internals. Therefore, the staff's concerns summarized in the original RAI question are not resolved. As a result, **RAI 374-2446, Question 03.09.05-27** was closed as unresolved. In a follow-up **RAI 663-4996, Question 03.09.05-28**, the staff requested the applicant to further justify why the hold-down spring is not considered a component contributing to support of the reactor core. In its response to **RAI 663-4996, Question 03.09.05-28**, dated January 21, 2011, the applicant reiterated the primary functions of the hold-down spring discussed above in its response to the original **RAI 374-2446, Question 03.09.05-27**, and added that even if the preload of the hold-down spring is lost, the core barrel flange would not lift during normal operation because the hydraulic lift force is smaller than the downward loads (e.g., dead weights and hold-down force of fuel assemblies). The applicant added that the core barrel may lift off from the vessel support ledge under abnormal or accident conditions, but the displacement is limited by the small stroke of the hold-down spring. Thus, the core support and restraint functions are accomplished by the lower and upper core support assemblies as discussed further in the response to **RAI 663-4996, Question 03.09.05-29**, dated January 21, 2011. The applicant further stated that the preload of the hold-down spring is required not for the core support function but for the long-term reliability of the reactor internals.

The staff finds the applicant's response acceptable because the applicant has clarified that the hold-down spring is required for the long-term reliability of the reactor internals and not for the core support function. Core support and restraint functions are accomplished by the lower and upper core support assemblies as discussed further in the response to **RAI 663-4996, Question 03.09.05-29**. Therefore, the staff's concerns are resolved, and **RAI 663-4996 Questions 03.09.05-28 is resolved**.

DCD Tier 2, Section 3.9.5.1 states that the materials to be used in the construction of the US-APWR reactor internals are discussed in DCD Tier 2, Section 4.5.2, "Reactor Internals and Core Support Materials"; DCD Tier 2, Table 4.5.2, "Summary of Reactor Internals and Core Support Materials," lists the materials of interest for reactor internals. The material aspects of reactor internals are evaluated in Section 4.5.2 of this report.

3.9.5.4.1.2 Design Arrangement - Upper Reactor Internals Assembly

The reactor internals of the US-APWR can be divided into two major assemblies, i.e., upper and lower. The hold-down spring in the reactor internals is a separate part, which is captured between the upper and lower reactor assembly flanges and which provides vertical preload and frictional restraint to the flanges. DCD Tier 2, Subsection 3.9.5.1.1 describes the design arrangement of the upper reactor internals assembly, including the functional requirements of each component. The major subassemblies of the upper reactor internals are the following:

- Upper core support assembly
- Upper core plate assembly
- Upper support column assemblies

- Top slotted columns and mixing devices
- Guide tube assemblies
- RV level instrumentation system assemblies
- ICIS detector guide thimbles and thimble assemblies; and
- Thermocouple conduit support column assemblies.

The upper core support assembly has an upper core support flange welded to the top of a cylindrical skirt which is welded at its bottom to the upper core support plate. The flange has flow holes to allow reactor coolant flow into the RV head plenum. Threaded roto-lock inserts in the upper core support flange are used for lifting the upper support assembly. Also, to guide the upper core assembly during installation, slots cut in the upper core support flange are engaged by head and vessel alignment pins attached to the core barrel flange. Holes in the upper core support plate allow installation of the lower guide tubes and upper support columns. Each guide tube assembly is secured to the upper core support plate by hold-down bolts, and each upper support column is secured with a large nut. The vertical cavity between the upper core support and the upper core plate is dimensionally controlled by upper support columns that are fastened at the bottom to the upper core plate and at the top to the top of the upper core support by a single extended tube with a treaded nut that bears on the upper core support. Some upper support columns have the detector guide thimbles or thermocouple conduits. DCD Tier 2, Section 3.9.5.1 states that on the periphery of the upper core plate there are several top slotted columns and mixing devices designed to provide a uniform exit flow and temperature distribution to the outlet loop pipes. There are also two RV level instrumentation support tubes that measure the water level in the RV. In the review of DCD Tier 2, Revision 1, Section 3.9.5.1, the staff found that the applicant did not provide sufficient information to allow the review of the supporting structures design and their liability to potential adverse flow effects. The DCD should explicitly state whether these structures and their operating environment are similar to those of the existing 4-loop reactor design. If this is not the case for some supporting structures, the applicant should explain the differences and provide appropriate flow-induced vibration analysis for those structures. In **RAI 374-2446, Question 03.09.05-1,** the staff requested the applicant to provide more details of the instrumentation supporting structures [e.g. thermocouple, water level sensor, ICIS, control and drive rod assembly] as well as the relevant flow-induced vibration analysis for these structures and to revise DCD Tier 2, Section 3.9.5 accordingly.

In its response to **RAI 374-2446, Question 03.09.05-1**, dated June 19, 2009, the applicant provided additional information about the instrumentation supporting structures. Regarding the supports for the thermocouple and the ICIS, the applicant stated that the upper core support columns are used as the guide structures in the upper plenum, and that these structures are similar to the upper support columns in the existing 4-loop reactor design. The response also included the requested information about the supporting structures of the water level sensors, stating that these supports are similar to the RCCA guide tubes in support conditions and vibration characteristics. For both structures, the top end is fixed on the upper core support plate with bolts and the bottom end is pin supported on the upper core plate. The applicant further added that assessments for the adverse flow effects such as the lock-in with vortex shedding or the fluid elastic instability of the RCCA guide tubes and the upper support columns were performed based on the ASME design guidelines. The analysis results show that both structures of the US-APWR have sufficient margins for the adverse flow effects as described in DCD Tier 2, Section 3.9.2.3, “Dynamic Response Analysis of Reactor Internals under Operational Flow Transients and Steady-State Conditions,” and Section 3.4 of the applicant’s

Technical Report MUAP-07027-P, Revision 1. The applicant proposed to include corresponding information in the DCD.

The staff finds the information provided by the applicant acceptable because the supporting structures are either similar to those used in existing 4-loop reactors, which have been in-service for many years, or are designed to avoid adverse flow effects as described in MUAP-07027-P, Revision 1. In addition, the staff confirmed that the applicant revised DCD Tier 2, Revision 3, Subsection 3.9.5.1.1, to include the requested information about the supporting structures of the instrumentation. The staff concerns are therefore resolved. Accordingly, **RAI 374-2446, Question 03.09.05-1, is resolved.**

DCD Tier 2, Subsection 3.9.5.1.1 states that the upper core support assembly is restrained vertically in the upward direction by the RV head flange and in the downward direction by the reactor internals hold-down spring. The preload in the hold-down spring during installation, is controlled by a fixed distance between the bottom of the upper core support flange and the top of the core barrel flange. The horizontal loads on the upper core support assembly due to flow, vibration, and seismic and pipe rupture events are transmitted from the upper core support flange to the RV head and hold-down spring by friction or direct contact with the RV flange; head and vessel alignment pins also transmit some of the horizontal loads. The staff's review of DCD Revision 1 indicated that the applicant did not discuss the potential loss of preload in the hold-down spring due to stress relaxation during service and its potential effect on the functional and structural integrity of upper core support assembly. In **RAI 374-2446, Question 03.09.05-2,** the staff requested the applicant to provide an assessment of the potential loss of preload of the hold-down spring due to stress relaxation during the design lifetime, and discuss its effect on the horizontal and vertical restraints of the upper core support and core barrel assemblies. The staff further requested the applicant to revise the DCD to include the requested information or, alternately, to provide a reference document where this information is available.

In its response **RAI 374-2446, Question 03.09.05-2**, dated July 17, 2009, the applicant stated that the hold-down spring in PWR operating plants have been observed in the industry to have some loss of preload from inelastic deformation of the contact surfaces during initial bolt-up and subsequent stress relaxation from plant operation. Since the material and design of the US-APWR hold-down spring is similar to that which has been successfully used in many operating PWR plants, it is not expected that the loss of preload will affect the hold-down spring functionality.

The applicant further stated that DCD Tier 2, Subsection 3.9.5.1.1 will be changed to explain how the primary function of the hold-down spring is to accommodate the thermal expansion differences between the RV and the reactor internals upper core support flange and core barrel flange. This includes details of the radial gap between the upper core support flange and the RV inside diameter. The gap is large enough to prevent contact from thermal expansion of the upper core support flange relative to the RV flange during normal operation. The applicant identified vertical and horizontal load transmissions details to be included in DCD Tier 2, Subsection 3.9.5.1.1 as well.

Regarding the effects of loss of preload in the hold-down spring on the horizontal and vertical restraints of the upper core support and core barrel assemblies, the applicant's response stated that the loss of preload is not expected to affect the hold-down spring functionality, but did not provide any technical basis for this statement. Furthermore, the applicant's response to **RAI 374-2446, Question 03.09.05-2,** seems to contradict the response to **RAI 374-2446, Question 03.09.05-27**, where the applicant stated this extreme example of complete loss of preload is undesirable from a functional standpoint because of the potential adverse effects on vibration and sliding of the reactor internals. The staff also noted that the proposed changes in DCD Tier 2, Subsection 3.9.5.1.1, basically describe the function of the hold-down spring and do not discuss the potential loss of preload of the hold-down spring and its effect on the horizontal and vertical restraints of the upper core support and core barrel assemblies. As a result, **RAI 374-2446, Question 03.09.05-2,** was closed as unresolved. In a follow-up **RAI 663-4996, Question 03.09.05-29**, the staff requested the applicant to respond to the original question, and discuss the effect of potential hold-down spring load relaxation on the horizontal and vertical restraint of the upper core support and core barrel assemblies, and ultimately the support of the reactor core.

In its response to **RAI 374-2446, Question 03.09.05-29**, dated January 21, 2011, the applicant stated that if the hold-down spring preload is completely lost, then there would be no possibility of core lift because the hydraulic forces would not exceed the downward load of the dead weight and vertical alignment of the upper and lower core support structures would be maintained because of the component alignment.

The applicant also stated that if the hold-down spring preload is completely lost, then under abnormal, seismic, or LOCA conditions, the core barrel flange may momentarily lift off. However, the applicant stated that there would be no damage to the reactor internals or preventing the shut-down of the reactor. The effect on the lower and upper core support structure would be absorbed by the stiffness of the fuel assembly hold down springs; therefore, the fuel assemblies would remain restrained.

The applicant further stated that the functions to support and restrain the core are accomplished by the upper core support structures (upper core support column and upper core plate) and the lower CSSs (the core barrel and lower core support plate) even when the preload of the hold-down spring is completely lost.

The staff finds the applicant’s response acceptable because the applicant has provided a detailed discussion regarding the effect of potential hold-down spring load relaxation on the support of the reactor core and explained that the component functions to support and restrain the core are accomplished by the lower and upper core support assemblies. Therefore, the staff’s concerns are resolved. Accordingly, **RAI 663-4996, Question 03.09.05-29, is resolved**.

DCD Tier 2, Subsection 3.9.5.1.1, describes the US-APWR upper reactor internals assembly design arrangement, including the manner of positioning and securing of these items and providing for axial and lateral retention and support. The staff reviewed DCD Tier 2, Revision 1, Subsection 3.9.5.1.1, and found that the applicant did not provide sufficient information to allow the review of the upper core plate design and its interfaces with other reactor components. In **RAI 374-2446, Question 03.09.05-3**, the staff requested the applicant to (1) provide sufficient details about the design of the upper core plate and its interface with the fuel assemblies, core barrel, upper support columns, and lower guide tubes; (2) explain any differences from the existing 4-loop design, and how these differences are evaluated against possible excitation mechanisms of flow-induced vibration; and (3) revise DCD Tier 2, Section 3.9.5 to include sufficient information about the design arrangement of the upper core plate and a discussion of the differences, if there are any, in the US-APWR’s loading conditions from the 4-loop reactor.

In its response to **RAI 374-2446, 03.09.05-3**, dated July 17, 2009, the applicant stated that the US-APWR upper core plate and its interface with the fuel assemblies, core barrel, upper support columns, and lower guide tubes are similar to those of the existing 4-loop design. Therefore, structural design changes around the upper core plate are expected to have little impact on the flow-induced vibration. The applicant added that further details and discussions about the design differences of the US-APWR reactor internals from current 4-loop and effects on the flow-induced vibration are described in Section 2.1 of MUAP-07027-P, Revision 1. Since the design of the upper core plate and its interface with other components is similar to that used in existing 4-loop reactors, the staff's concerns regarding the upper core plate design are resolved. However, the applicant did not commit to revising the DCD to include this information. Therefore, **RAI 374-2446, Question 03.09.05-3,** was closed as unresolved. In a follow-up **RAI 663-4996, Question 03.09.05-30**, the staff requested the applicant to revise DCD Tier 2, Section 3.9.5 to include this additional information.

In its response to **RAI 374-2446, Question 03.09.05-30**, dated January 21, 2011, the applicant agreed to add the requested information in the DCD Tier 2, Subsection 3.9.5.1.1. The staff reviewed the proposed changes to the DCD and found them to be acceptable. The staff finds the applicant's response acceptable because the DCD will be revised to include the additional information. **RAI 374-2446, Question 03.09.05-30 is being tracked as a Confirmatory Item** pending revision of the DCD.

DCD Tier 2, Revision 1, Subsection 3.9.5.1.1, stated that the guide tubes consist of two main assemblies, an upper and a lower guide tube, that provide horizontal restraint and guidance to the control rods and drive rod assembly, and allow parking of the drive rod during removal and installation after refueling. Both guide tubes have plates that guide the control rod spider during insertion and retraction of the RCCA. The staff determined that the applicant did not provide sufficient information about the control rod guide inside the upper and lower guide tubes. In **RAI 374-2446, Question 03.09.05-4**, the staff requested the applicant to provide design details together with relevant flow-induced vibration analysis (if needed) for the plates that guide the control rod spider inside the upper and lower guide tubes. In particular, the staff requested the applicant to explain, with the aid of technical drawing or sketches, the design of the control rod guide and to clarify any differences of this design from that of the existing 4-loop reactor. Also, the staff requested the applicant to explain the effects of any design differences on potential flow excitation mechanisms and to revise DCD Tier 2, Section 3.9.5 to provide the requested information.

In its response to **RAI 374-2446, Question 03.09.05-4**, dated July 17, 2009, the applicant stated that the US-APWR lower RCCA guide tube is similar in design to that of the current 4-loop reactor in the layouts and dimensions of the enclosure and guide plates. The upper RCCA guide tube is extended about 1 ft (0.3 m) from that for the 12 ft (3.7 m) core design as in the 14 ft (4.3 m) core 4-loop reactors existing in U.S. The applicant added that because both the flow conditions inside the guide tube and the mechanical interface with the RCCA spider are equivalent with existing 4-loop reactors, no impact is expected on the vibration of the RCCA spider.

The staff finds the applicant's response acceptable because the two assemblies of the guide tubes are similar in design to those used in current 4-loop reactors. In addition, the staff confirmed that as stated by the applicant in the response to **RAI 374-2446, Question 03.09.05-5**, this information is included in the revised version of the DCD. Therefore, the staff's concerns are resolved. Accordingly, **RAI 374-2446, Question 03.09.05-4, is resolved.**

DCD Tier 2, Subsection 3.9.5.1.1 describes the guide tube assemblies. The applicant stated that the upper and lower guide tube flanges are fastened together by hold-down bolts threaded to the top of the upper core support plate. The lower guide tube is inserted through holes in the upper core support and restrained in the horizontal direction by a small clearance between the lower guide tube flange and upper core support plate hole. Also, the bottom of the lower guide tube is fastened by two large support pins with flexible leaves that slide vertically with a small amount of friction force, but are horizontally preloaded against the upper core plate holes to prevent excessive vibration and wear. The applicant, however, did not include sufficient geometry and design details to allow the staff to evaluate the flow-induced response of the guide tubes.

In **RAI 374-2446, Question 03.09.05-5**, the staff requested the applicant to (1) provide details of the geometry and design of the lower and upper guide tubes indicating the differences from the guide tubes of the current 4-loop reactors; (2) explain the effect of these differences on the flow-induced structural response of the guide tubes; (3) substantiate the response to this RAI by referring to the flow-induced vibration analysis, which should be included in the response to this RAI by means of appropriate flow-induced vibration analysis for the guide tubes; and (4) revise the DCD to include additional details about the geometry and design of the lower and upper guide tubes in DCD Tier 2, Section 3.9.5.1, about their flow-induced vibration analysis in DCD Tier 2, Section 3.9.2.3, and also about their design bases in DCD Tier 2, Section 3.9.5.3.

In its response to **RAI 374-2446, Question 03.09.05-5**, dated June 19, 2009, the applicant stated that the lower RCCA guide tube of the US-APWR has a similar design to that of the current 4-loop reactor in the layouts and dimensions of the enclosure and guide structures. As discussed earlier, the upper RCCA guide tube is extended about 1 ft (0.3 m) from that for the 12 ft (3.7 m) core design, as in the 14-ft (4.3 m) core 4-loop reactors existing in U.S. So the vibration characteristics of the RCCA guide tubes for the US-APWR are equivalent with the existing guide tubes.

Regarding the liability of the RCCA guide tubes to flow-induced vibrations, the applicant added that because the lower RCCA guide tube is located in the cross flow field of the upper plenum, assessments for the adverse flow effects such as the lock-in with vortex shedding or the fluid elastic instability were performed based on the ASME design guidelines. The results indicate that the RCCA guide tube has sufficient margins for the adverse flow effects as described in DCD Tier 2, Section 3.9.2.3 and Section 3.4 of MUAP-07027-P, Revision 1.

The staff finds the applicant's response acceptable because it clarifies that the design of the RCCA guide tubes is similar to existing 4-loop reactors. The applicant also confirmed that flow-induced vibration analysis has been performed for the guide tubes exposed to cross-flow. In addition, the staff confirmed that this information is included in the revised version of the DCD. The staff's concerns regarding the design of the guide tubes are therefore resolved. Accordingly, **RAI 374-2446, Question 03.09.05-5 is resolved**.

DCD Tier 2 Subsection 3.9.5.1.1 describes the US-APWR upper reactor internals assembly design arrangement, including the manner of positioning and securing of these items and coolant flow through the reactor internal assemblies. The applicant stated that the exit flow core pressure difference between the fuel assemblies is limited by the design to an acceptable cross-flow velocity to prevent vibratory damage to the fuel rods, thimbles, or RCCAs. The staff's review of DCD Tier 2, Revision 1, Subsection 3.9.5.1.1, found that the applicant did not explain how the thermal-hydraulic design requirement regarding the fuel assembly exit core flow would be verified. As stated in DCD Tier 2, Subsection 3.9.5.3.2, "Thermal-Hydraulics Design-Basis," the thermal-hydraulic performance criteria require that the core outlet flows from the fuel assemblies are to be designed to minimize horizontal velocities that may contribute to vibration of the RCCA rodlets. In **RAI 374-2446, Question 03.09.05-6,** the staff requested the applicant to describe the procedure that is to be used to verify that the exit flow from the fuel assemblies does not lead to unacceptable cross-flow velocities that may cause vibration of the fuel rods, thimbles, or RCCAs and to revise DCD Tier 2 Subsection 3.9.5.1 to include the requested information.

The applicant's response to **RAI 374-2446, Question 03.09.05-6**, dated July 17, 2009, stated that the acceptability of the fuel assembly exit cross-flow velocity was based on operating experience of existing 4-loop plants with similar design features to the US-APWR fuel assemblies and upper internals. The applicant added that the design of the upper core plate flow holes, fuel assembly loss coefficients, and the fuel assembly design of the US-APWR are not significantly different from those of existing 4-loop plants. So the cross flow velocities at the core outlet are expected to be similar to the existing 4-loop plants.

The staff finds the response to this RAI unacceptable because it does not provide quantitative information about the design differences that may cause adverse flow effects of the fuel rods and thimbles. As a result, **RAI 374-2446, Question 03.09.05-6,** was closed as unresolved. In a follow-up **RAI 663 4996, Question 03.09.05-31**, the staff requested the applicant to provide a quantitative assessment of the effect of differences from existing 4-loop plants on the cross-flow excitation of the fuel rods, thimbles, and RCCAs of the US-APWR.

In its response to **RAI 374-2446, Question 03.09.05-31**, dated January 21, 2011, the applicant provided an assessment of the potential adverse cross-flow effect at the core outlet of US-APWR by comparison of total flow rate, flow rate per fuel assembly, and support span of the fuel assembly of the US-APWR with the existing 12 ft (3.7 m) fuel and 14 ft (4.3 m) fuel 4-loop PWR plants. A table is provided in the RAI response that shows the total flow rate, the calculated flow rate per assembly, physical dimensions of the fuel assembly and RCCAs, and the calculated support span of the fuel assembly. The applicant stated that although the total flow rate of US-APWR is larger than that of the existing 12 ft (3.7 m) and 14 ft (4.3 m) fuel plants, the flow rate per fuel assembly is lower due to its larger number of fuel assemblies (257 fuel assemblies vs. 193 fuel assemblies of the existing plants). The applicant also stated that the support span of the US-APWR fuel assembly is shorter than those of the existing plants. The applicant further stated that the dynamic characteristic of the US-APWR control rod are very close to that of the existing PWR plants because the material and physical dimensions are identical. The applicant concluded that in the currently operating 4-loop plants, there has been no observed high-cycle fatigue degradation of fuel assembly and control rod due to flow-induced vibration, therefore, the potential adverse effect due to cross-flow vibration on the fuel assembly and control rod of US-APWR design is not a concern.

The staff finds the applicant's response acceptable because the applicant has provided a quantitative assessment with calculations that demonstrate the APWR fuel assembly has lower flow rate and shorter support span and identical dynamic characteristic of control rod compared to these of the existing 4-loop PWR plants. The staff agrees that, based on the lower flow-rate and shorter support span of the fuel assembly, and operating experience of the existing PWR plants, the potential of adverse effect due to cross-flow induced vibration is not a concern for the fuel assembly, thimbles, and RCCAs in the APWR design. Therefore, the staff's concerns are resolved. Accordingly, **RAI 663-4996, Question 03.09.05-31, is resolved**.

3.9.5.4.1.3 Design Arrangement - Lower Reactor Internals Assembly

DCD Tier 2, Subsection 3.9.5.1.2, "Lower Reactor Internals Assembly Design Arrangement," describes the design arrangement of the lower reactor internals assembly, including the functional requirements of each component. The major subassemblies of the lower reactor internals are the following:

- Core barrel assembly,
- Lower core support assembly,
- Neutron reflector assembly,
- Irradiation specimen guide assembly; and
- Secondary core support assembly.

The core barrel assembly consists of an upper and a lower core barrel that are welded together, and a forged flange is welded to the upper core barrel. Flow nozzles are welded to the core barrel flange and provide flow of reactor coolant from the RV annulus to the RV head plenum. The core barrel assembly can be lifted by threading the lifting fixtures into the roto-lock inserts in the flange. The head and vessel alignment pins, bolted to the flange, are used to guide the core barrel assembly during installation and removal; they are guided and aligned by slots in the RV and RV head. Four core barrel outlet nozzles are welded to the upper core barrel to provide an exit flow path to the RV outlet nozzles. In addition, four safety injection pads are attached to the core barrel to divert the safety injection flow from directly impinging on the barrel during a safety injection event.

The lower core barrel receives maximum neutron fluence during normal operation. Irradiation specimen guides are fastened to the outside of the core barrel at specific locations by long socket head cap screws (to accommodate bending). The specimen capsules inside the specimen guides are held in place by springs and a threaded cap. RV surveillance test specimens are periodically removed during outage for examination of RV neutron radiation embrittlement. The core barrel flange also has holes for access to the irradiation specimens.

The lower core support assembly consists of a lower core support plate, six radial support keys, and fuel alignment pins. The applicant stated that the lower core support plate has orificed flow holes to reduce mal-distribution of the flow into the core. The safety analysis design requirements for US-APWR internals listed in DCD Tier 2, Subsection 3.9.5.3.1, "Safety Analysis Design-Basis," state that mal-distribution of the flow into the core should be limited so as not to impact core safety limits in DCD Tier 2, Chapter 15, "Transient and Accident Analyses." However, the applicant did not refer to any safety analysis that would ensure compliance with this safety requirement for the design of US-APWR core support structures and core internals. Therefore, in **RAI 374-2446, Question 03.09.05-7,** the staff requested the applicant to discuss the analysis performed and the measures undertaken to make sure that the mal-distribution of the flow into the core shall be limited so as not to impact the US-APWR core safety limits and to revise DCD Tier 2, Section 3.9.5 to provide the requested information or, alternately, provide a reference document where this information is available.

In its response to **RAI 374-2446, Question 03.09.05-7**, dated July 17, 2009, the applicant stated that the mal-distribution of flow into the core is limited by meeting several reactor internals design requirements, which include the allowable minimum and maximum fuel assembly inlet flow rate and the allowable difference in inlet flow rates between adjacent fuel assemblies. The design target values are specified for the minimum flow rate; the maximum flow rate; and difference between adjacent fuel assemblies. The applicant added that these design requirements are similar to those in operating 4-loop U.S. plants. The applicant further stated that confirmatory testing was performed for the US-APWR, and provided results for the minimum assembly flow, the maximum assembly flow, and the difference between adjacent fuel assemblies. With these results, the applicant concluded that the inlet core flow distribution meets the requirement to preclude adverse effects such as core tilt, flow starvation, or undesirable inlet cross-flow distribution.

The staff finds the applicant's response to this RAI partially acceptable because it indicates that well-defined design targets are established for the flow distribution into the core. These design targets are similar to those used for the operating 4-loop U.S. plants. In addition, confirmatory tests have been performed to validate that the design parameters of the US-APWR are within the acceptable range to avoid mal-distribution of flow into the core. However, in its response, the applicant cited a reference for the results but did not identify it. Therefore, **RAI 374-2446, Question 03.09.05-7,** was closed as unresolved. In a follow-up **RAI 663-4996, Question 03.09.05-32**, the staff requested the applicant to revise the DCD by identifying the reference, which includes confirmatory test results for the US-APWR core internals, and include it in the appropriate list of DCD references.

In its response to **RAI 663-4996, Question 03.09.05-32**, dated January 21, 2011, the applicant stated that as requested in this RAI, DCD Subsection 3.9.5.3.2 will be revised to include the design criteria required to limit mal-distribution of the flow into the core. The applicant stated also that reference to Technical Report MUAP-07022-P, "Reactor Vessel Lower Plenum 1/7 Scale Model Flow Test Report," Revision 0, issued June 2008, which includes the confirmatory testing, will be made in the DCD. The staff finds the applicant's response acceptable because the applicant provided design criteria and a reference to address flow mal-distribution and proposed to include them in the DCD. Therefore, the staff's concern is resolved. **RAI 663-4996, Question 03.09.05-32, is being tracked as a Confirmatory Item** pending revision of the DCD**.**

DCD Tier 2, Subsection 3.9.5.1.2, states that the lower core support plate supports the fuel assemblies and has two fuel alignment pins per fuel assembly for alignment and horizontal restraint of the bottom fuel nozzle. The fuel alignment pins are attached to the top of the lower core support plate and restrained by a locking device. The radial keys are attached to the outside rim of the lower core support plate, and engage the RV clevis inserts. The keys and clevis inserts provide alignment during installation, resistance to vibration from flow, and transmit asymmetric flow loads and dynamic loads from seismic and postulated LOCA events to the RV.

DCD Tier 2, Subsection 3.9.5.1, states that the horizontal fuel assembly cavity is defined by the neutron reflector inside surface, and the vertical fuel assembly cavity is defined by the upper core plate that compresses the fuel assembly hold-down springs and a lower core support plate that supports the fuel assemblies. DCD Tier 2, Subsection 3.9.5.1.2 states that the neutron reflector assembly offers an improvement in neutron reflectivity relative to current operating PWR plants, and a significant reduction in the number of threaded fasteners. The neutron reflector consists of multiple stacked ring blocks that are supported by the lower core support plate at the bottom and by tie rods and neutron reflector mounting bolts at the top; the bottom ring block is also secured to the lower core support plate by neutron reflector mounting bolts. The small gaps between ring blocks are aligned with the fuel assembly grids. The stacked ring blocks are connected to each other by ring block alignment pins mounted on their top and bottom surfaces for alignment and shear restraint. The neutron reflector upper and lower alignment pins are guided into position by clevises attached to the core barrel. This arrangement provides horizontal restraint for mechanical loads, similar to the upper core plate arrangement. Also, the ring blocks are designed with cooling holes to minimize void swelling and dimensional changes.

DCD Tier 2, Subsection 3.9.5.1.2, states that holes machined in the lower core support plate direct bypass cooling flow into the bottom ring block. These holes are orificed to provide a pressure drop that minimizes the pressure difference between the core and the neutron reflector flow paths, and are also sized to prevent debris from blocking the cooling holes in the ring blocks. Tie-rods provide vertical restraint for mechanical loads. The tie-rods pass through holes in the blocks and are threaded into the lower core support plate and are captured by a nut bearing on the top ring block. Fluence and temperature limits are imposed on the tie-rods to preclude excessive loss of pre-load from irradiation stress relaxation. The secondary core support assembly consists of secondary core support columns, diffuser plate support columns, a base plate, and energy absorber system. The diffuser plates are bolted to the diffuser plate support columns and those columns are fastened to the bottom of the lower core support plate. The energy absorber and base plate are supported by columns that are bolted to the bottom of the lower core support plate. DCD Tier 2, Subsection 3.9.5.1.2, further states that the energy absorber system and base plate have traditionally been used in PWR internals. Its purpose is to preclude overstressing the RV in the unlikely event of a failed core barrel weld. The drop distance between the bottom of the base plate and the energy absorber system RV bottom is carefully controlled to minimize the impact load and stresses on the RV bottom head. DCD Tier 2 Subsection 3.9.5.3.1, states that the safety analysis of this issue is a design requirement for the RV.

The staff reviewed DCD Tier 2, Revision 1, Section 3.9.5, and found that the applicant did not refer to any analysis of the impact load, which would result from a core drop event. In **RAI 374-2446, Question 03.09.05-8**, the staff requested the applicant to discuss the analysis performed and the measures undertaken to make sure that the impact load on the RV bottom head from a postulated core drop event would not adversely affect the integrity of the RV bottom head.

In its response to **RAI 374-2446, Question 03.09.05-8**, dated July 17, 2009, the applicant stated the core drop event is not a DBA condition. The applicant also stated that although the core drop event is not a DBA condition, the impact load and contact surface area are calculated using the sizing of the secondary core support structure and the gap clearance between the RV bottom head and base plate. The RV allowable impact load and contact surface area are provided by the RV designer and are to be complied with as a reactor internal interface design requirement.

The staff finds the response acceptable because the core drop event is considered beyond-design-basis event. Although this is not a regulatory requirement, the applicant added that the impact load and contact surface area are calculated and will be complied with as a reactor internal interface design requirements. Accordingly, **RAI 374-2446, Question 03.09.05-8, is resolved**.

DCD Tier 2, Subsection 3.9.5.1.3, describes the jurisdictional boundaries among core support structures, the reactor internal structures, and the interfacing components such as the RV, CRDMs, fuel assemblies and thermocouple, and ICIS. The DCD states that the jurisdictional boundaries of reactor internals follow the guidance for boundaries of jurisdiction in the ASME Code, Section III Subsection NG-1000 (these boundaries are illustrated in DCD Tier 2, Fig. 3.9-7, "Jurisdictional Boundary between Reactor Internals and RV"). The DCD summarizes the jurisdictional boundaries between the core support structure and threaded structural fasteners and the RV, fuel assemblies, ICIS, and the internal structures, as well as between internal structures and the interface components. Based on the jurisdictional boundaries following ASME Code, Section III Subsection NG-1000, the staff finds the description of the jurisdictional boundaries acceptable.

Pending the resolution of the aforementioned confirmatory items, the staff finds that the physical or design arrangements of all reactor internals structures, components, assemblies, and systems (including the manner of positioning and securing such items, providing for axial and lateral retention and support, and accommodating dimensional changes due to thermal and other effects) meet the relevant requirements of GDCs 1, 2, 4, and 10.

3.9.5.4.2 Summary of Loading Conditions

DCD Tier 2, Section 3.9.5.2, "Loading Conditions," describes US-APWR loading conditions, load combinations, and the acceptance criteria of the reactor internals, namely, the design, service limits, and displacement limits. All combinations of design and service loadings (e.g., operating differential pressure and thermal effects, potential adverse flow effects such as flow-induced vibration and acoustic resonance, seismic loads, and transient pressure loads of postulated LOCA) are included in the design of the reactor internals.

The loads of the reactor internals are categorized according to the design and service loading conditions for the plant. DCD Tier 2, Subsection 3.9.5.2.2, "Design and Service Limits," describes the load combinations for core support structures and threaded structural fasteners, as well as the stress categories and service limits, in ASME Code, Section III. The applicant states that, although the service limits for reactor internals other than the core support structures are not addressed in the ASME Code, Section III, because of their importance to the safe operation of the reactor internals, the stress limits for core support structures are also applied to the reactor internals. If the stress limits for the internal structure do not meet the ASME Code, Section III limits for the core support structures, then alternate acceptance criteria are employed based on validation by testing, sound engineering judgment, and experience with similar design. DCD Tier 2, Subsection 3.9.5.2.3, "Interface Load and Displacement Limits," describes the load and displacement limits for the reactor internals that affect the safety and operability of the interface components are also discussed.

3.9.5.4.2.1 Loading Conditions

DCD Tier 2, Revision 1, Section 3.9.5.2 identifies the loading conditions that have been considered in the design of the US-APWR core support structures and internal components. The list includes pressure differences due to coolant flow. However, the applicant did not provide any details regarding the method used to determine the pressure differences for reactor internal components during different operating conditions or to validate the calculated values. In **RAI 374-2446, Question 03.09.05-11**, the staff requested the applicant to provide a description and validation of the method for determining the maximum pressure differences for reactor internals during ASME Code, Section III, Level A, B, C, and D service conditions, and to revise the DCD to include the requested information or provide a reference document where the requested information is available.

In its revised response to **RAI 374-2446, Question 03.09.05-11**, dated February 28, 2013, the applicant stated that the maximum pressure difference across the core barrel thickness was determined from the total pressure loss for the downcomer, the lower plenum, the lower core support plate, the fuel assembly, the upper core plate, and the upper plenum. The applicant also stated that the formula and the pressure loss coefficients were determined based on the design handbooks and verified with the experience of the current operating plants. The applicant further described how the design difference pressure was determined, being based on mechanical design flow in hot conditions plus additional margin, which can represent the Level

A (normal operation) and Level B (up-set) service conditions. The pressure difference in Level C service conditions is also represented by the design conditions. For the Level D service condition, time histories of dynamic pressure differences in a LOCA event were obtained in the blow-down analysis and included in the response to **RAI 207-1577, Question 03.09.02-54**. The staff finds the response acceptable since the applicant described how the pressure differences were determined for the service conditions. Therefore, the staff's concerns regarding the maximum pressure differentials for the reactor internals are resolved. Accordingly, **RAI 374-2446, Question 03.09.05-11, is resolved**.

DCD Tier 2, Revision 1, Section 3.9.5.2, states that pressure differences due to the coolant flow have been taken into account in designing the US-APWR core support and internal structures. The complete list of loading conditions that have been considered in the reactor internals design is given in DCD Tier 2, Table 3.9-11, "Core Support Structures and Threaded Structural Fasteners Loading Conditions and Load Combinations." DCD Tier 2 Subsection 3.9.5.3.2, states that the thermal-hydraulic performance criteria require the pressure drops across the reactor internals to meet system requirements for all Level A and B service conditions. The staff reviewed DCD Tier 2, Section 3.9.5 but did not find where the applicant had evaluated the maximum pressure differentials for the reactor internals or verified that they meet the thermal-hydraulic performance requirements of DCD Tier 2, Subsection 3.9.5.3.2. In **RAI 374-2446, Question 03.09.05-12**, the staff requested the applicant to describe the system requirements for pressure differentials across reactor internals, and to assess the maximum pressure differentials for the reactor internals with respect to the design-basis system requirements. Also, the staff requested the applicant to revise the DCD to include the requested information.

In its response to **RAI 374-2446, Question 03.09.05-12**, dated July 17, 2009, the applicant stated that for the thermal hydraulic design of the reactor system, total pressure loss of the RV shall be 48.2 ± 4.8 psi (332 ± 33 kPa) as described in DCD Tier 2, Table 4.4-1, "Thermal-Hydraulic Comparison between US-APWR and Other Designs." The staff finds the response acceptable since the applicant identified where the maximum pressure differential is included in the DCD. Therefore, the staff's concerns regarding the maximum pressure differentials for the reactor internals are resolved. Accordingly, **RAI 374-2446, Question 03.09.05-12, is resolved**.

DCD Tier 2, Section 3.9.5.2 identifies the loading conditions that have been considered in the design of US-APWR core support and reactor internal components. The applicant stated that asymmetric LOCA loads for the reactor internals have been considered for the LOCA dynamic analysis. However, in DCD Tier 2, Section 3.9.5.2, the applicant did not confirm that such loads had been included in the reactor internals dynamic analysis of the reactor internals, and that they do not exceed the design limits. As stated in Section 3.9.5 of the SRP, the reactor internals should be designed to accommodate asymmetric blowdown loads from postulated pipe ruptures. Furthermore, the applicant's evaluation of such loads should demonstrate that these loads do not exceed the limits imposed by the applicable codes and standards.

In **RAI 374-2446, Question 03.09.05-13**, the staff requested the applicant to verify whether the asymmetric blowdown loadings on reactor internals due to pipe ruptures at postulated locations not excluded in LBB analyses have been evaluated in the design in accordance with the acceptance criteria of SRP Section 3.9.5, SRP Acceptance Criteria Subsection II.5. The staff also requested the applicant to revise DCD Tier 2, Section 3.9.5.2 to include the requested information.

In its response to **RAI 374-2446, Question 03.09.05-13**, dated June 19, 2009, the applicant stated that the asymmetric blowdown loadings on reactor internals due to pipe ruptures at postulated locations not excluded in leak before break analyses were evaluated as described in Technical Report MUAP-09002-P, "Summary of Seismic and Accident Load Conditions for Primary Components and Piping," Revision 1, issued March 2009. The applicant also stated that based on the dynamic response analysis and the stress evaluation it was confirmed that the reactor internals design withstood the blowdown loadings due to pipe rupture events; the summary of the stress analysis was reported in Technical Report MUAP-09004-P, "Summary of Stress Analysis Results for Core Support Structures," Revision 0, issued March 2009.

The applicant further stated that in DCD Tier 2, Section 3.9.5.2, the description of the rupture of the pipe will be revised to address asymmetric blowdown loadings due to pipe ruptures. The staff finds the response acceptable since the applicant addressed asymmetric blowdown loadings due to pipe ruptures. The staff confirmed that DCD Revision 3 was revised as stated. Therefore, the staff's concerns regarding the asymmetric blowdown loadings on reactor internals due to pipe ruptures at postulated locations are resolved. Accordingly, **RAI 374-2446, Question 03.09.05-13, is resolved**.

The staff determined that DCD Tier 2, Revision 1, Section 3.9.5.2, does not include any error analysis. This information is needed to evaluate the (minimum) margin of safety for the dynamic stress of various plant components. In **RAI 374-2446, Question 03.09.05-14,** the staff requested the applicant to provide a detailed analysis of expected bias errors and random uncertainties included in predicting the vibration responses of reactor core support and internal structure, SG internal components, and other plant systems and components. The staff requested the applicant to provide the total (or end-to-end) bias error and random uncertainties, and to substantiate the contributions of each of the following tasks to the total bias and uncertainties:

1. Modelling and validation of the forcing functions.

2. Modelling and validation of the acoustic environment using SYSNOISE.

3. Finite element modelling and validation of structural dynamic characteristics.

4. Combining the forcing functions and system dynamic characteristics to estimate the dynamic response of structures and components.

5. Experimental measurements which are used to validate models and analysis, whether these measurements are performed in-plant or in the laboratory by means of scale model testing.

The staff also requested the applicant to explain how the bias and uncertainties are implemented in the calculation of the minimum safety margin. The staff also requested the applicant to revise DCD Tier 2, Subsection 3.9.5.2 to include analysis of bias errors and random uncertainties.

In its response to **RAI 374-2446, Question 03.09.05-14**, dated July 17, 2009, the applicant stated that the bias error and uncertainties in the FIV assessment analysis was included in the verification process with the bench mark analysis of the APWR 1/5 scale model test, as discussed in Section 3.2 of the applicant's Technical Report MUAP-07027-P, Revision 1 (note the response referenced Technical Report MUAP-07023-P, "APWR Reactor Internals 1/5 Scale

Model Flow Test Report," Revision 1, issued May 2009 when clearly MUAP-07027-P, Revision 1 was intended as MUAP-07023-P, Revision 1 does not have a Section 3.2 and MUAP-07027-P, Revision 1, Section 3.2 discusses the FIV assessment analysis). The applicant also provided a list of various bias errors and random uncertainties for flow-induced forcing functions, RCP induced forcing functions, FE modeling, turbulence-induced response analysis, and experimental measurement.

In addition to details in this response, the applicant provided additional information in DCD Tier 2, Section 3.9.2, "Dynamic Testing and Analysis of Systems, Components, and Equipment," regarding the uncertainties in the predictions of the acoustic model, the structural models and the loading functions, respectively. In the absence of in-plant measurements for the US-APWR to validate these models, the applicant used sufficiently conservative model parameters and tested them against the results of the 1/5 and 1/7 scale model tests. For example, to ensure conservatism, the developed models neglect acoustic attenuation in the majority of the reactor components, assume smaller than expected structural damping, and overestimate the loading functions. In the light of this additional conservatism, the staff finds the applicant approach acceptable. The staff finds the response acceptable since the applicant described where the error analysis is included in the DCD or referenced technical reports and the approaches used. Accordingly, **RAI 374-2446, Question 03.09.05-14, is resolved.**

DCD Tier 2, Table 3.9-1, "RCS Design Transients," presents the complete list of design transients for all service conditions and the number of occurrences needed for stress analysis and fatigue analysis of reactor internal structures and components. Seismic loading and other mechanical loading, which are not included in DCD Tier 2, Table 3.9-1, are described in DCD Tier 2, Section 3.9.3, "ASME Code Class 1, 2, and 3 Components, Component Supports, and Core Support Structures." Also, the number of occurrences in DCD Tier 2, Table 3.9-1 is based on a 60-year design life. The DCD states that all combinations of design and service loadings (e.g., operating differential pressure and thermal effects, potential adverse flow effects such as FIV and acoustic resonance, seismic loads, and transient pressure loads of postulated LOCA) are accounted for in the design of the reactor internals. The loading combinations are summarized in DCD Tier 2, Table 3.9-11. For example, for the Level B loading conditions, the loads to be combined include pressure differences, weight and buoyant forces, lift and drag forces, superimposed internal loads due to other structures, one-third of the peak SSE inertial loads, external reaction restraints, thermal loads, and vibratory loads. The DCD also states that Level B service limits in the ASME Code, Subsection NG, should be larger than the component stress intensity and fatigue usage factors resulting from the combined loadings.

The reactor internal loads are categorized according to the design and service loading conditions for the plant. The ASME Code, Section III load combinations for core support structures and threaded structural fasteners are given in DCD Tier 2, Table 3.9-11, and the stress categories and service limits are given in DCD Tier 2, Table 3.9-12, "Core Support Structures Stress Categories and Stress Intensity Limits." In DCD Tier 2, Subsection 3.9.5.2.2, the applicant stated that the service limits for reactor internals other than the core support structures are not addressed in the ASME Code, Section III. However, because the structural integrity of the reactor internals is important-to-safety, the stress limits for core support structures are also applied to the reactor internals. If the stress limits for the internal structure do not meet the ASME Code, Section III limits for the core support structures, the applicant proposes to utilize alternate acceptance criteria based on validation by testing, sound engineering judgment, and experience with similar design. The staff's review of the DCD identified that the applicant neither provided sufficient information about the proposed alternate acceptance criteria nor on the resulting safety margin. In **RAI 374-2446, Question 03.09.05-15,** the staff requested the applicant to explain in more detail the meaning of the following statement, which is given in DCD Tier 2, Subsection 3.9.5.2.2: “However, if the stress limits for the internal structure do not meet the ASME Code, Section III limits for the core support structures, then alternate acceptance criteria are employed based on validation by testing, sound engineering judgment, and experience with similar designs.” The staff also requested the applicant to provide a list of all components, which did not meet the ASME Code for stress limits and explain the alternate design criteria used for these components. The staff also requested the applicant to revise DCD Tier 2, Section 3.9.5 to provide the requested information.

In its response to **RAI 374-2446, Question 03.09.05-15**, dated July 17, 2009, the applicant stated that the loading conditions and stress limit for the ASME Code, Section III Class CS were applied for the reactor internals, except the secondary core support structures. The applicant further stated that the function of the secondary core support assemblies was to limit the stroke of the drop and the impact force on the lower vessel head in the postulated core drop event. Therefore, the design of the secondary core support structures, including the lower diffuser plate, is determined with the impact force in the core drop event as a beyond-design-basis accident. The applicant also provided a table showing the load combination and acceptance criteria for the secondary core support structures, and stated that the stress limit for the core drop event was considered to be Level D of ASME Code, Section III Class CS components.

The staff finds the response acceptable with respect to the applicant providing the details regarding the stress limits and design criteria for the reactor internals. However, the applicant did not include this information in the revised DCD. Therefore, **RAI 374-2446, Question 03.09.05-15,** was closed as unresolved and in follow-up **RAI 663-4996, Question 03.09.05-33**, the staff requested the applicant to include this information in the next revision of the DCD.

In its response to **RAI 663-2446, Question 03.09.05-33**, dated January 21, 2011, the applicant stated that as requested, DCD Tier 2, Subsection 3.9.5.2.2 will be revised to include the information discussed in the response to **RAI 374-2446, Question 03.09.05-15**. The staff finds the applicant’s response acceptable because the APWR DCD will be revised to include the details regarding the stress limits and design criteria for the reactor internals and has provided corresponding markups. Accordingly, **RAI 663-2446, Question 03.09.05-33 is being tracked as a Confirmatory Item**.

The load and displacement limits for the reactor internals that affect the safety and operability of the interface components are summarized in DCD Tier 2, Table 3.9-2, “Reactor Internals Interface Load and Displacement Limits.” However, DCD Revision 1 did not give any details about how the deformation limits were determined or provide the technical basis for these deformation limits. As stated in SRP Section 3.9.5, SRP Acceptance Criterion 4, deformation limits for reactor internals should be established by the applicant and presented in the SAR, and the basis for these limits should be included. Also, the stresses for these displacements should not exceed the specified limits. In **RAI 374-2446, Question 03.09.05-17**, the staff requested the applicant to provide the technical basis for defining the displacement limits listed in DCD Tier 2, Table 3.9-2. The staff also requested the applicant to revise DCD Tier 2, Subsection 3.9.5.2.3 to include the requested information or provide a reference document where the requested information is available.

In its response to **RAI 374-2446, Question 03.09.05-17**, dated July 17, 2009, the applicant stated that the technical basis of the loads and deformation limits in DCD Tier 2, Table 3.9-2 are explained as follows.

(a) Allowable horizontal load of the RCCA guide tube should not impede insertion of the RCCA after the LOCA event.

Technical Basis: The horizontal load limit provides assurance that after a SSE plus LOCA combined event, the inelastic deformation of the guide tube is such that the control rods will be unimpeded during rod drop insertion. The horizontal load or displacement limit is determined from testing.

(b) Upper core barrel radial displacement to prevent impeding emergency core cooling flow in RV downcomer.

Technical Basis: The limit of the radial outward deformation of the upper core barrel, 60 mm (2.4 in.), is determined such that the flow area of the connection part of the inlet nozzle to the downcomer is not smaller than the inlet pipe section area.

(c) RV and upper head flange loads; lower radial key loads; and postulated core drop bottom of RV impact load and bearing area.

Technical Basis: Lower radial key loads are limited by the RV radial restraints. Postulated core drop bottom of RV impact load and bearing area are also limited by the RV vessel bottom head stresses.

(d) The maximum vertical displacement of the upper core plate relative to the upper support plate should preclude buckling of the guide tube.

Technical Basis: The maximum relative displacement between the upper core plate and the upper core support plate 3 mm (0.1 in.) is based on the axial clearance of the shoulder of guide tube support pin and the upper core plate to avoid the axial loading on the guide tube.

(e) Upper core barrel permanent displacement should not prevent loss of function of the RCCA by radial inwardly deforming the upper guide tube.

Technical Basis: The maximum inward radial deformation of the upper core barrel of 270 mm (10.6 in.) is determined based on the horizontal distance between the lower guide tube and the core barrel inside wall to prevent the interaction with the guide tube.

The staff finds that the applicant has provided the technical basis for defining the displacement limits listed in DCD Tier 2, Table 3.9-2, and the applicant's responses are acceptable with the exception of items (a) and (c). In item (a) of the response the applicant stated that the horizontal load or displacement limit is determined from testing but did not commit to providing this test report as a reference. In item (c) of the response the applicant discussed only the lower radial key loads and postulated core drop bottom of RV impact loads and bearing area, but not the RV and upper head flange loads. Therefore, **RAI 374-2446, Question 03.09.05-17,** is closed as unresolved and in follow-up **RAI 663-4996, Question 03.09.05-34**, the staff requested the applicant to provide (a) the test report, used for determining the horizontal load and displacement limits, for staff review and include it in the references, and (b) the technical basis for defining the loads and displacement limits for the RV and upper head flange.

In its revised response to **RAI 663-4996, Question 03.09.05-34**, dated October 5, 2012, the applicant stated that Technical Report MUAP-11012-P, "US-APWR RCCA Insertion Limit Load Test Report," Revision 0, issued March 2011, was previously transmitted, and will be added as a reference in DCD Tier 2, Section 3.9.10, "References." The applicant also stated that the loads and its bearing area at the RV and the upper head flange are limited by the bearing stress (i.e., contact force per bearing area) on the core barrel flange and the upper core support flange, and will update DCD Tier 2, Table 3.9-2, in the next DCD revision to reflect this. The bearing stress limit for each operating condition is specified in DCD Tier 2, Table 3.9-12. The staff reviewed the RCCA guide tube insertion limit test in MUAP-11012-P and found that the insertion test, which will be referenced in the next revision of the DCD, adequately demonstrates that allowable horizontal load of the RCCA guide tube will not impede insertion of the RCCA after a LOCA event. The staff finds the applicant's response acceptable since it defines the loads and displacement limits for the RV and upper head flange and corresponding changes will be included in the next DCD revision. Accordingly, **RAI 663-4996, Question 03.09.05-34, is being tracked as a Confirmatory Item** pending revision of the DCD.

According to the recommendations in SRP Section 3.9.5, Appendix A, the applicant is also expected to evaluate potential adverse flow effects on piping and components of plant systems. The staff reviewed DCD Tier 2, Revision 1, Section 3.9.5, and found that the applicant did not include an evaluation of these effects. In **RAI 374-2446, Question 03.09.05-19**, the staff requested the applicant to provide a detailed evaluation of potential adverse effects from flow-induced vibrations and acoustic resonances on piping and components of plant systems, including the reactor coolant, steam, feedwater, and condensate systems. Flow-induced vibrations of various sampling probes should also be evaluated. The staff also requested the applicant to substantiate any assumptions made in the analysis, particularly for damping coefficients of structural elements. The staff also requested the applicant to revise DCD Tier 2, Section 3.9.5.2 to include a detailed evaluation of potential adverse flow effects on piping and components of plant systems, including the sampling probes, or refer to this evaluation if it is included elsewhere in the DCD.

In its response to **RAI 374-2446, Question 03.09.05-19**, dated July 17, 2009, the applicant reiterated its response to previous RAIs and noted that the US-APWR piping is designed using the structural design rules based on years of empirical experience with similar plant piping systems which are already in operation for many years. DCD Tier 2, Section 3.9.2 addresses flow-induced vibrations and acoustic resonances on piping and components of plan systems. Section 3.9.2 of this report evaluates the applicant's treatment of flow-induced vibrations and acoustic resonances, including the concerns in **RAI 374-2446, Question 03.09.05-19**. Accordingly, **RAI 374-2446, Question 03.09.05-19, is resolved.**

SRP Section 3.9.5, Appendix A, also recommends that the applicant should maintain monitoring of potential adverse flow effects on plant systems and components for a sufficient time period to verify that adverse flow effects are not occurring (see Appendix A of SRP Section 3.9.5 for more details). The staff reviewed DCD Tier 2, Revision 1, Section 3.9.5 and found that that the applicant did not discuss monitoring of potential adverse flow effects. Therefore, in **RAI 374-2446, Question 03.09.05-20**, the staff requested the applicant to discuss the plans for monitoring potential adverse flow effects in the plant after the initial start-up period. Previous plant experience has shown that adverse flow effects might not appear for an extended period of time following initial start-up. The staff also requested the applicant to revise DCD Tier 2, Section 3.9.5.2 to include adequate information about monitoring of potential adverse flow effects on plant systems and components.

In its response to **RAI 374-2446, Question 03.09.05-20**, dated July 17, 2009, the applicant stated that the vibration measurement will be conducted before core loading as discussed in DCD Tier 2, Sections 3.9.2.3 and 3.9.2.4, “Preoperational Flow-Induced Vibration Testing of Reactor Internals.” More details of this assessment are described in MUAP-07027-P, Revision 1, Section 3.3. The vibration measurement plan is included in MUAP-07027-P, Revision 1, Section 4. The applicant also referred to the response to **RAI 206-1576, Question 03.09.02-43,** on DCD Tier 2, Section 3.9.2.4, which addresses the need of vibration measurement after the core loading.

Regarding vibration monitoring, the applicant referred to DCD Tier 2, Section 4.4.6.3, “Other Monitoring,” explaining that the ex-core neutron detectors can be used to provide continuous monitoring of the core vibration. The detected signals are recorded and analysed by means of a spectrum analyser. Since the vibration frequencies and amplitudes are measured in a preoperational test, the correlation between the detected signals and the core vibration characteristics can be established.

The staff reviewed the above response as well as the applicant’s response to **RAI 206-1576, Question 03.09.02-43** and agrees that since the start-up tests will be performed without the core in place, for which the forcing functions on the reactor internals are higher than those with the core in place, it is adequate to use the ex-core neutron detectors to provide continuous monitoring of the core vibration after the initial start-up tests. Therefore, the applicant has addressed the staff concerns about monitoring potential adverse flow effects. Accordingly, **RAI 374-2446, Question 03.09.05-20, is resolved.**

The staff’s review also identified that neither DCD Tier 2, Revision 1, Section 3.9.2 nor Section 3.9.5, provide any values of damping coefficient values used in the assessment of the dynamic response of the reactor and steam generator internals. Instead, the DCD Tier 2, Subsection 3.9.2.3.3, states that “damping coefficient smaller than the best estimate damping value” is used. The reliability and associated bias and uncertainty errors of the dynamic analysis of the reactor internals and steam generator internals depend on the damping coefficient assumed for various structural components. The use of appropriate damping values is therefore necessary to ensure that the reactor and SG internal structures are designed to quality standards commensurate with the importance of their safety functions. In **RAI 374-2446, Question 03.09.05-21,** the staff requested the applicant to provide and substantiate the damping coefficient values used in the dynamic analysis of the reactor and SG internals, and to support the response to this RAI by referring to available in-plant measurements of damping values for the current 4-loop reactors and SGs. The staff requested the applicant to discuss the damping values used in the following situations, together with the methods used to validate these values and the expected bias error and random uncertainties:

1. Calculations of the vibratory response of the scale model internals and comparison with the measured values for damping coefficient.

2. Calculations of the vibratory response of the US-APWR and comparison with the damping measured for the current 4-loop reactors.

3. Calculations of the vibratory response of the SG internals and comparison with the measured values from operational SGs.

The staff also requested the applicant to revise the applicable subsections of the DCD to include the damping values used in the analysis as outlined above.

In its response to **RAI 374-2446, Question 03.09.05-21**, dated June 19, 2009, the applicant provided the following damping values and explanations:

1. Reactor internals scale model test and analysis:

   (a) Measured values of the damping ratio were identified to the critical damping for the core barrel and the neutron reflectors as described in Table 6-1, “Comparison of Vibration Characteristics of Test Condition and J-APWR Reactor Internals (77GTs),” of MUAP-07023-P, Revision 1.

   (b) In the calculation of the vibration response of the scale model internals, the ratio to the critical damping was applied as described in Table 3.2.3-1, “J-APWR SMT Benchmark Analysis Conditions,” of MUAP-07027-P, Revision 1. This was based on the measured value as mentioned above.

2. Calculations of the US-APWR reactor internals:

   (a) In the calculations of the vibratory response of the US-APWR, the ratio to the critical damping was applied as described in Table 3.3.3-1, “Analysis Matrix (US-APWR Analysis Conditions),” of MUAP-07027-P Revision 1.

   (b) No measured damping ratio is available in the current 4-loop reactors.

3. SG internals:

   The damping ratio used in the SG tube vibration analysis is identified, which is suggested in ASME Code, Section III Appendix-N 1331-3 as conservative value based on the laboratory test data base for avoiding fluid-elastic instabilities of tube arrays.

The staff reviewed the applicant’s response together with MUAP-07023-P, Revision 1 and MUAP-07027, Revision 1, which include the proprietary damping values. The responses to items 1 and 2 are acceptable because the quoted damping values are sufficiently conservative, especially for structures in water flow. The response to item 3 is acceptable because the applicant used a damping value that is in compliance with the recommendations of ASME. Accordingly, **RAI 374-2446, Question 03.09.05-21, is resolved.**

Pending resolution of the aforementioned confirmatory items, the staff finds that the loading conditions on reactor internals are in conformance with the relevant requirements of GDC 1, 2, 4, and 10 as follows.

> The specified design transients, design and service loadings, and combination of loadings as applied to the design of the reactor internals structures and components provide reasonable assurance that in an earthquake or a system transient during normal plant operation the consequent deflections and stresses imposed on these structures and components would not exceed allowable stresses and deformation limits for the materials of construction. Limitation of stresses and deformations under such loading combinations is an acceptable basis for the design of these structures and components to withstand the most adverse loading events postulated to occur during service lifetime without loss of structural integrity or impairment of function.

The reactor internals have been designed to accommodate asymmetric blowdown loads from postulated pipe ruptures. Potential adverse effects of FIV and acoustic resonances on reactor internals, including piping and components of plant systems and internal components in steam generators, have been adequately addressed in accordance with relevant criteria stated in the Appendix A of SRP Section 3.9.5.

3.9.5.4.3 Summary of Design Bases

DCD Tier 2, Section 3.9.5.3 describes the design bases for the manufacture and installation, as well as the operability, of the US-APWR core support structures and internal structures. The rules for materials, design, fabrication, examination, and preparation of reports for the manufacture and installation of the US-APWR core support structures and internal structure follow those in Section III, Subsection NG of the ASME B&PV Code, 2001 Edition up to and including 2003, Addenda. DCD Tier 2, Section 3.9.5.3 states that additional codes, standards, regulations, and guidelines from the NRC and the Utility Requirements Document are adhered to, and are listed in the owner’s design specification.

DCD Tier 2, Section 3.9.5.3 states that the design bases for the operability of the US-APWR core internals are listed and discussed in detail under the following sections: (1) safety analysis; (2) thermal-hydraulic performance; (3) core loading pattern; (4) environmental conditions including radiation; (5) RCS transients; and (6) interface design requirements. DCD Tier 2, Subsection 3.9.5.3.2 describes the reactor coolant flow path for the internals to address the thermal-hydraulic performance requirements. DCD Tier 2, Subsection 3.9.5.3.4, “Environmental Conditions Design-Basis,” identifies the effects of reactor water chemistry and fast neutron fluence considered in the design of US-APWR core internals materials, including (1) corrosion; (2) stress corrosion cracking (SCC); (3) fatigue strength reduction; (4) irradiated assisted SCC; (5) irradiation stress relaxation; (6) irradiation embrittlement; (7) gamma heating; (8) radiation exposure of the RV; and (9) void swelling. DCD Tier 2, Subsection 3.9.5.3.12, “PSI and ISI Plans,” describethe pre-service and ISI plans for the reactor internals.

3.9.5.4.3.1 Design Bases - General

DCD Tier 2, Subsection 3.9.5.3 describes the design bases for the US-APWR reactor internals. To ensure that ASME components meet the service level stress and functionality requirements, ASME Code, Section III, NCA-3000 requires that design specifications and corresponding design reports be prepared. DCD Tier 2, Section 3.9.3, states that the design specifications for ASME Code, Section III, Class 1, 2, and 3 components, supports, and appurtenances are prepared under administrative procedures that meet or exceed the ASME Code, Section III rules. The ASME Code also requires a design report for safety-related components, to demonstrate that the component design meets the requirements of the relevant ASME design specification and the applicable ASME Code, Section III requirements. DCD Tier 2, Section 3.9.5.3, states that the rules for materials, design, fabrication, examination, and preparation of reports for the manufacture and installation of the US-APWR core support structures and internal structures follow those in Section III, Subsection NG of the ASME B&PV Code (2001 Edition up to and including 2003 Addenda).

DCD Tier 2, Section 3.9.5.3 describes the design bases for the operability of the US-APWR internals, including: (1) safety analysis; (2) thermal-hydraulic performance; (3) core loading pattern; (4) environmental condition; and (5) other design bases.

3.9.5.4.3.2 Design Bases - Safety Analysis

DCD Tier 2, Subsection 3.9.5.3.1 states that the safety analysis design requirements and limits for the US-APWR core internals are as follows:

- Mal-distribution of flow to the core should be limited so as not impact core safety limits in Chapter 15.

- RCCA drop times or insertion during normal service conditions should not be adversely affected.

- RCCA are to be inserted without impediment after an unanticipated accident, or a seismic and postulated LOCA event.

- There should be no impediment of the reactor internals to the emergency core cooling flow after a seismic event and postulated LOCA event.

- The impact load on the RV bottom head from a postulated core drop event should not adversely affect the integrity of the RV bottom head.

- The reactor internals are to provide fast neutron fluence protection to the RV to preclude excessive embrittlement.

- The water volume is to be monitored at all times.

The staff's review of the DCD indicates that DCD Tier 2, Section 3.9.4.2, "Applicable CRDS Design Specifications," addresses the safety-analysis design requirement and limit for RCCA drop times, which are evaluated in Section 3.9.4 of this report. DCD Tier 2, Section 3.9.2.5, "Dynamic System Analysis of the Reactor Internals under Faulted Conditions," addresses the safety-analysis design requirement for RCCA insertion and emergency core cooling flow without impediment after an unanticipated accident or a seismic event and postulated LOCA, which are evaluation in Section 3.9.4 of this report. Therefore, the staff finds the description of the safety analysis design-basis acceptable.

3.9.5.4.3.3 Design Bases - Thermal Hydraulics

DCD Tier 2, Subsection 3.9.5.3.2 states that the US-APWR core internal components are to be designed for the following thermal-hydraulic performance parameters:

- The flow conditions are: thermal design flow, best estimate flow, mechanical design flow, hot pump overspeed, and hot functional testing.

- Pressure drops across the reactor internals are to meet system requirements for all Level A and B service flow conditions.

- Bypass flow is to be minimized and must not exceed system requirements for normal operation.

- Distribution of main coolant inlet flow into the fuel assemblies during normal operation must meet core inlet requirements for the fuel assemblies.

- The core outlet flows from the fuel assemblies are to be designed to minimize horizontal velocities that may contribute to vibration of the RCCA rodlets.

- Main coolant flow into the outlet piping during normal operation is to meet system requirements, specifically (1) to minimize exit fluid temperature striations, and (2) meet the velocity criteria to prevent erosion.

- The bulk temperature of the main coolant flow is not to exceed the pressurized water saturation temperature.

- Fluid temperature increase in the bypass flow may be credited to the reactor power output.

DCD Tier 2, Subsection 3.9.5.3.2 describes in depth the reactor coolant flow path for the reactor internals. DCD Tier 2, Figure 3.9-8, "Reactor Internals RCS Flow and Bypass Flow Paths," shows the reactor coolant flow paths. The applicant stated that the primary coolant flows down through the down-comer and at the bottom of the RV it turns upward, flowing past the diffuser plates and distributing into the orificed holes of the lower core support plate. The orifices are designed to control the flow into the fuel assemblies and to minimize uneven flow distributions and hot spots. As noted above, DCD Tier 2, Subsection 3.9.5.3.2 states that the distribution of main coolant inlet flow into the fuel assemblies during normal operation must meet the requirements of the fuel assembly core inlet. However, the DCD does not provide any details about these requirements or how compliance with the requirements is verified. In **RAI 374-2446, Question 03.09.05-23**, the staff requested the applicant to provide additional details regarding the fuel-assembly core inlet requirements to explain how compliance with these design requirements during service is verified. The staff also requested the applicant to revise the applicable subsections of the DCD to include the requested information.

In its response to **RAI 374-2446, Question 03.09.05-23**, dated July 17, 2009, the applicant referred to the response to **RAI 374-2446, Question 03.09.05-7**. As discussed above, the staff finds the response to **RAI 374-2446, Question 03.09.05-7** partially acceptable because it indicates that well defined design targets are established for the flow distribution into the core. These design targets are similar to those used for the operating 4-loop U.S. plants. In addition, confirmatory tests have been performed to validate that the design parameters of the US-APWR are within the acceptable range to avoid mal-distribution of flow into the core. However, in its response the applicant cited a reference for the confirmatory test results for the US-APWR core internals but did not identify it. Additionally, the applicant did not revise the DCD. Therefore, the staff closed as unresolved **RAI 374-2446, Questions 03.09.05-7 and 03.09.05-23**. In a follow-up **RAI 663-4996, Question 03.09.05-32**, the staff requested the applicant to identify the reference and revise the DCD to include the requested information. As discussed above in 3.9.5.4.1.3, the applicant provided an acceptable response and **RAI 663-4996, Question 03.09.05-32 is being tracked as a Confirmatory Item** pending revision of the DCD.

DCD Tier 2, Subsection 3.9.5.3.2 states that the coolant is heated in each fuel assembly to a fluid temperature depending on its location in the core loading pattern. To preclude bulk-boiling in the main coolant flow, the fuel-assembly exit temperature is not allowed to exceed the water saturation temperature. As stated above, this is also a thermal-hydraulics performance requirement for the design of reactor internals.

After exiting the fuel assemblies, the coolant flows through flow holes in the upper core plate. The upper core plate has two types of flow holes. One type is circular in shape and the other type is rectangular in shape. The circular shape is for open exit flow or exit flow below the upper support columns. The rectangular shape is for exit flow below the guide tubes. Most of the main coolant that enters the guide tube exits through "windows" into the upper plenum cavity. Some of the coolant exits through a controlled gap between the bottom of the guide tube flange and the top of the upper core plate. DCD Tier 2, Subsection 3.9.5.3.2 states that the guide tubes and support columns are carefully configured to minimize the pressure drop and cross-flow from the core exit fluid. The thermal-hydraulic performance criteria, listed above, require that the core outlet flows from the fuel assemblies are to be designed to minimize horizontal velocities that may contribute to vibration of the RCCA rodlets.

In **RAI 374-2446, Question 03.09.05-6,** the staff requested the applicant to describe the procedure that is to be used to verify that the exit flow from the fuel assemblies does not lead to unacceptable cross-flow velocities that may cause vibration of the fuel rods, thimbles, or RCCAs. As discussed above, the staff finds the response to **RAI 374-2446, Question 03.09.05-6,** as unresolved. However, the applicant provided an acceptable assessment of cross-flow in the response to the follow-up **RAI 663 4996, Question 03.09.05-31**. As discussed above, **RAI 663 4996, Question 03.09.05-31, is resolved** and the staff finds the thermal-hydraulic performance criteria related to cross-flow acceptable.

DCD Tier 2, Subsection 3.9.5.3.2, stated that the main coolant flow then mixes in the upper plenum and exits from the core-barrel outlet nozzles at an average fluid temperature of T_{hot}. The applicant further stated that special flow columns are spaced on the periphery of the upper core plate near the core barrel outlet nozzles to improve mixing and minimize mal-distribution of the outlet fluid temperature. DCD Tier 2, Subsection 3.9.5.3.2, requires that the main coolant flow into the outlet piping during normal operation meets the system requirements; specifically, the exit fluid temperature striations are minimized and velocity criteria to prevent erosion are met. DCD Revision 1 did not provide any details on how compliance with these system requirements is verified. In **RAI 374-2446, Question 03.09.05-24**, the staff requested the applicant to provide additional details regarding the system requirements for exit fluid velocity and temperature striations, and to describe the procedure used to verify compliance with these requirements during service. The staff also requested the applicant to revise the applicable subsections of the DCD to include the requested information.

In its response to **RAI 374-2446, Question 03.09.05-24**, dated July 17, 2009, the applicant stated that although it is considered to be good design practice to minimize the exit fluid temperature striations, no design criterion is specified for the exit fluid temperature striations. The applicant further stated that the outlet nozzle diameter was determined on the basis of the increased flow rate of the APWR design relative to the current 4-loop design, to assure the equivalent margin to erosion. The staff finds the response acceptable because the applicant has qualitatively designed the core internals to achieve an acceptably low exit temperature striation value. Therefore, the staff's concerns regarding the system requirements for exit fluid velocity and temperature striations are resolved. Accordingly, **RAI 374-2446, Question 03.09.05-24, is resolved**.

DCD Tier 2, Subsection 3.9.5.3.2 states that some percentage of the main coolant flow is bypass flow, which is either for cooling metal or leakage between gaps. The bypass flows from gap leakages are as follows: small gap between the core-barrel outlet nozzle and RV outlet nozzle, neutron-reflector ring block inside surface and peripheral fuel assembly grids and nozzles, and neutron-reflector small gaps between the ring blocks. However, the applicant did not assess the liability of the core barrel flange to leakage flow-induced vibration. In **RAI 374-2446, Question 03.09.05-25,** the staff requested the applicant to discuss the liability of the core barrel flange to flow-induced vibration caused by the leakage (or bypass) flow between the outlet nozzle of the core barrel flange and the RV exit nozzle. Since the diameter of the core barrel flange is larger than that of the 4-loop reactors, its shell modes have lower frequencies. In addition, the leakage flow rate is higher in the US-APWR than in the 4-loop reactors. The staff requested the applicant to provide evidence showing that the leakage flow between the outlet nozzle of the core barrel flange and the RV exit nozzle will not cause excessive vibration of the core barrel flange. The staff also requested the applicant to revise DCD Tier 2, Section 3.9.5, to include an assessment of the leakage flow effects on the core barrel flange.

In its response to **RAI 374-2446, Question 03.09.05-25**, dated June 19, 2009, the applicant stated that based on the experience from previous plants there has been no reported evidence of nozzle gap by-pass flow being a major contributor to core barrel vibration response. Further, the applicant explained that the bypass flow in the gap between the core barrel and RV has little effect on the core barrel vibration because the flow rate and the flow contact area of the gap are much smaller than those of the downcomer. However, the applicant did not address the staff concern about possible vibration due to leakage flow instability. Therefore, the staff closed as unresolved **RAI 374-2446, Question 03.09.05-25**, and in follow-up **RAI 646-5065, Question 03.09.02-92,** the staff requested the applicant to provide evidence or a basis for stating that leakage flow vibration is not a concern in the US-APWR. As discussed in Section 3.9.2 of this report, the applicant addressed the leakage flow vibration issue in the response to **RAI 646-5065, Question 03.09.02-92.** Therefore, the concern is resolved.

3.9.5.4.3.4 Design Bases - Core Loading Pattern and Axial Power Distribution

DCD Tier 2, Subsection 3.9.5.3.3, “Core Loading Pattern and Axial Power Distribution Design-Basis,” states that the core loading pattern and the axial power distribution have a design impact on the gamma heating, fluid and metal temperatures, and neutron fluence of the reactor internals. The reactor internals that are affected by the core loading pattern and fluence include the following: core barrel, upper core plate, lower core support plate, neutron-reflector ring blocks, neutron-reflector ring blocks alignment pins, neutron-reflector tie rods and mounting bolts, and irradiation specimen guide and bolts. The staff finds the description acceptable since it identifies those reactor internals that may be impacted by the core loading pattern.

3.9.5.4.3.5 Design Bases - Environmental Condition

DCD Tier 2, Subsection 3.9.5.3.4, describes the effects of reactor water chemistry and fast neutron fluence that are considered in the design of US-APWR core internals. The environmental effects on reactor internals materials include: (1) corrosion, (2) stress corrosion cracking, (3) fatigue strength reduction, (4) irradiated assisted SCC, (5) irradiation stress relaxation, (6) irradiation embrittlement, (7) gamma heating, (8) radiation exposure of the RV, and (9) void swelling. The materials to be used in the construction of the US-APWR reactor internals are described in DCD Tier 2, Section 4.5.2.

In its review of DCD Tier 2, Revision 1, Section 3.9.5.3 the staff found that the DCD does not adequately address aforementioned environmental effects. In DCD Tier 2, Subsection 3.9.5.3.4 the applicant stated that corrosion, SCC, radiation embrittlement, and degradation of fatigue strength are not considered to be an issue for the operating conditions of the US-APWR, and that the potential for irradiated assisted SCC of the US-APWR core internals is very low. The applicant further stated that void swelling from neutron irradiation was a concern for components with high dose of neutron fluence, such as neutron reflector ring blocks, but they are cooled to keep metal temperature low and minimize void swelling. However, the applicant did not provide estimates of temperature and end-of-life neutron fluence for the various reactor internal components or identify the components where void swelling, radiation embrittlement, irradiated assisted SCC, and degradation in fatigue strength are likely to be significant. The PWSCC of Ni-alloys, such as X-750, is also not addressed in the DCD. In addition, the DCD does not provide an assessment of environmental effects on the structural and functional integrity of reactor internals. Therefore, in **RAI 374-2446, Question 03.09.05-9**, the staff requested the applicant to (a) describe the environmental conditions, including estimates of the temperature and end-of-life neutron fluence, for the various reactor internal components, and (b) either provide an evaluation to verify that, under the operating conditions of the US-APWR, the effects of corrosion, SCC, irradiated assisted SCC, PWSCC, degradation of fatigue strength, radiation embrittlement, and void swelling on the structural and functional integrity of the reactor internal components are not a concern during the design life of 60 years, or define an acceptable program for investigating and managing these environmental effects on reactor internals. The applicant was also requested to revise the DCD to include the requested information or provide a reference where this information is available or alternately, the applicant can provide a reference document where the requested information is available.

In its response to **RAI 374-2446, Question 03.09.05-9**, dated July 17, 2009, the applicant stated that the effects of irradiation fluence and PWR water chemistry on the functional and structural integrity of the reactor internal components have been assessed. The applicant stated that the effects of corrosion and PWSCC are not expected to be a concern for reactor internals because the water chemistry is tightly controlled and most of the reactor internals are fabricated from austenitic stainless steel material which has shown to be robust in a PWR water chemistry environment. The applicant also stated that other materials such as water quenched Inconel X-750 for support pins, and Type 403 material for the hold-down spring have been successfully used in 4-Loop operating plants under stress and environmental conditions similar to those for the US-APWR design. The staff agrees that current operating experience does not suggest that general corrosion and PWSCC will be a concern for the US-APWR internals. The staff finds the applicant's assessment of general corrosion and PWSCC to be acceptable because it meets the criteria of GDC 4 for the design of the reactor internals compatible with environmental conditions associated with the operation of the reactor for a design life of 60 years.

In its response to **RAI 374-2446, Question 03.09.05-9**, the applicant also stated that the effects of neutron irradiation such as irradiated assisted SCC, embrittlement, and void swelling have been evaluated for those components subjected to high dose rates, such as the neutron reflector block; neutron reflector block alignment pin; neutron reflector tie-rod, and core barrel. The applicant also provided the estimated end-of-life irradiation neutron dose and the metal temperature of the components subjected to high neutron dose. The applicant's analysis of each of these topics (void swelling, embrittlement, and irradiated assisted SCC) is evaluated individually as follows.

In its response to **RAI 374-2446, Question 03.09.05-9**, the applicant stated that the effects of void swelling were evaluated in accordance with the guidelines of EPRI Report 1012081, "Materials Reliability Program: PWR Internal Material Aging Degradation Mechanism Screening and Threshold Values (MRP-175)," issued December 2005. The applicant further stated that in response to **RAI 269-2155, Question 04.05.02-2**, the void swelling of the neutron reflector, which represents the bounding value for all reactor internals, was evaluated. The applicant also stated that in accordance with MRP-175 no functionality evaluation is required if void swelling is less than the threshold value of 2.5 percent. The applicant determined that there is a wide margin between the calculated void swelling and the threshold value. There is also a wide margin between the threshold value and value which operating experience and research have shown to be a problem (5 to 10 percent). Therefore, the staff agrees that the estimated void swelling of neutron reflector is insignificant and not likely to have a significant effect on dimensional change of the component. The staff finds the applicant's assessment of void swelling to be acceptable because it meets the criteria of GDC 4 for the design of the reactor internals compatible with environmental conditions associated with the operation of the reactor for a design life of 60 years.

However, the staff further noted that the applicant did not provide an assessment of neutron embrittlement of the reactor internal components. The data presented in MRP-175 indicate significant reduction in fracture toughness of austenitic stainless steels and welds exposed to neutron dose above 1.5 displacements per atom (dpa).

Therefore, the staff closed as unresolved **RAI 374-2446, Question 03.09.05-9**. In follow-up **RAI 971-6681, Question 03.09.05-35,** the staff requested the applicant to a) describe how it will evaluate the effect of neutron embrittlement on components subjected to high dose rates, and b) describe the ISI plan (type of inspection, inspection area, frequency, acceptance criteria, etc.) for the components subjected to high dose rates to ensure that neutron embrittlement will not adversely affect the integrity of these components.

In its amended response to **RAI 971-6881, Question 03.09.05-35**, dated March 29, 2013, the applicant evaluated the effect of neutron embrittlement on the neutron reflector ring block, neutron reflector block alignment pin, neutron reflector tie-rod, and core barrel which are internal components receiving a lifetime irradiation level exceeding 1.5 dpa. The applicant's analysis of these components was performed using the criteria of MRP-175 Appendix F and showed that the calculated stress intensity factor (K) is less than the lower bound fracture toughness (K_{JC}). Current operating experience data as discussed in MRP-175 would not indicate that irradiation embrittlement would be a concern for components with stress intensity factor calculated to be below the lower bound fracture toughness. The staff finds the applicant's assessment of irradiation embrittlement to be acceptable because it meets the criteria of GDC 4 for the design of the reactor internals compatible with environmental conditions associated with the operation of the reactor for a design life of 60 years. Therefore the staff finds the response to **RAI 971-6881, Question 03.09.05-35,** to be acceptable. Accordingly, **RAI 971-6881, Question 03.09.05-35, is resolved.**

In its response to **RAI 374-2446, Question 03.09.05-9**, dated July 17, 2009, the applicant stated that the assessment of irradiated assisted SCC was performed in accordance with the guidance of MRP-175. The staff evaluated the applicant's assessment of irradiated assisted SCC in Section 4.5.2 of this report. The staff finds the applicant's assessment of irradiated assisted SCC to be acceptable because it meets the criteria of GDC 4 for the design of the reactor internals compatible with environmental conditions associated with the operation of the reactor for a design life of 60 years.

Based on the above, the staff finds the applicant's response to **RAI 374-2446 Question 03.09.05-9,** to be acceptable regarding the effects of corrosion, SCC, irradiated assisted SCC, PWSCC, degradation of fatigue strength, and void swelling on the structural and functional integrity of the reactor internal components since the response showed that these effects are not a concern during the design life of the reactor internal components. As discussed above, the response to **RAI 374-2446 Question 03.09.05-9**, did not fully address neutron embrittlement, which was addressed by the follow-up **RAI 971-6881, Question 03.09.05-35.**

DCD Tier 2, Subsection 3.9.5.3.4 includes the potential effects of irradiation stress relaxation in the list of environmental effects on reactor core internal materials caused by long term exposure to fast neutron irradiation. DCD Tier 2, Subsection 3.9.5.1.2, states that neutron fluence and temperature limits are imposed on the tie-rods to preclude excessive loss of preload from irradiation stress relaxation. The staff reviewed the DCD but did not find where the applicant had provided an evaluation of the loss of preload in various threaded fasteners due to irradiation stress relaxation or a reference where this information was available. Also, the applicant did not identify the fasteners where the effect of irradiation stress relaxation is expected to be significant. Moreover, it is not clear how the pre-stress will be maintained in the preloaded components such as the tie-rods of the ring block or the hold-down bolts of the guide tube. The flow-induced structural response of the guide tubes can be affected by loss of preload in the hold-down bolts that secure the guide tube assembly to the upper core support plate, or in the guide tube support pins and the flexible leaves that are horizontally preloaded against the upper core plate holes. Also, loss in preload in the tie-rods of the neutron reflector ring block can compromise the structural integrity of the neutron reflector assembly. Therefore, in **RAI 374-2446, Question 03.09.05-10**, the staff requested the applicant to assess potential loss of preload due to irradiation stress relaxation in various threaded fasteners, in particular, the guide tube hold-down bolts, guide tube support pins and the flexible leaves, and the neutron reflector tie-rods, and to examine its effect on the structural and functional integrity of the components. The staff requested the applicant to revise the DCD to include the requested information or provide a reference where the requested information is available.

In its response to **RAI 374-2446, Question 03.09.05-10**, dated July 17, 2009, the applicant stated, "The irradiation level of the various threaded fasteners is much smaller than tie-rod, and the volume of the potential loss is small." The applicant also stated, "The neutron level of the tie-rod is comparable to the neutron reflector block, and the preload of the tie-rod have been decreased for 60 years." The applicant further added, "The function of the tie-rod is to restrain the axial response of the neutron reflector blocks in SSE and LOCA events if the tie-rod is not fastened, the function requirement of tie-rod is acceptable." The staff was unable to evaluate this response since it was unclear and there was no objective assessment of irradiation assisted stress relaxation. Therefore, in **RAI 784-5887, Question 04.05.02-25**, the staff requested the applicant to address the following areas:

- Functionality of tie-rod fasteners.
- Screening criteria for irradiation assisted stress relaxation.
- Evaluation of internal fasteners functionality based on screening criteria.
- ISI guidance for neutron reflector/tie rod assembly.

In the second amended response to **RAI 784-5887, Question 04.05.02-25**, dated April 2, 2012, the applicant clarified that:

- The tie-rod must be fastened and maintain some pre-load to restrain vertical lift-off of the neutron reflector blocks during seismic and LOCA events. However, if pre-load of the tie-rod fasteners is reduced due to radiation effects, the functionality of the tie-rod fasteners is not degraded. The applicant has stated in DCD Tier 2, Section 3.9.5.3.4 that preloaded tension on the tie-rod is reduced as a result of irradiation generated during normal plant operation. Because of this, the tie-rod is designed to ensure some preload is left at the end of a 60 year plant life in order to ensure that its functional requirements are met.

- On September 5, 2012, the applicant amended the response to **RAI 374-2446, Question 03.09.05-10**, and included a figure of the neutron reflector assembly with tie-rods, updated descriptions in DCD Tier 2, Section 3.9.5.1.2 clarifying the arrangement, and updated DCD Tier 2, Subsection 3.9.5.3.4 to be consistent with its response above to **RAI 784-5887, Question 04.05.02-25**. The staff reviewed the proposed changes to the DCD and found that the applicant has addressed preload in the design of the tie-rod fastener that meets a design life of 60 years. Therefore the staff finds the revised response to **RAI 374-2446, Question 03.09.05-10** to be acceptable**.**

- A two stage design screening criteria for irradiation assisted stress relaxation would be used. For neutron fluence less than or equal to 0.2 dpa, no assessment is performed. For neutron fluence above 0.2 dpa, the applicant performed a simple assessment and determined that critical preload would not be lost as long as neutron fluence was less than 0.6 dpa. Allowing for uncertainty, the applicant proposes that for neutron fluence exceeding 0.5 dpa, the applicant would perform a special assessment since irradiation is assumed to reduce fastener preload. The staff finds the screening criteria acceptable because it conforms to ASME Code NG and would ensure fasteners maintain preload for a design life of 60 years.

  - The applicant provided a list of reactor internal fasteners and assessed the neutron radiation exposure of these fasteners. For those fasteners, with neutron exposures in excess of the 0.2 dpa screening criteria, the applicant performed an evaluation to determine that no critical fastener of reactor internals is identified as losing its function due to irradiation stress relaxation. For fasteners exceeding 0.5 dpa, a special assessment was performed.

  - Four types of fasteners exceeded the 0.2 dpa screening criteria but not the 0.5 dpa screening criteria. These fasteners were evaluated and found to be acceptable with ASME NG and not to lose critical preload.

  - The Neutron Reflector tie rod may see a fluence of up to 4.1 dpa. The applicant performed an assessment and determined that irradiation assisted stress relaxation would decrease preload on the tie rods over the life of the plant. However, the applicant's assessment also determined that the loss in preload would not negatively affect the function of the tie-rods which are to capture the neutron reflector blocks and prevent vertical lift off during seismic and LOCA events. The applicant also stated that the tie-rod is designed to ensure some preload is left at the end of life.

  - The fuel alignment pin may see a fluence of up to 1.6 dpa. However, since the pin is designed as a lateral support against shear loading and is not a threaded fastener, there is no preload and thus the preload requirements are not applicable.

The applicant further proposed to modify DCD Tier 2, Section 4.5.2 to state that the assessment of stress relaxation of reactor internals fasteners is conducted using the screening criteria of MRP-175 based on irradiation analysis, and that irradiation induced stress relaxation does not cause a loss of functionality during the 60 year life of the reactor. The staff finds the screening criteria and results acceptable because it conforms to ASME Code NG and ensures preload through a period of 60 years. Therefore, this portion of the response to **RAI 784-5887, Question 04.05.02-25,** is acceptable.

- Inspection guidance from EPRI Report 1016596, "Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227)," issued December 2008, (which was applicable to baffle former bolt assemblies) is not applied for the US-APWR neutron reflector tie-rods since MRP-227 is intended for highly stressed assemblies that are susceptible to irradiated assisted SCC. The applicant further proposed to modify DCD Tier 2, Section 4.5.2 to state that the inspection guidance of MRP-227 is not applied to the US-APWR neutron reflector tie-rods. The applicant also proposed to add MRP-175 and MRP-227 as references in DCD Tier 2, Section 4.5.4, "References." The staff has evaluated and accepted that the neutron reflector tie-rods are below the stress threshold for irradiated assisted SCC. This evaluation is further discussed in Section 4.5.2 of this report. Therefore, this portion of the response to **RAI 784-5887, Question 04.05.02-25,** is acceptable.

As described above, based on the applicant's revised response to **RAI 374-2446, Question 03.09.05-10** and its second amended response to **RAI 784-5887, Question 04.05.02-25,** the staff finds that the effects of irradiation assisted stress relaxation are not a concern during the design life of the reactor internal components. Therefore, the staff finds the responses acceptable. **RAI 374-2446, Question 03.09.05-10 and RAI 784-5887, Question 04.05.02-25 are being tracked as a Confirmatory Items** pending revision of the DCD.

3.9.5.4.3.5 Design Bases – Other Design Bases

The interface design requirements, and RCS transients and other design bases are described in DCD Tier 2, Subsections 3.9.5.3.5, "RCS Transient Design-Basis," to 3.9.5.3.11, "Mechanical Design Criteria for the Reactor Internals." DCD Tier 2, Subsection 3.9.5.3.5 states that the RCS transient design-basis is discussed in DCD Tier 2, Section 3.9.1.1, "Design Transients." DCD Tier 2, Section 3.9.1.1 states that to assure high quality for the RCS piping and components, the transient conditions are based on the conservative estimates of the magnitude and frequency of the temperature and pressure transients resulting from normal, upset, emergency, and faulted-service condition and test condition. To a large extent, the specific transients considered for design are based on engineering judgment and experience. Design transients are evaluated in Section 3.9.1 of this report.

DCD Tier 2, Subsection 3.9.5.3.6, "Reactor Internals Vibration Design-Basis," states that the reactor internals vibrations loads come from a dynamic computer model that inputs the pressure difference across components and the pump rotating speed and pump-induced vibration effects. The mechanical loads and displacements from the vibration analysis are used as input to the structural analysis of the reactor internals. The validation of the FE analysis and computer codes are evaluated in Section 3.9.2 of this report.

DCD Tier 2, Subsection 3.9.5.3.7, “Seismic Design-Basis,” states that the seismic design-basis is discussed in DCD Tier 2, Section 3.9.2.5. The seismic analysis methodology is based on static and dynamic mathematical models and uses general-purpose FE computer code. DCD Tier 2, Subsection 3.9.5.3.8, “LOCA Design-Basis,” states that the input for the LOCA design-basis is discussed in Subsection 3.9.2.5 of the DCD. These two topics are evaluated in Section 3.9.2 of this report.

DCD Tier 2, Subsection 3.9.5.3.9, “Interface Components Design-Basis,” states that the interface components design-basis consists of those design parameters and design requirements that affect the design of the core support and internal structures. The applicant states that the parameters and requirements for interface components (such as the RV, fuel assemblies, CRDM, and RCCA drive-line system, thermocouple instrumentation, and ICIS) are included in the design specification.

DCD Tier 2, Subsection 3.9.5.3.10, “Reactor Internals Computational Methods and Verification of Input,” Computational methods such as FE analysis are used to determine stresses and displacements in the reactor internal components. Validation of the computation methods includes (a) the comparison of results with similar designs or (b) testing for the natural frequencies, mode shapes, and frequency response functions with experimental or plant results. The validation of the FE analysis and computer codes are evaluated further in Section 3.9.2 of this report.

DCD Tier 2, Subsection 3.9.5.3.11, states that the mechanical design criteria for the reactor internals is included in the design specifications.

DCD Tier 2, Subsection 3.9.5.3.12, describes the pre-service inspection and ISI plans for the US-APWR core internals. The applicant stated that the pre-service inspection plan as well as the ISI plan follows the rules of ASME Code, Section XI. This area of review is evaluated in Section 5.2.4 of this report.

The staff finds the description of the other design bases identified above acceptable as it is consistent with the guidelines of SRP Section 3.9.5 Subsection I.2.

3.9.5.4.4 Inspections, Tests, Analyses, and Acceptance Criteria

The ITAAC for reactor internals are evaluated in Section 14.3.4 of this report.

3.9.5.5 Combined License Information Items

There are no COL information items for this subsection.

3.9.5.6 Conclusions

For the reasons set forth above, the staff concludes that the design bases for the mechanical design of the RV internals, including design bases for the operability of the US-APWR core internals and the pre-service inspection and ISI plans, are in conformance with the relevant requirements of GDCs 1, 2, 4, and 10, 10 CFR 50.55a, and 10 CFR 52.47(b)(1) upon satisfactory completion of the following confirmatory items:

1. **RAI 374-2446, Question 03.09.05-10**, for revision to DCD Tier 2, Subsections 3.9.5.1.2 and 3.9.5.3.4.

2. **RAI 374-2446, Question 03.09.05-30**, for revision to DCD Tier 2, Subsection 3.9.5.1.1.

3. **RAI 663-4996, Question 03.09.05-32** for revision of DCD Tier 2, Subsection 3.9.5.3.2, to include the design criteria that limit mal-distribution of flow into the core.

4. **RAI 374-2446, Question 03.09.05-33**, for revision to DCD Tier 2, Subsection 3.9.5.2.2.

5. **RAI 663-4996, Question 03.09.05-34**, for revision to DCD Tier 2, Section 3.9.10, and Table 3.9-2.

6. **RAI 784-5887, Question 04.05.02-25**, for revision to DCD Tier 2, Sections 4.5.2 and 4.5.4.

3.9.6 Functional Design, Qualification, and Inservice Testing Programs for Pumps, Valves, and Dynamic Restraints

3.9.6.1 Introduction

This section evaluates the functional design, qualification, and IST programs for safety-related pumps, valves, and dynamic restraints (snubbers) described in Revision 3 to the US-APWR DCD.

3.9.6.2 Summary of Application

DCD Tier 1: There are no DCD Tier 1 requirements specific to IST programs for the US-APWR design. The system-based descriptions of DCD Tier 1, Chapter 2 address regulatory requirements for design and qualification of system components.

DCD Tier 2: The applicant has provided a DCD Tier 2 system description in DCD Tier 2, Section 3.9.6, “Functional Design, Qualification, and Inservice Testing Programs for Pumps, Valves, and Dynamic Restraints,” which is summarized below:

> DCD Tier 2, Section 3.9.6 includes in DCD Tier 2, Section 3.9.6.1, “Functional Design and Qualification of Pumps, Valves, and Dynamic Restraints,” provisions for the functional qualification of pumps, valves, and dynamic restraints. DCD Tier 2, Section 3.9.3.3, “Pump and Valve Operability Assurance,” includes provisions for the qualification of safety-related pumps and valves. DCD Tier 2, Section 3.9.3.4, “Component Supports,” includes provisions for the qualification of dynamic restraints.
>
> DCD Tier 2, Section 3.9.6 describes the IST program for safety-related pumps, valves, and dynamic restraints typically designated as ASME B&PV Code Class 1, 2, or 3. DCD Tier 2, Section 3.9.6 states that the operational readiness of ASME B&PV Code Class 1, 2 and 3 safety-related pumps, valves, and dynamic restraints will be verified according to the IST requirements set forth in 10 CFR 50.55a. The DCD specifies that IST provisions for safety-related pumps, valves, and dynamic restraints are incorporated into the design

of the US-APWR. DCD Tier 2, Section 3.9.6.1 references other sections in DCD Tier 2 for design provisions related to testing pumps, valves, and dynamic restraints.

DCD Tier 2, Section 3.9.6.2, “IST Program for Pumps,” and DCD Tier 2, Table 3.9-13, “Pump IST,” specify IST provisions for safety-related pumps in support of the US-APWR DC. DCD Tier 2, Table 3.9-13 includes test frequency and acceptance criteria for pumps within the IST program in the US-APWR.

DCD Tier 2, Section 3.9.6.3, “IST Program for Valves,” and DCD Tier 2, Table 3.9-14, “Valve Inservice Test Requirements,” specify IST provisions for safety-related valves in support of the US-APWR DC. DCD Tier 2, Table 3.9-14 includes the valve type, safety positioning requirements, safety functions, and IST categories and testing frequencies in the ASME *Code for Operation and Maintenance of Nuclear Power Plants* (OM Code) for each valve within the IST program in the US-APWR.

The IST program is described for each of the following valve types for the US-APWR:

- Power-operated valves (POVs), such as motor-operated valves (MOVs), air-operated valves (AOVs), hydraulic-operated valves (HOVs), and solenoid-operated valves (SOVs).
- Check valves.
- Pressure and CIVs.
- Safety and relief valves.
- Manual valves.

DCD Tier 2, Section 3.9.6.4, “IST Program for Dynamic Restraints,” specifies provisions for preservice and inservice examinations for snubbers described in Subsection ISTD, “Preservice and Inservice Examination and Testing of Dynamic Restraints (Snubbers) in Light-Water Reactor Nuclear Power Plants,” of the ASME OM Code.

DCD Tier 2, Section 3.9.6.5, “Relief Request and Authorization to ASME OM Code,” states that experience has been used in designing and locating pumps, valves, and dynamic restraints to permit access for performing preservice testing (PST) and IST required by the ASME OM Code. The DCD specifies that relief requests will be prepared where full compliance with the ASME OM Code testing requirements cannot be satisfied.

ITAAC: There are no ITAAC specific to the IST program. The system-based descriptions contained in the DCD Tier 1, Chapter 2 address regulatory requirements related to functional design and qualification of system components.

TS: TS 5.5.8, “Inservice Testing Program,” in DCD, Chapter 16, “Technical Specifications,” requires that licensees perform IST activities in accordance with the frequency specified in the ASME OM Code.

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

Technical Reports: There are no technical reports for this area of review.

Cross-cutting Requirements (TMI, USI/ GSI, Op Ex): There are no cross-cutting requirements for this area of review.

US-APWR Interface Issues Identified in the DCD: See DCD Tier 2, Table 1.8-2, "Compilation of All Combined License Applicant Items for Chapters 1-19."

Site Interface Issues Identified in the DCD: See DCD Tier 2, Table 1.8-1, "Significant Site Specific Interfaces with the Standard US-APWR Design."

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

3.9.6.3 Regulatory Basis

The relevant requirements of the Commission's regulations for this area of review, and the associated acceptance criteria are listed in Section 3.9.6, "Functional Design, Qualification, and Inservice Testing Programs for Pumps, Valves, and Dynamic Restraints," Revision 3, issued March 2007, of NUREG-0800 and are summarized below. Review interfaces with other SRP sections can be found in Section 3.9.6 of NUREG-0800.

1. 10 CFR 50.55a and GDC 1 require that pumps, valves, and dynamic restraints important to safety be designed, fabricated, tested, and inspected to quality standards commensurate with the importance of the safety functions to be performed.

2. GDC 2 requires that pumps, valves, and dynamic restraints important to safety be designed to withstand the effects of natural phenomena combined with the effects of normal and accident conditions without loss of capability to perform their safety functions.

3. GDC 4 requires that pumps, valves, and dynamic restraints important to safety be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents.

4. GDC 14 requires that pumps, valves, and dynamic restraints that are part of the reactor coolant boundary be designed to have an extremely low probability of abnormal leakage, rapidly propagating failure, and gross rupture.

5. GDC 15 requires that pumps, valves, and dynamic restraints within the RCS be designed with sufficient margin to assure that the design conditions are not exceeded.

6. GDC 37 requires that the ECCS be designed to permit periodic pressure and functional testing to assure leak-tight integrity and performance of its active components.

7. GDC 40 requires that periodic functional testing of the containment heat-removal system assure leak-tight integrity and performance of its active components, and of the operability of the system as a whole.

8. GDC 43 requires that the containment atmospheric cleanup systems be designed to permit periodic functional testing to assure leak-tight integrity and performance of the active components, and of the operability of the system as a whole.

9. GDC 46 requires that the cooling water system be designed to permit periodic functional testing to assure leak-tight integrity and performance of the active components, and of the operability of the system as a whole.

10. GDC 54 requires that piping systems penetrating containment be designed with the capability to test periodically the operability of the isolation valves and determine valve leakage acceptability.

11. 10 CFR 50.34 and Appendix B to 10 CFR Part 50 requires the establishment of a QAP for the design, fabrication, construction, and testing of safety-related pumps, valves, and dynamic restraints.

12. 10 CFR 50.55a(c), (d), and (e) incorporate by reference the ASME B&PV Code, Section III, for the qualification of mechanical equipment and supports.

13. 10 CFR 50.55a incorporates by reference the ASME OM Code, which establishes requirements for PST and IST of components to assess their operational readiness.

14. 10 CFR 50.55a(b)(3)(ii) requires the establishment of a program to periodically verify the design-basis capability of safety-related MOVs.

15. 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 has been constructed 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 are discussed in SRP Section 3.9.6. Additional considerations for the NRC staff review include:

1. The staff considered guidance provided in applicable Commission SECY papers, Commission staff requirements memoranda, generic letters (GLs), RGs, and regulatory issue summaries.

2. The Commission's Staff Requirements Memorandum (SRM) dated September 11, 2002, for Commission Paper SECY-02-0067, "Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC) for Operational Programs (Programmatic ITAAC)," stated that ITAAC for an operational program are unnecessary if the program and its implementation are fully described in a COLA and found to be acceptable by the NRC. In its SRM dated May 14, 2004, for SECY-04-0032, "Programmatic Information Needed for Approval of a Combined License without Inspections, Tests, Analyses, and Acceptance Criteria," the Commission defined "fully described" as when the program is clearly and sufficiently described in terms of the scope and level of detail to allow a reasonable assurance finding of

acceptability. The Commission also noted that required programs should always be described at a functional level and at an increasing level of detail where implementation choices could materially and negatively affect the program effectiveness and acceptability. Commission Paper SECY-05-0197, "Review of Operational Programs in a Combined License Application and Generic Emergency Planning Inspections, Tests, Analyses, and Acceptance Criteria," dated October 28, 2005, summarizes the NRC position regarding the full description of operational programs to be provided by COL applicants. RG 1.206, "Combined License Applications for Nuclear Power Plants (LWR Edition)," provides guidance for COL applicants with respect to fully describing plant operational programs.

3. The US-APWR DC applicant intended that its DCD fully describe the functional design, qualification, and IST programs for pumps, valves, and dynamic restraints to allow COL applicants to incorporate by reference the descriptions of these programs in their COLAs with only plant-specific information needed to supplement the program descriptions. Therefore, the staff followed the guidance in SECY-05-0197 and RG 1.206, in addition to SRP Section 3.9.6, in reviewing the descriptions of the functional design, qualification, and IST programs for pumps, valves, and dynamic restraints to be used in the US-APWR design.

3.9.6.4 Technical Evaluation

In accordance with 10 CFR Part 52, the staff reviewed the design aspects of the functional design, qualification, and IST programs for pumps, valves, and dynamic restraints described in the DCD. The staff also reviewed the US-APWR DC application to determine whether the description of the functional design, qualification, and IST programs for pumps, valves, and dynamic restraints to be used in a US-APWR nuclear power plant is sufficient to allow a COL applicant to reference the DCD as part of satisfying the requirement to fully describe these programs in support of the COLA. Therefore, in addition to design aspects, the staff evaluated DCD Tier 2, Section 3.9.6 and its associated sections to determine whether the DCD provisions describe the functional design, qualification, and IST programs for pumps, valves, and dynamic restraints to be used in the US-APWR sufficiently to satisfy the requirements of NRC regulations and the ASME OM Code as incorporated by reference in the regulations. As part of this review, the staff assessed the adequacy of the US-APWR design to ensure that it provided access to allow IST activities for pumps, valves, and snubbers.

3.9.6.4.1 Functional Design and Qualification of Pumps, Valves, and Dynamic Restraints

DCD Tier 2, Section 3.9.6 states that safety-related pumps, valves, and dynamic restraints conform to the requirements of 10 CFR 50.55a in that such components are designed in accordance with ASME B&PV Code, Section III requirements. The NRC regulations in GDC 1 require that SSCs important to safety be designed to quality standards commensurate with the importance of the safety functions to be performed. Where generally recognized codes and standards are used, they shall be identified and evaluated to determine their applicability, adequacy, and sufficiency as necessary to assure a quality product in keeping with the required safety function.

The staff reviewed DCD Tier 2, Sections 3.9.3.3, 3.9.3.4, and 3.9.6.1 to evaluate whether the design and qualification process for safety-related pumps, valves, and dynamic restraints to be used in the US-APWR design incorporates lessons learned related to component design and qualification from operating nuclear power plants. For example, ASME Standard QME-1-2007, "Qualification of Active Mechanical Equipment Used in Nuclear Power Plants," incorporates lessons learned to ensure that pumps, valves, and dynamic restraints are functionally designed and qualified to perform their safety functions. The staff accepted the use of ASME QME-1-2007 in RG 1.100, "Seismic Qualification of Electrical and Active Mechanical Equipment and Functional Qualification of Active Mechanical Equipment for Nuclear Power Plants," Revision 3, issued September 2009, with specific conditions.

In **RAI 288-2274, Question 03.09.06-1**, the staff requested that the applicant describe the functional qualification program for safety-related pumps, valves, and dynamic restraints. In its response to **RAI 288-2274, Question 03.09.06-1**, dated May 25, 2009, the applicant stated that, as outlined in DCD Tier 2, Section 3.10, "Seismic and Dynamic Qualification of Mechanical and Electrical Equipment," the design and qualification requirements with respect to safety-related pumps, valves, and dynamic restraints will adhere to the requirements of ASME QME-1-2007. The applicant stated that DCD Tier 2, Section 3.9.6.1 would be revised to include a reference to DCD Tier 2, Section 3.10 for the design and qualification requirements. The staff confirmed that DCD Tier 2, Revision 2 (and Revision 3) includes the reference to DCD Tier 2, Section 3.10 in DCD Tier 2, Section 3.9.6.1. However, DCD Tier 2, Section 3.10 applies to the seismic and dynamic qualification of mechanical and electrical equipment. DCD Tier 2, Section 3.9.3.3 addresses the functional qualification of pumps and valves used in the US-APWR design.

The NRC regulations in 10 CFR 52.79(a)(11) require COL applicants to provide a description of the programs and their implementation necessary to ensure that systems and components meet the requirements of the ASME B&PV Code and OM Code in accordance with 10 CFR 50.55a. The US-APWR DC applicant has indicated that a sample of design and procurement specifications will be prepared for staff audit review. The staff will perform an audit of the US-APWR design and procurement specifications to evaluate implementation of the functional design, qualification, and IST programs in accordance with the requirements of 10 CFR 52.79(a)(11). Therefore, pending completion of the audit, **RAI 288-2274, Question 03.09.06-1, is being tracked as an Open Item**.

In follow-up **RAI 801-5897, Question 03.09.06-49** the staff requested that the applicant specify the application of ASME QME-1-2007 as accepted in to RG 1.100, Revision 3 in DCD Tier 2, Section 3.9.3.3 or DCD Tier 2, Section 3.9.6.1. In its response to **RAI 801-5897, Question 03.09.06-49**, dated November 2, 2011, the applicant indicated that DCD Tier 2, Section 3.9.6.1 would be modified to specify that the functional design and qualification of safety-related pumps, valves, and dynamic restraints will be performed in accordance with ASME QME-1-2007 as accepted in RG 1.100, Revision 3. The staff finds that the planned revision to the DCD will provide an acceptable methodology for the functional design and qualification of safety-related pumps, valves, and dynamic restraints and therefore the response is acceptable. **RAI 801-5897, Question 03.09.06-49 is being tracked as a Confirmatory Item**.

GDC 37, 40, 43, and 46 address the design requirements for the capability to test pumps and valves at design-basis (flow, pressure, temperature) conditions. In **RAI 288-2274, Question 03.09.06-4**, the staff requested that the applicant describe testing of pumps and valves at design-basis conditions. In its response to **RAI 288-2274, Question 03.09.06-04**, dated May 25, 2009, the applicant responded that testing of pumps and valves in the US-APWR is consistent with the requirements of GDC 37, 40, 43, and 46. However, the staff identified that IST provisions should be described separate from provisions for the functional design and qualification of pumps, valves, and dynamic restraints. Therefore, the staff closed as unresolved **RAI 288-2274, Question 03.09.06-4** and in follow-up **RAI 801-5897, Question**

03.09.06-52, the staff requested that the applicant relocate the IST provisions in DCD Tier 2, Section 3.9.6.1 to an IST section in the DCD, and specify provisions for the functional design and qualification of pumps, valves, and dynamic restraints in Section 3.9.6.1.

In its response to **RAI 801-5897, Question 03.09.06-52** dated November 2, 2011, the applicant stated that DCD Tier 2, Section 3.9.6 would be revised to specify the applicable GDC for the functional design and qualification of safety-related pumps, valves, and dynamic restraints. The applicant also stated that DCD Tier 2, Section 3.9.6.1 would be revised to reference the use of ASME QME-1-2007 as accepted in RG 1.100, Revision 3, and other applicable sections of the DCD. The staff finds that the planned modifications to the DCD will provide an acceptable methodology for the functional design and qualification of pumps, valves, and dynamic restraints and therefore the response is acceptable. **RAI 801-5897, Question 03.09.06-52 is being tracked as a Confirmatory Item**.

3.9.6.4.2 Inservice Testing Program

DCD Tier 2, Section 3.9.6 (through Revision 3) specified the ASME OM Code, 1995 Edition through 2003 Addenda, in Reference 3.9-13 of DCD Tier 2, Section 3.9.10, "References," as the basis for the US-APWR IST program for ASME B&PV Code, Section III, Class 1, 2 and 3 safety-related pumps, valves, and dynamic restraints. DCD Tier 2, Table 3.9-13 indicates that the 2004 Edition of the ASME OM Code is used in the IST program for the US-APWR. In **RAI 288-2274, Question 03.09.06-2**, the staff requested that the applicant verify the code edition and addenda to be used as the basis for the IST program. In its response to **RAI 288-2274, Question 03.09.06-2**, dated May 25, 2009, the applicant stated that the reference in DCD Tier 2, Table 3.9-13 should be to the latest edition and addenda of the ASME OM Code incorporated by reference in NRC regulations. The reference to a specific ASME OM Code edition was removed from Table 3.9-13 in Revision 2 to the DCD Tier 2. However, Revision 3 to DCD Tier 2, Section 3.9.10 specified ASME OM Code, 1995 Edition through the 2003 Addenda, in Reference 3.9-13.

Therefore, the staff closed as unresolved **RAI 288-2274, Question 03.09.06-2** and in follow-up **RAI 801-5897, Question RAI 03.09.06-50**, the staff requested that the applicant establish a section in the DCD that specifies the overall provisions for the IST program for pumps, valves, and dynamic restraints. For example, the staff requested that the applicant address the applicable edition of the ASME OM Code for the IST program for pumps, valves, and dynamic restraints used in the US-APWR design. The staff also indicated that the new DCD section should specify that the US-APWR will be designed to allow accessibility for the performance of IST activities for pumps, valves, and dynamic restraints.

In its response to **RAI 801-5897, Question 03.09.06-50**, dated November 2, 2011, the applicant stated that the overall provisions for the IST program for pumps, valves, and dynamic restraints would be provided in DCD Tier 2, Section 3.9.6. The applicant also stated that the DCD would specify the applicability of the 2004 Edition through the 2006 Addenda of the ASME OM Code for the IST program for pumps, valves, and dynamic restraints used in the US-APWR design. The applicant also stated that the DCD would specify that the US-APWR will be designed to allow accessibility for the performance of IST activities for pumps, valves, and dynamic restraints. The staff finds that the planned modifications to the DCD will establish overall provisions for the IST program that are acceptable for reference by COL applicant and therefore the response is acceptable. The staff notes that the NRC regulations in 10 CFR 50.55a require the COL licensee to apply the latest edition and addenda of the ASME OM Code incorporated by reference in 10 CFR 50.55a 12 months before initial fuel load (or the optional ASME Code cases listed in RG 1.192 as incorporated by reference in 10 CFR 50.55a), subject to the conditions specified in 10 CFR 50.55a, for the initial 10-year interval of the IST program. **RAI 801-5897, Question 03.09.06-50 is being tracked as a Confirmatory Item**.

DCD Revision 1 referenced ISI requirements in DCD Tier 2, Section 3.9.6.1 for ASME B&PV Code Section III, Class 1, 2 and 3 pumps, valves, and dynamic restraints. In **RAI 288-2274, Question 03.09.06-3**, the staff requested that the applicant clarify the reference to ISI requirements in this DCD section. In its response to **RAI 288-2274, Question 03.09.06-3**, dated May 25, 2009, the applicant stated that DCD Tier 2, Section 3.9.6.1 would be clarified. Subsequently, Revisions 2 and 3 to the DCD Tier 2, Section 3.9.6.1 specified IST requirements for ASME B&PV Code Section III, Class 1, 2 and 3 pumps, valves, and dynamic restraints. However, the staff noted that the scope of DCD Tier 2, Section 3.9.6.1 applies to the functional design and qualification of pumps, valves, and dynamic restraints and therefore the DCD should be revised to address functional design and qualification in this section and IST provisions in a separate section.

Therefore, the staff closed as unresolved **RAI 288-2274, Question 03.09.06-3,** and in follow-up **RAI 801-5897, Question 03.09.06-51**, the staff requested that the applicant relocate the discussion of IST requirements to the applicable section in the DCD. In its response to **RAI 801-5897, Question 03.09.06-51**, dated November 2, 2011, the applicant stated that the DCD would be modified to relocate the IST provisions from DCD Tier 2, Section 3.9.6.1 to DCD Tier 2, Section 3.9.6. The staff finds that the planned modifications to the DCD will provide the IST provisions in the applicable section and therefore the response is acceptable. **RAI 801-5897, Question 03.09.06-51 is being tracked as a Confirmatory Item**.

3.9.6.4.2.1 Inservice Testing Program for Pumps

DCD Tier 2, Section 3.9.6.2 describes the IST program for pumps to be used in the US-APWR. The staff reviewed the description of the IST program for pumps in the US-APWR design that are required to perform a specific function in shutting down the reactor to a safe-shutdown condition, maintaining the safe-shutdown condition, or mitigating the consequence of an accident. For example, the staff reviewed test schedules and parameters for pumps that are operated continuously or routinely during operation, cold shutdown, or refueling operations (Group A pumps), and pumps that are not operated routinely except for testing (Group B pumps).

In **RAI 288-2274, Question 03.09.06-7**, the staff requested that the applicant provide a full description of the IST program for pumps that complies with the ASME OM Code, or specify that the COL applicant will need to supplement the DCD to provide a full description of the IST program for pumps as part of the COLA. In its response to **RAI 288-2274, Question 03.09.06-7**, dated May 25, 2009, the applicant proposed to clarify the DCD regarding the responsibility of the COL applicant to provide a full description of the IST program for pumps. However, the applicant did not clarify whether it was intending to provide a full IST program description for pumps. Therefore, the staff closed as unresolved **RAI 288-2274, Question 03.09.06-7** and in follow-up **RAI 801-5897, Question 03.09.06-53**, the staff requested that the applicant clarify whether the DCD is intended to fully describe the IST program for pumps used in the US-APWR, or the COL applicant must supplement the provisions in the DCD to fully describe the IST program for pumps in its COLA. For example, the staff indicated that the DCD would need to provide additional specifications, such as a summary of the ASME OM Code Subsection ISTA/ISTB overall requirements, and pump reference values and their maintenance, to provide a full description of the pump IST program.

In its response to **RAI 801-5897, Question 03.09.06-53**, dated November 2, 2011, as revised in its submittal dated March 8, 2012, the applicant indicated that the DCD would fully describe the IST program for pumps used in the US-APWR. The applicant provided planned modifications to DCD Tier 2, Section 3.9.6.2 to describe the pump IST program and to refer to the IST program in COL Information Item 3.9(8) rather than an IST program plan. For example, DCD Tier 2, Section 3.9.6 will state that the COL applicant is to administratively control the IST program for pumps, valves, and dynamic restraints, and to control the ASME OM Code edition and addenda to be used for the IST program. Further, DCD Tier 2, Section 3.9.9, “Combined License Information,” will specify in COL Information Item 3.9(8) that the COL applicant is to administratively control the edition and addenda to be used for the IST program, and to provide a full description of their IST program for pumps, valves, and dynamic restraints. The staff finds that the DCD with the planned modifications in response to this and other RAIs will provide a description of the pump IST program for the US-APWR that is acceptable for reference by a COL applicant and therefore the response is acceptable. **RAI 801-5897, Question 03.09.06-53 is being tracked as Confirmatory Item**.

In **RAI 288-2274, Question 03.09.06-8**, the staff requested that the applicant provide a flow diagram depicting the SFP demineralizer train valve arrangements or describe how the pump test flow path will be established to facilitate ASME OM Code pump testing. In its response to **RAI 288-2274, Question 03.09.06-8**, dated May 25, 2009, the applicant clarified the flow path and indicated that the test flow rate was controlled by the flow control valves. The staff finds that the applicant has clarified the test arrangements for the SFP demineralizer train and therefore the response is acceptable. Accordingly, **RAI 288-2274, Question 03.09.06-8 is resolved**.

The staff reviewed the description of the IST program for safety-related ASME B&PV Code Class 1, 2, and 3 pumps. DCD Tier 2, Chapter 16, TS 5.5.8, states that the IST program provides controls for testing ASME B&PV Code Class 1, 2, and 3 components. In **RAI 288-2274, Question 03.09.06-9**, the staff requested that the applicant provide the basis for not including the following pumps in DCD Tier 2, Table 3.9-13:

- Class 2 SI Auxiliary Oil Pumps.
- Class 3 Boric Acid Transfer Pumps.
- Class 3 Emergency Gas Turbine Fuel Oil Transfer Pumps.

In its response to **RAI 288-2274, Question 03.09.06-9**, dated May 25, 2009, the applicant stated the Emergency Gas Turbine Fuel Oil Transfer Pumps would be added to DCD Tier 2, Table 3.9-13. The applicant clarified that the SI Auxiliary Oil Pumps are attached to the SI Pumps and tested concurrently, and therefore are not included in DCD Tier 2, Table 3.9-13. The Boric Acid Transfer Pumps are nonsafety pumps as identified in DCD Tier 2, Table 3.2-2, “Classification of Mechanical and Fluid Systems, Components, and Equipment,” and therefore are not included in DCD Tier 2, Table 3.9-13. The staff finds this response to be acceptable in clarifying the information on the specific pumps. However, in reviewing Revision 2 to DCD Tier 2, Table 3.9-13, the staff found that the new entries for the Emergency Gas Turbine Fuel Oil Transfer Pumps did not match the RAI response (such as the pump tag numbers and specified tests). In reviewing Revision 3 to DCD Tier 2, Table 3.9-13, the staff found that the Emergency Gas Turbine Fuel Oil Transfer Pump numbers were changed to GTS-MPP-001A to D, and 002A to D.

Therefore, the staff closed as unresolved **RAI 288-2274, Question 03.09.06-9,** and in follow-up **RAI 801-5897, Question 03.09.06-54**, the staff requested that the applicant resolve the differences between the RAI response and DCD Tier 2, Table 3.9-13 for the Emergency Gas Turbine Fuel Oil Transfer Pumps. In its response to **RAI 801-5897, Question 03.09.06-54**, dated November 2, 2011, the applicant reported that the tag numbers for the Emergency Gas Turbine Fuel Oil Transfer Pumps had been corrected in Revision 3 to the DCD. The staff has confirmed this change and therefore the response is acceptable. Accordingly, **RAI 801-5897, Question 03.09.06-54, is resolved**.

3.9.6.4.2.2 Inservice Testing Program for Valves

DCD Tier 2, Section 3.9.6.3 includes provisions for the IST program for valves to be used in the US-APWR, including valves that are required to perform a specific function in shutting down the reactor to the safe shutdown condition, maintaining the safe shutdown condition, or mitigating the consequences of an accident. DCD Tier 2, Table 3.9-14 lists valves to be included in the IST program with the type of valve and safety function provided for each valve. As discussed below, the staff reviewed the description of the IST program for valves provided in the DCD.

In **RAI 288-2274, Question 03.09.06-10**, the staff requested that the applicant provide a full description of the IST operational program for valves in the US-APWR, or specify that the COL applicant will need to supplement the DCD provisions to provide a full description of the IST program for valves as part of the COLA. In its response to **RAI 288-2274, Question 03.09.06-10**, dated May 25, 2009, the applicant referred to the regulatory requirement that the IST program to be developed by the COL licensee must satisfy the ASME OM Code incorporated by reference into 10 CFR 50.55a 12 months before fuel loading. However, the applicant did not clarify whether it was intending to provide a full IST program description for valves. Therefore, the staff closed as unresolved **RAI 288-2274, Question 03.09.06-10** and in follow-up **RAI 801-5897, Question 03.09.06-55**, the staff requested that the applicant clarify whether the DCD is intended to fully describe the IST program for valves in the US-APWR, or that the COL applicant must supplement the provisions in the DCD to fully describe the IST program for valves in its COLA. For example, the staff indicated that the applicant would need to provide additional specification, such as a summary of the ASME OM Code Subsection ISTA/ISTC overall requirements, valve reference values and their maintenance, and avoidance of preconditioning, to provide a full description of the valve IST program.

In its response to **RAI 801-5897, Question 03.09.06-55**, dated November 2, 2011, as revised in its submittal dated March 8, 2012, the applicant indicated that DCD Tier 2, Section 3.9.6.3 is intended to fully describe the IST program for valves used in the US-APWR. The staff finds that the DCD with the planned modifications in response to this and other RAIs will provide a description of the valve IST program for the US-APWR that is acceptable for reference by a COL applicant, and therefore the response is acceptable. **RAI 801-5897, Question 03.09.06-55, is being tracked as a Confirmatory Item**.

Inservice Testing Program for Motor-Operated Valves

Revisions 0 and 1 to DCD Tier 2, Section 3.9.6.1 referred to ASME OM Code, Subsection ISTC 4.2 for the IST program for MOVs in the US-APWR. In **RAI 288-2274, Question 03.09.06-5**, the staff requested that the applicant provide the basis for this reference to a specific IST provision in the ASME OM Code. In its response to **RAI 288-2274, Question 03.09.06-5**, dated May 25, 2009, the applicant agreed that this reference was not necessary. Subsequently, the staff confirmed Revision 2 (and Revision 3) to the DCD Tier 2, Section 3.9.6.1, deleted the reference to ASME OM Code, Subsection ISTC 4.2. Accordingly, **RAI 288-2274, Question 03.09.06-5, is resolved**.

DCD Tier 2, Subsection 3.9.6.3.1, "IST Program for MOVs," provides information on the IST program for MOVs to be used in the US-APWR. In **RAI 288-2274, Question 03.09.06-13**, the staff requested that the applicant provide a full description of the MOV testing operational program, or specify that the COL applicant will need to supplement the DCD to provide a full description of the MOV testing program as part of the COLA. In its response to **RAI 288-2274, Question 03.09.06-13**, dated May 25, 2009, the applicant provided a planned revision to DCD Tier 2, Section 3.9.6.3.1, that included additional information on the IST program for MOVs in the US-APWR. The staff confirmed that Revision 2 (and Revision 3) to DCD Tier 2, Section 3.9.6.3.1 included the changes specified in the RAI response.

Based on its review, the staff found that Revision 3 to the DCD did not provide a full description of the IST program for MOVs. Therefore, the staff closed as unresolved **RAI 288-2274, Question 03.09.06-13** and in follow-up **RAI 801-5897, Question 03.09.06-58**, the staff requested that the applicant provide a full description of the IST program for MOVs in the DCD, or specify that the COL applicant will be responsible for supplementing the DCD in support of the COLA. The staff indicated specific aspects to be described, including, for example, specification that the MOV program description will address the IST requirements in the ASME OM Code, MOV periodic verification in accordance with 10 CFR 50.55a, MOV operating experience, temperature effects on MOV output, and periodic verification of MOV actuator output.

In its response to **RAI 801-5897, Question 03.09.06-58** dated November 2, 2011, as revised in its submittal dated March 8, 2012, the applicant stated that the DCD would be modified to fully describe the IST program for MOVs used in the US-APWR. In particular, the applicant stated that the description of the IST program for MOVs would specify that the MOV program will satisfy the IST testing requirements in the ASME OM Code and also specify the requirement for periodic verification of MOVs in accordance with 10 CFR 50.55a(b)(3)(ii). The applicant also indicated that the description of the IST program for MOVs will specify that either in-plant valve operation of prototype valve testing at system flow and pressure, or system differential pressure, to verify correct MOV actuator sizing and control settings will satisfy 10 CFR 50.55a(b)(3)(ii). The MOV program description will reference the Joint Owners Group (JOG) Program on MOV Periodic Verification that is accepted by the NRC staff in a SER, "Final Safety Evaluation on Joint Owners' Group Program on Motor-Operated Valve Periodic Verification (TAC Nos. MC2346, MC2347, and MC2348)," dated September 25, 2006, and its supplement, "Final Supplement to Safety Evaluation for Joint Owners' Group Motor-Operated Valve Periodic Verification Program (TAC Nos. MD8920 and MD8921)," dated September 18, 2008. The MOV program description will specify that the MOV program will implement ASME OM Code Case OMN-1, "Alternative Rules for Preservice and Inservice Testing of Certain Electric Motor-Operated Valve Assemblies in Light-Water Reactor Power Plants OM Code-1995, Subsection ISTC," Revision 0, 1999, that is accepted for use by the NRC with conditions in RG 1.192, "Operation and Maintenance Code Case Acceptability, ASME OM Code," issued June 2003. The MOV program description also will specify the consideration of temperature effects on MOV output, indicate that MOV operating experience is incorporated, and address periodic verification of MOV actuator output. The staff finds that the DCD with the planned modifications will provide a description to the IST program for MOVs that may be referenced by a COL applicant and therefore the response is acceptable. **RAI 801-5897, Question 03.09.06-58, is being tracked as a Confirmatory Item**.

The staff compared DCD Tier 2, Table 6.2.4-3, “List of Containment Penetrations and System Isolation Positions,” and DCD Tier 2, Table 3.9-14 for consistency regarding the CIVs and IST program valves. In **RAI 288-2274, Question 03.09.06-23**, the staff requested that the applicant address several findings from the staff review for these tables. The applicant submitted its response on May 25, 2009. The staff findings and the RAI response provided by the applicant are as follows:

(a) Table 3.9-14 identifies valves NCS-MOV-511 and NCS-MOV-517 as motor operated while Revision 0 to DCD Tier 2, Table 6.2.4-3 identified the valves as air operated. In its RAI response, the applicant indicated that these valves are motor operated. The staff finds the response acceptable with the clarification of the valve types. Subsequently, the staff confirmed that Revision 1 (and Revisions 2 and 3) to DCD Tier 2, Table 6.2.4-3 specifies these valves as motor operated. Accordingly, **RAI 288-2274, Question 03.09.06-23(a) is resolved**.

(b) Revision 0 to DCD Tier 2, Table 3.9-14 identified valve PSS-AOV-071 as an air-operated valve while DCD Tier 2, Table 6.2.4-3 identifies the valve as motor operated. In its RAI response, the applicant indicated that the valve is motor operated and its correct identification is PSS-MOV-071. The staff finds the response acceptable with the clarification of the valve type. Subsequently, the staff confirmed that Revision 2 (and Revision 3) to DCD Tier 2, Table 3.9-14 identifies this MOV as PSS-MOV-071. Accordingly, **RAI 288-2274, Question 03.09.06-23(b) is resolved**.

(c) DCD Tier 2, Table 6.2.4-3 included CIVs FSS-VLV-001, 003, and 006; FSS-MOV-004; CAS-VLV-101 and 103; RMS-VLV-005; RMS-MOV-001, 002, and 003; IGS-AOV-001 and 002; and LTS-VLV-001 and 002, which were not listed in DCD Tier 2, Table 3.9-14. In its RAI response, the applicant indicated that these valves would be included in the IST table in the DCD. The staff found the valves to be included in Revision 2 (and Revision 3) to DCD Tier 2, Table 3.9-14, with the exception of FSS-VLV-001. Therefore, the staff closed as unresolved **RAI 288-2274, Question 03.09.06-23(c)** and in follow-up **RAI 801-5897, Question 03.09.06-61**, the staff requested that the applicant clarify the specification of FSS-VLV-001 in DCD Tier 2, Table 3.9-14. In its response to **RAI 801-5897, Question 03.09.06-61**, dated November 2, 2011, the applicant stated that the valve tag number for FSS-VLV-001 was changed to FSS-AOV-001 in Revision 3 to the DCD and therefore the response is acceptable. The staff confirmed that DCD Tier 2, Revision 3 Table 3.9-14 includes the corrected valve tag number for FSS-AOV-001. Accordingly, this part of **RAI 801-5897, Question 03.09.06-61, is resolved**.

(d) DCD Tier 2, Table 6.2.4-3 lists containment penetrations P262R and P262L, but does not provide information on their isolation. In its RAI response, the applicant clarified that these penetrations are used for instrument lines with seal sensors without the need for isolation valves. The staff finds the response acceptable with the clarification of the containment penetrations. Accordingly, **RAI 288-2274, Question 03.09.06-23(d) is resolved**.

(e) DCD Tier 2, Table 6.2.4-3 did not include units for the operating time for the CIV. In its RAI response, the applicant indicated that units for closure time would be included in DCD Tier 2, Table 6.2.4-3. However, the applicant did not specify the time units. Therefore, the staff closed as unresolved **RAI 288-2274, Question 03.09.06-23(e)** and in follow-up **RAI 801-5897, Question 03.09.06-61**, the staff requested that the applicant clarify the specification of the stroke time units in DCD Tier 2, Table 6.2.4-3. In its response to **RAI 801-5897, Question 03.09.06-61**, dated November 2, 2011, the applicant stated that DCD Tier 2, Table 6.2.4-3 would be revised to specify that the units for valve closure time would be specified as seconds. The staff finds that the planned modifications to DCD Tier 2, Table 6.2.4-3 will clarify the units for valve closure time and therefore the response is acceptable. This part of **RAI 801-5897, Question 03.09.06-61, is being tracked as a Confirmatory Item**.

(f) DCD Tier 2, Table 6.2.4-3 specifies that valves VCS-AOV-304, 305, 306, and 307 will have a closure time of five seconds and the staff requested clarification on how the closure time will be satisfied. In its RAI response, the applicant clarified that the pneumatic actuators for these valves will be designed to provide closure in five seconds. The staff finds the response acceptable. Accordingly, **RAI 288-2274, Question 03.09.06-23(f) is resolved**.

(g) Revision 0 to DCD Tier 2, Table 6.2.4-3 specified NA (not applicable) for the closure time of CIVs CSS-MOV-004A to D. In its RAI response, the applicant stated that the closure time for these valves is 40 seconds, and that the table would be corrected. The staff finds the response acceptable. Subsequently, the staff confirmed that Revision 2 (and Revision 3) to DCD Tier 2, Table 6.2.4-3 specified a closure time of 40 for CSS-MOV-004A to D. The units for closure time were addressed as part of paragraph (e) above. Accordingly, **RAI 288-2274, Question 03.09.06-23(g) is resolved.**

As noted in paragraph (e) above, **RAI 801-5897, Question 03.09.06-61, is being tracked as a Confirmatory Item.**

Revision 0 to DCD Tier 2, Table 3.9-14 specified the safety functions of valves EWS-MOV-503A to D as maintain open and close, and to transfer open and close. However, DCD Tier 1, Table 2.7.3.1-2, "Essential Service Water System Equipment Characteristics," identified the active safety function of these valves as transfer open. In **RAI 288-2274, Question 03.09.06-33**, the staff requested that the applicant clarify the safety function for these valves. In its response to **RAI 288-2274, Question 03.09.06-33**, dated May 25, 2009, the applicant indicated that the active safety function of the valves is transfer open. The staff finds the response acceptable. The staff confirmed that Revision 1 (and Revisions 2 and 3) to DCD Tier 2, Table 3.9-14 specifies the safety-related missions of EWS-MOV-503A to D as maintain open and close, and transfer open. Accordingly, **RAI 288-2274, Question 03.09.06-33, is resolved**.

DCD Tier 2, Table 3.9-14 identifies CCW valves NCS-MOV-237A and B and NCS-MOV-232A and B as ASME OM Category B valves without leakage criteria. In **RAI 288-2274, Question 03.09.06-36**, the staff requested that the applicant clarify the ASME OM Code categorization of these valves. In its response to **RAI 288-2274, Question 03.09.06-36**, dated May 25, 2009, the applicant indicated that NCS-MOV-232A and B are used to establish bypass flow and isolate CCW supply headers. Therefore, no specific maximum amount of seat leakage in the closed position is applied to the valves. The staff finds the clarification of the OM categorization of NCS-MOV-232A and B to be acceptable.

However, the applicant did not address valves NCS-MOV-237A and B. Therefore, the staff closed as unresolved **RAI 288-2274, Question 03.09.06-36** and in follow-up **RAI 801-5897, Question 03.09.06-64**, the staff requested in RAI 03.09.06-64 that the applicant discuss the ASME OM Code categorization of NCS-MOV-237A and B. In its response to **RAI 801-5897, 03.09.06-64**, dated November 2, 2011, the applicant stated that valves NCS-MOV-237A and B did not exist in the CCW system. The applicant stated that DCD Tier 2, Table 3.9-14 would be revised to correct the description of valves NCS-MOV-232A and B. The staff finds the correction of the RAI response and the planned clarification of the DCD to be acceptable. **RAI 801-5897, Question 03.09.06-64 is being tracked as a Confirmatory Item**.

Revision 0 to DCD Tier 1, Table 2.7.6.7-1, "Process and Post-Accident Sampling System Equipment Characteristics," specified the loss of motive power position of valve PSS-MOV-006 as fail closed. In **RAI 288-2274, Question 03.09.06-41**, the staff requested that the applicant address the loss of motive power position of this MOV. In its response to **RAI 288-2274, Question 03.09.06-41**, dated May 25, 2009, the applicant stated that the table was incorrect and would be modified. Subsequently, Revision 2 (and Revision 3) to DCD Tier 1, Table 2.7.6.7-1 specifies the loss of motive power position of PSS-MOV-006 to be fail as-is. The staff finds the revised table to be acceptable as the failure mode of this MOV is correctly indicated to be fail as-is and therefore the response is acceptable. Accordingly, **RAI 288-2274, Question 03.09.06-41, is resolved**.

Inservice Testing Program for Thermal Relief Valves

DCD Tier 2, Section 3.9.6 states that pressure relief valves used for protecting systems or portions of systems that perform a function in shutting down the reactor to a safe shutdown condition, in maintaining a safe shutdown condition, or in mitigating the consequences of an accident, are subject to IST activities. In **RAI 288-2274, Question 03.09.06-11**, the staff requested that the applicant clarify the IST program description for testing thermal relief valves. In its response to **RAI 288-2274, Question 03.09.06-11, dated May 25, 2009**, the applicant stated that safety-related thermal relief valves are used in the US-APWR design and that testing of these relief valves will be included in the IST program. The applicant also noted that thermal relief valve testing will be conducted in accordance with paragraphs I-1340 and I-1390 in Appendix I, "Inservice Testing of Pressure Relief Devices in Light-Water Reactor Nuclear Power Plants," to the ASME OM Code. The applicant stated that the IST provisions for valves NCS-VLV-406A to D, NCV-VLV-513, NCS-VLV-533, and NCS-VLV-035A and B would be included in DCD Tier 2, Table 3.9-14.

However, the staff identified that some of the valve numbers in DCD Tier 2, Table 3.9-14 needed clarification. Therefore, the staff closed as unresolved **RAI 288-2274, Question 03.09.06-11,** and in follow-up **RAI 801-5897, Question 03.09.06-56**, the staff requested that the applicant clarify the numbering of these valves as incorporated into DCD Tier 2, Table 3.9-14. In its response to **RAI 801-5897, Question 03.09.06-56**, dated November 2, 2011, the applicant stated that the type designation for these valves was changed from VLV to SRV. The applicant stated that Revision 3 to the DCD had corrected the valve type designation and certain typographical errors in the previous RAI response and DCD markup. The staff has confirmed that Revision 3 to the DCD includes the specified corrections. Accordingly, **RAI 801-5897, Question 03.09.06-56, is resolved.**

DCD Tier 2, Section 9.2.2, "Component Cooling Water System," indicates that relief valves will be used in the CCW system. In **RAI 288-2274, Question 03.09.06-25**, the staff requested that the applicant discuss the design of the CCW system in satisfying the IST requirements for pressure relief valves. In its response to **RAI 288-2274, Question 03.09.06-25, dated May 25, 2009**, the applicant stated that the IST requirements for thermal relief valves would be included in DCD Tier 2, Table 3.9-14, as discussed its response to **RAI 288-2274, Question 03.09.06-11**. The staff agrees that **RAI 288-2274, Question 03.09.06-25** is addressed by the response to **RAI 288-2274, Question 03.09.06-11.** Accordingly, **RAI 801-5897, Question 03.09.06-56, is resolved**.

Inservice Testing Program for POVs Other than MOVs

Revision 0 to DCD Tier 2, Subsection 3.9.6.3.2, "IST Program for POVs Other Than MOVs," provided information on the IST program for POVs other than MOVs to be used in the US-APWR. The staff found that Revision 0 to the DCD did not provide a full description of the IST program for POVs. Therefore, in **RAI 288-2274, Question 03.09.06-14**, the staff requested that the applicant provide a full description of the operational program for POVs other than MOVs, or specify that the COL applicant must supplement the DCD to provide a full description of the IST program for POVs as part of the COLA. In its response to **RAI 288-2274, Question 03.09.06-14**, dated May 25, 2009, the applicant stated that DCD Tier 2, Section 3.9.6.3.2 would describe the application of MOV lessons learned in developing the IST program for POVs, such as discussed in NRC Regulatory Issue Summary (RIS) 2000-03, "Resolution of Generic Safety Issue 158: Performance of Safety-Related Power-Operated Valves Under Design-Basis Conditions." The applicant provided a planned modification to DCD Tier 2, Section 3.9.6.3.2. The staff confirmed Revision 2 (and Revision 3) to DCD Tier 2, Subsection 3.9.6.3.2 incorporated the modifications discussed in the applicant's response to **RAI 288-2274, Question 03.09.06-14**.

The staff reviewed the description of the IST program for POVs other than MOVs provided in Revision 3 to DCD Tier 2, Section 3.9.6.3.2. However, it was not clear whether the applicant intended to provide a full IST program description for POVs other than MOVs. Therefore, the staff closed as unresolved **RAI 288-2274, Question 03.09.06-14** and in follow-up **RAI 801-5897, Question 03.09.06-59**, the staff requested that the applicant clarify whether the DCD is intended to provide a full description of the IST operational program for POVs other than MOVs, or that the COL applicant is responsible for supplementing the DCD to provide a full description of the POV program. For example, the staff indicated that the POV program description needed to address provisions that specify critical parameters, consideration of uncertainties in diagnostic analysis, and POV testing acceptance criteria specified in RG 1.206. The staff also noted that POV program description should also specify testing for all safety-related POVs regardless of their safety significance.

In its response to **RAI 801-5897, Question 03.09.06-59**, dated November 2, 2011, as revised in its submittal dated March 8, 2012, the applicant indicated that DCD Tier 2, Subsection 3.9.6.3.2 is intended to fully describe the IST program for POVs used in the US-APWR. The applicant provided planned modifications to the DCD to be consistent with the guidance in RG 1.206 for POVs. The planned revision to the US-APWR will provide a description of the IST program for POVs other than MOVs that specifies the implementation of the IST requirements in the ASME OM Code. The POV program description will also specify attributes for a successful periodic verification program described in RIS 2000-03 to incorporate lessons learned from valve operating experience at nuclear power plants and industry and regulatory research programs. In addition, the POV program description will include provisions that specify critical parameters, consideration of uncertainties in diagnostic analysis, POV testing acceptance criteria, and testing for all safety-related POVs regardless of their safety significance. The staff finds that the DCD with the planned modifications will provide a description of the IST program for POVs other than MOVs that may be referenced by a COL applicant as part of providing a full description of the IST operational program. Therefore, the staff finds the response acceptable. **RAI 801-5897, Question 03.09.06-59 is being tracked as a Confirmatory Item**.

In **RAI 288-2274, Question 03.09.06-15**, the staff requested that the applicant describe the IST provisions for SOVs. In its response to **RAI 288-2274, Question 03.09.06-15**, dated May 25, 2009, the applicant stated that the DCD would be revised to specify that SOVs will be verified, to the extent practical, to be capable of performing their safety functions for the electrical power supply amperage and voltage at design-basis extremes. Subsequently, the staff confirmed that Revision 2 (and Revision 3) to DCD Tier 2, Subsection 3.9.6.3.2 includes the IST provisions for SOVs specified in the RAI response. The staff finds the IST provisions for SOVs in DCD Tier 2, Section 3.9.6.3.2, to be acceptable as consistent with the guidance in RG 1.206. Therefore, the response is acceptable. Accordingly, **RAI 288-2274, Question 03.09.06-15, is resolved**.

Inservice Testing Program for Check Valves

DCD Tier 2, Subsection 3.9.6.3.3, "IST Program for Check Valves," specifies IST provisions for check valves in the US-APWR. In **RAI 288-2274, Question 03.09.06-16**, the staff requested that the applicant clarify the testing provisions in DCD Tier 2, Subsection 3.9.6.3.3 for check valves in series. In its response to **RAI 288-2274, Question 03.09.06-16**, dated May 25, 2009, the applicant stated that the US-APWR will have series-installed check valves in the SI and RHR systems. The applicant stated that these valves are pressure isolation valves (PIVs) that will be leak tested individually.

However, the test connections for check valve testing were not clearly described. Therefore, the staff closed as unresolved **RAI 288-2274, Question 03.09.06-16** and in follow-up **RAI 801-5897, Question 03.09.06-60**, the staff requested that the applicant describe the test connections for these check valves to allow testing in both directions consistent with Commission guidance (for example, RG 1.206) for check valve testing for new plants.

In its response to **RAI 801-5897, Question 03.09.06-60,** dated November 2, 2011, as revised in its submittal dated March 8, 2012, the applicant stated that a temporary test connection is not required to perform a test of the check valves in the SI and RHR systems. The applicant noted that opening of these check valves can be verified by confirming the presence of flow from the discharge of the applicable system pumps. The applicant stated that permanent test lines identified in piping diagrams in the DCD are used to perform tests in the close direction of the check valves. The staff finds that the applicant has clarified the DCD regarding the testing of these check valves and therefore the response is acceptable. Accordingly, **RAI 801-5897, Question 03.09.06-60, is resolved**.

The NRC regulations in 10 CFR 50.55a through incorporation by reference of Subsection ISTC-3610 of the ASME OM Code specify requirements for the leakage rates for ASME OM Category A valves. In **RAI 288-2274, Question 03.09.06-17**, the staff requested that the applicant clarify the leakage rate testing requirements for ASME OM Category A check valves in DCD Tier 2, Subsection 3.9.6.3.3. In its response to **RAI 288-2274, Question 03.09.06-17**, dated May 25, 2009, the applicant stated that PIVs have maximum leakage requirements included in the surveillance requirements for DCD Tier 2, Chapter 16, "Technical Specification" 3.4.14, "RCS Pressure Isolation Valve (PIV) Leakage," and that CIV leakage rates will be specified in accordance with 10 CFR Part 50, Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors." In that IST leakage testing is required to satisfy the requirements of ASME OM Code, Subsection ISTC-3600, the staff finds the clarification of the leakage testing provisions for ASME OM Category A check valves in the US-APWR to be acceptable. Therefore the response is acceptable. Accordingly **RAI 288-2274, Question 03.09.06-17, is resolved**.

The IST program for check valves needs to address the effects of rapid pump starts and stops as expected for system operating conditions and other reverse flow conditions that might occur during expected system operating conditions. In **RAI 288-2274, Question 03.09.06-18**, the staff requested that the applicant provide additional information on check valve testing to address pump operation. In its response to **RAI 288-2274, Question 03.09.06-18**, dated May 25, 2009, the applicant stated that the DCD would be revised to include provisions for check valve testing to address pump operation. Subsequently, the staff confirmed that Revision 2 (and Revision 3) to DCD Tier 2, Subsection 3.9.6.3.3 states that the effects of rapid pump starts and stops will be considered in check valve testing if expected for system operating conditions. The DCD also states that other reverse flow conditions will be considered in the testing if it may occur during expected system operating conditions. The staff finds that the DCD specifies acceptable provisions for the consideration of pump operation and other reverse flow conditions during check valve testing. Therefore the response is acceptable. Accordingly, **RAI 288-2274, Question 03.09.06-18, is resolved.**

DCD Tier 2, Section 9.5, “Other Auxiliary Systems,” describes the GTG system, which is the source of emergency power in the US-APWR design. In **RAI 288-2274, Question 03.09.06-28**, the staff requested that the applicant provide additional information regarding the inclusion of pumps and valves in the GTG system as part of the IST program for the US-APWR. In its response to **RAI 288-2274, Question 03.09.06-28**, dated May 25, 2009, the applicant stated that pumps and valves in the GTG system perform safety functions and will be included in the IST program. The applicant referred to **RAI 288-2274, Question 03.09.06-9,** for incorporation of the fuel oil transfer pumps of the GTGs in DCD Tier 2, Table 3.9-13. The staff confirmed that Revision 2 (and Revision 3) to DCD Tier 2, Table 3.9-14 includes POVs, check valves, and relief valves in the GTG system with their IST parameters and frequencies as indicated in the RAI response in accordance with the ASME OM Code. Therefore the response is acceptable. Accordingly, **RAI 288-2274, Question 03.09.06-28, is resolved**.

DCD Tier 2, Section 10.4.7, “Condensate and Feedwater System,” indicates that check valves are used to isolate the normal feedwater system from the EFW system in the US-APWR design. In **RAI 288-2274, Question 03.09.06-29**, the staff requested that the applicant provide additional information regarding the inclusion of these check valves in the IST program. In its response to **RAI 288-2274, Question 03.09.06-29**, dated May 25, 2009, the applicant stated that these check valves have a safety function to close and to isolate normal feedwater and EFW during emergency operations. The applicant stated that the DCD would be revised to include these valves in the IST program. Subsequently, Revision 2 (and Revision 3) to DCD Tier 1, Table 2.7.1.9-1, “Condensate and Feedwater System Location of Equipment and Piping,” and DCD Tier 1, Table 2.7.1.9-2, “Condensate and Feedwater System Equipment Characteristics,” includes MFW check valves NFS-VLV-511A to D. Revision 2 (and later revisions) to DCD Tier 2, Table 3.9-14 includes check valves NFS-VLV-511A to D with IST parameters and frequencies that meet the ASME OM Code requirements. Therefore the response is acceptable. Revision 3 to the DCD renumbered the MFW check valves to FWS-VLV-511A to D. Accordingly, **RAI 288-2274, Question 03.09.06-29, is resolved**.

DCD Tier 2, Section 10.4.9, “Emergency Feedwater System,” describes the EFW system for the US-APWR design. In **RAI 288-2274, Question 03.09.06-30**, the staff requested that the applicant provide additional information on the leakage function of check valves EFS-VLV-018A to D. In its response to **RAI 288-2274, Question 03.09.06-30**, dated May 25, 2009, the applicant stated that the function of these check valves is to prevent backflow from the steam generator to the EFW system. The applicant stated that no specific maximum amount of seat leakage in the closed position is applicable to these ASME OM Category B valves. DCD Tier 2, Subsection 10.4.9.2.4, "Instrumentation Requirements," specifies that EFW discharge line temperature upstream of the EFW flow control valves will be monitored with a high temperature alarm in the control room to indicate back-leakage of the check valve. The staff finds the RAI response acceptable to clarify the leakage function of the check valves EFS-VLV-018A to D that satisfies the ASME OM Code. Accordingly, **RAI 288-2274, Question 03.09.06-30, is resolved**.

DCD Tier 1, Section 2.7.1.9, "Condensate and Feedwater System," addresses the condensate and feedwater system in the US-APWR design. In **RAI 288-2274, Question 03.09.06-37**, the staff requested that the applicant provide additional information regarding the isolation features between the normal feedwater system and EFW system. In its response to **RAI 288-2274, Question 03.09.06-37**, dated May 25, 2009, the applicant stated that the MFW check valves provide isolation between these systems. The applicant referred to its response to **RAI 288-2274, Question 03.09.06-29,** for the incorporation of these valves into DCD Tier 1, Figure 2.7.1.9-1, "Feedwater System," and DCD Tier 2, Table 3.9-14. The applicant referred to its response to **RAI 288-2274, Question 03.09.06-34,** for modifications to DCD Tier 1, Table 2.7.3.1-2, regarding these valves. The applicant indicated that the ESW pump motor cooling water piping and valves would be listed in DCD Tier 1, Table 2.7.3.1-1, "Essential Service Water System Location of Equipment and Piping," and Table 2.7.3.1-3, "Essential Service Water System Piping Characteristics." The staff finds that the applicant has incorporated the ESW pump motor cooling water piping and valves in Revision 2 (and Revision 3) to DCD Tier 1, Tables 2.7.3.1-1 and 3. Therefore the response is acceptable. Accordingly, **RAI 288-2274, Question 03.09.06-37, is resolved**.

DCD Tier 1, Section 2.7.6.8, "Equipment and Floor Drainage Systems," addresses the equipment and floor drainage systems, including drainage and flood protection for rooms containing safety equipment, in the US-APWR design. Drain systems from ESF equipment rooms are designed to prevent flooding due to backflow by virtue of a difference in elevation of the ESF rooms and the collective sump. Additionally, isolation valves are provided on the ESF room drainage piping in order to protect against flooding due to backflow. In **RAI 288-2274, Question 03.09.06-40**, the staff requested that the applicant discuss prevention of backflow-induced flooding. In its response to **RAI 288-2274, Question 03.09.06-40**, dated May 25, 2009, the applicant stated that the ESF equipment rooms have isolation valves installed in the drain piping preventing in-flow of water into the room by means of the floor drains. It was further noted the isolation valves are normally closed and thus no active operation is necessary for the valves to perform their function. The potential differential pressure across the valves would be low, thus leakage past the closed isolation valve would be minimal. For these reasons, the applicant determined that no IST requirements apply to the ESF room drain isolation valves. However, the basis for excluding these valves from the IST Program was unclear. Therefore, the staff closed as unresolved **RAI 288-2274, Question 03.09.06-40,** and in follow-up **RAI 801-5897, Question 03.09.06-65**, the staff requested that the applicant address whether opening these valves manually is a credited safety function and clarify whether the valves should be included in the IST program.

In its response to **RAI 801-5897, 03.09.06-65**, dated November 2, 2011, the applicant stated that the safety function of these normally closed manual valves is to prevent flooding of the ESF room due to backflow, and that opening the valves manually is not a credited safety function but used only for maintenance of ESF equipment. In addition, the applicant indicated in its response dated December 17, 2012, to **RAI 242-2153, Question 14.03.03-16** that DCD Tier 1, Table 2.7.6.8-1, “Equipment and Floor Drainage Systems Inspections, Tests, Analyses and Acceptance Criteria,” will be revised to include ITAAC to confirm the leak tightness of the manual ESF equipment room drain isolation valves. The staff finds that the applicant has clarified the intended function of the manual valves in the ESF equipment rooms. Therefore, the response is acceptable. Accordingly, **RAI 801-5897, Question 03.09.06-65, is resolved.** To confirm the planned revision to DCD Tier 1, Table 2.7.6.8-1, **RAI 242-2153, Question 14.03.03-16 is being tracked as a Confirmatory Item**.

Pressure Isolation Valve Leak Testing

DCD Tier 2, Section 9.2.2 discusses the function of the CCW system, including the ability to isolate the nonsafety-related portion of the system from the safety-related portion, in the US-APWR design. In **RAI 288-2274, Question 03.09.06-26**, the staff requested that the applicant provide additional information on the isolation features of the CCW system, the significance of leakage between safety-related and nonsafety-related portions of the system, and the basis for the categorization of the valves NCS-MOV-232A and B, 233A and B, and 020A and B, and NCS-VLV-033A. In its response to **RAI 288-2274, Question 03.09.06-26**, dated May 25, 2009, the applicant indicated that valves NCS-MOV-232A and B, and 233A and B are used to establish flow bypass, and that no specific maximum amount of seat leakage applies to these valves. The applicant also noted that valve NCS-VLV-033A has no safety function relating to IST. The staff finds that the applicant has clarified the leakage function of valves NCS-MOV-232A and B, 233A and B, and 020A and B, and NCS-VLV-033A in the CCW system that satisfies the ASME OM Code requirements. Therefore, the response is acceptable. Accordingly, **RAI 288-2274, Question 03.09.06-26, is resolved**.

DCD Tier 1, Section 2.7.1.11, “Emergency Feedwater System,” addresses the EFW system in the US-APWR design. The staff found that Revision 0 to DCD Tier 1, Figure 2.7.1.11-1, “Emergency Feedwater System,” did not list the isolation valves between the SG and the EFW system. In **RAI 288-2274, Question 03.09.06-31**, the staff requested that the applicant discuss the means of achieving isolation, whether the isolation should be shown on the figure, and the need to establish leakage criteria. In its response to **RAI 288-2274, Question 03.09.06-31**, dated May 25, 2009, the applicant indicated that valves EFS-VLV-018A to D had been included in Figure 2.7.1.11-1. The staff found valves EFS-VLV-018A to D to be identified in Revision 2 (and Revision 3) to DCD Tier 1, Section 2.7.1.11, including Figure 2.7.1.11-1. Therefore, the response is acceptable. Accordingly, **RAI 288-2274, Question 03.09.06-31, is resolved.**

DCD Tier 1, Section 2.7.6.3, “Spent Fuel Pit Cooling and Purification System,” addresses the SFPCS in the US-APWR design. In **RAI 288-2274, Question 03.09.06-39**, the staff requested that the applicant specify the IST requirements for valves SFS-VLV-101A and B, and 133A and B, which isolate the safety-related spent fuel cooling system from the non-code purification system. In its response to **RAI 288-2274, Question 03.09.06-39**, dated May 25, 2009, the applicant stated that these valves would be included in the IST program. Subsequently, Revision 2 (and Revision 3) to DCD Tier 2, Table 3.9-14 includes valves SFS-VLV-101A and B, and 133A and B, and their IST provisions. The staff finds that the IST provisions in Table 3.9-14 for valves SFS-VLV-101A and B, and 133A and B satisfy the ASME OM Code. Therefore, the response is acceptable. Accordingly, **RAI 288-2274, Question 03.09.06-39, is resolved**.

Inservice Testing Program for Safety and Relief Valves

DCD Tier 2, Section 3.9.6.3.6, “IST Program for Safety and Relief Valves,” addresses the IST program for pressure relief devices in the US-APWR design. In **RAI 288-2274, Question 03.09.06-19**, the staff requested that the applicant confirm that the safety and relief valves would be tested in accordance with ASME OM Code, Appendix I. In its response to **RAI 288-2274, Question 03.09.06-19**, dated May 25, 2009, the applicant stated that the DCD would be revised to clarify the testing of the safety and relief valves. Subsequently, Revision 2 (and Revision 3) to DCD Tier 2, Section 3.9.6.3.6 includes a provision that pressure relief devices that provide a safety-related function in shutting down the reactor, in mitigating the consequence of an accident, or in protecting equipment in systems that perform a safety-related function will be tested in accordance with the ASME OM Code. DCD Tier 2, Section 3.9.6.3.6 also states that the inservice tests for these valves are identified in ASME OM Code, Appendix I. The staff finds that the DCD specifies provisions for pressure relief devices that satisfy the ASME OM Code. Therefore, the response is acceptable. Accordingly, **RAI 288-2274, Question 03.09.06-19 is resolved**.

DCD Tier 2, Section 3.9.6.3.6 states the test frequency for safety and relief valves is every five years for ASME B&PV Code, Section III, Class 1 valves and MS line safety valves, and every 10 years for ASME B&PV Code, Section III, Class 2 and 3 valves. In **RAI 288-2274, Question 03.09.06-20**, the staff requested that the applicant address the requirement in ASME OM Code, Appendix I, for periodic sampling of these valves. In its response to **RAI 288-2274, Question 03.09.06-20**, dated May 25, 2009, the applicant stated that the DCD would be revised to address this Appendix I requirement. Subsequently, Revision 2 (and Revision 3) to DCD Tier 2, Section 3.9.6.3.6 specifies that 20 percent of the valves from each valve group will be tested within any 24-month interval for Class 1 valves and MS line safety valves, and within any 48-month interval for Class 2 and 3 valves. The staff finds the DCD to specify sampling of the safety and relief valves that satisfies the ASME OM Code, Appendix I requirements. Therefore, the response is acceptable. Accordingly, **RAI 288-2274, Question 03.09.06-20, is resolved**.

DCD Tier 2, Section 9.3.4, "Chemical and Volume Control System," discusses pressure relief valves in the CVCS in the US-APWR design. In **RAI 288-2274, Question 03.09.06-27**, the staff requested that the applicant identify the pressure relief valves in the CVCS that are to be included in the IST program. In its response to **RAI 288-2274, Question 03.09.06-27**, dated May 25, 2009, the applicant indicated that the pressure relief valves in the CVCS are included in DCD Tier 2, Table 3.9-14. The staff finds that the RAI response has clarified the IST requirements for the CVCS pressure relief valves in the DCD. Accordingly, **RAI 288-2274, Question 03.09.06-27, is resolved**.

Inservice Testing Program for Manual Valves

DCD Tier 2, Table 3.9-14 identifies manual CIV RCS-VLV-140 as a vacuum-venting line check valve bypass valve. In **RAI 288-2274, Question 03.09.06-21**, the staff requested that the applicant discuss the exercising requirements for this valve. In its response to **RAI 288-2274, Question 03.09.06-21**, dated May 25, 2009, the applicant clarified that RCS-VLV-140 is a locked-closed valve with no active safety function, and is only opened during refueling outages to perform a vacuum-venting operation. The staff finds that the RAI response clarifies that exercising of this passive valve is not required by the ASME OM Code. Accordingly, **RAI 288-2274, Question 03.09.06-21, is resolved**.

Revision 0 to DCD Tier 2, Table 3.9-14 specified that valves SIS-VLV-114 and DWS-VLV-004 were manual CIVs. In **RAI 288-2274, Question 03.09.06-22**, the staff requested that the applicant address the exercising requirements for these valves. In its response to **RAI 288-2274, Question 03.09.06-22**, dated May 25, 2009, the applicant stated that Revision 1 to DCD

Tier 2, Table 3.9-14 specified valve SIS-VLV-114 as a power-operated valve (identified as SIS-AOV-114) with full-stroke exercise testing performed at a cold shutdown interval. The applicant clarified that manual valve DWS-VLV-004 is normally locked closed with no active safety function, and is only opened during refueling outages to provide demineralized water for maintenance. The staff finds that DCD Tier 2, Table 3.9-14 specifies full-stroke exercising during cold shutdown condition for valve SIS-AOV-114. The staff also finds the RAI response to clarify that the ASME OM Code does not require exercising of manual valve DWS-VLV-004. Therefore, **RAI 288-2274, Question 03.09.06-22, is resolved**.

Inservice Testing Program for Pyrotechnic-Actuated Valves

The US-APWR design does not include pyrotechnic-actuated valves.

Review of Valve IST Provisions in DCD Tier 2, Table 3.9-14

The staff reviewed DCD Tier 2, Table 3.9-14 to confirm that the valve testing provisions are consistent with the ASME OM Code requirements and other sections of the DCD. In **RAI 288-2274, Question 03.09.06-12**, the staff requested the applicant to provide additional information on the following items (Items (a) through (aa)). The applicant submitted its response **RAI 288-2274, Question 03.09.06-12** on May 25, 2009. The statement of the staff concern, the applicant's response, and the staff evaluation for each item follows.

(a) The staff requested that the applicant provide additional information regarding position indication testing addressed in Note 1 to DCD Tier 2, Table 3.9-14 for valves required by ASME OM Code, ISTC-3700. In its response the applicant stated that Note 1 would be clarified regarding position indication sensors. Subsequently, Revision 2 (and Revision 3) to DCD Tier 2, Table 3.9-14, states in Note 1 that position indication sensors for pressurizer safety valves and MS safety valves will be tested during set-pressure testing required by ASME OM Code, Appendix I. The staff finds that Note 1 in DCD Tier 2, Table 3.9-14 clarifies position indication testing for the applicable valves and therefore the response is acceptable. Accordingly, **RAI 288-2274, Question 03.09.06-12(a) is resolved**.

(b) The staff requested that the applicant provide additional information regarding an operability test for specific valves in DCD Tier 2, Table 3.9-14. In its response the applicant indicated that the operability test relates to functional testing specified in DCD Tier 2, Section 3.9.6.3.1. The staff finds that the RAI response clarifies the operability test provision in DCD Tier 2, Table 3.9-14 and therefore the response is acceptable. Accordingly, **RAI 288-2274, Question 03.09.06-12(b) is resolved**.

(c) The staff requested that the applicant provide additional information on leakage criteria for RCS PIVs RCS-MOV-116A and B, and 117A and B, which are identified as ASME OM Category B valves. In its response the applicant stated that any leakage past these valves is to the pressurizer relief tank (PRT), which does not present a potential for system overpressurization due to the leakage. However, any leakage limits for the system through these valves were not clearly addressed. Therefore, the staff closed, as unresolved, **RAI 288-2274, Question 03.09.06-12(c)** and in follow-up **RAI 801-5897, Question 03.09.06-57**, the staff requested that the applicant clarify any leakage limits for the system through

RCS-MOV-116A and B, and 117A and B. In its response to **RAI 801-5897, Question 03.09.06-57**, dated November 2, 2011, as revised in its submittal dated March 8, 2012, the applicant stated that the leakage limits for the RCS pressure boundary valves RCS-MOV-116A and B, and 117A and B will be the same as the leakage limits for PIVs as specified in the TS 3.4.14, "RCS Pressure Isolation Valve (PIV) Leakage," DCD Tier 2, Chapter 16. The applicant stated that DCD Tier 2, Table 3.9-14 would be revised to categorize these valves as Category A, and to specify leak testing on a refueling outage frequency for these valves with a new note specifying that TS SR 3.4.14.1 will be used for the valve leakage acceptance criteria. The staff finds that the planned modifications to the DCD will include acceptable categorization and leak test provisions for RCS-MOV-116A and B, and 117A and B. Therefore, the response to this part of **RAI 801-5897, Question 03.09.06-57** is acceptable. **RAI 801-5897, Question 03.09.06-57, is being tracked as a Confirmatory Item**.

(d) The staff requested that the applicant provide additional information on the leakage criteria for valves RCS-MOV-118, 119, 002A and B, and 003A and B, which are maintained closed to preserve the RCS pressure boundary. In its response the applicant stated that no leakage criteria are applicable to these valves because any leakage from these valves is discharged to the PRT. However, any leakage limits for the system through these valves were not clearly addressed. Therefore, the staff closed, as unresolved, **RAI 288-2274, Question 03.09.06-12(d)** and in follow-up **RAI 801-5897, Question 03.09.06-57**, the staff requested that the applicant clarify any leakage limits for the system through RCS-MOV-118, 119, 002A and B, and 003A and B. In its response to **RAI 801-5897, Question 03.09.06-57** dated November 2, 2011, as revised in its submittal dated March 8, 2012, the applicant stated that the leakage limits for RCS pressure boundary valves RCS-MOV-118, 119, 002A and B, and 003A and B will be the same as the leakage limits for pressure isolation valves as specified in TS 3.4.14. The applicant stated that Table 3.9-14 would be revised to include RCS-MOV-118 and 119. The applicant also stated that DCD Tier 2, Table 3.9-14 would be revised to categorize RCS-MOV-118, 119, 002A and B, and 003A and B as Category A valves, and to specify leak testing on a refueling outage frequency for these valves with a new note specifying that TS SR 3.4.14.1 will be used for the valve leakage acceptance criteria. The staff finds that the planned modifications to the DCD will include acceptable categorization and leak test provisions for RCS-MOV-118, 119, 002A and B, and 003A and B. Therefore, the response to this part of **RAI 801-5897, Question 03.09.06-57** is acceptable. **RAI 801-5897, Question 03.09.06-57 is being tracked as a Confirmatory Item**.

(e) The staff requested that the applicant provide additional information regarding the basis for not specifying quarterly testing of valves RCS-AOV-132, 138, 147, and 148. In its response the applicant stated that these valves can be exercised quarterly and that the DCD would be revised to reflect this IST frequency. Subsequently, Revision 2 (and Revision 3) to DCD Tier 2, Table 3.9-14 specifies quarterly exercising of RCS-AOV-132, 138, 147, and 148. The staff finds the revised IST frequency for these valves to satisfy the ASME OM Code. Therefore the response is acceptable. Accordingly, **RAI 288-2274, Question 03.09.06-12(e), is resolved**.

(f) The staff requested that the applicant provide additional information on the type of actuator for valves listed in DCD Tier 2, Table 3.9-14 as having a "remote" valve type. In its response the applicant stated that DCD Tier 2, Table 3.9-14 would be revised to specify valve and actuator type. Subsequently, Revision 2 (and Revision 3) to DCD Tier 2, Table 3.9-14 includes the valve and actuator type for valves within the scope of the US-APWR IST program. The staff finds DCD Tier 2, Table 3.9-14 to clarify the valve and actuator type for valves in the IST program for the US-APWR. Therefore the response is acceptable. Accordingly, **RAI 288-2274, Question 03.09.06-12(f) is resolved**.

(g) The staff requested that the applicant provide additional information regarding the leakage criteria for CIVs SIS-MOV-001A to D, and 009A to D, and SIS-VLV-010A to D listed in DCD Tier 2, Table 3.9-14. In its response the applicant indicated that DCD Tier 2, Table 3.9-14 would be revised to specify leakage testing for valves SIS-MOV-009A to D. The applicant clarified that 10 CFR Part 50, Appendix J, Type C testing is not required for valves SIS-MOV-001A to D as they are installed in closed systems outside containment and have a fluid seal. Revision 2 to DCD Tier 2, Table 3.9-14 specified leakage testing for valves SIS-MOV-009A to D, but deleted other IST requirements. Revision 3 to DCD Tier 2, Table 3.9-14 reinstated the exercise full stroke/quarterly operability test for these valves. However, DCD Tier 2, Table 3.9-14 did not include the remote position indication test for these valves. Therefore, the staff closed, as unresolved, **RAI 288-2274, Question 03.09.06-12(g)** and in follow-up **RAI 801-5897, Question 03.09.06-57**, the staff requested that the applicant resolve the deletion of the remote position indication test for SIS-MOV-009A to D from Table 3.9-14. The staff also requested that the applicant confirm that DCD Tier 2, Table 3.9-14 requires a remote position indication test for other valves in accordance with the ASME OM Code. In its response to **RAI 801-5897, Question 03.09.06-57**, dated November 2, 2011, as revised in its submittal dated March 8, 2012, the applicant stated that a remote position indication test every two years would be added to DCD Tier 2, Table 3.9-14 for SIS-MOV-009A to D. The applicant indicated that DCD Tier 2, Table 3.9-14 was reviewed for other valves requiring a remote position indication test in accordance with the ASME OM Code. As a result, the applicant stated DCD Tier 2, Table 3.9-14 would be revised to specify a remote position indication test every two years for SIS-MOV-001A to D. The applicant also indicated that the provision for remote position indication for RWS-VLV-023 would be deleted as this is a check valve without remote position indication capability. The staff finds that the planned modifications to DCD Tier 2, Table 3.9-14 to specify appropriate remote position indication provisions are acceptable. Therefore, the response to this part of **RAI 801-5897, Question 03.09.06-57** is acceptable. **RAI 801-5897, Question 03.09.06-57, is being tracked as a Confirmatory Item**.

(h) The staff requested that the applicant clarify the identification of valve SIS-VLV-012A as a PIV. In its response the applicant clarified the specification of valve SIS-VLV-012A as a RCPB PIV in Table 3.9-14. The staff finds the RAI response to clarify the valve specification. Accordingly, **RAI 288-2274, Question 03.09.06-12(h) is resolved**.

(i) The staff requested that the applicant clarify the allowable leak rate limits for RCPB PIVs SIS-MOV-014A to D. In its response the applicant stated that SIS-

MOV-014A to D isolate the RCS from an attached SI system. The applicant stated that leakage from these valves would be discharged to the refueling water storage pit. Therefore, the applicant did not consider leakage limits to be necessary. However, any leakage limits for the system through these valves were not clearly addressed. Therefore, the staff closed, as unresolved, **RAI 288-2274, Question 03.09.06-12(i)** and in follow-up **RAI 801-5897, Question 03.09.06-57**, the staff requested that the applicant clarify whether any leakage limitations from the RCS are applicable to these valves. In its response to **RAI 801-5897, Question 03.09.06-57** dated November 2, 2011, as revised in its submittal dated March 8, 2012, the applicant stated the leakage limits for the RCS pressure boundary valves SIS-MOV-014A to D will be the same as the leakage limits for pressure isolation valves as specified in TS 3.4.14. The applicant stated that Table 3.9-14 would be revised to categorize these valves as Category A, and to specify leak testing on a refueling outage frequency for these valves with a new note indicating that TS SR 3.4.14.1 will be used for the valve leakage acceptance criteria. The staff finds that the planned modifications to the DCD will include acceptable categorization and leak test provisions for SIS-MOV-014A to D. Therefore, this part of the response to **RAI 801-5897, Question 03.09.06-57** is acceptable. **RAI 801-5897, Question 03.09.06-57, is being tracked as a Confirmatory Item**.

(j) The staff requested that the applicant clarify the allowable leak rate limits for certain RCS pressure boundary valves. In its response the applicant stated that SIS-MOV-031A, 031D, 032A, and 032D isolate the RCS from an attached SIS. The applicant stated that any leakage would be discharged to the RWSP. Therefore, the applicant did not consider leakage limits to be necessary. However, any leakage limits for the system through these valves were not clearly addressed. Therefore, the staff closed, as unresolved, **RAI 288-2274, Question 03.09.06-12(j)** and in follow-up **RAI 801-5897, Question 03.09.06-57**, the staff requested that the applicant clarify whether any leakage limitations from the RCS are applicable to these valves. In its response to **RAI 801-5897, Question 03.09.06-57** dated November 2, 2011, as revised in its submittal dated March 8, 2012, the applicant stated the leakage limits for the RCS pressure boundary valves SIS-MOV-031A, 031D, 032A, and 032D will be the same as the leakage limits for pressure isolation valves as specified in TS 3.4.14. The applicant stated that Table 3.9-14 would be revised to categorize these valves as Category A, and to specify leak testing on a refueling outage frequency for these valves with a new note indicating that TS SR 3.4.14.1 will be used for the valve leakage acceptance criteria. The staff finds that the planned modifications to the DCD will include acceptable categorization and leak test provisions for SIS-MOV-031A, 031D, 032A, and 032D. Therefore, this part of the response to **RAI 801-5897, Question 03.09.06-57** is acceptable. **RAI 801-5897, Question 03.09.06-57, is being tracked as a Confirmatory Item**.

(k) The staff requested that the applicant provide additional information on the basis for testing check valves SIS-VLV-102A to D and SIS-VLV-103A to D by alternate means as indicated in DCD Tier 2, Table 3.9-14 rather than by non-intrusive means. In its response the applicant agreed that these check valves could be tested by non-intrusive means and that DCD Tier 2, Table 3.9-14 would be modified accordingly. Subsequently, Revision 2 (and Revision 3) to DCD Tier 2, Table 3.9-14 removed the reference to alternative means for testing SIS-VLV-

102A to D and SIS-VLV-103A to D. The staff finds DCD Tier 2, Table 3.9-14 to specify testing of these check valves by non-intrusive means consistent with nuclear power plant operating experience. Therefore the response is acceptable. Accordingly, **RAI 288-2274, Question 03.09.06-12(k) is resolved**.

(l) The staff requested that the applicant provide additional information regarding the basis for omitting leak tests of RCS pressure boundary CIVs RHS-MOV-002A to D, which are water sealed in a closed system and closed when in modes above hot shutdown. In its response the applicant discussed the function of the valves, and the configuration and operation of the associated system. Revision 2 (and Revision 3) to DCD Tier 2, Table 3.9-14, includes Note 10 that discusses the justification for omitting leak tests of these valves. However, the justification for omitting leak tests of these valves was not clear. Therefore, the staff closed, as unresolved, **RAI 288-2274, Question 03.09.06-12(l)** and in follow-up **RAI 801-5897, Question 03.09.06-57**, the staff requested that the applicant clarify the potential loss of the water seal and if leak testing of these valves is necessary to address this scenario. In its response to **RAI 801-5897, Question 03.09.06-57**, dated November 2, 2011, as revised in its submittal dated March 8, 2012, the applicant stated that the containment penetrations and CIVs, including valves RHS-MOV-002A to D, are tested as Type A under 10 CFR Part 50, Appendix J. The applicant stated that RHS-MOV-001A to D and 002A to D are subject to the RCPB leak test as specified in DCD Tier 2, Table 3.9-14. As described in Note 10 to DCD Tier 2, Table 3.9-14, the applicant indicated that the justification for the categorization of these valves is that the specific system is a closed system outside containment design, and the valves do not constitute a potential containment atmosphere leak path. The applicant provided planned modifications to DCD Tier 2, Table 3.9-14 to clarify the reference to the applicable notes. The staff finds that the RAI response and planned modifications to the DCD will clarify the leak tests for RHS-MOV-002A to D. Therefore, this part of the response to **RAI 801-5897, Question 03.09.06-57** is acceptable. **RAI 801-5897, Question 03.09.06-57, is being tracked as a Confirmatory Item**.

(m) The staff requested that the applicant provide additional information on the leakage categorization of valves RHS-MOV-026A to D. In its response the applicant clarified that RHS-MOV-026A to D are isolated from the RCS by check valves RHS-VLV-027A to D, and by downstream check valves RHS-VLV-028A to D. The check valves are identified as PIVs and are ASME OM Category A/C valves. Therefore, valves RHS-MOV-026A to D do not have specific leakage limitations and are categorized as ASME OM Category B. The staff finds that the RAI response clarifies the leakage categorization of RHS-MOV-026A to D. Accordingly, **RAI 288-2274, Question 03.09.06-12(m) is resolved**.

(n) The staff requested that the applicant provide additional information on the leakage categorization of CIVs EFS-MOV-019A to D in the EFW system. In its response the applicant clarified that these EFW valves are installed in piping that is considered to be an extension of the SG secondary system and, thus, not subject to ASME OM Code Category A leak testing. The staff finds that the RAI response has clarified the leakage categorization of EFS-MOV-019A to D. Accordingly, **RAI 288-2274, Question 03.09.06-12(n) is resolved**.

(o) The staff requested that the applicant provide additional information on the leakage categorization of CIVs NFS-VLV-512A to D in the MFW system. In its response the applicant clarified that similar to the response to item (n) of this RAI, these MFW system valves (identified as NFS-SMV-512A to D in Revision 2 and FWS-SMV-512A to D in Revision 3 to the DCD) are installed in piping that is considered an extension of the secondary side of the SG and, thus, not subject to ASME OM Code Category A leak testing. The staff finds the RAI response has clarified the leakage categorization of FWS-SMV-512A to D. Accordingly, **RAI 288-2274, Question 03.09.06-12(o) is resolved**.

(p) The staff requested that the applicant clarify the valve type for MFW isolation valves NFS-VLV-512A to D to be able to determine the appropriate IST requirements. In its response the applicant stated that the valve type and IST requirements would be specified in DCD Tier 2, Table 3.9-14. Subsequently, Revision 2 to DCD Tier 2, Table 3.9-14 identified the MFW isolation valves as NFS-SMV-512A to D with IST requirements as Remote Position Indication with Exercise every two years, Exercise Full Stroke at Cold Shutdown, and Operability Test. In Revision 3 to the DCD, these valves are identified as FWS-SMV-512A to D. However, the revised DCD appeared to be inconsistent with the RAI response. Therefore, the staff closed, as unresolved, **RAI 288-2274, Question 03.09.06-12(p)** and in follow-up **RAI 801-5897, Question 03.09.06-57**, the staff requested that the applicant clarify the apparent difference in the valve/actuator type indicated in the RAI response and DCD. In its response to **RAI 801-5897, Question 03.09.06-57**, dated November 2, 2011, as revised in its submittal dated March 8, 2012, the applicant stated that the valve tag numbers NFS-VLV-512A to D had been changed in Revision 3 to the DCD to FWS-SMV-512A to D. The staff finds that DCD Tier 2, Table 3.9-14 includes the corrected valve numbers. Therefore, this part of the response to **RAI 801-5897, Question 03.09.06-57** is acceptable.

(q) The staff requested that the applicant clarify the exercising requirement for the MFW isolation valves. In its response the applicant stated that the exercising requirements for these valves would be addressed as indicated in response to **RAI 288-2274, Question 03.09.06-12(p)**. The staff confirmed that the applicant addressed this item through **RAI 288-2274, Question 03.09.06-12(p)** as discussed above. Therefore the response is acceptable. Accordingly, **RAI 288-2274, Question 03.09.06-12(q) is resolved**.

(r) The staff requested that the applicant specify the valve and actuator type for valves NMS-HCV-3625, 3635, and 3645. In its response the applicant stated that DCD Tier 2, Table 3.9-14 would be revised to specify the valve and actuator type for these valves. However, the staff did not find these valve numbers in DCD Revision 3. Therefore, the staff closed, as unresolved, **RAI 288-2274, Question 03.09.06-12(r)** and in follow-up **RAI 801-5897, Question 03.09.06-57**, the staff requested that the applicant identify these valves in DCD Tier 2, Table 3.9-14. In its response to **RAI 801-5897, Question 03.09.06-57**, dated November 2, 2011, as revised in its submittal dated March 8, 2012, the applicant stated that the valve tag numbers for NMS-HCV-3615, 3625, 3635, and 3645 were changed in Revision 3 to the DCD to MSS-HCV-565, 575, 585, and 595. The staff finds that DCD Tier 2, Table 3.9-14 includes the corrected valve numbers. Therefore, this part of the response to **RAI 801-5897, Question 03.09.06-57** is acceptable.

(s) The staff requested that the applicant provide additional information on the leakage requirements for CIVs CSS-MOV-004A to D. In its response the applicant stated that the leakage criteria for these valves would be specified. Subsequently, Revision 2 (and Revision 3) to DCD Tier 2, Table 3.9-14 specifies CSS-MOV-004A to D as ASME OM Category A valves with leakage testing every refueling outage consistent with the ASME OM Code. Therefore the response is acceptable. Accordingly, **RAI 288-2274, Question 03.09.06-12(s) is resolved**.

(t) The staff requested that the applicant provide additional information regarding the basis for alternate exercise methods specified for check valves CSS-VLV-005A to D in DCD Tier 2, Revision 0, Table 3.9-14 rather than nonintrusive means. In its response the applicant stated that these valves could be tested by nonintrusive means. Subsequently, Revision 2 to DCD Tier 2, Table 3.9-14 specifies exercise testing of valves CSS-VLV-005A to D every refueling outage. In Revision 3 to DCD Tier 2, Table 3.9-14, the applicant identified valves CSS-VLV-005A to D as ASME OM Category A/C check valves. However, any leak testing provisions of these valves were not clearly addressed. Therefore, the staff closed, as unresolved, **RAI 288-2274, Question 03.09.06-12(t)** and in follow-up **RAI 801-5897, Question 03.09.06-57**, the staff requested that the applicant specify the leak testing provisions (and remote position indication if equipped with remote indicators) for these valves in DCD Tier 2, Table 3.9-14. In its response to **RAI 801-5897, Question 03.09.06-57** dated November 2, 2011, as revised in its submittal dated March 8, 2012, the applicant stated that Note 4 in DCD Tier 2, Table 6.2.4-3 provides the justification for CSS-VLV-005A to D in that these valves are located in a closed system outside containment, and do not constitute a potential containment atmosphere leak path during or following a LOCA with a single active failure of a system component. The applicant also noted that these valves do not possess remote position indication. The staff finds that the applicant has clarified the leak testing provisions for CSS-VLV-005A to D. Therefore, this part of the response to **RAI 801-5897, Question 03.09.06-57,** is acceptable.

(u) The staff requested that the applicant clarify the function of valves EWS-VLV-602A to D. In its response the applicant indicated that the function of these valves is to open and close the ESW pump motor cooling water path according to the cooling water supply conditions. In Revision 3 to DCD Tier 2, the applicant deleted these valves from DCD Tier 2, Table 3.9-14. Therefore, the staff closed, as unresolved, **RAI 288-2274, Question 03.09.06-12(u)** and in follow-up **RAI 801-5897, Question 03.09.06-57**, the staff requested that the applicant clarify the basis for the deletion of these valves from the IST program. In its response to **RAI 801-5897, Question 03.09.06-57**, dated November 2, 2011, as revised in its submittal dated March 8, 2012, the applicant stated that EWS-VLV-602A to D were deleted from the table because the motor cooling method was changed from water to air cooling. The staff finds that the applicant has clarified the basis for the deletion of EWS-VLV-602A to D from DCD Tier 2, Table 3.9-14. Therefore, this part of the response to **RAI 801-5897, Question 03.09.06-57,** is acceptable.

(v) The staff requested that the applicant provide additional information on the basis for testing the following valves on a cold shutdown frequency: LMS-AOV-060, 056, 055, 053, and 052; LMS-LCV-1000A and 1000B; LMS-AOV-105 and 104; PSS-AOV-003; PSS-MOV-006, 013, 023, 031A, 031B, 052A, and 052B; PSS-AOV-062B, 062C, 062D, 063, and 071; PSS-VLV-072; and SGS-AOV-031A to D. In its response the applicant indicated that the testing frequency for these valves (except PSS-VLV-072) would be changed to quarterly. The applicant indicated that the exercise test frequency for check valve PSS-VLV-072 will be retained as Exercise/Refueling outage because it is impractical to test the valve during normal operation inside the containment vessel. The staff finds the response acceptable because the applicant clarified the testing frequency of the specific valves consistent with the ASME OM Code. Subsequently, the staff confirmed that Revision 2 (and Revision 3) to DCD Tier 2, Table 3.9-14 specifies quarterly exercising for LMS-AOV-060, 056, 055, 053, and 052; LMS-LCV-010A and B (previously identified as LMS-LCV-1000A and 1000B); LMS-AOV-105 and 104; PSS-AOV-003; PSS-MOV-006, 013, 023, 031A, 031B, 052A, and 052B; PSS-AOV-062B, 062C, 062D, 063, and 071; and SGS-AOV-031A to D. DCD Tier 2, Table 3.9-14 also removes the reference to Note 6 for these valves relating to cold shutdown exercising. Accordingly, **RAI 288-2274, Question 03.09.06-12(v) is resolved**.

(w) The staff requested that the applicant provide additional information on the IST requirements for containment isolation check valve DWS-VLV-005. In its response the applicant stated that DWS-VLV-005 is the CIV in the demineralized water supply line, which is not used during normal operation but only during refueling outages for maintenance activities. DCD Tier 2, Table 3.9-14 specifies check valve DWS-VLV-005 as a passive valve. However, the staff noted that check valves are considered active valves within the IST program with testing in the open and close direction to verify the integrity of the valve disk. Therefore, the staff closed, as unresolved, **RAI 288-2274, Question 03.09.06-12(w)** and in follow-up **RAI 801-5897, Question 03.09.06-57**, the staff requested that the design applicant discuss the basis for categorizing check valves in the IST program (such as DWS-VLV-005, RWS-VLV-003, and any other check valves in Table DCD Tier 2, 3.9-14) as passive valves. In its response to **RAI 801-5897, Question 03.09.06-57**, dated November 2, 2011, as revised in its submittal dated March 8, 2012, the applicant stated that DCD Tier 2, Table 3.9-14 would be revised to indicate that check valves RWS-VLV-003, DWS-VLV-005, CAS-VLV-103, and FSS-VLV-006 are considered active valves in the IST program with testing in the open and close direction to verify the integrity of the valve disk. DCD Tier 2, Table 3.9-14 will be revised to specify Category A/C for these valves and to require exercise testing on a refueling outage frequency with reference to the applicable note justifying the test frequency. The staff finds that the planned modifications to the DCD will provide categorization and testing provisions for RWS-VLV-003, DWS-VLV-005, CAS-VLV-103, and FSS-VLV-006 in conformance with the ASME OM Code. Therefore, this part of the response to **RAI 801-5897, Question 03.09.06-57** is acceptable. **RAI 801-5897, Question 03.09.06-57, is being tracked as a Confirmatory Item**.

(x) The staff requested that the applicant provide additional information on the basis for several valves listed in DCD Tier 2, Table 3.9-14 used to process essential chilled water necessary for controlling environmental conditions in various plant areas to be exercised on a cold shutdown frequency. In its response the applicant stated that the exercise frequency would be changed to quarterly for these valves. The staff finds the response acceptable because the applicant revised the testing frequency of the specific valves consistent with the ASME OM Code. Subsequently, the staff confirmed that Revision 2 (and Revision 3) to DCD Tier 2, Table 3.9-14 specifies quarterly exercising for those valves. Accordingly, **RAI 288-2274, Question 03.09.06-12(x) is resolved**.

(y) The staff requested that the applicant provide additional information on the basis for Note 11 in Revision 0 to DCD Tier 2, Table 3.9-14 regarding partial stroke tests of valves. In its response the applicant stated that Revision 1 to DCD Tier 2, Table 3.9-14 specified exercising full stroke at cold shutdown for the MS isolation valves and main feed isolation valves. The applicant also stated that Note 11 had been modified to reflect the revised exercising frequency. However, the plant conditions for the testing of these specific valves were not clearly addressed. Therefore, the staff closed, as unresolved, **RAI 288-2274, Question 03.09.06-12(y)** and in follow-up **RAI 801-5897, Question 03.09.06-57**, the staff requested that the applicant clarify the discussion in Note 11 regarding hot standby testing compared to the cold shutdown frequency specified for these valves in DCD Tier 2, Table 3.9-14. In its response to **RAI 801-5897, Question 03.09.06-57**, dated November 2, 2011, as revised in its submittal dated March 8, 2012, the applicant stated that DCD Tier 2, Table 3.9-14 would be revised to specify that full stroke testing of MS isolation valves MSS-SMV-515A to D and main feed isolation valves FWS-SMV-512A to D will be conducted at hot standby conditions (rather than cold shutdown) consistent with Note 11 in DCD Tier 2, Table 3.9-14. The staff finds that the planned modifications to the DCD will specify IST provisions for MSS-SMV-515A to D and FWS-SMV-512A to D that satisfy the ASME OM Code. Therefore, this part of the response to RAI 03.09.06-57 is acceptable. **RAI 801-5897, Question 03.09.06-57, is being tracked as a Confirmatory Item**.

(z) The staff requested that the applicant discuss the use of alternate test methods specified for accumulator injection line check valves SIS-VLV-102A to D and 103A to D, containment spray CS containment isolation check valves CSS-VLV-005A to D, and MS line check valves NMS-VLV-516A to D, rather than non-intrusive testing. In its response the applicant stated that non-intrusive means can be employed to test the accumulator injection line check valves and containment spray header containment isolation check valves as indicated in response to **RAI 288-2274, Question 03.09.06-12(k)** and **RAI 288-2274, Question 03.09.06-12(t)**. The applicant also indicated that non-intrusive testing can be applied to the turbine driven EFW pump steam supply line drain line check valves. However, the applicant stated that the MS check valves (identified as MSS-VLV-516A to D in DCD Revisions 2 and 3) cannot be tested on line and, therefore, will be tested during the cold shutdown of a refueling outage. Subsequently, Revision 2 (and Revision 3) to DCD Tier 2, Table 3.9-14, Note 12 specifies alternative testing for the MS check valves. The staff finds that the modifications to DCD Tier 2, Table 3.9-14 for testing of the accumulator injection line check valves, containment spray header containment isolation check valves, and turbine driven EFW pump steam supply line drain line check valves to satisfy the ASME OM Code. However, the plant conditions for testing these specific valves were not clearly addressed. Therefore, the staff closed, as unresolved,

RAI 288-2274, Question 03.09.06-12(z) and in follow-up **RAI 801-5897, Question 03.09.06-57**, the staff requested that the applicant clarify the provision for testing the MS line check valves when cold shutdown conditions for a refueling outage are established, rather than for any cold shutdown. In its response to **RAI 801-5897, Question 03.09.06-57**, dated November 2, 2011, as revised in its submittal dated March 8, 2012, the applicant stated that DCD Tier 2, Table 3.9-14 would be revised to specify that testing of MS check valves MSS-VLV-515A to D will be conducted when the unit is at cold shutdown conditions during an outage of sufficient duration (rather than a refueling outage frequency). The staff finds that the planned modifications to the DCD will specify IST provisions for MSS-VLV-515A to D that satisfy the ASME OM Code. Therefore, this part of the response to **RAI 801-5897, Question 03.09.06-57** is acceptable. **RAI 801-5897, Question 03.09.06-57, is being tracked as a Confirmatory Item**.

(aa) The staff requested that the applicant clarify the leak rate limitations for several reactor pressure boundary isolation valves. In its response the applicant referred to its responses to **RAI 288-2274, Question 03.09.06-12(c)** and **RAI 288-2274, Question 03.09.06-12(j)**. This staff confirmed that the issue was addressed as part of those RAIs. Therefore, the response is acceptable. Accordingly, **RAI 288-2274, Question 03.09.06-12(aa) is resolved**.

DCD Tier 2, Section 9.2.1, “Essential Service Water System,” describes the ESW system for the US-APWR design. In **RAI 288-2274, Question 03.09.06-24**, the staff requested that the applicant provide additional information on the design and operation of the ESW system, and the need for pressure relief devices in the system. In its response to **RAI 288-2274, Question 03.09.06-24**, dated May 25, 2009, the applicant stated that the ESW system is designed to withstand maximum operating pressure, taking into account maximum pump discharge and static head in the system. Therefore, the applicant did not plan to include pressure relief devices in the ESW system. However, it was not clear whether thermal expansion effects had been addressed. Therefore, the staff closed as unresolved **RAI 288-2274, Question 03.09.06-24** and in follow-up **RAI 801-5897, Question 03.09.06-62**, the staff requested in that the applicant clarify the potential for thermal expansion and the possibility of system or component damage due to thermal expansion effects.

In its response to **RAI 801-5897, Question 03.09.06-62**, dated November 2, 2011, as revised in its submittal dated March 8, 2012, the applicant stated that the ESW system includes manual valves that are locked open. The applicant indicated that plant components are isolated for maintenance after the temperature indication of the ESW outlet line equals the temperature indication of the ESW inlet line. The staff finds that the applicant has clarified the process for isolating components in the ESW system. Therefore, the response is acceptable. Accordingly, **RAI 801-5897, Question 03.09.06-62, is resolved**.

DCD Tier 1, Section 2.7.1.11 describes the EFW system for the US-APWR design. In **RAI 288-2274, Question 03.09.06-32**, the staff requested that the applicant discuss the safety function of the valves provided in DCD Tier 1, Table 2.7.1.11-2, “Emergency Feedwater System Equipment Characteristics,” and DCD Tier 2, Table 3.9-14. In its response to **RAI 288-2274, Question 03.09.06-32**, dated May 25, 2009, the applicant clarified that DCD Tier 1, Table 2.7.1.11-2 presents only the active safety function for valves while DCD Tier 2, Table 3.9-14 presents both active and passive missions. The staff finds the RAI response acceptable in that it clarifies the information presented in DCD Tier 1, Table 2.7.1.11-2 and DCD Tier 2, Table 3.9-14. Accordingly, **RAI 288-2274, Question 03.09.06-32, is resolved**.

DCD Tier 1, Table 2.7.3.1-2 specifies performance characteristics for ESW equipment, including applicable valves, in the US-APWR design. In **RAI 288-2274, Question 03.09.06-34**, the staff requested that the applicant provide additional information regarding valves EWS-VLV-502A to D, and 602A to D that are listed in DCD Tier 2, Table 3.9-14 but not in DCD Tier 1, Table 2.7.3.1-2 (Revision 0). In its response to **RAI 288-2274, Question 03.09.06-34**, dated May 25, 2009, the applicant stated that DCD Tier 1, Table 2.7.3.1-2 would be revised to include the applicable information for the subject valves. Subsequently, Revision 2 to DCD Tier 1, Table 2.7.3.1-2 includes EWS-VLV-502A to D, and 602A to D, and their applicable performance characteristics as indicated in the RAI response. However, Revision 3 to DCD Tier 1, Table 2.7.3.1-2 does not include EWS-VLV-602A to D.

Therefore, the staff closed, as unresolved, **RAI 288-2274, Question 03.09.06-34** and in follow-up **RAI 801-5897, Question 03.09.06-63**, the staff requested that the applicant provide the basis for the provisions for EWS-VLV-502A to D, and 602A to D in DCD Tier 1, Table 2.7.3.1-2. In its response to **RAI 801-5897, Question 03.09.06-63**, dated November 2, 2011, as revised in its submittal dated March 8, 2012, the applicant stated that the motor cooling method for this system had been changed from water-cooling to air-cooling and, therefore, valves EWS-VLV-602A to D had been deleted from Revision 3 of DCD Tier 1, Table 2.7.3.1-2 and DCD Tier 2, Table 3.9-14. The applicant noted that valves EWS-VLV-502A to D are included in DCD Tier 1, Table 2.7.3.1-2 and DCD Tier 2, Table 3.9-14, and that those valves do not have remote position indication. The staff finds that the applicant has clarified the DCD regarding the valves included in the ESW system. Therefore, the response is acceptable. Accordingly, **RAI 801-5897, Question 03.09.06-63, is resolved**.

DCD Tier 1, Table 2.4.5-2, "Residual Heat Removal System Equipment Characteristics," specifies the performance characteristics for equipment in the RHR system in the US-APWR design. In **RAI 288-2274, Question 03.09.06-35**, the staff requested that the applicant clarify the safety functions for valves RHS-VLV-022A to D as identified in DCD Tier 1, Table 2.4.5-2 and DCD Tier 2, Table 3.9-14. In its response to **RAI 288-2274, Question 03.09.06-35**, dated May 25, 2009, the applicant stated that valves RHS-VLV-022A to D have both transfer open and transfer closed functions. Therefore, the applicant stated that DCD Tier 1, Table 2.4.5-2 would be modified to specify both of these functions for the applicable valves. Subsequently, Revision 2 (and Revision 3) to DCD Tier 1, Table 2.4.5-2 indicates that valves RHS-VLV-022A to D have transfer open and transfer closed functions. The staff finds that the DCD has clarified the safety functions for valves RHS-VLV-022A to D. Therefore, the response is acceptable. Accordingly, **RAI 288-2274, Question 03.09.06-35, is resolved**.

DCD Tier 1, Table 2.7.3.5-2, "Essential Chilled Water System Equipment Characteristics," specifies the safety mission of valves VWS-TCV-2845, 2855, 2865, 2875, 2784, 2794, 2804, 2814, 2574, 2584, 2594, 2604, 2671, 2676, 2681, 2868, 2721A to D, 2726A to D, 2731A to D, 2741A to D, 2331, 2336, 2341, and 2346 in the essential chilled water system. In **RAI 288-2274, Question 03.09.06-38**, the staff requested that the applicant clarify the safety function for these valves identified in DCD Tier 1, Table 2.7.3.5-2 and DCD Tier 2, Table 3.9-14. In its response to **RAI 288-2274, Question 03.09.06-38**, dated May 25, 2009, the applicant confirmed the safety function of these valves is to transfer open to provide full flow of chilled water upon receipt of a high-temperature alarm as indicated in DCD Tier 1, Table 2.7.3.5-2. The applicant stated that DCD Tier 2, Table 3.9-14 would be revised to correct the safety mission for these valves. Subsequently, Revision 2 to DCD Tier 2, Table 3.9-14 specified the safety function of valves VWS-TCV-2845, 2855, 2865, 2875, 2784, 2794, 2804, 2814, 2574, 2584, 2594, 2604, 2671, 2676, 2681, 2868, 2721A to D, 2726A to D, 2731A to D, 2741A to D, 2331, 2336, 2341, and 2346 to be transfer open. Revision 3 to DCD Tier 2, Table 3.9-14 modified these valves numbers to be VWS-TMV-141, 151, 161, 171, 206, 226, 246, 266, 304, 314, 324, 334, 402, 412, 422, 432, 502, 512, 522, 532, 542, 552, 562, 572, 582, 592, 602A and B, 612A and B, 622, 632, 642, 652, and 662A and B, respectively. The staff finds the modification to the DCD to be acceptable. Therefore, the response is acceptable. Accordingly, **RAI 288-2274, Question 03.09.06-38, is resolved**.

DCD Tier 1, Section 2.4.6, "Chemical and Volume Control System," describes the CVCS for the US-APWR design. In **RAI 288-2274, Question 03.09.06-42**, the staff requested that the applicant clarify the safety functions for CVCS valves indicated in DCD Tier 1, Table 2.4.6-2, "Chemical and Volume Control System Equipment Characteristics," and DCD Tier 2, Table 3.9-14. In its response to **RAI 288-2274, Question 03.09.06-42**, dated May 25, 2009, the applicant indicated that DCD Tier 2, Table 3.9-14 had been corrected in Revision 1 to the DCD to specify the active functions of the CVCS valves. Subsequently, the staff confirmed that Revisions 2 and 3 to the DCD Tier 2, Table 3.9-14 specify transfer open and close, and maintain open and close as the safety functions for these valves. Therefore, the response is acceptable. Accordingly, **RAI 288-2274, Question 03.09.06-42, is resolved**.

3.9.6.4.2.3 Inservice Testing Program for Dynamic Restraints

DCD Tier 2, Subsection 3.9.3.4.2.9, "Snubber Examination and Testing," and DCD Tier 2, Section 3.9.6.4 provide information on the IST program for dynamic restraints to be used in the US-APWR. COL applicants are responsible for providing a full description of their snubber IST program in their COLA as discussed in SECY-05-0197. In **RAI 288-2274, Question 03.09.06-43**, the staff requested that the applicant provide additional information regarding whether the DCD would provide a full description of the IST program for dynamic restraints, or would specify that the COL applicant will need to supplement the DCD to provide a full description of the IST program for dynamic restraints as part of the COLA. In its response to **RAI 288-2274, Question 03.09.06-43**, dated May 25, 2009, the applicant noted that, as indicated in its letter, "Transmittal of COL Information Update for US-APWR Design Control Document Revision 1," dated November 7, 2008, the DCD would be revised to expand the description of the IST program for dynamic restraints in DCD Tier 2, Section 3.9.6.4, including new Subsections 3.9.6.4.1, "Design and Operating Information," through 3.9.6.4.4, "Service Life Monitoring." In addition, the applicant noted that COL Information Item COL 3.9(6) in DCD Tier 2, Section 3.9.9 would be modified to require the COL applicant to provide the IST program plan for dynamic restraints in accordance with Nonmandatory Appendix A, "Preparation of Test Plans," to the ASME OM Code. Subsequently, Revision 2 (and Revision 3) to DCD Tier 2, Section 3.9.6.4 included these modifications.

However, it was not clear whether the applicant was intending to provide a full IST program description for dynamic restraints. Therefore, the staff closed, as unresolved, **RAI 288-2274, Question 03.09.06-43** and in follow-up **RAI 801-5897, Question 03.09.06-66**, the staff requested that the applicant provide additional information regarding compliance of the IST program for dynamic restraints in the US-APWR design with the requirements of the ASME OM Code, Subsection ISTD. For example, the staff requested that the applicant clarify whether the DCD was intended to fully describe the IST program for dynamic restraints as discussed in SECY-05-0197, or that the COL applicant must fully describe the IST program for dynamic restraints in accordance with the requirements in ASME OM Code, Subsection ISTD. The staff also requested that the applicant clarify the reference to Nonmandatory Appendix A to the

ASME OM Code, which only applies to test plans rather than the program description.

In its response to **RAI 801-5897, Question 03.09.06-66**, dated November 2, 2011, as revised in its submittal dated March 8, 2012, the applicant stated that DCD Tier 2, Section 3.9.6.4 would be revised to fully describe the IST program for dynamic restraints used in the US-APWR. As part of this modification, the applicant indicated that the reference to Nonmandatory Appendix A to the ASME OM Code would be removed from the text and COL Information Item 3.9(6). Further, the applicant stated that the COL Information Item 3.9(6) would reference the IST program rather than only a program plan. The staff finds that the planned modifications to the DCD will provide a description of the IST program for dynamic restraints that may be referenced by a COL applicant as part of a full description of the IST program. Therefore, the response to **RAI 801-5897, Question 03.09.06-57,** is acceptable. **RAI 801-5897, Question 03.09.06-66, is being tracked as a Confirmatory Item**.

In **RAI 288-2274, Question 03.09.06-44**, the staff requested that the applicant provide additional information on dynamic restraint testing for specific systems. In its response to **RAI 288-2274, Question 03.09.06-44**, dated May 25, 2009, the applicant referred to its response to **RAI 288-2274, Question 03.09.06-43**. The staff confirmed that this issue was addressed as part of **RAI 288-2274, Question 03.09.06-43**. Therefore, the response is acceptable. Accordingly, **RAI 288-2274, Question 03.09.06-44, is resolved**.

In **RAI 288-2274, Question 03.09.06-45**, the staff requested that the applicant clarify whether the US-APWR design includes any dynamic restraints not categorized as ASME B&PV Code Class 1, 2, or 3 that are safety-related. In its response to **RAI 288-2274, Question 03.09.06-45**, dated May 25, 2009, the applicant stated that the US-APWR design does not use safety-related dynamic restraints that are not categorized as ASME B&PV Code Class 1, 2, or 3. The staff finds this response to clarify the scope of dynamic restraints in the US-APWR design. Therefore, the response is acceptable. Accordingly, **RAI 288-2274, Question 03.09.06-45, is resolved**.

In **RAI 288-2274, Question 03.09.06-46**, the staff requested that the applicant provide additional information regarding the reference to a component support inspection and testing program plan in Revision 0 to DCD Tier 2, Section 3.9.6.4. In its response to **RAI 288-2274, Question 03.09.06-46**, dated May 25, 2009, the applicant referred to its response to **RAI 288-2274, Question 03.09.06-43** and stated that the DCD would be modified to clarify this reference. The staff confirmed that this issue was addressed as part of **RAI 288-2274, Question 03.09.06-43**. Therefore, the response is acceptable. Accordingly, **RAI 288-2274, Question 03.09.06-46, is resolved**.

3.9.6.4.3 Relief Requests and Alternative Authorizations to ASME OM Code

DCD Tier 2, Section 3.9.6.5 states that experience was used in designing and locating pumps, valves, and dynamic restraints to permit access for performing PST and IST required by the ASME OM Code. In **RAI 288-2274, Question 03.09.06-47**, the staff requested that the applicant clarify the plans regarding relief from the ASME OM Code. In its response to **RAI 288-2274, Question 03.09.06-47**, dated May 25, 2009, the applicant provided a planned modification to Section 3.9.6.5 to indicate that relief from the testing requirements of the ASME OM Code will be requested when full compliance with the requirements of the ASME OM Code is not practical. Subsequently, Revision 2 (and Revision 3) to DCD Tier 2, Section 3.9.6.5 includes this modification. However, DCD Tier 2, Section 3.9.6.5 does not address alternatives to the ASME OM Code. Therefore, the staff closed, as unresolved, **RAI 288-2274, Question**

03.09.06-47 and in follow-up **RAI 801-5897, Question 03.09.06-67**, the staff requested that the design applicant clarify whether any alternatives to the ASME OM Code are planned for the US-APWR IST program.

In its response to **RAI 801-5897, Question 03.09.06-67** dated November 2, 2011, the applicant stated that it had not identified planned alternatives to the ASME OM Code. The applicant indicated that, in the future, if full compliance with the requirements in the ASME OM Code is not practical, it will revise the DCD to include alternatives. The staff notes that in its response to **RAI 801-5897, Question 03.09.06-58**, dated March 8, 2012, the applicant stated that the MOV program description will specify that the MOV program will implement ASME OM Code Case OMN-1 (Rev. 0, 1999) that is accepted for use with conditions in RG 1.192. The staff finds that the planned use of alternatives to the ASME OM Code has been addressed in a sufficient manner for the review of the application. Therefore, the response is acceptable. Accordingly, **RAI 801-5897, Question 03.09.06-67 is resolved**.

3.9.6.4.4 Technical Specifications

The NRC regulations in 10 CFR 50.55a specify that the IST program be included as part of the TS. In **RAI 288-2274, Question 03.09.06-6**, the staff requested that the applicant provide additional information regarding the applicability of TS to the IST program for the US-APWR. In its response to **RAI 288-2274, Question 03.09.06-6**, dated May 25, 2009, the applicant stated that the DCD would be revised to address the relationship of the IST program to the TS. Subsequently, Revision 2 (and Revision 3) to DCD Tier 2, Section 3.9.6.1, states that the requirements of the IST program are included in TS 5.5.8 of Section 5.5, “Programs and Manuals.” The staff finds that the DCD clarifies the relationship of the IST program to the TS. Therefore, the response is acceptable. Accordingly, **RAI 288-2274, Question 03.09.06-6, is resolved**.

3.9.6.4.5 Inspections, Tests, Analyses, and Acceptance Criteria

DCD Tier 1 is intended to include ITAAC for verifying the design-basis capability of specific safety-related components. In **RAI 896-6269, Question 03.09.06-69**, the staff requested that the applicant confirm that the DCD Tier 1 includes ITAAC to verify the functional design and qualification for all safety-related pumps and valves to be capable of performing their intended function for the full range of operating conditions up to design-basis conditions. For such ITAAC, the staff indicated that the Design Commitment should specify that pumps and valves identified in the applicable Tier 1 table will be functionally designed and qualified such that each pump and valve is capable of performing its intended function for a full range of system differential pressure and flow, ambient temperatures, and available voltage (as applicable) under conditions ranging from normal operating to DBA conditions. The Inspections, Tests, and Analyses should specify that tests or type tests of the pumps and valves listed in the applicable Tier 1 table will be conducted to demonstrate that the pumps and valves function under conditions ranging from normal operating conditions to DBA conditions. The Acceptance Criteria should specify that a test report exists and concludes that the pumps and valves listed in the applicable Tier 1 table function under conditions ranging from normal operating conditions to DBA conditions.

In its response to **RAI 896-6269, Question 03.09.06-69**, dated April 3, 2012, the applicant stated that safety-related active mechanical equipment functional design and qualification is performed in accordance with DCD Tier 2 (Section 3.9.6) and the US-APWR equipment qualification program described in DCD Tier 2 (Section 3.11) and Technical Report MUAP-

08015, "US-APWR Equipment Environmental Qualification Program." The applicant asserted that the equipment qualification data summary report documents the qualification data package that assures that safety-related pumps and valves are capable of performing their intended function for the full range of operating conditions up to design-basis conditions. The applicant referenced ITAAC for specific valves in the DCD Tier 1 that are intended to verify the ability of those valves to perform their design-basis safety functions. The applicant provided planned modifications to the DCD Tier 1 to incorporate ITAAC to provide assurance that specific pumps are capable of performing their safety functions under design conditions.

In its amended response to **RAI 896-6269, Question 03.09.06-69**, dated December 28, 2012, the applicant provided a planned revision of the DCD Tier 1 to specify ITAAC and related provisions for the functional design and qualification of safety-related pumps and valves to be used in the US-APWR. The staff finds that the proposed ITAAC are consistent with the guidance provided in RG 1.206 for ITAAC to confirm that pumps and valves are capable of performing their safety functions under expected ranges of fluid flow and other applicable parameters up to and including design-basis conditions. Therefore, the response is acceptable. **RAI 896-6269, Question 03.09.06-69, is being tracked as a Confirmatory Item**.

3.9.6.5 Combined License Information Items

The following is a list of COL item numbers and descriptions from DCD Tier 2, Table 1.8-2 related to the functional design, qualification, and IST programs for pumps, valves, and dynamic restraints.

Table 3.9.6-1
US-APWR Combined License Information Items

Item No.	Description	Section
3.9(1)	The COL Applicant is to assure snubber functionality in harsh service conditions, including snubber materials (e.g., lubricants, hydraulic fluids, seals).	3.9.3.4.2.5
3.9(2)	The first COL Applicant is to complete the vibration assessment program, including the vibration test results, consistent with guidance of RG 1.20. Subsequent COL Applicants need only provide information in accordance with the applicable portion of position C.3 of RG 1.20 for Non-Prototype internals.	3.9.2.4.1
3.9(6)	The COL Applicant is to provide the program plan for IST of dynamic restraints in accordance with Nonmandatory Appendix A of ASME OM Code.	3.9.6.4
3.9(8)	The COL Applicant is to administratively control the edition and addenda to be used for the IST program plan, and to provide a full description of their IST program plan for pumps, valves, and dynamic restraints.	3.9.6
3.9(10)	The COL Applicant is to identify the site-specific active pumps.	3.9.3.3.1
3.9(11)	The COL Applicant is to provide site-specific, safety-related pump IST parameters and frequency.	3.9.6.2
3.9(12)	The COL Applicant is to provide type of testing and frequency of site-specific valves subject to IST in accordance with the ASME Code.	3.9.6.3

The staff reviewed these COL Information Items applicable to the functional design, qualification, and IST programs for pumps, valves, and dynamic restraints listed in DCD Tier 2, Table 1.8-2 and Section 3.9.9.

Regarding COL Information Item 3.9(6), in its responses to **RAI 801-5897, Questions 03.09.06-66 and 03.09.06-68,** dated November 2, 2011, as revised in its submittal dated March 8, 2012, the applicant stated that COL Information Item 3.9(6) would be revised to state that the COL applicant is to provide the program for IST of dynamic restraints in accordance with the ASME OM Code. **RAI 801-5897, Question 03.09.06-66** is addressed in Section 3.9.6.4.2.3 of this report. **RAI 801-5897, Question 03.09.06-66** is addressed below. **RAI 801-5897, Questions 03.09.06-66 and 03.09.06-68, are being tracked as Confirmatory Items**.

Regarding COL Information Item 3.9(8), in its response to **RAI 801-5897, Questions 03.09.06-53, 55, and 68** dated November 2, 2011, as revised in its submittal dated March 8, 2012, the applicant stated that COL Information Item 3.9(8) would be revised to state that the COL applicant is to administratively control the edition and addenda to be used for the IST program and to provide a full description of their IST program for pumps, valves, and dynamic restraints. **RAI 801-5897, Question 03.09.06-53** is addressed in Section 3.9.6.4.2.1 of this report. **RAI 801-5897, Question 03.09.06-55** is addressed in Section 3.9.6.4.2.2 of this report. **RAI 801-5897, Question 03.09.06-68** is addressed below. **RAI 801-5897, Questions 03.09.06-53, 55, and 68, are being tracked as Confirmatory Items**.

In **RAI 288-2274, Question 03.09.06-48**, the staff requested that the applicant revise the DCD to specify that the COL applicant must provide a full description of the IST operational program for pumps, valves, and dynamic restraints, and MOV testing operational program. In its response to **RAI 288-2274, Question 03.09.06-6**, dated May 25, 2009, the applicant stated that its response to **RAI 288-2274, Question 03.09.06-7** clarified that the COL applicant must provide a full description of the IST program for pumps, valves, and dynamic restraints. However, the applicant did not clarify whether it was intending to provide a full IST program description for pumps, valves, and dynamic restraints. Therefore, the staff closed, as unresolved, **RAI 288-2274, Question 03.09.06-48** and in follow-up **RAI 801-5897, Question 03.09.06-68**, the staff requested that the applicant clarify whether it intends that the DCD provide a full description of the IST program for pumps, valves, and dynamic restraints. The staff also requested that the applicant clarify the intent of COL Information Items COL 3.9(6) and COL 3.9(8). In its response to **RAI 801-5897, Question 03.09.06-68**, dated November 2, 2011, as revised in its submittal dated March 8, 2012, the applicant stated that COL Information Items COL 3.9(6) and COL 3.9(8) would be revised to remove the reference to Nonmandatory Appendix A to the ASME OM Code and to refer to the IST program rather than a program plan. The staff finds that the planned modification to the DCD will provide acceptable COL Information Items specifying that the COL applicant is responsible for providing a full description of the IST program for pumps, valves, and dynamic restraints. As discussed in this SE section, the staff notes that a COL applicant may reference the provisions in the DCD as part of its full description of the IST program for pumps, valves, and dynamic restraints. Therefore, the response to **RAI 801-5897, Question 03.09.06-68** is acceptable. **RAI 801-5897, Question 03.09.06-68, is being tracked as a Confirmatory Item**.

3.9.6.6 Conclusions

The staff has reviewed the US-APWR DC application for compliance with NRC regulations and the applicable edition and addenda of the ASME Code for the functional design, qualification, and IST programs for pumps, valves, and dynamic restraints to be used in US-APWR nuclear power plants. The staff also evaluated the DC application for the consideration of lessons learned based on operating experience from plant components at current nuclear power plants and the results of NRC and industry research programs. Based on its review, the staff concludes that the description of the functional design, qualification, and IST programs for pumps, valves, and dynamic restraints to be used in the US-APWR satisfies the NRC regulations and ASME Code, and addresses operating experience and research results in an adequate manner for a DC application. However, the staff cannot conclude that the DCD is acceptable for reference by a COL applicant as part of providing a full description of the functional design, qualification, and IST programs for pumps, valves, and dynamic restraints for the DC because of the open and confirmatory items identified in this SE.

The remaining open and confirmatory items for the staff review of the functional design, qualification, and IST programs for pumps, valves, and dynamic restraints for the DC application are as follows:

Open Item:

RAI 288-2274, Question 03.09.06-1 (Audit of procurement specifications).

Confirmatory Items:

RAI 801-5897, Question 03.09.06-49 (Use of ASME QME-1-2007).
RAI 801-5897, Question 03.09.06-50 (Specify ASME OM Code edition and addenda).
RAI 801-5897, Question 03.09.06-51 (Clarification of general IST provisions).
RAI 801-5897, Question 03.09.06-52 (Clarification of functional qualification discussion).
RAI 801-5897, Question 03.09.06-53 (Description of pump IST program).
RAI 801-5897, Question 03.09.06-55 (Description of valve IST program).
RAI 801-5897, Question 03.09.06-57 (IST Table 3.9-14 corrections).
RAI 801-5897, Question 03.09.06-58 (Description of MOV IST program).
RAI 801-5897, Question 03.09.06-59 (Description of POV IST program).
RAI 801-5897, Question 03.09.06-61 (Clarify Table 6.2.4-3 operating time units).
RAI 801-5897, Question 03.09.06-64 (Correction of Table 3.9-14 for CCW valves).
RAI 801-5897, Question 03.09.06-66 (Description of dynamic restraint IST program).
RAI 801-5897, Question 03.09.06-68 (Revision of COL Information Items 3.9(6) and (8)).
RAI 896-6269, Question 03.09.06-69 (Pump and valve functional design and qualification ITAAC).
RAI 242-2153, Question 14.03.03-16 (Manual ESF equipment room drain valves ITAAC).

As part of the review of a COLA for a US-APWR nuclear power plant, the staff will evaluate the full description of the functional design, qualification, and IST programs for pumps, valves, and dynamic restraints provided by the COL applicant through incorporation by reference of the DCD and additional information to supplement the program description provided in the DCD.

3.10 Seismic and Dynamic Qualification of Mechanical and Electrical Equipment

3.10.1 Introduction

The purpose of this section is to review the information provided by the applicant for the US-APWR standard design that is employed to ensure the functionality of mechanical and electrical equipment, including instrumentation and controls, under the full range of normal and accident loadings (including seismically induced loadings). The review addresses mechanical and electrical equipment associated with systems that are essential to emergency reactor shutdown, containment isolation, reactor core cooling, and containment and reactor heat removal or are otherwise essential in preventing significant release of radioactive material to the environment. It also addresses instrumentation that is needed to assess plant and environmental conditions during and after an accident.

3.10.2 Summary of Application

DCD Tier 1: Tier 1 information associated with this section is found in DCD Tier 1, Sections 2.4, “Reactor Systems,” 2.5, “Instrumentation and Controls,” 2.6, “Electrical Systems,” 2.7, “Plant Systems,” and 2.11, “Containment Systems,” which include requirements for seismic qualification of electrical and mechanical equipment. Seismic qualification of equipment is addressed in the system descriptions for seismic Category I equipment, and ITAAC are included to verify that seismic Category I equipment will withstand design-basis loads.

DCD Tier 2: The applicant has provided in DCD Tier 2, Section 3.10, “Seismic and Dynamic Qualification of Mechanical and Electrical Equipment,” a description of its program to ensure the seismic and dynamic qualification of mechanical and electrical equipment, summarized here in part, as follows.

> The applicant presents or references the specific seismic Category I equipment and supports; seismic qualification criteria, standards, and performance requirements; methods and procedures for analysis and testing, including loads and load combinations; and qualification documentation.
>
> The applicant states that the following classes of equipment require seismic qualification:
>
> - Equipment essential to emergency shutdown, containment isolation, core cooling, containment heat removal, or prevention of significant radioactive release to the environment.
> - Accident monitoring instrumentation as described in RG 1.97, “Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants,” Revision 4, issued June 2006.
>
> The applicant stated that the foregoing equipment classes include manually and automatically actuated equipment as well as equipment whose failure could prevent accomplishing the required safety functions. This includes Class 1E equipment as well as mechanical components (excluding piping), such as pumps, valves, fans, dampers, and valve operators, and snubbers.
>
> The applicant states that seismic qualification of mechanical and electrical equipment will meet IEEE Standard 344-2004 under the conditions of acceptance of this standard as described in RG 1.100, “Seismic Qualification of Electric and Mechanical Equipment for Nuclear Power Plants,” Revision 3, issued September 2009. Qualification will be done by either analysis or testing, or a combination of analysis and testing. The applicant states that analysis without testing is acceptable only if the design function can be achieved by structural integrity alone. Experience based qualification is not used.

The applicant states that their operability program for active valves follows the guidance of ASME QME-1-2007, and is described in DCD Tier 2, Section 3.9.3, “ASME Code Class 1, 2, and 3 Components, Component Supports, and Core Support Structures,” and Section 3.9.6, “Functional Design, Qualification, and Inservice Testing Programs for Pumps, Valves, and Dynamic Restraints.”

Specific seismic Category I SSCs are presented in DCD Tier 2, Table 3.2-2, “Classification of Mechanical and Fluid Systems, Components, and Equipment,” and DCD Tier 2, Appendix 3D “Equipment Qualification List Safety and Important to Safety Electrical and Mechanical Equipment.” For seismic Category I active mechanical components, the performance requirements are defined in the associated equipment specifications, together with the functional requirements described in DCD Tier 2, Sections 3.2, “Classification of Structures, Systems, and Components,” 3.9, “Mechanical Systems and Components,” and various DCD sections describing associated systems.

Safety-related components are qualified to withstand seismic loadings in combination with concurrent dynamic loading effects under the SSE conditions as defined in DCD Tier 2, Section 3.7.1, “Seismic Design Parameters,” for the standard plant design or as defined by the COL applicant for the site-specific design. Deformation of supports and structures is proposed as acceptable at SSE levels, provided the functioning of safety-related equipment is not compromised.

DCD Tier 2, Section 1.5.2.3, “Gas Turbine Generator,” describes the technical information on verification and confirmation tests for the class 1E GTG (class 1E GTG), which is used as emergency power source for the US-APWR. The applicant states that a class 1E GTG has not been used in conventional PWR plants for emergency power source. The applicant has developed a class 1E GTG qualification program, which includes the evaluation of seismic capability. The class 1E GTG qualification program is described in MUAP-07024, “Qualification and Test Plan of Class 1E Gas Turbine Generator System,” Revision 2, issued October 2010. The results of the class 1E GTG qualification program are provided in MUAP-10023, “Initial Type Test Results of Class 1E Gas Turbine Generator System,” Revision 3, issued September 2011.

ITAAC: ITAAC for this area of review are found in DCD Tier 1, Sections 2.4, “Reactor Systems,” 2.5, “Instrumentation and Controls,” 2.6, “Electrical Systems,” 2.7, “Plant Systems,” and 2.11, “Containment Systems,”.

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

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

Technical Reports: The technical report associated with DCD Tier 2, Section 3.10 is:

1. MUAP-07024-P, "Qualification and Test Plan of Class 1E Gas Turbine Generator (GTG) System," Revision 0, issued December 2007.

2. MUAP-08015, “US-APWR Equipment Environmental Qualification Program,” Revision 0, issued February 2009.

3. MUAP-08015, “US-APWR Equipment Qualification Program,” Revision 1, issued November 2009.

4. MUAP-10023, "Initial Type Test Result of Class 1E Gas Turbine Generator System," Revision 3, issued September 2011.

Cross-cutting Requirements (TMI, USI/ GSI, Op Ex): There are no cross-cutting requirements for this area of review.

US-APWR Interface Issues Identified in the DCD: See DCD Tier 2, Table 1.8-2, “Compilation of All Combined License Applicant Items for Chapters 1-19.”

Site Interface Issues Identified in the DCD: See DCD Tier 2, Table 1.8-1, “Significant Site Specific Interfaces with the Standard US-APWR Design.”

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

3.10.3 Regulatory Basis

The relevant requirements of the Commission regulations for this area of review, and the associated acceptance criteria, are given in Section 3.10, “Seismic and Dynamic Qualification of Mechanical and Electrical Equipment,” Revision 3, issued March 2007, of NUREG-0800 and are summarized below. Further details and review interfaces with other SRP sections also can be found in Section 3.10 of NUREG-0800.

1. GDC 1 and 30, as they relate to qualifying equipment to appropriate quality standards commensurate with the importance of the safety functions to be performed.

2. GDC 2 and Appendix S to 10 CFR Part 50, as they relate to designing equipment to withstand the effects of natural phenomena such as earthquakes.

3. GDC 4, as it relates to qualifying equipment as capable of withstanding the dynamic effects associated with external missiles and internally-generated missiles, pipe whip, and jet impingement forces.

4. GDC 14, as it relates to qualifying equipment associated with the reactor coolant boundary so that there is an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture.

5. 10 CFR Part 50, Appendix B, as it relates to qualifying equipment using the QA criteria provided.

6. 10 CFR Part 50, Appendix B, Criterion III, as it relates to verifying and checking the adequacy of design, such as by the performance of a suitable test program, among other things, and which specifically requires that a test program used to verify the adequacy of a specific design feature shall include suitable qualifications testing of a prototype unit under the most adverse design conditions.

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 has been constructed and will operate in accordance with the DC, the provisions of the Atomic Energy Act of 1954, and the NRC regulations.

Acceptance criteria adequate to meet the above requirements include:

1. Acceptance criteria described in SRP Section 3.10.

2. RG 1.61, "Damping Values for Seismic Design of Nuclear Power Plants," Revision 1, issued March 2007, as it relates to the selection of damping values for equipment to be qualified.

3. RG 1.92, "Combining Modal Responses and Spatial Components in Seismic Response Analysis," Revision 2, issued July 2006, as it relates to the use of multimodal and multidirectional responses for qualification.

4. RG 1.97, "Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants," Revision 4, issued June 2006, as it relates to accident monitoring equipment.

5. RG 1.100, "Seismic Qualification of Electric and Mechanical Equipment for Nuclear Power Plants," Revision 3, issued September 2009, as it relates to the seismic and dynamic qualification of equipment, which includes the endorsement of IEEE Standard 344-2004, with conditions.

6. IEEE Standard 344-2004, "IEEE Recommended Practice for Seismic Qualification of Class 1E Equipment for Nuclear Generating Stations," and ASME QME-1-2007, "Qualification of Active Mechanical Equipment Used in Nuclear Power Plants," as they relate to the seismic qualification of equipment used in nuclear power plants.

7. ANSI B 16.41, ANSI N 41.6, ANSI/ASME N551.1, ANSI/ASME N551.2, ANSI N45 N551.4, as they relate to the qualification of equipment used in nuclear power plants.

3.10.4 Technical Evaluation

3.10.4.1 Review of Tier 1 Information and ITAAC

In accordance with SRP Section 3.10, the staff reviewed the applicant's DCD Tier 1 information concerning proposed ITAAC that are necessary in providing reasonable assurance that a plant incorporating the US-APWR DC has been constructed and will operate in accordance with the DC and NRC regulations. Relevant to seismic qualification, the staff's evaluation regarding ITAAC focused on the seismic performance, as qualified, and preservation of the safety functions of electrical, instrumentation and control (I&C), and mechanical equipment under seismic loads and is documented in Section 14.3.3 of this SE. Accordingly, the staff finds the

US-APWR meets the requirements of 10 CFR 52.47(b)(1) regarding ITAAC related to seismic qualification.

3.10.4.2 Seismic and Dynamic Qualification Criteria for Mechanical and Electrical Equipment

The staff reviewed DCD Tier 2, Section 3.10, in accordance with SRP Section 3.10 and RG 1.206, “Combined License Applications for Nuclear Power Plants (LWR Edition),” issued June 2007. The functional performance of the seismic Category I mechanical and electrical equipment was reviewed to confirm a safe reactor shutdown, and to ensure conformance with the requirements of GDC 1, 2, 4, 14, and 30. DCD Tier 2, Section 3.10.1.1, “Qualification Standards,” states that the US-APWR mechanical and electrical equipment seismic qualification will meet ASME QME-1-2007 and IEEE Standard 344-2004, as endorsed by RG 1.100, Revision 3, for qualification by either analysis, testing, or a combination of testing and analysis.

DCD Tier 2, Section 3.10.1, “Seismic Qualification Criteria,” states that in accordance with Appendix S to 10 CFR Part 50, the OBE for the standard plant is set at 1/3 or less of the SSE, eliminating the OBE from the design of SSCs for the standard plant. The DCD Tier 2, Section 3.10.1 also states that, per COL Information Item 3.10(8), for the design of seismic Category I and II components that are not part of the standard plant, the COL applicant can similarly eliminate the OBE, or optionally set the OBE higher than 1/3 SSE, provided the design of the non-standard plant components are analyzed for the chosen OBE. However, for seismic qualification of all US-APWR safety-related mechanical and electrical equipment, with the elimination of OBE, the evaluation for fatigue effects for a smaller earthquake is performed at an equivalent fraction of the SSE as identified in SECY-93-087, “Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs,” issued April 2, 1993, and the associated SRM, issued July 21, 1983.

DCD Tier 2, Chapter 3, “Design Of Structures, Systems, Components, and Equipment,” describes the function of seismic Category I components and specify requirements pertaining to their materials, design, inspection, and testing prior to and during service. The loading combinations and corresponding stress limits for ASME Code design are defined for the Design Condition, ASME Code, Section III Service Levels A, B, C and D (also known as normal, upset, emergency, and faulted conditions), and test conditions. The staff’s evaluation of loading combinations for mechanical components is provided in Section 3.9.3 of this report.

DCD Tier 2, Section 3.10.2.1.2, “Test and Analysis,” states that the US-APWR utilizes a combination of test and analysis to qualify seismic Category I instrumentation and electrical equipment. The test methods utilized in DCD Tier 2, Section 3.10 are similar to those described for type testing along with static and/or dynamic analysis. These methods establish input response requirements at sub-component locations. This approach can justify the extrapolation of tests on a single electrical cabinet, or a small number of connected cabinets, to qualify an assembly. Analysis can be used to: explain unexpected behavior during a test; obtain a better understanding of the dynamic behavior of the equipment so that the proper test can be defined; or obtain a measure of expected response before a test. DCD Tier 2, Section 3.10.2, “Methods and Procedures for Qualifying Mechanical and Electrical Equipment and Instrumentation,” states that the testing is performed in the proper sequence as indicated in IEEE Standard 323-1974, “IEEE Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations,” and the testing identifies and accounts for significant aging mechanisms. The equipment is demonstrated to be capable of performing its safety-related function throughout its qualified life including its functional operability during and after a SSE at the end of that qualified life. The equipment to be tested is mounted in a manner that simulates the intended service mounting, and the fixture design is such that it does not cause any extraneous dynamic coupling to the test component. The dynamic coupling effect of electrical connections, conduit, sensing lines, and any other interfaces are considered and included in the test unless otherwise justified. The method chosen for testing depends upon the nature of the expected vibration environment and also on the nature of the equipment.

Section C.I.3.10.4 of RG 1.206 states that if the seismic and dynamic qualification testing is incomplete at the time of application, the applicant should include an implementation program, including milestones and completion dates with appropriate information submitted for staff review and approval prior to installation of equipment. Therefore, in **RAI 216-1749, Question 03.10-1**, the staff requested the applicant to provide information as described above for the staff's review and approval.

In its response to **RAI 261-1749, Question 03.10-1**, dated March 25, 2009, the applicant stated that MHI recognizes the seismic and dynamic qualification testing will be incomplete at the time of application. Therefore, an implementation program, including milestones and completion dates with appropriate information, will be submitted for staff review and approval prior to installation of equipment as part of the US-APWR Equipment Environmental Qualification (EQ) Program. COL Information Item 3.10(1) indicates that it is the COL Applicant's responsibility to implement the US-APWR Equipment EQ Program and to provide milestones and completion dates for its implementation. The applicant further explained the implementation of the US-APWR Equipment EQ Program.

The applicant stated that procurement activities and the associated seismic and dynamic qualification testing for both US-APWR standard plant and project-specific equipment were not complete at the time of the DC application. The applicant's technical report MUAP-08015, "US-APWR Equipment Environmental Qualification Program," Revision 0, February 2009, established and defined the generic US-APWR Equipment EQ Program. The MUAP-08015, Revision 0 supplements the DCD and includes testing requirements and criteria for qualification of standard plant and site-specific electrical and instrumentation equipment, mechanical equipment, and inline fluid system components. Implementation of the EQ Program is dependent upon the unique schedule for each US-APWR plant site, including procurement activities and associated qualification testing. Although the COL applicant is not responsible for the design of the US-APWR standard plant equipment, the COL applicant is responsible for project-specific implementation of the US-APWR Equipment EQ Program. This includes but is not limited to procurement and associated testing of both standard and project specific equipment, and associated documentation. Implementation also includes development and maintenance of equipment qualification files. Therefore, it is the COL Applicant's responsibility to ensure that equipment is qualified as applicable for seismic and dynamic loadings according to the criteria and requirements of the equipment qualification program described within the US-APWR Equipment EQ Program, and to provide implementation milestones and completion dates with appropriate information for staff's review and approval. The staff considers the applicant's response to be acceptable since it explained the implementation of the US-APWR Equipment EQ Program. Accordingly, **RAI 216-1749 Question 03.10-1, is resolved**.

DCD Tier 2, Section 3.10.2, states that per COL Information Item 3.10(5), for components that have been previously tested to IEEE Standard 344-1971, the COL applicant is to re-qualify the components using biaxial test input motion unless the applicant provides justification for using a single-axis input motion. This process may not be adequate because the adequacy of seismic qualification of US-APWR equipment within the DCD scope should be addressed and approved within the DCD review and the COL Information Item 3.10(5) may not be acceptable. Therefore, in **RAI 216-1749, Question 03.10-8**, the staff requested the applicant to clarify this issue.

In its response to **RAI 261-1749, Question 03.10-8**, dated March 25, 2009, the applicant stated that the DCD will be revised to state that guidelines for qualifying components with respect to single-axis and biaxial test input motion are included in the procedures of the US-APWR Equipment EQ program. The applicant further explained the implementation of the US-APWR Equipment EQ Program.

The applicant stated that procurement activities and the associated seismic and dynamic qualification testing for both US-APWR standard plant and project-specific equipment are not complete at the time of the DC application. MUAP-08015, Revision 0 established and defined the generic US-APWR Equipment EQ Program. The US-APWR Equipment EQ Program supplements the DCD and includes testing requirements and criteria for qualification of standard plant and site-specific electrical and instrumentation equipment, mechanical equipment, and inline fluid system components. The US-APWR Equipment EQ Program criteria define specific conditions for when single-axis input motion can be justified and also define requirements for biaxial test input motion. Since procurement of equipment does not occur under the scope of the DC application, it is therefore the COL applicant's responsibility to ensure that any equipment previously tested to IEEE Standard 344-1971 is re-qualified as necessary according to the criteria in the US-APWR Equipment EQ Program. The staff considers the applicant’s response to be acceptable because this issue will be resolved in accordance with US-APWR Equipment EQ Program. Accordingly, **RAI 216-1749, Question 03.10-8, is resolved**.

In MUAP-08015, Revision 0, Attachment B, Section B.7, the applicant discussed the consideration of OBE and the application of OBE ISRS for equipment seismic qualification. In **RAI 486-3861, Question 03.10-10**, the staff raised the following two questions:

(a) The first paragraph of Section B.7, Attachment B of MUAP-0815, indicated that the OBE will be set on a site-specific basis by the COL applicant for each individual US-APWR project and it must be enveloped by 1/3 of the standard plant CSDRS. Given that CSDRS-based OBE ISRS are developed based on CSDRS-based SSE ISRS, the applicant is requested to clarify the last sentence of the fourth paragraph, “When considering OBE ISRS for purpose of fatigue during seismic qualification testing of US-APWR standard plant equipment, it is acceptable to obtain OBE spectra by scaling directly from the site-specific SSE ISRS,” which is contrary to the above statement.

(b) According to SRP Section 3.10, item (i) of SRP Acceptance Criteria (1)(A), the applicant should demonstrate that the equipment can withstand the equivalent effect of five OBE excitation. In the fifth paragraph of Section B.7, the applicant indicated that it is acceptable to consider fewer than five site-specific OBE events when technical justification is provided. The staff understands that compliance to SRP is not required; however, the applicant is required to evaluate how the proposed alternatives to the SRP acceptance criteria provide acceptable methods of compliance with the NRC regulation. Thus, the applicant is requested to revise the statement indicating that technical justification will be provided and subjected to staff review and approval.

In its response to **RAI 486-3861, Question 03.10-10**, dated December 9, 2009, the applicant stated regarding Part (a) that the last sentence of the fourth paragraph in Section B.7 of

Attachment B has been clarified in MUAP-08015, “US-APWR Equipment Qualification Program,” Revision 1, issued November 2009, to state it is acceptable to obtain OBE spectra by scaling directly from the standard-plant SSE ISRS. In addition, the second sentence of the fourth paragraph in Section B.7 has been clarified in MUAP-08015, Revision1 by deleting the phrase "(based on the standard plant SSE or site-specific SSE, as applicable)." In its response to Part (b) of the RAI, the applicant modified the last sentence of the fifth paragraph in Section B.7 of MUAP-08015 by adding to the end of the sentence "and subject to NRC staff review and approval."

Based on the above responses, the staff considers that the response to Part (b) to be acceptable because site-specific OBE fewer than five will be justified and approved by the NRC. However, the response to Part (a) is not acceptable and the issue is unresolved because of the inconsistency in the determination of OBE. Therefore, to ensure the standard plant equipment is qualified for the site, **RAI 486-3861, Question 03.10-10** was closed, as unresolved, and in **RAI 1019-7043, Question 03.10-20**, the staff requested the applicant to revise MUAP-08015 Report on determination of OBE by adding that "If the site-specific SSE ISRS exceeds the standard plant SSE ISRS, the OBE should be scaled from the site-specific SSE ISRS." The staff also requested the applicant to identify and clarify any similar statements regarding the seismic qualification of standard plant equipment in the DCD and MUAP-08015.

In its response to **RAI 1019-7043, Question 03.10-20**, dated June 18, 2013, the applicant deleted the last sentence of the fourth paragraph in Section B.7 of MUAP-08015, and stated that the OBE will be set on a site-specific basis by the COL applicant for each individual US-APWR project and it must be enveloped by 1/3 of the standard plant CSDRS as stated in MUAP-08015. This is acceptable to the staff because the OBE ISRS for a site-specific plant should be scaled from the site-specific SSE ISRS. Therefore, **RAI 1019-7043, Question 03.10-20, is resolved.**

In MUAP-08015, Attachment B, Section B.14, the applicant addressed high-frequency exceedance of earthquake ground motion. It stated that additional equipment evaluation by screening and subsequent qualification testing must be performed when exceedances occur in the 20-50 Hz frequency range. In **RAI 486-3861, Question 03.10-11**, the staff requested the applicant to explain why concern about the exceedance is limited to only in the "20-50 Hz range" instead of the entire frequency range of the GMRS-based ISRS.

In its response to **RAI 486-3861, Question 03.10-11**, dated December 9, 2009, the applicant stated that the range 20-50 Hz was used for reference since this is generally the range of concern when high frequency exceedances occur. However, as required by the US-APWR Equipment EQ Program procedures, if equipment cannot be "screened out" with respect to the potential for high frequency sensitivity, then additional high-frequency qualification testing is required in order to cover the high frequency range, which includes the 20-50 Hz range and beyond. The applicant stated that the second sentence of the fourth paragraph in Section B.14 of Attachment B has been clarified in of MUAP-08015 Revision 1 to state that as per the guidance of RG 1.100 and NRC interim staff guidance, such evaluations must be performed when exceedances occur at 20 Hz and above. The applicant also revised statements in DCD Tier 2, Section 3.10 that mention the range of high frequency exceedances to provide consistency with MUAP-08015, Revision 1. The staff considers the applicant’s response to be acceptable since the applicant clarified the treatment of high frequency exceedances. The staff confirmed that the changes were incorporated into DCD Revision 3, and MUAP-08015, Revision 1. Accordingly, **RAI 216-1749, Question 03.10-11, is resolved**.

DCD Tier 2, Section 3.11, “Environmental Qualification of Mechanical and Electrical Equipment,” and MUAP-08015 Revision 0 indicated that some equipment important-to-safety that is required to be environmentally qualified may be purchased as commercial-grade items and its qualification would become part of the commercial-grade dedication process. DCD Tier 2, Section 3.10 did not contain a similar statement and it is not clear to the staff how the applicant had planned this process regarding seismic qualification. In **RAI 486-3861, Question 03.10-12**, the staff requested the applicant to provide additional information to clarify its position on procurement of all seismic Category I equipment as basic components as defined in and in accordance with 10 CFR Part 21 or if procurement of such equipment as commercial-grade items and qualifying it as part of the dedication process is envisioned. If such a process is envisioned, the applicant is requested to provide information, specifically in DCD Tier 2, Section 3.10 to describe the process it will employ to accomplish seismic qualification through commercial-grade dedication.

In its response to **RAI 486-3861, Question 03.10-12**, dated December 9, 2009, the applicant stated that it is its intent to procure and utilize all seismic Category I and important to safety equipment as basic components as defined in and in accordance with 10 CFR Part 21, with accompanying seismic qualifications meeting the requirements of DCD Tier 2, Section 3.10. The applicable seismic qualification standards are identified in DCD Tier 2, Section 3.10.1.1 and the applicable methods of seismic qualification (analysis, test or by a combination of test and analysis) are identified in DCD Tier 2 Section 3.10.2. If procurement of a seismic Category I or important to safety equipment as a basic component is not possible or feasible (such as potential manufacturers/vendors are no longer available, or potential manufacturers/vendors cannot provide an acceptable QAP meeting the requirements of 10 CFR Part 50 Appendix B, etc.), the equipment will be procured as a commercial-grade item and a commercial-grade dedication process will be used to seismically qualify this equipment for use on the US-APWR. The applicant modified DCD Tier 2, Section 3.10.2 to clarify this intention. The staff confirmed that the changes were incorporated into DCD Revision 3. The staff considers the response to be acceptable since the applicant clarified its commercial-grade dedication process. Accordingly, **RAI 216-1749, Question 03.10-12, is resolved**.

3.10.4.3 Seismic and Dynamic Qualification of Safety-Related Active Mechanical Equipment

DCD Tier 2, Section 3.10.1.1 states that the qualification of the design of US-APWR safety-related, seismic Category I mechanical equipment to assure structural integrity of pressure boundary components follows the guidance provided in the ASME B&PV Code, Section III. DCD Tier 2, Section 3.10.1 states that the SSE term is applicable to either the site-independent earthquake or the SSE as defined in DCD Tier 2 Section 3.7.1. Therefore, the expression “SSE” as used for seismic qualification of SSCs refers to equipment qualified for either the standard plant design or the site-specific design.

DCD Tier 2, Section 3.10.2.2, “Seismic and Operability Qualification of Active Mechanical Equipment,” states that seismic Category I active mechanical equipment is designed to withstand seismic and dynamic loads, including the intended service load conditions identified in the equipment’s design specification, in accordance with the requirements in ASME Code, Section III as described in DCD Tier 2, Section 3.9. Such service loads include: normal, upset, emergency, faulted, testing, and other conditions. Other loads include, as applicable, internal pressure, operator thrust, dynamic transients, flow induced vibration, degraded flow conditions, reciprocating and rotating equipment vibrations, and nozzle loads. Load combinations are described in DCD Tier 2, Section 3.9.3 and documented in the equipment qualification file including equipment qualification summary data sheet (EQSDS).

DCD Tier 2, Section 3.10.2 uses the recommended guidelines given in IEEE Standard 344-2004 and RG 1.100, Revision 3, for the development and implementation of methods and procedures for seismic qualification of mechanical and electrical equipment. The methods and guidance in ASME QME-1-2007, "Qualification of Active Mechanical Equipment Used in Nuclear Power Plants," including Appendix QR-A with exceptions provided in RG 1.100, Revision 3 are also used. DCD Tier 2, Section 3.10.4, "Test and Analyses Results and Experience Database," addresses the requirements of GDC 1 and 10 CFR Part 50, Appendix B, Criteria XVII to establish records concerning the qualification of equipment and maintaining their qualification files. These files describe the qualification method used for equipment and the tests and analyses results in sufficient detail to document the degree of compliance with the equipment seismic qualification requirements. DCD Tier 2, Section 3.10.2.2 states that DCD Tier 2, Section 3.9 includes dynamic testing and analysis of mechanical systems, components and equipment, seismic analysis, and qualification of safety-related mechanical equipment, load combinations, and IST programs. The applicant also states that if the dynamic testing of a pump or valve assembly proves to be impractical, static testing of the assembly is acceptable provided that the end loadings are conservatively applied and are equal to or greater than the postulated event loads, dynamic amplification effects are accounted for, the component is in the operating mode during and after the application of loads, and an adequate analysis is made to show the validity of the static application of loads. DCD Tier 2, Section 3.10.2.2 states that the methods and procedures used for qualifying US-APWR active mechanical equipment (i.e., valves, pumps, and dampers) are described in DCD Tier 2, Section 3.9, and Section 3.10.2. Analysis, test, or a combination of test and analysis are used for qualification of seismic Category I active mechanical equipment to show it maintains structural integrity (including pressure retention), and operability. The methods used assure equipment functionality and operability for its intended safety-related function under required plant conditions.

Based on the staff's review of the applicant's approach for qualifying US-APWR equipment, the staff finds that the criteria, methods, and procedures used for qualifying US-APWR active mechanical equipment are adequate and acceptable.

3.10.4.4 Seismic and Dynamic Qualification of Safety-Related Electrical and Instrumentation and Control Equipment

DCD Tier 2, Section 3.10.2 states that the qualification of safety-related seismic Category I electrical equipment by testing is the preferred method for complex equipment which must perform an active function during the SSE. The relays and other electrical and I&C devices whose output signals could be affected by high frequency excitation, are potentially sensitive to high frequency motion and can be damaged by high frequency exceedances of the design spectra. The applicant described methods to identify, evaluate, and qualify or eliminate SSCs that are potentially sensitive to high frequency exceedances.

DCD Tier 2, Section 3.10.2 states that the US-APWR seismic Category I equipment is qualified to show that it can perform its safety-related function during and after a postulated earthquake. The seismic qualification considers interfaces and the effects of the amplification within the equipment due to the interfaces and supporting structure. The potential failure modes of the high frequency-sensitive component types and assemblies are considered in order to demonstrate the suitability of the US-APWR equipment for high-frequency seismic environments. The generic failure modes involving inadvertent change of state, contact chatter, signal change/drift, and connection problems due to high frequency effects are the main focus of the high frequency qualification testing. High frequency failures resulting from improper design of mounting, inadequate design connections and fasteners, mechanical misalignment/binding of parts and the rare case of failure of a component part, will result from the same structural failure modes as those experienced during low frequency content spectra qualification testing in accordance with IEEE Standard 344-2004. Because the safety-related equipment will experience higher stresses and deformations when subjected to the low frequency excitation, these failure modes are more likely to occur under the low frequency testing. Failure modes related to improper mounting, inadequate securing of connections, poor quality joints (cyclic strain effects), etc., are precluded for US-APWR design by QA inspection and process/design controls. Potentially high frequency sensitive components include: electro-mechanical relays; electro-mechanical contactors; circuit breakers; auxiliary contacts; control switches; transfer switches; process switches and sensors; potentiometers; and digital/solid-state devices (mounting and connections only).

In DCD Tier 2, Section 3.10.2, the applicant discussed briefly the equipment seismic issues related to hard rock high frequency seismic excitation. The acceptable methods used by the applicant for resolving such high frequency concerns, where site-specific in-structure response spectra generated for the COLA results in high frequency exceedances of the standard design in-structure response spectra, include: 1) review existing equipment qualification test data for adequate high frequency input motion; 2) review circuits containing potentially sensitive items for inappropriate system actions due to intermediacy or set point drifts; and 3) screening test to confirm equipment does not have high frequency vulnerabilities. The staff noticed that the applicant's criteria and procedures used for addressing the hard rock high frequency issues in the DCD are not consistent with staff guidance (DC/COL-ISG-1, "Interim Staff Guidance on Seismic Issues Associated with the High Frequency Ground Motion in Design Certification and Combined License Applications"). In particular, the use of sine beat at 1/3 octave for screening test, and the statement that "the above testing is not a qualification test" are not acceptable. Per Section 4.0 of DC/COL-ISG-1, sine beat at 1/6 octave should be used for screening test instead of 1/3 octave. For screened-in equipment and/or components (equipment potentially sensitive to high frequency excitation), the test procedure shall be consistent with the guidelines of IEEE Standard 344-2004 as endorsed by RG 1.100, Revision 3. This provides an acceptable means to meet the NRC regulations, Appendix A to 10 CFR Part 100 and Appendix S to 10 CFR Part 50, and the criterion and procedures delineated in SECY-93-087. Therefore, in **RAI 216-1749, Question 03.10-2**, the staff requested the applicant to provide detailed criteria and procedures to address the hard rock high frequency issues in accordance with the staff's guidance DC/COL-ISG-1 for staff review and approval.

In its response to **RAI 261-1749, Question 03.10-2**, dated March 25, 2009, the applicant stated that the US-APWR Equipment EQ Program supplements the DCD. To be consistent with current requirements, including that of the US-APWR Equipment EQ Program, the applicant proposed to revise DCD Tier 2, Section 3.10.2 to change the sine beat octave to 1/6 for the screening tests. The screening tests described in DCD Tier 2, Section 3.10.2 are not intended to be qualification tests but instead are intended to determine if equipment is potentially sensitive to high-frequency excitation. If the screening tests determine that the equipment is potentially sensitive to high-frequency excitation ("screened-in"), then full-scale qualification testing including testing over the range of high-frequency exceedances is required. The statement in the DCD that "the above testing is not a qualification test" was modified to clarify this point. The applicant also stated that detailed criteria and procedures to address the hard rock high frequency issues in accordance with DC/COL-ISG-1 are provided in the US-APWR Equipment EQ Program. The qualification criteria, documented in the procedure, incorporate the guidelines of IEEE Standard 344 and are consistent with the guidance of RG 1.100, and SECY-93-087.

In its response to **RAI 261-1749, Question 03.10-2,** the applicant also indicated that sine sweep, sine beat at 1/6 octave spacing, band-limited white noise, or random multi-frequency time history may be used as a high frequency screening test. The staff had the following concerns:

(i) The applicant did not identify the acceleration level (g) of those tests. The relationship of the acceleration levels between the screening test and the site-specific spectrum is lacking.

(ii) The applicant did not explain clearly the acceptance criteria of the screening tests that "demonstrate lack of sensitivity to high frequency vibrations."

(iii) The applicant did not define the term "full-scale qualification testing" in the ninth paragraph of Subsection 3.10.2. This term should be related to the OBE and SSE of the hard-rock high-frequency site and TRS must envelop the RRS according to SRP Section 3.10 guidance.

(iv) The staff recognized that the value of 50 Hz is used in the ISG mentioned in the response. However, the applicant should include a statement indicating that the 50 Hz upper bound could be adjusted upward if the site-specific required response spectra show frequency exceedance beyond 50 Hz.

Based on the above concerns, the staff closed as unresolved **RAI 216-1749, Question 03.10-2**, and in follow-up **RAI 486-3861, Question 03.10-13**, the staff requested the applicant to address these concerns and to provide detailed criteria and procedure to address the hard rock high frequency issues in accordance with the ISG for staff review and approval.

In its response to **RAI 486-3861, Question 03.10-13**, dated December 25, 2009, the applicant addressed the staff's above questions as follows:

(i) The applicant stated that a 0.2 g peak acceleration will be used for the high frequency screening test of standard plant and site-specific safety-related components. Such a magnitude had conventionally been used for resonant searches for equipment and devices throughout the nuclear industry in the past. Such a magnitude is reasonably chosen when compared to the 0.3 g acceleration at zero period acceleration (ZPA) for the standard plant at the foundation level. It is expected that the use of 0.2 g for the screening test will not damage the equipment or the device. The applicant proposed to revise the eighth paragraph in DCD Tier 2, Section 3.10.2 to clarify this information. The staff finds this to be acceptable since it is consistent with industry practice.

(ii) The applicant stated that structural resonances are normally detected by observing amplifications of the input motion in the test item. Phase relationships between the sinusoidal input signal and the structural response at the point of measurement will also help in defining resonances. Therefore, for each screening test, the transfer function and phase data shall be generated by performing Fast Fourier Transform analysis of excitation and response time histories. These results are to be used to judge whether equipment or devices are sensitive to the high-frequency excitation. The applicant proposed to revise DCD Tier 2, Subsection 3.10.2 to include these requirements. The staff finds this inclusion to be acceptable since the applicant clarified its acceptance criteria for screening tests to determine high frequency sensitivity of the equipment.

(iii) The applicant proposed to revise DCD Tier 2, Section 3.10.2 to include the following two sentences: "In addition, the start of the ZPA range at 50 Hz are adjusted upward if the site-specific required response spectra show frequency exceedance beyond 50 Hz. In performing the equipment seismic qualification via testing, the TRS envelopes the RRS in all frequency ranges." The staff finds this inclusion to be acceptable since it meets the guidelines of IEEE Standard 344.

(iv) The applicant referred to the response to **RAI 486-3861, Question 03.10-15,** which addresses this concern, and notes that, Revision 1 of MUAP-08015 Section B.10 was revised to clarify rigidity definition of 50 Hz. The staff finds the response acceptable since the staff agrees that the response to **RAI 486-3861, Question 3.10-15** addresses the concern**.**

The staff confirmed that the proposed DCD changes were incorporated into DCD Revision 3. Accordingly, **RAI 486-3861, Question 03.10-13, is resolved**.

DCD Tier 2, Section 3.10.2 states that, with the elimination of the OBE from design considerations, two alternatives exist that essentially maintain the guidelines provided in IEEE Standard 344-2004 to qualify equipment: (1) the equivalent of five OBE events followed by one SSE event (with ten maximum stress cycles per event) and (2) five 1/2 SSE events followed by one full SSE event (with ten maximum stress cycles per event). Of these alternatives, the applicant selected the equipment qualification option of five 1/2 SSE events followed by one full SSE event (with ten maximum stress cycles per event). The applicant further stated that, in terms of maximum stress cycles for fatigue analysis, in accordance with SECY-93-087, this is equivalent to any of the following:

- 20 cycles of SSE,
- 50 cycles of 1/2 SSE and 10 cycles of SSE,
- 150 cycles of 1/3 SSE and 10 cycles of SSE,
- 300 cycles of 1/3 SSE,
- 100 cycles of 1/2 SSE.

The applicant’s interpretation of SECY-93-087 related to seismic qualification of US-APWR mechanical and electrical equipment is not entirely correct. The last two bullets are not acceptable because there is no full SSE level qualification involved. Therefore, in **RAI 216-1749 Question 03.10-3**, the staff requested the applicant to correct this interpretation in the DCD.

In its response to **RAI 261-1749, Question 03.10-3**, dated April 22, 2009, the applicant proposed to revise DCD Tier 2, Section 3.10.2 to delete the last two bullets cited above. The staff finds the response acceptable since the applicant eliminated the two inappropriate bullets. The staff confirmed that applicant incorporated the proposed changes to DCD Tier 2, Section 3.10.2 into DCD Revision 3. Accordingly, **RAI 216-1749, Question 03.10-3, is resolved**.

DCD Tier 2, Section 3.10.2 states that the seismic qualification testing inputs and methods for the qualification of mechanical and electrical equipment are in accordance with the guidelines provided in IEEE Standard 344-2004, Section 8. DCD Tier 2, Revision 1, Section 3.10.2, under "Testing," stated that: "The TRS envelopes the RRS except for equipment not sensitive to high frequency motion with exceedances in the 25-50 Hz range." The staff noted that SRP Section 3.10, SRP Acceptance Criteria, 1.A.iv states that for seismic and dynamic loads, the actual test input motion should be characterized in the same manner as the required input motion, and the conservatism in amplitude and frequency content should be demonstrated (i.e., the TRS should closely resemble and envelop the RRS over the critical frequency range). Therefore, in **RAI 216-1749, Question 03.10-4**, the staff requested the applicant to justify that the US-APWR mechanical and electrical equipment seismic qualification (including high frequency (HF) excitations) meets the guidance as described in SRP Section 3.10 SRP Acceptance Criteria and SECY 93-087.

In its response to **RAI 261-1749, Question 03.10-4**, dated March 25, 2009, the applicant stated that, with limited exceptions such as the one discussed in this response, the US-APWR Equipment EQ Program requires that the actual test input motion be characterized in the same manner as the required input motion, and that conservatism in amplitude and frequency content are demonstrated [i.e., the TRS should closely resemble and envelop the RRS over the critical range of frequencies]. The DCD statement quoted in the question above is intended to indicate the exception that it would not be required for the TRS to envelope the RRS in the high frequency range if the "critical frequency range" for the equipment/components does not include frequencies within the high frequency range. The "critical frequency range" of equipment/ components, as referred to in SRP 3.10, is established by testing (either screening tests or qualification tests). This exception applies to equipment/ components for which the measured resonance frequencies are outside the high frequency exceedance range. The test input motion requirements for equipment qualification, including enveloping requirements for the TRS with regard to the RRS, are described in detail in the US-APWR Equipment EQ Program procedures. The procedural requirements are intended to be in accordance with the acceptance criteria in SRP Section 3.10, SECY 93-087, IEEE Standard 344, IEEE Standard 323, and RG 1.100.

The applicant's response did not fully address the staff questions. Therefore, **RAI 216-1749, Question 03.10-4** was closed, as unresolved. In follow-up **RAI 486-3861, Question 03.10-14**. the staff requested the applicant to address the situations when the RRS of the equipment indicated amplified spectral acceleration exceedance over 50 Hz and/or the resonance frequencies of the equipment are higher than 50 Hz.

In its response to **RAI 468-3861, Question 03.10-14**, dated December 25, 2009, the applicant indicated that equipment potentially sensitive to high frequency excitation include the electro-mechanical relays, electro-mechanical contactors, circuit breakers, auxiliary contacts, control switches, transfer switches, process switches and sensors, potentiometers, and digital/solid-state devices (mounting and connections only). For equipment with resonance frequencies greater than 50 Hz and the equipment is not considered sensitive to high frequency excitation, the qualification is not affected because the dynamic responses in the high frequency range would only be reflective of minor modes that would have insignificant contribution to the overall dynamic response. For equipment potentially sensitive to high frequency excitation with resonant frequencies higher than 50 Hz, then additional high frequency screening and possibly subsequent high frequency testing would be needed as required by the US-APWR Equipment EQ Program procedures.

The staff finds the response acceptable since the applicant addressed the issue of equipment potentially sensitive to high frequency excitation. Accordingly, **RAI 486-3861, Question 03.10-14, is resolved**.

In DCD Tier 2, Subsection 3.10.2.1.1, “Type Testing,” the applicant uses IEEE Standard 382-1996, "IEEE Standard for Qualification of Actuators for Power-operated Valve Assemblies with Safety-Related Functions for Nuclear Power Plants,” for qualification of line mounted equipment. It is noted that the testing frequency range used in IEEE Standard 382-1996 may not be adequate for US-APWR equipment. Therefore, in **RAI 216-1749, Question 03.10-5(a)**, the staff requested the applicant to identify all components that are being addressed using IEEE Standard 382-1996, and justify adequacy.

In DCD Tier 2, Section 3.10.2.2, a rigid valve is defined as the valve with natural frequency equaling or exceeding 33 Hz. Figure 6 of IEEE Standard 382-1996 is used as the required input motion (up to 32 Hz) for qualification of a valve. The definition of a rigid valve, the determination of the equivalent static load from the dynamic analysis of the valve, and the use of Figure 6 of the IEEE standard might not be adequate for the hard rock high frequency RRS with exceedance. For hard rock high frequency spectra, the definition of a rigid valve depends on the frequency at the beginning of ZPA of the RRS for the valve. Therefore, in **RAI 216-1749, Question 03.10-5(b)**, the staff requested the applicant to explain why the use of Figure 6 (frequency ends at 32 Hz) of IEEE Standard 382-1996 is still adequate for qualification of valves, and provide methodologies that would be acceptable for the case of hard rock high frequency RRS with exceedance.

In its response to **RAI 261-1749, Question 03.10-5**, dated April 22, 2009, the applicant adequately addressed the concern for Part (a) but not for Part (b). Regarding **RAI 216-1749, Question 03.10-5(a)**, the applicant stated that all actuators for power-operated valves listed in DCD Tier 2, Table 3.2-2 or DCD Tier 2, Appendix 3D of Chapter 3, will be qualified under the US-APWR Equipment EQ Program. The applicant stated that the US-APWR Equipment EQ Program, as documented in MUAP-08015, encompasses IEEE Standards 323, 344, and 382. The US-APWR Equipment EQ Program addresses, in Section B.14 "High-Frequency Exceedances of Earthquake Ground Motion," the case of hard rock high frequency RRS not covered in IEEE Standard 382. The applicant also propose to revise the DCD in accordance with the response to question **RAI 216-1749 Questions 03.10-2,** to include statements that clarify the use of the US-APWR Equipment EQ Program in screening and qualification of SSCs exposed to high-frequency exceedances. The US-APWR Equipment EQ Program procedures also contain criteria documents as listed in MUAP-08015, Section B.8 that address qualification requirements for inline fluid system components, including establishment of the RRS used for qualification. In accordance with those US-APWR Equipment EQ Program procedures and as discussed in DCD Tier 2, Subsection 3.10.2.1.1, for line-mounted equipment that is not qualified to the generic level of 6.0g, the seismic input motion is determined from the response of the system analysis on which it is mounted.

The staff finds the response to be acceptable because the applicant clarified how the use of US-APWR Equipment EQ program will adequately address the issue. Accordingly, **RAI 216-1749, Question 03.10-5(a) is resolved**.

Regarding **RAI 216-1749, Question 03.10-5(b)**, the applicant stated that DCD Tier 2, Section 3.10.2.2, does not give the criteria explicitly defining a valve as rigid or not rigid. However, DCD Tier 2, Section 3.10.2 states that:

> ...for the purpose of qualification of equipment by analysis, the rigid range is defined as having a natural frequency greater than 50 Hz. For the purpose of testing of equipment that is not sensitive to response levels caused by high frequency ground motions, rigid is defined as equipment with a natural frequency greater than 33 Hz. If the equipment, to be tested, is sensitive to response caused by high frequency ground motions, then rigid is defined as equipment having a natural frequency greater than 50 Hz.

Additionally, mechanical equipment (including valves) is qualified within the US-APWR Equipment EQ Program referenced in response Part (a) with guidance from ASME QME-1-2007. The staff finds that the response was acceptable regarding the frequency ranges specified but the applicant needed to clarify the definition of rigid equipment. Therefore, **RAI 216-1749, Questions 03.10-5(b)** was closed, as unresolved. In follow-up **RAI 486-3861, Question 03.10-15**, the staff requested the applicant to clarify and provide justification for the following statement which appears both in the response to **RAI 216-1749, Question 03.10-5(b)** and in a statement in MUAP-08015, Revision 0, Section B.10 (Qualification by Analysis):

> If the equipment to be tested is sensitive to response caused by high frequency ground motion, then rigid is defined as equipment having a natural frequency greater than 50 Hz.

The staff noted that, for hard rock high frequency spectra, the definition of rigid equipment should depend on the frequency at the beginning of ZPA of the RRS for the equipment location.

In its response to **RAI 468-3861, Question 03.10-15**, dated December 25, 2009, the applicant stated that MUAP-08015, Revision 1, Section B10, has clarified the rigidity definition of 50 Hz. For the US-APWR, 50 Hz is chosen as the definition of rigid because it matches the frequency at the beginning of the ZPA range for the standard plant CSDRS and associated ISRS, which serve to define the RRS for equipment locations. For equipment which is not sensitive to high frequency excitation, setting the ZPA frequency of the RRS at 50 Hz will not yield any meaningful new dynamic testing results. For equipment which may be sensitive to high frequency, the provisions of the US-APWR Equipment EQ Program procedures already require additional screening and/or qualification testing for all high frequency exceedances above 20 Hz, as explained in the response to **RAI 486-3861, Question 03.10-11**. Therefore, for equipment potentially sensitive to high frequency, any exceedances above 50 Hz are captured by the US-APWR Equipment EQ Program. This situation may occur at certain hard rock high frequency sites. In these cases, the applicant noted that the site-specific ZPA range may then actually be at frequencies greater than 50 Hz, and the RRS would then also need to be adjusted accordingly.

The staff finds the response acceptable since the applicant clarified the definition of rigid equipment in MUAP-08015, Revision 1. Accordingly, **RAI 486-3861, Question 03.10-15, is resolved**.

DCD Tier 2, Section 3.10.2.2, under “Valves" states that the seismic Category I active valves are analyzed using the requirements and stress limits of ASME Code, Section III, and DCD Tier 2, Section 3.9.3. The method for qualification of active valve assemblies that can provide an acceptable level of assurance of functional operability provided in ASME QME-1-2007 is also used as guidance. This method of qualification is based on tests and analysis demonstrating the ability of the valve assembly to perform its function under extreme adverse conditions of pressure, mechanical loading, flow dynamics, temperature and vibration.

While DCD Tier 2, Revision 1, Section 3.10.2.2, states that ASME QME-1-2007 will be used, DCD Tier 2, Section 3.10.2 and DCD Tier 2, Section 3.10.6, “References,” refer to ASME QME-1-2002. Therefore, in **RAI 216-1749 Revision 1, Question 03.10-6**, the staff requested the applicant to correct this inconsistency, with ASME QME-1-2007 being the correct reference.

In its response to **RAI 261-1749, Question 03.10-6**, dated March 25, 2009, the applicant stated that the reference to ASME QME-1-2002 and DCD Tier 2, Section 3.10.2 and DCD Tier 2, Section 3.10.6 have been corrected in DCD Revision 2 to incorporate the proper reference to ASME QME-1-2007. The staff confirmed that the correction was made in DCD Revision 2. Therefore, the staff considers the applicant’s response to be acceptable. Accordingly, **RAI 216-1749, Question 03.10-6, is resolved**.

3.10.4.5 Analysis or Testing of Mechanical and Electrical Equipment Supports

Regarding DCD Tier 2, Section 3.10.1.3, ‘Performance Criteria,” the staff identified that the applicant’s statement that the deformation of component supports and structures is acceptable at the SSE levels requires justification. Therefore, in **RAI 216-1749, Question 03.10-7**, the staff requested the applicant to provide additional information and to revise the DCD Tier 2, Section 3.10 of the DCD to explicitly clarify and justify the level, and location/situations, of deformation that will be allowable according to the proposed approaches for seismic qualification of equipment. The staff requested that the applicant specify the permissible extent and degree of inelasticity at the SSE design level for various categories of equipment, types of equipment supports, the criteria for successful performance of the equipment safety function during and after seismic events, and the applicant’s basis (whether implicit or explicit) for assuring adequate beyond-design-basis margin with respect to both inelastic capacity reserve and equipment functionality reserve.

In its response to **RAI 261-1749, Question 03.10-7**, dated April 22, 2009, the applicant stated that deformation is to be limited to prevent interaction with other systems and equipment, and to demonstrate that functionality is maintained. Deformation of component supports and structures at SSE levels is limited in the design through conformance to applicable codes and criteria as described in various DCD Tier 2 Sections, including Sections 3.8.4, “Other seismic Category I Structures,” 3.9.2, “Dynamic Testing and Analysis of Systems, Components, and Equipment,” 3.9.3, 3.12.6, “Piping Support Design Criteria,” and Appendices 3A, “Heating, Ventilation, and Air Conditioning Ducts and Duct Supports,” 3F, “Design of Conduit and Conduit Supports,” and 3G, “Seismic Qualification of Cable Trays and Supports.” For design and analysis, the permissible extent and degree of inelasticity at the SSE design level for equipment is implicitly defined by the codes and criteria that are described and defined in those DCD Tier 2 Sections. Those codes and criteria identify stress and deformation limits and load combinations. As described in DCD Tier 2 Section 3.10.2.2, the acceptance criteria in the seismic qualification of active mechanical equipment (valves, pumps and dampers) is that the allowable stresses permitted by ASME B&PV Code, Section III, for Class 1, 2 and 3 active components as identified in DCD Tier 2, Section 3.9.3 are not exceeded. Similarly for other mechanical equipment, the applicable code is ASME Code, Section III. DCD Tier 2, Section 3.10.3, “Methods and Procedures of Analysis or Testing of Supports of Mechanical and Electrical Equipment and Instrumentation,” refers to DCD Tier 2, Appendices 3A, 3F and 3G for seismic qualification of HVAC duct supports, conduit supports and cable tray supports respectively. DCD Tier 2, Section 3.10.4 also states that mechanical equipment supports and instrument lines supports addressed in DCD Tier 2 Subsection 3.12.6 are seismically designed per ASME Code, Section III.

The US-APWR Equipment EQ Program supplements DCD Tier 2, Section 3.10. The US-APWR Equipment EQ Program includes qualification criteria for analysis and testing to assure successful performance of the equipment safety function in addition to structural integrity. MUAP-08015, Section B.11 includes the following requirement with respect to all seismic qualification by testing:

> Testing must exhibit a 10 percent margin above those acceleration requirements established at the mounting point of equipment unless otherwise justified in the seismic qualification report (Reference Section 6.3.1.6 of IEEE Standard 323).

The US-APWR Equipment EQ Program procedures also incorporate other pertinent equipment qualification guidelines of IEEE Standard 323, and guidelines of IEEE Standard 344 and ASME QME-1 with respect to assuring the performance of safety functions of plant equipment. The margin described above provides potential capacity and functionality reserve. Based on above discussion, the applicant stated that equipment and supports will perform their designated safety-related functions.

The staff found the response acceptable regarding mechanical equipment supports since it is implicitly defined by the applicable sections of ASME B&PV code and criteria. However, the response was unacceptable regarding electrical and instrumentation and control equipment. Therefore, **RAI 216-1749, Question 03.10-7** was closed, as unresolved. In follow-up **RAI 486-3861, Question 03.10-16**, the staff requested the applicant to clarify and justify the level of deformation that will be allowable according to the proposed approaches for seismic qualification test and/or analysis by specifying the permissible extend and degree of inelasticity at the SSE design level for electrical equipment.

In its response to **RAI 468-3861, Question 03.10-16**, dated December 25, 2009, the applicant stated that seismic Category I electrical and I&C equipment, such as motor control centers, control consoles, electrical panels, pressure switches, level transmitters, sensors, etc. are normally manufactured and installed within a steel enclosure. Seismic Category I I&C components such as pressure sensors, temperature sensors, flow monitors, transmitters, etc. are also designed and manufactured for installation on piping and HVAC duct systems. The main concerns for the electrical, including, I&C equipment qualification evaluation are their functionality and operability under the seismic effect. Due to the sensitivity of this type of equipment to the earthquake motion, its seismic qualification is generally obtained through testing instead of analysis. As stated in DCD Tier 2, Section 3.10.2, the preferred method of seismic qualification is by testing for complex electrical equipment, which must perform an active function during and after the SSE. During seismic qualification testing, the electrical equipment is inspected to ensure that there is no physical deformation or damage on the equipment and it is mounting to the testing equipment, the electrical components themselves and their mounting on and within the steel enclosure. In addition, the 10 percent margin noted by the applicant in their original response to **RAI 216-1749, Question 3.10-7** is stated in MUAP-08015, Section B.11, as applicable to all seismic qualification by testing, electrical and I&C equipment. Seismic Category I electrical equipment such as electrical boxes and panels may be "hard mounted equipment (floor and/or wall mounted)" with or without an intervening support structure (i.e., steel frame or concrete support pad). Seismic Category I I&C components such as valves, transmitters, etc., can be steel rack mounted and then "hard mounted" to the building structure floor or walls. For these situations where maintaining structural integrity alone can assure the design-intended function of the electrical equipment, excessive deformation of the supporting structure (concrete or steel) and the anchorage system to the building structure are prevented in order not to damage the supported electrical and I&C components and to prevent interaction with other system supports as applicable. The steel and concrete support structures are designed as "Other Seismic Category I Structures" in accordance with AISC N690 and ACI-349 codes, as applicable and as discussed in DCD Tier 2, Section 3.8.4. The load, load combinations, and acceptance criteria for steel structures are listed in DCD Tier 2, Table 3.8.4-4, "Load Combinations and Load Factors for Seismic Category I Concrete Structures," and in DCD Tier 2, Table 3.8.4-3, "Load Combinations and Load Factors for Seismic Category I Steel Structures," for concrete structures. For in-line electrical and instrumentation equipment mounted to piping, the equipment supports are qualified using load combinations, factors, and allowable stresses for the supports as given in ASME Code, Section III, Subsection NF, as stated in DCD Tier 2, Section 3.12.6.12, "Instrumentation Line Support Criteria."

The staff finds the response acceptable because the applicant has adequately addressed the equipment support issues for both mechanical and electrical equipment. Accordingly, **RAI 486-3861, Question 03.10-16, is resolved**.

DCD Tier 2, Section 3.10.3 states that the qualification of US-APWR safety-related seismic Category I electrical and mechanical equipment supports is performed by either tests or analyses to assure their structural capability, including anchorage, to withstand seismic excitation characterized by the RRS at the support mounting location.

DCD Tier 2, Section 3.10.3 states that electrical equipment and instrumentation supports (including instrument racks, control consoles, cabinets, and panels) are tested with the equipment installed or an equivalent dummy simulating the equivalent equipment inertial mass effects and dynamic coupling to the support. The input motion for the test is determined by the inservice mounting location of the support. The method for testing supports is the same as described for equipment in DCD Tier 2, Section 3.10.2. If the equipment is installed in a non-operational mode for the support test, the response of the support in the test at the equipment mounting location is monitored and characterized as a RRS to be used for functional qualification of the equipment separately, as described in DCD Tier 2, Section 3.10.2. The TRS must be shown to envelope the RRS to qualify the support for structural integrity.

DCD Tier 2, Section 3.10.3 states that if the electrical equipment supports are qualified by analysis using the methods in DCD Tier 2, Section 3.10.2, the input motion takes into account the interface requirements. The analytical results include the required input motions to the mounted equipment as obtained and characterized by a RRS, which are used to qualify the equipment separately as described in DCD Tier 2, Section 3.10.2 and the allowable stress criteria is appropriately used for the support material.

DCD Tier 2, Section 3.10.3 states that for mechanical equipment supports (including pumps and valves) the design and service load combinations and stress limits for ASME Code, Section III, is given in DCD Tier 2, Section 3.9.3 and the support is qualified by showing that these stress limits are not exceeded. The analysis results and equipment mounting and interface requirements are identified in the EQSDSs for the equipment.

DCD Tier 2, Section 3.10.3 states that batteries and battery rack supports are mounted to the building floor and seismically qualified for applicable seismic loads using the floor RRS given in DCD Tier 2, Section 3.7.2, "Seismic System Analysis," and the methods in DCD Tier 2, Section 3.10.2. For procured equipment and supports, the interface requirements are provided to the supplier in the equipment specification which is part of the purchase order. Equipment supports used in the US-APWR are generally designed to be rigid and are qualified by analysis or tests as described above and in DCD Tier 2, Section 3.10.2, and include the interface with the supporting equipment. The structural integrity of the supports is assured by showing that stresses are below the applicable code allowable stresses.

DCD Tier 2, Section 3.10.3 states that the criterion for instrumentation line supports is addressed in DCD Tier 2, Section 3.12.6 using the criteria from ASME Code, Section III, Subsection NF for Equipment Class 1, 2, and 3 supports. Based on the staff's review of the applicant's response to the associated issues, the staff finds the analysis or testing of mechanical and electrical equipment supports as described above acceptable because the criteria and methods used meet the guidelines of ASME QME-1-2007 or IEEE Standard 344-2004.

3.10.4.6 Gas Turbine Generator

The function of gas turbines for US-APWR design is to provide standby power system for each unit and each system will be used to supply onsite power to safely shut down the reactor in case the off-site power becomes unavailable. In Section 6, "Seismic Analysis," of the applicant's technical report, MUAP-07024-P, "Qualification and Test Plan of Class 1E Gas Turbine Generator (GTG) System," Revision 0, issued December 2007, the seismic analysis includes the power section (Gas Turbine Engine Assembly), and the Reduction Gearbox. The seismic capability of the GP6000-type Gas Turbine (including the power section and Reduction Gearbox) was evaluated using 1g shock loads in the axial and radial directions of the shaft. However, shock loading tests are not the same as the loading subject to the earthquake RRS input motions at the GTG system location. Therefore, in **RAI 216-1749, Question 03.10-9**, the staff requested the applicant to provide adequate detail of the seismic qualification of Class 1E GTG system in accordance with the criteria and procedures delineated in the DCD Tier 2, Section 3.10.2.

In its response to **RAI 261-1749, Questions 03.10-9**, dated April 22, 2009, the applicant stated that the purpose of MUAP-07024-P, Revision 0, is to present general information that shows that the GTG system is highly reliable and dependable and very well suited to perform its safety functions as required by US codes and standards. MUAP-07024, Section 6, merely presents some considerations for the seismic analysis of a commercial product (GTG system), in order to provide a general understanding on the feasibility of the GTG system for nuclear applications. These preliminary seismic analysis data indicates inherent margin such that the GTG has enough seismic capability. The GTGs are required to be procured and qualified (seismically and environmentally) in accordance with the requirements of the DCD and the US-APWR Equipment EQ Program in the same manner that emergency diesel generators would be procured and qualified. The seismic qualification process is to include consideration of the loading induced by the appropriate earthquake RRS input motions for the GTG system, as required by the DCD and US-APWR Equipment EQ Program.

As part of the Class 1E GTG dedication process, the applicant has performed a seismic test for a part of the components of Class 1E GTG units of US-APWR. The results are documented in the applicant's Technical Report MUAP-10023-P, "Initial Type Test Result of Class 1E Gas Turbine Generator System," Revision 3, dated September 2011.

MUAP-10023-P, Revision 3, provides results of shaker table testing of the GTG system based on the RRS input motions established at that time for the GTG system location. However, the applicant indicated in the report that, because of the seismic reanalysis for the US-APWR standard plant design, the RRS for the GTG system will be regenerated. The report also indicated that, some components of the GTG system will be redesigned and tested, some components will be re-evaluated, some components will be redesigned and reanalyzed. The response was unacceptable because of the impact of the seismic reanalysis and potential redesign of the GTG system components and lack of details in the response. Therefore, **RAI 216-1749, Question 03.10-9** was closed, as unresolved. In follow-up **RAI 486-3861, Question 03.10-17**, the staff requested the applicant to (1) provide a list of components in the GTG System to be seismically qualified with the method of seismic qualification specified for each component and estimated qualification schedules so that the staff will have opportunities to witness the testing; and (2) describe, in DCD Tier 2, Section 3.10, the seismic qualification criteria and procedures including referenced report number for related electrical and mechanical components of the GTG System.

In its response to **RAI 468-3861, Question 03.10-17**, dated December 25, 2009, the applicant indicated that:

(1) A detailed list of components in the GTG system, the method of seismic qualification for each component, and the overall qualification schedule, will become available as equipment procurement and seismic qualification proceeds. The seismic qualification of components within the GTG system will be provided by one of the following methods:

   a) Numerical Analysis

      Numerical analysis will be performed in accordance with IEEE Standard 344-2004 Section 7.0, Analysis.

   b) Tri Axial Seismic Test

      Tri axial seismic test will be performed in accordance with IEEE Standard 344-2004 Section 8.0, Testing.

   c) ASME Code, Section III Component

      ASME Code, Section III components will be procured from vendor as designed and qualified to ASME Code, Section III, Class 3. The vendor will perform the seismic analysis in accordance with IEEE Standard 344-2004 Section 7.0, Analysis.

(2) Upon completion of the seismic qualification of the components in the GTG System, details of the GTG System seismic qualification will be described in a report which will be added as a reference in a future revision of DCD Tier 2 Section 3.10.

The staff finds that the applicant did not adequately clarify the components to be qualified. In addition, the applicant has subsequently provided multiple updates between 2010, and 2013, regarding its plan to revise reports impacted by seismic reanalysis. Its latest plan "Updated Closure Plan for US-APWR Seismic and Structural Analyses – Schedule Improvement," dated February 15, 2013, describes a planned revision to MUAP-10023. The MUAP-10023 revision will provide the information about the re-evaluation of the test results and stress analysis results based on Soil-Structure Interaction (SSI) Analysis for the PS/B, which has been incorporated into the R/B complex.

Based on the above, **RAI 486-3861, Question 03.10-17 is being tracked as an Open Item** pending the submittal and the staff review of MUAP-10023 Revision. The staff issued a related **RAI 951-6587, Question 03.10-18**, as follows.

As a result of seismic reanalysis for US-APWR, the plant layout, equipment location, and the input loadings for the mechanical and electrical equipment (i.e., ISRS) in the plant might have been changed. The staff is concerned about its impact on DCD Tier 2, Section 3.10, seismic and dynamic qualification of mechanical and electrical equipment. As a result, the staff requested the applicant to provide the following information:

1. Using DCD Tier 2, Table 3.2-2 as a base, identify the mechanical and electrical equipment that are impacted by the seismic reanalysis and indicate the specific method of seismic qualification (by analysis, testing, or a combination of analysis and testing) for each item of the seismic Category I and II equipment in the Table. Reevaluate the seismic qualification of all mechanical and electrical equipment, as identified, which are impacted by the increase in the seismic input loadings (ISRS). Provide a summary of reevaluation results.

2. Confirm the validity of the previous responses to each RAI for DCD Tier 2, Section 3.10, as a result of seismic reanalysis. If necessary, provide supplemental response information to each RAI to address the issues caused by the increase in the seismic input loadings (ISRS).

In its response to **RAI 951-6587, Question 03.10-18**, dated October 15, 2012, the applicant addressed each part of the question. In its response to part 1, the applicant stated that the schedule for the update to the current plan for completing the resolution of seismic issues related to the US-APWR has been provided to the NRC in the applicant's Seismic Closure Plan letter submitted on August 29, 2012 [as noted above, it was updated on February 15, 2013]. In that plan, development of the revised ISRS for the US-APWR Standard Plant, and revision of DCD Tier 2, Appendix 3I, "In-Structure Response Spectra," reflecting the new results, is discussed. This revision will include new and revised ISRS for the revised foundation plan for the US-APWR Standard Plant, which includes the combined nuclear island structure, comprising the R/B complex, A/B and east and west PS/B. As a result of the new and revised ISRS, the seismic performance requirements of mechanical and electrical equipment within the scope of the US-APWR Standard Plant as described in DCD Tier 2 Section 3.10 may be impacted. This includes the seismic Category I mechanical and electrical equipment (including instrumentation, but excluding piping), and, where applicable, their supports, as listed in DCD Tier 2, Table 3.2-2. This equipment will be seismically qualified in accordance with the US-APWR Equipment EQ Program (MUAP-08015) referenced in DCD Tier 2, Section 3.10. The seismic performance requirements will be specified in the corresponding EQSDS, which are developed during the detailed design or procurement phase. The equipment qualification file and EQSDS identify the TRS and the RRS derived from the ISRS for the seismic qualification. The TRS is required to envelope the RRS for qualification of equipment. It should be noted that the EQSDS, and the seismic qualification results, are not available during certification of the DCD since detailed design and as-procured information is necessary. The seismic Category I capability will be verified as part of seismic Category I ITAAC in DCD Tier 1. In addition, as stated in DCD, Tier 2 Section 3.10.2, seismic Category II equipment, as defined in DCD Tier 2, Section 3.2.1, "Seismic Classification," is designed and analyzed for the SSE event, using the same methods as specified for seismic Category I equipment, to demonstrate structural integrity so as not to collapse on, or adversely interfere with seismic Category I equipment. Seismic

Category I equipment is protected from non-seismic equipment by isolation or the use of barriers when possible. If isolation is not possible, then the equipment is designed and analyzed as seismic Category II to maintain structural integrity to withstand an SSE event.

The seismic qualification requirements for ASME Code, Section III PSC are impacted by the seismic reanalysis results. The design specifications for risk-significant ASME Code, Section III PSC will be available for audit during the DCD application phase. In this case, the design specification is affected by this seismic reanalysis, and the proposed completion plan for ASME Code, Section III PSC was previously provided in the applicant's PSC Design Completion Plan letter submitted on May 12, 2011. The applicant provided an updated completion plan, "Revised Design Completion Plan for US-APWR Piping Systems and Components," on December 7, 2012. The applicant identified a list of risk-significant components by letter, "List of Risk-Significant ASME Code, Section III Piping Systems and Components Associated with Revised Design Completion Plan for US-APWR Piping Systems and Components," dated March 1, 2013. The applicant will complete design specifications for these components prior to DC. However, the remaining design will be provided as part of PSC ITAAC as described in DCD Tier 2, Chapter 14, Appendix 14B and DCD Tier 1 Section 2.3, and there is no effect during the DCD application phase. The new and revised ISRS are implemented in the detailed design phase for each ASME Code, Section III PSC. Tables 1-1 and 1-2 in the response to **RAI 951-6587, Question 03.10-18** show a summary of the impact of the seismic reanalysis on the seismic Category I or II mechanical and electrical equipment derived from DCD Tier 2, Table 3.2-2 and DCD Tier 2, Table 3D-2, "US-APWR Environmental Qualification Equipment List." These tables group the equipment in accordance with the type of equipment or the timing when the seismic qualification results for the equipment will be available. These tables also indicate the specific method of seismic qualification (by analysis, testing, or a combination of analysis and testing). As previously stated, the EQSDS, and the seismic qualification results, are not available during the DCD application phase since detailed design and as-procured information is necessary, but will be developed in accordance with the US-APWR Equipment Qualification Program.

The staff finds the response to part 1 of **RAI 951-6587, Question 03.10-18**, acceptable because the applicant has provided an adequate closure plan for seismic qualification of mechanical and electrical equipment which may be impacted by the seismic reanalysis.

In its response to part 2 of **RAI 951-6587, Question 03.10-18**, the applicant stated that it has evaluated the validity of the previous responses to each Section 3.10 RAI as a result of the in-progress seismic reanalysis and included the evaluation in Table 2 of the response. The applicant also stated that the planned DCD changes do not impact the previous responses to the listed RAI questions.

The staff considers that the applicant's response to part 2 of **RAI 951-6587, Question 3.10-18**, is not acceptable because the statement in the response, "previous responses are still materially accurate," is not clear. For example, the applicant's previous responses to **RAI 216-1749, Question 03.10-9** and **RAI 768-3861, Question 03.10-17** are either not valid or incomplete. Since part 2 is unresolved, the staff closed, as unresolved, **RAI 951-6587, Question 3.10-18** and issued follow-up **RAI 1019-7043, Question 03.10-19**, as follows.

Based on the original review by the staff of the seismic reanalysis and its impact on the seismic qualification of mechanical and electrical equipment, with the pending revision to the DCD, the staff cannot make a conclusion on the acceptability of the seismic and dynamic qualification of mechanical and electrical equipment for the US-APWR. In **RAI 1019-7043, Question 03.10-19**, the staff requested the applicant to provide, once the DCD has been updated, an assessment of the impact of the changes from the DCD to RAI responses previously provided to the staff, particularly where the revised ISRS exceeds the original ISRS and, therefore, additional qualification is required to conform to the regulatory guidance.

In its response to **RAI 1019-7043, Question 3.10-19**, dated June 18, 2013, the applicant reviewed the previous RAI responses for each DCD Tier 2, Section 3.10 RAI and evaluated the need for additional qualification based on the revised ISRS (from the seismic reanalysis) exceeding the original ISRS. The applicant found that no previous RAI responses are impacted by the seismic reanalysis or the revised ISRS exceeding the original ISRS. The staff finds the applicant's response acceptable because there is no impact from seismic reanalysis for each RAI response with the new or revised ISRS. Therefore, **RAI 1019-7043, Question 3.10-19 is resolved.**

3.10.5 Combined License Information Items

The following is a list of combined license information items from DCD Tier 2, Table 1.8-2 related to seismic and dynamic qualification of mechanical and electrical equipment:

Table 3.10-1
US-APWR Combined License Information Items

Item No.	Description	Section
3.10(1)	The COL Applicant is to document and implement an equipment qualification program for seismic Category I equipment and provide milestones and completion dates.	3.10.4.1
3.10(3)	The COL Applicant is to develop and maintain an equipment qualification file that contains a list of systems, equipment, and equipment support structures, as defined above, and summary data sheets referred to as an equipment qualification summary data sheet (EQSDS) of the seismic qualification for each piece of safety-related, seismic Category I equipment (i.e., each mechanical and electrical component of each system), which summarize the component's qualification.	3.10
3.10(5)	Components that have been previously tested to IEEE Standard 344-1971 prior to submittal of the DCD are reevaluated to justify the appropriateness of the input motion and re-qualify the equipment, if necessary. The COL Applicant is to re-qualify the component using biaxial test input motion unless the applicant provides justification for using a single-axis test input motion.	3.10.2
3.10(8)	For design of seismic Category I and II SSCs that are not part of the standard plant, the COL applicant can similarly eliminate the OBE, or optionally set the OBE higher than 1/3 SSE, provided the design of the non-standard plant's SSCs are analyzed for the chosen OBE.	3.10.1
3.10(9)	The COL Applicant is to investigate if site-specific in-structure response spectra generated for the COL application may exceed the standard US-APWR design's in-structure response spectra in the high-frequency range. Accordingly, the COL applicant is to consider the functional performance of vibration-sensitive components, such as relays and other instrument and control devices whose output could be affected by high frequency excitation.	3.10.2

The staff finds the above listing in the table concerning seismic and dynamic qualification of mechanical and electrical equipment to be complete. Also, the list adequately describes actions necessary for the COL applicant to take. No additional COL information items were identified that need to be included in DCD Tier 2, Table 1.8-2 regarding seismic and dynamic qualification of mechanical and electrical equipment.

3.10.6 Conclusions

As a result of the open item for **RAI 486-3861, Question 03.10-17**, the staff is unable to finalize its conclusions on Section 3.10 related to seismic and dynamic qualification of mechanical and electrical equipment, in accordance with NRC regulations.

3.11 Environmental Qualification of Mechanical and Electrical Equipment

3.11.1 Introduction

This section addresses the environmental qualification (EQ) of US-APWR plant equipment (mechanical, electrical, and I&C, including digital I&C) that is important to safety. EQ is the design verification process by which important to safety equipment is demonstrated to remain capable of performing its design functions during and after exposure to DBA in a harsh environment. Electrical equipment (including I&C) important to safety, as defined in 10 CFR 50.49, “Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants,” includes: (1) safety-related equipment that must remain capable of performing its design safety functions under all normal environmental conditions, AOOs, and accident and post-accident environmental conditions, (2) nonsafety-related equipment whose failure under the postulated environmental conditions could prevent satisfactory performance of safety functions, and (3) certain PAM equipment.

EQ of mechanical equipment is addressed under the provisions of GDC 4 which states, in part, that "Structures, systems, and components important to safety shall be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant accidents." Compliance with the environmental design provisions of GDC 4 for nonmetallic components of mechanical equipment is achieved by demonstrating that nonmetallic components are suitable for both the internal service conditions of the system and the external environmental conditions. The nonmetallic components shall be qualified to the most adverse of these conditions.

During plant construction, US-APWR COL applicants implement the plant-specific EQ program. This consists of selecting and procuring the equipment required to be qualified, ensuring that it is qualified in accordance with the EQ process approved during DC, installing and testing the equipment in a manner consistent with its qualification, and that auditable records of that qualification are maintained for the life of the equipment. In addition, the operational EQ program specifies the replacement frequencies of affected safety-related equipment in harsh environments and it governs periodic tests and inspections over the qualified life of the equipment to ensure that the equipment has not degraded to an unacceptable level during normal service before scheduled replacement or requalification at its end of qualified life.

DCD Tier 2, Section 3.11 describes the EQ of nonmetallic components of mechanical equipment (e.g., seals, gaskets, lubricants, fluids for hydraulic systems, and diaphragms). Functional and seismic qualifications of metallic components of mechanical equipment are not addressed in Section 3.11. DCD Tier 2 provides for the functional and seismic qualification of mechanical equipment in DCD Tier 2, Sections 3.9 and 3.10 by its reference to ASME B&PV Code, Section III, as incorporated by reference in 10 CFR 50.55a, and ASME Standard QME-1-2007, as accepted in RG 1.100, "Seismic Qualification of Electric and Mechanical Equipment for Nuclear Power Plants," Revision 3, issued September 2009.

3.11.2 Summary of Application

DCD Tier 1: Tier 1 requirements for EQ of electrical and mechanical equipment are contained in Sections 2.4, "Reactor Systems," 2.5 "Instrumentation and Controls," 2.6, "Electrical Systems," 2.7, "Plant Systems," and 2.11, "Containment Systems." These specific subsections are enumerated below in the discussion of ITAAC. ITAAC must verify that important to safety equipment selected, procured and installed in the plant is qualified in accordance with applicable regulations and guidance. ITAAC must also verify that important to safety equipment is installed and tested in a manner consistent with its acceptance criteria.

DCD Tier 2: The applicant has provided the Tier 2 description in Section 3.11, "Environmental Qualification of Mechanical, and Electrical Equipment." This section describes the implementation of the EQ program, which is generic to all US-APWR COL applicants. It provides a programmatic basis to identify, document, and confirm compliance with applicable portions of the relevant regulations.

DCD Tier 2, Appendix 3D, "Equipment Qualification List Safety and Important to Safety Electrical and Mechanical Equipment," introduces mechanical and electrical equipment that is qualified for service in the US-APWR in accordance with the requirements delineated in the US-APWR equipment EQ program. It identifies EQ equipment, its location, purpose, operational duration, and environmental conditions, radiation condition, qualification process, and seismic category. DCD Tier 2, Table 3D-2, "US-APWR Environmental Qualification Equipment List," lists all EQ equipment qualified for the US-APWR standard plant.

The applicant stated that active and passive mechanical equipment is included in the EQ program. Active mechanical equipment qualification is discussed in DCD Tier 2, Section 3.9.3, "ASME Code Class 1, 2, and 3 Components, Component Supports, and Core Support Structures," and DCD Tier 2, Appendix 3D. Qualification of nonmetallic components such as gaskets, seals, and lubricants for mechanical equipment that are safety-related or important to safety are to be included in the EQ program. The applicant stated that non-active mechanical equipment serving primarily as a safety-related support or pressure boundary will be qualified to the requirements of the ASME B&PV Code, Section III.

The applicant's Technical Report MUAP-08015, "US-APWR Equipment Qualification Program," Revision 1, issued November 2009, describes how to implement the US-APWR EQ program.

ITAAC: As noted in the summary of DCD Tier 1, a table is included below of all the ITAAC that pertain to EQ equipment.

Table 3.11-1
US-APWR ITAAC for Environmental Qualification of

Electrical and Mechanical Equipment

DCD Section Number	DCD Table Number	Design Commitment Number
2.4.1	2.4.1-2	10
2.4.2	2.4.2-5	9.a
2.4.4	2.4.4-5	6.a
2.4.5	2.4.5-5	6.a
2.4.6	2.4.6-5	6.a
2.5.1	2.5.1-6	6
2.5.4	2.5.4-2	3
2.6.1	2.6.1-3	6.a
2.6.2	2.6.2-2	2
2.6.3	2.6.3-3	3
2.6.8	2.6.8-1	7
2.7.1	2.7.1.2-5	6.a
2.7.1	2.7.1.9-5	6.a
2.7.1	2.7.1.10-4	12
2.7.1	2.7.1.11-5	6.a
2.7.3	2.7.3.3-5	6.a
2.7.6	2.7.6.7-5	6.a
2.7.6	2.7.6.13-3	3
2.11.2	2.11.2-2	6.a
2.11.3	2.11.3-5	6.a

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

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

Technical Reports: The technical reports associated with DCD Tier 2, Section 3.11 are:

1. MUAP-08015, "US-APWR Equipment Environmental Qualification Program," Revision 0, issued February 2009.

2. MUAP-08015, "US-APWR Equipment Qualification Program," Revision 1, issued November 2009.

Cross-cutting Requirements (TMI, USI/ GSI, Op Ex): There are no cross-cutting requirements for this area of review.

US-APWR Interface Issues Identified in the DCD: See DCD Tier 2, Table 1.8-2, "Compilation of All Combined License Applicant Items for Chapters 1-19."

Site Interface Issues Identified in the DCD: See DCD Tier 2, Table 1.8-1, "Significant Site Specific Interfaces with the Standard US-APWR Design."

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

3.11.3 Regulatory Basis

The relevant requirements of the Commission regulations for this area of review, and the associated acceptance criteria, are given in Section 3.11, “Environmental Qualification of Mechanical and Electrical Equipment,” Revision 3, issued March 2007, of NUREG-0800 and are summarized below. Further details and review interfaces with other SRP sections also can be found in Section 3.11 of NUREG-0800.

1. 10 CFR 50.49 requires that the applicant establish a program for qualifying electrical equipment important to safety located in a harsh environment.

2. GDC 1 requires that components important to safety be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety function to be performed.

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

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

5. GDC 23 requires that protection systems be designed to fail in a safe state, or in a state demonstrated to be acceptable on some other defined basis, if conditions such as postulated adverse environments (e.g., extreme heat or cold, pressure, steam, water, or radiation) are experienced.

6. 10 CFR Part 50, Appendix B, Criterion III, requires that measures be established to ensure that applicable regulatory requirements and the associated design bases are correctly translated into specifications, drawings, procedures and instructions. Criterion III also requires measures for verifying and checking the adequacy of design, such as by the performance of a suitable test program. The measures specifically requiring that a test program used to verify the adequacy of a specific design feature shall include suitable qualifications testing of a prototype unit under the most adverse design conditions.

7. 10 CFR Part 50, Appendix B, Section XI, requires that a test control plan be established to ensure that all tests needed to demonstrate a component's capability to perform satisfactorily in service be identified and performed in accordance with written procedures that incorporate the requirements and acceptance limits contained in applicable design documents.

8. 10 CFR Part 50, Appendix B, Section XVII, requires that sufficient records be maintained to furnish evidence of activities affecting quality.

Regulatory guidance provided for the above requirements includes:

1. RG 1.89, “Environmental Qualification of Certain Electrical Equipment Important to Safety for Nuclear Power Plants,” Revision 1, issued June 1984, provides the principal guidance for implementing the requirements and criteria of 10 CFR

50.49 for EQ of electrical equipment that is important to safety and located in a harsh environment.

2. NUREG-0588, “Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment,” provides staff positions applicable to existing plants for assessing the compliance of an environmental qualification program with 10 CFR 50.49 and conformance with RG 1.89. This guidance may be used if relevant guidance is not provided in RG 1.89.

3. RG 1.40, “Qualification Tests of Continuous-Duty Motors Installed inside the Containment of Water-Cooled Nuclear Power Plants,” endorses IEEE Standard 334, "IEEE Standard for Qualifying Continuous Duty Class 1E Motors for Nuclear Power Generating Stations," Revision 0, issued March 1973.

4. RG 1.63, “Electrical Penetration Assemblies in Containment Structures for Nuclear Power Plants,” endorses IEEE Standard 317, "IEEE Standard for Electric Penetration Assemblies in Containment Structures for Nuclear Power Generating Stations," Revision 3, issued February 1987.

5. RG 1.73, “Qualification Tests of Electric Valve Operators Installed Inside the Containment of Nuclear Power Plants,” endorses IEEE Standard 382, "IEEE Trial Use Guide for Type Test of Class 1E Electric Valve Operators for Nuclear Power Generating Stations," issued January 1974.

6. RG 1.97, "Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants," Revision 4, issued June 2006, provides guidance acceptable to the staff for the EQ of the PAM equipment described in subsection I, Item 1(f), of this SRP section, as well as instruments and controls for the equipment.

7. Draft RG 1.131, “Qualification Tests of Electric Cables and Field Splices for Light-Water-Cooled Nuclear Power Plants,” endorses IEEE Standard 383-1974, "Standard for Type Test of Class 1E Electric Cables and Field Splices for Nuclear Power Generating Stations," issued April 2009. Since then RG 1.131 was replaced by RG 1.211, “Qualification of Safety-Related Cables and Field Splices for Nuclear Power Plants,” issued April 2009, endorses IEEE Standard 383-2003, "Standard for Qualifying Class 1E Electric Cables and Field Splices for Nuclear Power Generating Stations."

8. RG 1.156, “Environmental Qualification of Connection Assemblies for Nuclear Power Plants,” issued November 1987, endorses IEEE Standard 572, "IEEE Standard for Qualification of Class 1E Connection Assemblies for Nuclear Power Generating Stations."

9. RG 1.158, "Qualification of Safety-Related Lead Storage Batteries for Nuclear Power Plants," issued February 1989, endorses IEEE Standard 535, "IEEE Standard for Qualification of Class 1E Lead Storage Batteries for Nuclear Power Generating Stations."

10. RG 1.180, "Guidelines for Evaluating Electromagnetic and Radio- Frequency Interference in Safety-Related Instrumentation and Control Systems," Revision 1,

October 2003, provides guidance acceptable to the staff for determining electromagnetic compatibility for I&C equipment during service.

11. RG 1.183, "Alternative Radiological Source Terms for Evaluating Design-Basis Accidents at Nuclear Power Reactors," issued July 2000, provides guidance acceptable to the staff for determining the radiation dose and dose rate for equipment during postulated accident conditions. These criteria, as supplemented by those of RG 1.89, should be used to evaluate the accident source term used in the environmental design and qualification of equipment important to safety.

12. RG 1.209, “Guidelines for Environmental Qualification of Safety-Related Computer-Based Instrumentation and Control Systems in Nuclear Power Plants,” issued March 2007, provides guidance acceptable to the staff for determining the EQ procedures for safety-related computer-based I&C systems for service within nuclear power plants.

13. SECY-05-0197, “Review of Operational Programs in a Combined License Application and Generic Emergency Planning Inspections, Tests, Analyses, and Acceptance Criteria,” provides the use of a license condition for operational program implementation milestones that are fully described or referenced in the FSAR.

3.11.4 Technical Evaluation

The staff reviewed DCD Tier 2, Revision 3, Section 3.11, which provides the description of the US-APWR EQ program for satisfying 10 CFR 50.49 requirements pertaining to the EQ of equipment located in a harsh environment. The staff review evaluated whether the applicant’s information presented in the DCD Tier 2, Section 3.11 is sufficient to support the conclusion that all items of equipment that are important to safety are capable of performing their design safety functions under: (1) normal environmental conditions, (2) AOOs, (3) DBAs, and post-accident environmental conditions.

The staff has also reviewed the implementation of the US-APWR EQ program that is described in MUAP-08015, Revision 1. This report includes information on: (1) the applicant’s approach and method for identifying equipment required to be environmentally qualified, (2) the DBA harsh environmental conditions to which such equipment could be exposed, and the bases thereof, during and after which DBAs such equipment must remain functional, and (3) its equipment qualification methods and processes.

In addition, the staff reviewed DCD Tier 1, Sections 2.4, 2.5, 2.6, 2.7 and 2.11, for information pertaining to ITAAC to verify EQ of equipment required to be qualified by 10 CFR 50.49. The staff review was based on SRP Section 3.11. The staff also reviewed the EQ of mechanical equipment as described in Section 3.11.4.3 of this report.

3.11.4.1 Environmental Qualification of Electrical and Instrumentation and Control Equipment

Electrical, mechanical, and I&C (both analog and digital) equipment designated as important to safety, that is located in a harsh environment, are addressed in the US-APWR EQ program to verify it is capable of performing its design functions under all anticipated service conditions. SRP Section 3.11 for the electrical, mechanical, and I&C equipment requires satisfying the following requirements:

- 10 CFR 50.49.
- GDCs 1, 2, 4, and 23.
- 10 CFR Part 50, Appendix B, Criteria III, XI, and XVII.

3.11.4.1.1 Compliance with 10 CFR 50.49

10 CFR 50.49 requires that the EQ of US-APWR plant equipment (electrical, and I&C) located in a harsh environment shall be designed to have the capability of performing its design safety functions under all anticipated operational occurrences and normal, accident, and post-accident environments, and for the length of time its function is required and shall be demonstrated by appropriate type testing, testing supported by analyses, or analyses supported by experience data and/or partial test data. This equipment qualification information shall be documented and maintained in an auditable form. The staff has reviewed the US-APWR EQ information described in DCD Tier 2, Section 3.11, DCD Tier 2, Appendix 3D, and MUAP-08015, Revision 1.

Since the EQ equipment must operate without a loss of its safety function, for the time required to perform its engineered safeguards function(s), it must be identified and qualified to withstand such environmental conditions that would exist before, during, and following a DBA. The environmental conditions include time dependent temperature and pressure profiles, humidity, chemical effects, radiation, aging, submergence, and those synergistic effects, which have a significant effect on the equipment performance.

As a part of electrical and I&C equipment important to safety, 10 CFR 50.49(b) requires the applicant to identify: (1) safety-related equipment that is required to remain functional during and after postulated DBAs, (2) nonsafety-related equipment whose failure under postulated environmental conditions could prevent satisfactory accomplishment of safety functions specified by the safety-related equipment, and (3) certain PAM equipment delineated in RG 1.97, Revision 4.

10 CFR 50.49(d) requires the applicant to prepare a list of electrical (and I&C) equipment important to safety within each defined environmental zone that must remain functional during and after DBAs. The equipment also must not fail in a manner that could impact other functions important to safety nor mislead the operator, and provide the appropriate PAM functions. Finally, the functional performance requirements and associated electrical service conditions must be defined for the affected equipment.

DCD Tier 2, Table 3D-2, lists the electrical (and I&C) equipment that is identified by system code and component type. A safety-related function is an action relied upon during and following a DBE to provide for: (1) integrity of the reactor coolant system, (2) the capability to shut down and maintain the reactor in a safe-shutdown condition, and (3) the capability to prevent or mitigate the consequences of an accident that could result in the potential for offsite exposure pursuant to the requirements delineated in 10 CFR Part 100. The typical DBEs considered by the EQ Program MUAP-08015, Revision 1 include:

1. Normal operating conditions (e.g., refueling, shutdown, startup, operating).

2. AOOs (e.g., plant trips, testing).

3. DBAs (e.g., LOCA, HELB).

4. External events (e.g., LOOP).

5. Natural phenomena (e.g., earthquake, tornado).

In DCD Tier 2, Section 3.11, the applicant considered the following environmental conditions in accordance with 10 CFR 50.49(e):

1. Environment (temperature, pressure, spray, submergence, and humidity).

2. Seismic (mechanical shock).

3. Chemical effects (reactions and composition issues).

4. Radiation (60-year normal plus accident dose).

5. Performance (voltage, load, aging effects, allowable margins, etc.).

6. Synergistic effects (e.g., system interactions, testing stresses).

In implementing the US-APWR EQ program, the applicant referred to MUAP-08015, Revision 1. This report identifies applicable regulatory basis and supporting industry standards for equipment qualification as well as the EQ process including the EQ criteria and methodology for the US-APWR. The report describes the applicant's proposed methods and processes for EQ of electrical and I&C, and mechanical equipment important to safety. The EQ process also includes detailed testing during procurement, construction, and startup phases. Section 2.0, “Scope,” of the report indicated “The equipment qualification process is required for the life of the facility (i.e., ~60 years).” For the electrical and mechanical equipment, the applicant stated that a 60-year life is used for the design-basis unless otherwise stated. The ability of this equipment to operate over this period is verified by periodic inspection and testing. Equipment that does not have a 60-year service life is expected to be replaced or otherwise evaluated during the life of the plant on a scheduled basis.

DCD Tier 2, Table 3D-2 lists qualified US-APWR EQ equipment for harsh environments, including: equipment description, location (building and zone), operation duration, environmental and radiation conditions, influence of submergence, equipment classification (electrical or mechanical), and seismic category.

The applicant's approach and process for identification of equipment required to be environmentally qualified involves first defining various environmental zones within the plant, i.e., segregated spaces that will experience relatively homogeneous environmental parameters during the extremes of normal operational occurrences and the harsh environments resulting from DBAs. Then the applicant performed calculations and determined the time-dependent profile of the severity of the harsh environmental parameters in those spaces, including temperature, pressure, humidity, aging, chemical spray, submergence, radiation (including dose rate effects), and synergistic effects in accordance with 10 CFR 50.49(e).

The applicant stated that the Class 1E equipment qualified for a harsh environment is designed to withstand the environmental conditions that would exist before, during, and following a DBE without loss of safety function for the time required to perform the safety function according to 10 CFR 50.49(d). The electrical equipment EQ program under 10 CFR 50.49(e)(5) includes equipment qualifying by testing for aging, but it does not include analysis. However, DCD Tier 2, Section 3.11.2, “Qualification Tests and Analyses,” includes a statement that “The equipment is qualified for aging by test or analysis, which considers natural or artificial (accelerated) aging to its end-of-installed life condition.” This description was unclear whether aging for EQ qualification testing includes analysis also. However, the use of analysis alone is not allowed for EQ electrical equipment in a harsh environment according to the qualification method cited in 10 CFR 50.49(f).

In **RAI 511-3739, Question 03.11-25**, the staff requested that the applicant revise DCD Tier 2, Subsection 3.11.2 to delete "or analysis" from the description of EQ test specimen for aging or revise it to make clear that aging requirement of 10 CFR 50.49(e)(5) will be met.

In its response to **RAI 511-3739, Question 03.11-25**, dated February 2, 2010, and the June 25, 2010, supplemental response, the applicant stated that the requirements of 10 CFR 50.49(e)(5) apply only to those important to safety SSCs that may be impacted by the effects of aging. Therefore, it is not necessary to delete “or analysis” statement because the qualification of these SSCs as equipment requiring age testing will be preconditioned as appropriate for the expected aging guidance provided in various standards IEEE Std. 323, and ASME QME-1, “Qualification of Active Mechanical Equipment Used in Nuclear Power plants.” The applicant further stated that techniques to address the effects of aging will include operating experience, testing, analysis, in-service surveillance, condition monitoring and maintenance activities.

Based on the applicant’s response to preconditioning and statement about the age testing which includes combination of operating experience, testing, analysis, in-service surveillance, condition monitoring, and maintenance activities, the applicant assures that aging process does not depend on the use of analysis alone. Therefore, the staff finds the response acceptable and the current statement is acceptable as written (i.e., leaving in “or analysis”) in DCD Tier 2, Section 3.11.2. Accordingly, **RAI 511-3739, Question 03.11-25, is resolved.**

In Section 4.0, "Qualification Criteria," of MUAP-08015, “US-APWR Equipment Environmental Qualification Program,” Revision 0, issued February 2009, the aging was further discussed in Subsection 4.2, “Aging,” and Subsection 4.5.2, “Aging.” These subsections described the aging test that is used to determine the service life and the qualified life that is demonstrated by calculation for an SSC important to safety. Since 10 CFR 50.49(e)(5) is clear that aging testing should include preconditioning by natural or artificial (accelerated) aging to its end-of-installed life, the staff needed clarification as to how US-APWR EQ program determines their qualified life.

In **RAI 511-3739, Question 03.11-20**, the staff requested the applicant to provide additional information on: (1) how their EQ program provides for verification that the assumptions used in qualified life calculations remain valid, (2) what adjustments are to be made if they are found not to be valid, and (3) how components will be examined periodically to determine if they are aging faster than predicted in a manner that could shorten qualified life, and how to deal with that situation.

In its response to **RAI 511-3739, Question 03.11-20**, dated February 2, 2010, and its June 25, 2010, supplemental response, the applicant referred to Subsection 6.5.1.1, “Aging” of Section

6.5, “Combination of Methods,” in MUAP-08015, Revision 1 discussion on the qualification process and the specific aging program requirements. The applicant committed to include additional guidance for the use of aging techniques in the next revision of MUAP-08015, Revision 1. The applicant stated that the past qualification test results along with existing data will be considered in the development of test plans and analysis procedures. The applicant further stated that “Aging analysis is used to determine qualified life, shelf life and design life. Identified aging mechanisms include in service thermal, chemical, vibration, radiation and cyclic conditions. The most limiting of these is used to set the qualified life of an SSC in a harsh environment.”

The applicant stated that setting the qualified life will follow the guidance in IEEE Std. 323, "IEEE Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations," IEEE Std. 1205, "IEEE Guide for Assessing, Monitoring, and Mitigating Aging Effects on Class 1E Equipment Used in Nuclear Power Stations,” and ASME QME-1 for addressing aging and qualified life analysis requirements. The licensee’s aging monitoring programs will be implemented to identify and resolve with premature aging effects to SSCs.

Based on additional information on aging tests that described: (1) preconditioning of SSCs before any aging tests, (2) simulated aging analysis will be performed in accordance with guidelines provided in IEEE Std. 323, IEEE Std. 1205, and ASME QME-1, and (3) implementation of age monitoring programs to identify and resolve issues with premature aging effects, the staff finds that the approach used by the US-APWR EQ program to determine the qualified life for SSCs acceptable.

The incorporation of the proposed language in Subsection 6.5.1.1 to include additional guidance for the use of aging techniques will be verified upon review of the next revision to MUAP-08015, Revision 1. Since the applicant proposed technical report changes, **RAI 511-3739, Question 03.11-20, is being tracked as a Confirmatory Item.**

Under 10 CFR 50.49(f), each EQ equipment must be qualified by appropriate type testing, testing supported by analyses, or analyses supported by experience data or partial test data. Subsection 6.2, “Analysis,” of MUAP-08015, Revision 1 under Section 6.0, “Equipment Qualification methods,” included using “Similarity” and “Substitution.”

Subsection 6.2.1, "Similarity," of MUAP-08015, Revision 1, states that: "If the qualified life of one module can be established, then modules of similar types will have an equivalent qualified life if the modules have similar failure mechanisms." Subsection 6.2.1 then delineates the attributes that are to be compared to define and establish similarity under the EQ program. These attributes listed as: (1) type of technology used to design and manufacture the module, (2) type of critical components, (3) packaging, mounting and type of connections, (4) service conditions, and (5) safety functions.

However, the staff finds that these attributes are not sufficient to establish similarity in terms of durability and satisfactory application-specific performance in a harsh environment at end-of-life conditions, because they lack consideration of material properties that determine the critical materials' durability, aging characteristics, and application-specific harsh environment performance in end-of-life condition. For example, it is not sufficient to consider only failure mechanisms when using similarity analysis for qualified life comparison.

In **RAI 511-3739, Question 03.11-18**, the staff requested that the applicant revise Subsection 6.2.1 of MUAP-08015, Revision 0, to include consideration of key material properties and aging characteristics (e.g., application/failure mode-specific activation energy), known exposure sequence effects, known radiation type/dose rate/configuration effects, and known synergistic effects for all application-relevant environmental stressors, that can affect accelerated aging equivalent degradation and end-of-life harsh environment durability and performance.

In its June 25, 2010, supplemental response to **RAI 511-3739, Question 03.11-18,** the applicant stated that Subsection 6.2.1 of MUAP-08015, Revision 1 will be revised in a future revision to provide additional clarification on the use of similarity analysis as part of the component qualification process for qualifying similar components during the design, procurement and construction phases of a US-APWR Project. The applicant submitted the proposed revision and the planned change to this section, that will include an evaluation and analysis for mechanical and electrical equipment according to guidance provided in IEEE Std. 323, IEEE Std. 344, “IEEE Recommended Practice for Seismic Qualification of Class 1E equipment for Nuclear Power Generating Stations” and ASME QME-1, “Qualification of Active Mechanical Equipment Used in Nuclear Power Plants.”

The applicant further stated that “Supporting analysis is used to demonstrate that the results of previous tests can be appropriately used to demonstrate the qualification of similar equipment.”

The staff has reviewed the proposed list of qualification parameters for mechanical and electrical equipment that will be used for the evaluation of similarity analysis and finds it is acceptable.

The incorporation of the proposed revision in Subsection 6.2.1 to include additional clarification on the use of similarity analysis as part of the component qualification process for qualifying similar components during the design, procurement and construction phases of a US-APWR Project will be verified in the next revision of MUAP-08015, Revision 1. Since the applicant proposed technical report changes, **RAI 511-3739, Question 03.11-18, is being tracked as a Confirmatory Item.**

For Subsection 6.2.2, “Substitution” of MUAP-08015, Revision 1, the applicant stated that substitution of parts or materials to demonstrate qualification is acceptable if a comparison or analysis of their fit, form, and function supports the conclusion that the equipment performance is equal to or better than the originally qualified equipment. The staff finds that the identification of fit, form, and function alone are not adequate to establish qualification, to control the design in accordance with 10 CFR 50.49(f), and 10 CFR Part 50, Appendix B, Criterion III, “Design Control” review for suitability of application, and 10 CFR Part 21. Thus, the staff determines that form, fit, and function are not sufficient because they do not take materials or manufacturing process into account, both of which have the most significant effect on equipment performance in a harsh environment, especially prolong exposure to elevated temperatures, moisture and radiation.

In **RAI 511-3739, Question 03.11-17**, the staff requested that the applicant revise Subsection 6.2.2 of MUAP-08015, Revision 1, to reflect the analysis of substitute parts or materials that must have the material properties required in a harsh environment, and the manufacturing processes that could affect equipment performance in a harsh environment into account, or the use of partial test data (or applicable operating experience data) to support the analyses as required by 10 CFR 50.49(f) when analysis is used in combination with other methods for qualification.

In its June 25, 2010, supplemental response to **RAI 511-3739, Question 03.11-17,** the applicant stated “Section 6.2.2 of MUAP-08015, Revision 1 will be revised in a future revision to provide additional clarification regarding equipment qualification methodologies for substitute of components during the procurement and construction and phases of a US-APWR Project.”

The applicant further stated that the planned change to this section will include that the improvement for a new US-APWR substitution or like-for-like replacement of qualified components. This action will be taken only if the original as-designed components are no longer available. In this case, the procurement and design documents would be suitably revised to reflect the use of a substitute component. During the substitute equipment qualification process, parameters that would be analyzed will include materials, manufacturing process, manufacturers’ qualify program, design, and form, fit, and function.

The staff has reviewed the proposed revision to MUAP-08015, Subsection 6.2.2 and finds the change to be acceptable since the language proposed by the applicant outlines the analysis of substitute parts or materials that takes into consideration not only fit, form, or function of the substitute part but it also considers the material properties required in a harsh environment, and the manufacturing processes that could affect equipment performance in a harsh environment.

The incorporation of the proposed revision to Subsection 6.2.2 to provide additional clarification regarding equipment qualification methodologies for substitute of components during the procurement and construction and phases of a US-APWR Project will be verified upon review of the next revision to the MUAP-08015, Revision 1. Since the applicant proposed technical report changes, **RAI 511-3739, Question 03.11-17, is being tracked as a Confirmatory Item.**

For electrical equipment qualification methods, 10 CFR 50.49(f) lists type testing, operating experience, and combined method (i.e., combination of type test, operating experience and analysis) for qualifying important to safety electrical equipment.

Subsection 3.1.1, “10 CFR 50.49 Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants,” of MUAP-08015, Revision 1, states: "An alternate methodology for qualifying equipment in harsh environments is to follow commercial dedication procedures, where applicable, as outlined in EPRI and NRC approved EPRI topical reports." Also, Subsection 3.7 of the same document, states, in part: "EPRI commercial grade dedication methodologies, as approved by the NRC, are encompassed in the US-APWR EQP." The subsection further states that "NUPIC [Nuclear Procurement Issues Committee] commercial dedication methodologies, as approved by the NRC, are encompassed in the US-APWR EQP."

However, with the exception of NRC's SERs, which approved the use of EPRI TR-106439 and TR-107330 for mild-environment qualification of a specific vendor's digital I&C equipment, there are no topical (or technical) reports on commercial-grade dedication produced by EPRI or NUPIC that the NRC has approved for use specifically as a method of harsh environmental or dynamic qualification. In addition, while the cited references provide general guidance on commercial-grade dedication, they do not provide specific guidance on demonstrating the equipment qualification of each commercial-grade item production unit (i.e., designed and built without the benefit of a 10 CFR Part 50, Appendix B, QAP) without degrading or destructive type tests on each unit.

While describing general acceptance methods from EPRI Report NP-5652, “Guideline for the Utilization of Commercial-Grade Items in Nuclear Safety-Related Applications (NCIG-07),” 1988, conditionally endorsed by the NRC for screening of fraudulent components in NRC Generic

Letter (GL) 89-02, “Actions to Improve the Detection of Counterfeit and Fraudulently Marketed Products,” issued March 21, 1989, the GL was not intended to provide the EQ of electrical equipment important to safety under 10 CFR 50.49 in a harsh environment. These documents could be applicable in general, but insufficient, because they do not contain sufficient information specifically pertinent to EQ. The process followed in EPRI TRs 106439 and 107330 is for dedicating digital I&C equipment in a mild environment. While that is acceptable for digital equipment in a mild environment, these processes are not applicable for EQ of electrical equipment (including I&C) in a harsh environment under 10 CFR 50.49.

In **RAI 511-3739, Question 03.11-19**, the staff requested that the applicant revise Subsection 3.7, “NSSS Industry Standards” of MUAP-08015, to describe how the applicant's EQ program will provide for the qualification of commercial-grade items, especially those that will be located in a harsh environment.

In its February 2, 2010, response, and the June 25, 2010, supplemental response to **RAI 511-3739, Question 03.11-19,** the applicant cited the following reference documents pertaining to commercial-grade dedication for US-APWR EQ program: (1) EPRI NP-5652 (2) NRC Inspection Procedure (IP) 38703, “Commercial-Grade Dedication,” (3) IP 43004, “Inspection of Commercial-Grade Dedication Programs,” which describes the use of commercial-grade items for harsh environments, and (4) EPRI TR-102260, “Supplemental Guidance for the Application of EPRI Report NP-5652 on the Utilization of Commercial Grade Items.” The applicant provided the revision of Section 3.1.1, of MUAP-08015 to further explain the use of commercial-grade dedication in the US-APWR EQ program.

However, some aspects of the applicant’s approach to commercial-grade dedication remained unclear. Therefore, the staff closed as unresolved **RAI 511-3739, Question 03.11-19,** and in follow-up **RAI 650-5093, Question 03.11-39**, the staff requested the applicant to further account for the use of commercial dedication methods to meet the 10 CFR 50.49 requirements.

In its response to **RAI 650-5093, Question 03.11-39**, dated February 17, 2011, the applicant asserted that the application of commercial-grade dedication for qualifying basic components as defined by 10 CFR Part 21 is supported by present statutory and regulatory positions. The applicant quoted the definition from 10 CFR 21.3 of the term “basic component” under 10 CFR Part 50 or Part 52, which means a SSC, or part thereof that affects its safety function necessary to assure “the integrity of the reactor coolant pressure boundary, the ability to shut down the reactor and maintain it in a safe shutdown condition, and to prevent or mitigate the consequences of accidents that could result in the offsite release of radioactivity....” The applicant also quoted the definition of dedication from 10 CFR 21.3, i.e., which states, in part, that “dedication is an acceptance process undertaken to provide reasonable assurance that a commercial-grade item to be used as a basic component will perform its intended safety function and, in this respect, is deemed equivalent to an item designed and manufactured under a 10 CFR Part 50, Appendix B, quality assurance program.” This assurance is achieved by: (1) identifying the critical characteristics of the item and (2) verifying their acceptability by inspections, tests, or analyses performed by the purchaser or third-party dedicating entity after delivery. The response also included the following NRC references:

- Generic Letter (GL) 91-05, “Licensee Commercial-Grade Procurement and Dedication Programs,” issued April 9, 1991.
- NRC Inspection Procedure (IP) 38703, “Commercial-Grade Dedication.”

- NRC Inspection Procedure (IP) 43004, “Inspection of Commercial-Grade Dedication Programs.”
- NRC Inspection Procedure 88108, “Quality Assurance: Control of Materials, Equipment, and Services (Pre-licensing and Construction).”

By citing the above NRC referenced GL and IPs, the applicant provided details on the commercial-grade item, critical characteristics, dedicating entity, and dedication process. The applicant further explained how the staff accepted commercial-grade dedication as an acceptable method of verifying an SSC for qualification for service in a harsh environment.

The staff determined that the applicant’s proposed commercial-grade dedication process does not address all EQ requirements specified in 10 CFR 50.49 for electrical equipment and the guidelines of ASME QME-1-2007, Appendix QR-B for nonmetallic components of mechanical equipment. The staff notes that commercial-grade dedication is a process where a commercial-grade item is deemed equivalent to an item designated and manufactured under a 10 CFR 50, Appendix B, QAP. Commercial-grade dedication is a separate process from EQ. Therefore, **RAI 650-5093, Question 03.11-39, is being tracked as an Open Item.**

A mild environment is one in which the environmental conditions would at no time be significantly more severe than the environment that would occur during normal plant operation, including AOOs. For electrical and mechanical equipment located in a mild environment, the staff’s position is that acceptable environmental design can be demonstrated by the "design/purchase" specification process for the equipment. The specifications must contain a description of the functional requirements for a specific environmental zone during normal environmental conditions and AOOs. A well-supported maintenance/surveillance program, in conjunction with a good preventive maintenance program, should ensure that equipment that meets the design/purchase specifications is qualified for the designed life.

The applicant defined a mild radiation environment for such electronic equipment as a total integrated dose of less than 10 Gy (1E3 rad), and a mild radiation environment for other equipment is less than 100 Gy (1E4 rad). For the EQ of safety-related computer-based I&C equipment located in a mild environment, the applicant stated that the guidance in RG 1.209, will be used to satisfy the above mild environmental requirements. The equipment will be also tested for potential electromagnetic interference and radio frequency interference (EMI/RFI) compatibility to verify acceptable operation and analyzed to satisfy the mild EQ requirements according to the guidance provided in RG 1.180. As an example, DCD Tier 1, Table 2.5.1-6, “RT System and ESF System Inspections, Tests, Analyses, and Acceptance Criteria,” Item 7 includes ITAAC to demonstrate that the subject equipment is qualified for EMI/RFI.

In **RAI 880-6142, Question 03.11-45**, the staff requested the applicant to clarify testing of the EMI/RFI qualification of electrical and I&C equipment located in a mild environment described in Sections 4.2, 4.2.10.1, and 10.3.1 of MUAP-08015, Revision 1.

In its response to **RAI 880-6142, Question 03.11-45**, dated March 23, 2012, the applicant stated that Sections 4.2, 4.2.10.1 and 10.3.1 of MUAP-08015, Revision 1, will be revised to clarify testing of the EMI/RFI qualification of electrical and I&C equipment in accordance with RG 1.180 and RG 1.209. The staff has reviewed the proposed changes that will be revised and finds it is acceptable.

The incorporation of the proposed revision to Sections 4.2, 4.2.10.1 and 10.3.1 of MUAP-08015, Revision 1 to include additional clarification on the EMI/RFI testing as part of the electrical and I&C qualification will be verified upon review of the next revision to the MUAP-08015, Revision 1. Since the applicant proposed technical report changes, **RAI 880-6142, Question 03.11-45 is being tracked as a Confirmatory Item.**

As for qualification testing, the applicant stated that EQ of the equipment listed in DCD Tier 2, Appendix 3D may rely on testing in conjunction with the verification process. The testing will be conducted following written test procedures in compliance with the requirements of 10 CFR 50, Appendix B, Criterion XI, "Test Control." The testing applicable to aging, seismic, radiation, or EQ parameters will be performed by the manufacturers and qualified laboratories during construction and startup phases as part of the ITAAC process.

Equipment qualification methods for important-to-safety equipment list type test, analysis which includes similarity and substitution, operating experience, and combination of methods. Subsection 6.5, "Combination of Methods," of MUAP-08015 stated "if analysis is used, justification includes identifying a test or experience bases, and addressing concerns related to departure from the required type test sequence." This position is consistent with 10 CFR 50.49(f).

Equipment qualification records through the design phase will be maintained in an auditable form to permit verification that each item of mechanical, electrical, and I&C equipment is qualified for its application and meets its specified performance requirements when subjected to the environmental conditions identified above in accordance with 10 CFR 50.49(j).

For chemical environment, the applicant stated that the adverse effects of various chemicals used within the plant are normally contained by design. However, during a DBA, various chemicals can be released into the equipment's environment, which could impair the ability of the equipment to operate properly. The impact of the various chemicals used in the plant is factored into the design and EQ process. The potential for exposure to hydrogen, based on a 100 percent fuel-clad, metal-water reaction (10 CFR 50.44(c)), is also included as part of the chemical exposure analysis. The applicant stated that mechanical and electrical equipment will be evaluated its requirements to be able to withstand the effects of chemical exposure and it will be factored into the procurement process.

Synergistic effects are also evaluated to verify that these effects do not adversely impact the qualification of the electrical equipment as equipment is installed and tested. Onsite testing for electromagnetic and radio-frequency interference is performed that complies with the guidance provided in RG 1.180. Other tests which include thermal expansion tests, vibration tests, and process interaction tests are conducted to verify satisfactory performance of mechanical and electrical systems in their installed environments.

3.11.4.1.1.1 Environmental Qualification for Radiation Exposure

For radiation dose and dose rate used to determine the radiation environment for electrical and mechanical equipment qualification, the applicant stated that they are based on an NRC staff-approved source term and methodology in accordance with NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants – Final Report," issued February 1995. Radiation doses (i.e., normal operating and consequent 60 year design expected doses) associated with postulated accidents are determined by analytical computer codes and follow the appropriate guidance in RG 1.183 as described in DCD Tier 2, Chapter 15, “Transient and Accident Analyses.”

The expected levels of radiation exposure for the qualification of mechanical and electrical equipment are factored into the design process and are based on the type of radiation. The total dose expected during normal operation (assuming an expected 60 years of continuous operations) over the installed life of the equipment, and the radiation environment associated with the most severe DBA during or following an accident in which the equipment is required to remain functional, including the radiation resulting from re-circulating fluids for equipment located near the re-circulating lines and including dose-rate effects. For equipment that is only located in areas considered harsh by the potential presence of radiation, this equipment is qualified by analysis and partial test data with the appropriate considerations for margins and aging effects.

DCD Tier 2, Section 3D.1.7, “Determination of Radiation Exposure Requirements,” noted that equipment radiation doses for accidents are determined by the analytical codes as described in the DCD Tier 2 Chapter 15. MUAP-08015, Revision 0, Section 5.5.1.1, specifically noted that the guidance of RG 1.183 is incorporated into the dose analysis. However, neither DCD Tier 2, Chapter 15, nor DCD Tier 2, Chapter 12, “Radiation Protection,” provide a description of the methodologies used to perform these analyses. As a result of this observation, in **RAI 358-2642, Question 03.11-1, Item 1,** the staff requested that the applicant describe the methods, models and assumptions used to determine the Total Integrated Dose (TID) to equipment in the plant.

In its response to **RAI 358-2642, Question 03.11-1, Item 1,** dated July 10, 2009, the applicant described the methodology that would be used to calculate TID and stated that MUAP-08015 would be revised to state that the source term assumptions described in DCD Tier 2, Section 12.2.1.3, “Sources for the Design-Basis Accident,” would be used as the input to MicroShield to calculate the gamma dose rate to equipment, and that the beta dose rate would be calculated based on the effective beta energy per disintegration. However, because the information provided by the applicant did not provide sufficient information to allow confirmation of stated TID by the staff, **RAI 358-2642, Question 03.11-1, Item 1**, is considered closed, but unresolved. Therefore, in follow up **RAI 512-3893, Question 03.11-29** the staff requested the applicant to provide additional information about the MicroShield input parameters used to determine the stated gamma TID.

In its response to **RAI 512-3893, Question 03.11-29**, dated January 28, 2010, the applicant stated that the cumulative accident gamma dose in containment is determined using a cylindrical model of containment that corresponds to the actual containment inner diameter and free volume. Since MicroShield does not allow estimation of absorbed dose inside the source, the surface dose of a half-height cylinder is calculated. The response also provided a narrative description of some of the MicroShield input parameters and two tables of integrated gamma ray and beta source strengths in the containment vessel following a LOCA, but it did not commit to including this information in the DCD. Because insufficient information was available to the staff to allow confirmation of the TID values provided in MUAP-08015, Revision 1, Table 5-5 "Total Integrated Dose for Zone," **RAI 512-3893, Question 03.11-29** is considered closed, but unresolved. Therefore, in follow up **RAI 589-4536, Question 03.11-36,** the staff requested the applicant to describe (1) the computer codes and revisions used to calculate TID; (2) the methodology used to address plate out of radioactive material; and (3) the method used for interpolating TID from the tables provided in response to **RAI 512-3893, Question 03.11-29**; and to include this information in the DCD or in MUAP-08015.

In its response to **RAI 589-4536 Question 03.11-36,** dated July 8, 2010, the applicant stated that the MicroShield calculation assumes an input source term of 1.0E+0 (Photon/sec) for every energy group. With the use of Microsoft EXCEL, the "actual" dose rates are calculated through a process of multiplying the MicroShield output dose rates for each group by the "actual" source term for each group as previously provided in Table-1 “Integrated gamma ray and beta source strengths in the CV after a LOCA,” and Table-2 “Integrated gamma ray and beta source strengths in the recirculation water after a LOCA,” of the response to **RAI 512-3893 Question 03.11-29**. During the staff review of the response, the applicant indicated that it intended to revise the response. Therefore, **RAI 589-4536 Question 03.11-36 is being tracked as an Open Item**.

RAI 358-2642, Question 03.11-1, Item 1, also requested additional information regarding the methodology and assumptions used to calculate the TID to equipment. In its response to **RAI 358-2642, Question 03.11-1, Item 1**, the applicant also described the method used to establish the source term for the containment airborne activity concentration, which is not consistent with the analysis methodology used in DCD Tier 2, Chapter 15. Based on the methodology described in this response, the Containment Airborne Activity values stated in DCD Tier 2 Table 15A-15, "Peak Concentration in Containment during LOCA," would not reflect the activity values used for equipment qualification TID. The applicant did not present CV airborne activity concentrations in this response, in MUAP-08015, or in DCD Tier 2, Section 3.11. Therefore, **RAI 358-2642, Question 03.11-1, Item 1**, was closed, as unresolved, and in follow up **RAI 512-3893 Question 03.11-30,** the staff requested the applicant to revise and update MUAP-08015 or DCD Tier 2, Section 3.11 to provide the airborne activity concentrations used to determine equipment gamma and beta TID.

In its response to **RAI 512-3893 Question 03.11-30**, dated January 28, 2010, the applicant stated that reductive factors, other than radioactive decay, were not considered in the source terms utilized for the containment TID calculations. Radioactive decay, as well as CV leakage, CV spray, and the reduction effect of spontaneous deposition, is considered in the calculation of airborne radioactivity in the containment vessel in the DCD Tier 2, Chapter 15 dose analysis, making it more realistic. The applicant stated that the source terms used is shown in Table 1, “Integrated gamma ray and beta source strengths in the CV after a LOCA,” of the response to **RAI 512-3893, Question 03.11-29**. The effective energy of beta radiation used is from Appendix A, "Nuclear Decay Data" in Federal Guidance Report No.12. However, since the effective beta energy is dependent on the assumed isotopic concentration, insufficient information was provided by the applicant to allow confirmation of the reported energy distributions and resultant dose rates and TID. Therefore, **RAI 512-3893, Question 03.11-29** was closed, as unresolved, and in follow up **RAI 589-4536, Question 03.11-37**, the staff requested the applicant to revise and update MUAP-08015, Revision 1, or DCD Tier 2, Revision 2, Section 3.11, to provide the airborne activity concentrations used to determine equipment gamma and beta TID.

In its response to **RAI 589-4536, Question 03.11-37**, dated July 8, 2010, the applicant stated that the beta ray source term in Table-1 “Integrated gamma ray and beta source strengths in the CV after a LOCA” and Table-2 “Integrated gamma ray and beta source strengths in the recirculation water after a LOCA” of the response to **RAI 512-3893, Question 03.11-29** is calculated by multiplying the amount of radioactivity in Table-3 “Radioactivity (Ci) at Typical Times after LOCA (for airborne)” and Table-4 “Radioactivity (Ci) at Typical Times after LOCA (for recirculation water)” of the response to **RAI 589-4536 Question 03.11-36** by the effective energy of each beta ray. During the staff review of the response, the applicant indicated that it intended to revise the response. Therefore, **RAI 589-4536 Question 03.11-37, is being tracked as an Open Item**.

RAI 358-2642, Question 03.11-1, Item 1, also requested additional information regarding the methodology and assumptions used to calculate the TID to equipment. In its response to **RAI 358-2642, Question 03.11-1, Item 1**, the applicant also described the method used to establish the source term for the containment airborne activity concentration. In this response the applicant did not commit to revising MUAP-08015, DCD Tier 2, Section 3.11, or DCD Tier 2, Section 12.2, "Radiation Sources," to include the containment airborne activity source term. Therefore, the staff closed, as unresolved, **RAI 358-2642, Question 03.11-1, Item 1**, and in follow up **RAI 512-3893, Question 03.11-31,** the staff requested the applicant to revise and update MUAP-08015, or DCD Tier 2, Section 3.11 to provide the description of the airborne activity concentrations used to determine equipment gamma and beta TID. In its response to **RAI 512-3893, Question 03.11-31**, dated January 28, 2010, the applicant stated that information discussed in the response to **RAI 358-2642 Question 03.11-1, Item 1,** would be included in MUAP-08015 and the DCD. The staff found the response acceptable since the applicant agreed to include the requested information in MUAP-08015 and the DCD. The staff confirmed that DCD Tier 2, Revision 3 and MUAP-08015, Revision 1 have been updated to include information describing how total integrated dose is determined. Accordingly, **RAI 512-3893 Question 03.11-31, is resolved**.

MUAP-08015, Section 5.5.1.2, "Radiation Environment - Steam Line Break Accident," described some of the parameters used to determine TID for some equipment located in a harsh environment in the MS Line area. However, neither DCD Tier 2, Chapter 15, nor DCD Tier 2, Chapter 12, provide a description of the methods, models and assumptions used to perform these analyses. As a result of this observation, in **RAI 358-2642, Question 03.11-1, Item 2**, the staff requested that the applicant describe the methods, models and assumptions used to determine the TID to equipment located in the MS Line area. In its response to **RAI 358-2642, Question 03.11-1, Item 2**, dated July 10, 2009, the applicant presents a table titled, "Radioactivity Release into the MS /Feedwater piping area during MSLB" that indicates that iodine and other activities in the secondary coolant will be 16 percent of primary coolant. However, justification for the use of these values is not present in DCD Tier 2, Section 3.11 or in DCD Tier 2, Chapter 15. Therefore, the staff closed, as unresolved, **RAI 358-2642, Question 03.11-1, Item 2** and in follow up **RAI 512-3893, Question 03.11-32**, the staff requested the applicant to revise and update MUAP-08015 or DCD Tier 2, Section 3.11 to provide clarification of the methodology used to determine equipment gamma and beta source term for a MS Line Break (MSLB). In its response to **RAI 512-3893 Question 03.11-32**, dated January 28, 2010, the applicant stated that the dose rate in the MS /Feedwater piping area during an MSLB is assumed to be the same as that in the CV during a LOCA. Actually, the dose rate during an MSLB will not increase to the same level as in the CV during a LOCA, making this assumption conservative. The analysis assumes 350 gallons (1324 L) of primary-to-secondary leakage for 14 hours results in a leakage of about 0.5 percent of the primary coolant into the secondary system. All of the noble gases in the leaked primary coolant are assumed to be released into the MS /Feedwater piping area. Iodine and other nuclides that account for 16 percent of the radioactivity in the primary coolant are assumed to be released into the MS /Feedwater piping area. This assumption takes into account contributions from all other radioactive nuclides in the SG where the MSLB occurs, as well as in the remaining three SGs, and therefore is very conservative. The staff finds the response acceptable since the information provided in the response sufficiently describes the source term for a MSLB event. The staff confirmed that MUAP-08015 Revision 1contains this information. Accordingly, **RAI 512-3893 Question 03.11-32, is resolved**.

DCD Tier 2, Section 3.11.4, "Loss of Ventilation," noted that equipment in conditioned spaces was evaluated for loss of HVAC type events. However, neither DCD Tier 2, Chapter 15, nor DCD Tier 2, Chapter 12, provide a description of the methodologies used to perform the TID analyses following a loss of ventilation. As a result of this observation, in **RAI 358-2642, Question 03.11-1, Item 3**, the staff requested that the applicant describe the methods, models and assumptions used to determine the TID to equipment in the plant as a result of a loss of ventilation. In its response to **RAI 358-2642, Question 03.11-1, Item 3**, dated July 10, 2009, the applicant stated that because the ventilation system would be restored to service as soon as possible, the TID consequences would be minimal, so there would be no impact on equipment qualification. However, based on the information provided in MUAP-08015, Revision 1, Table 5-5 “Total Integrated Dose for Zone,” sheet 4 of 7, the Zone 6 beta dose (an indication of airborne activity present in the area), may be higher than the total (operational plus accident) gamma dose in Zone 6, even with the assumption of ventilation system operation. Since the applicant did not provide any criteria for the restoration of ventilation system, or limits for airborne radioactivity concentration or the resultant dose rates following a loss of ventilation system, insufficient information was available to allow the NRC staff to confirm that the loss of ventilation system would result in an insignificant impact on equipment TID. Therefore, the staff closed, as unresolved, **RAI 358-2642, Question 03.11-1, Item 3** and in follow up **RAI 512-3893, Question 03.11-33,** the staff requested the applicant to revise and update MUAP-08015 or DCD Tier 2, Section 3.11, to provide clarification with respect to the allowable unavailability of the ventilation system during DBEs. In its response to **RAI 512-3893 Question 03.11-33**, dated January 28, 2010, the applicant stated that that it is not necessary to postulate a complete loss of HVAC cooling function during a DBA because of the redundant, independent safety-related design of the HVAC for areas containing safety-related equipment. The staff finds the response acceptable since the information provided in the response sufficiently describes how the ventilation system controls TID to equipment. Accordingly, **RAI 512-3893 Question 03.11-33**, **is resolved**.

10 CFR 50.49 and 10 CFR 52.79 require applicants to develop an EQ program for equipment important to safety. SRP Section 3.11 notes that the applicant is to provide the conceptual approach, including the environmental design bases for identified equipment. SRP Section 3.11 and RG 1.206, “Combined License Applications for Nuclear Power Plants,“ issued June 2007, note that applicants should identify equipment located in harsh environments. DCD Tier 2, Section 3.11.5.2, “Radiation Environment,” stated that radiation dose rates and integrated doses of neutrons, beta, and gamma radiation for harsh environmental conditions for various plant areas and systems, are presented in DCD Tier 2, Appendix 3D. RG 1.206, Section C.I.3.11.5 “Estimated Chemical and Radiation Environment,” states that the applicant should indicate whether airborne activity contributes to the estimated TID. DCD Tier 2, Section 3.11.4, states that equipment in conditioned spaces is evaluated for loss of HVAC type events and that equipment that may be impacted is identified during the design process. However, because this information was not presented, in **RAI 358-2642, Question 03.11-2, Item 1**, the staff requested the applicant to identify in DCD Tier 2, Table 3D-2, which pieces of equipment could have the TID estimate impacted by a loss of HVAC. In its response to **RAI 358-2642 Question 03.11-2**, **Item 1**, dated July 10, 2009, the applicant stated that this item was addressed by the response to **RAI 358-2642, Question 03.11-1, Item 3**. Since the follow up RAIs to **RAI 358-2642, Question 03.11-1, Item 3**, addressed HVAC impact on TID, the staff considers **RAI 358-2642, Question 03.11-2**, **Item 1, to be resolved**.

RG 1.206, Section C.I.3.11.1, “Equipment Location and Environmental Conditions,” states that the applicant should specify both the normal and accident environmental conditions for each item of equipment, including submergence. MUAP-08015, Revision 0 notes that, while generally precluded by design, one of the anticipated environmental conditions is submergence. However, because this information was not presented, in **RAI 358-2642, Question 03.11-2, Item 2**, the staff requested the applicant to identify in DCD Tier 2, Table 3D-2, which pieces of equipment could have the TID estimate impacted by submergence. In its response to **RAI 358-2642, Question 03.11-2**, **Item 2**, dated July 10, 2009, the applicant proposed to add a column for the Influence of Submergence for Total Integrated Dose to DCD Tier 2, Table 3D-2. The staff finds the response acceptable since the column illustrates whether there is an impact to TID when components are submerged in case of HELB including LOCA. The staff confirmed that DCD Revision 2 incorporates the changes to DCD Tier 2, Table 3D-2 proposed in the RAI response. Accordingly, **RAI 358-2642, Question 03.11-2, Item 2, is resolved**.

RG 1.206, CI.3.11.1, states that the applicant should specify the location of each piece of equipment. Because DCD Tier 2, Table 3D-2, provided only the building containing the equipment, in **RAI 358-2642, Question 03.11-2, Item 3**, the staff requested the applicant to provide sufficient location information in DCD Tier 2, Table 3D-2, to allow an accurate determination of the radiological environment of the listed equipment. In its response to **RAI 358-2642, Question 03.11-2**, **Item 3**, dated July 10, 2009, the applicant stated that columns for the zone numbers and radiation condition would be added to DCD Tier 2, Table 3D-2, and the location of the zone numbers would be provided in the new DCD Tier 2, Table 3D-3, "Location for Zone." However, in some cases, the EQ Zones listed in DCD Tier 2, Table 3D-2 are inconsistent with dose rate data provided in DCD Tier 2, Figures 12.3-1, "Radiation Zones for Normal Operation/Shutdown," and 12.3-3, "Post Accident Radiation Zone MAP: 1hour After Accident." For instance, DCD Tier 2, Table 3D-2, Sheet 4 of 64 shows RHS-PT-010 in EQ Zone 13.3 (R/B Passage), and a Harsh Radiation Condition. In contrast, DCD Tier 2, Figure 12.3-3 Sheet 1 of 10 lists the post-accident dose rate in the passage as < 15 mrem/h (150 uSv/h) and DCD Tier 2, Figure 12.3-1 Sheet 4 of 34, lists the operational dose rate in that same area as < 2.5 mrem/h (25 uSv/h). The dose rates from Figures 12.3-1 and 12.3-3 do not support the conclusion that the piece of equipment is located in a Harsh Radiation Condition. The applicant has not stated any other criteria (e.g., extra conservatism for some equipment, or other radiation sources), for classification of the Radiation Condition for equipment. Therefore the staff closed, as unresolved, **RAI 358-2642, Question 03.11-2**, **Item 3**, and in follow up **RAI 512-3893, Question 03.11-34**, the staff requested the applicant to revise MUAP-08015 or DCD Tier 2, Section 3.11 to clarify the criteria for equipment Radiation Condition stated in DCD Tier 2, Table 3D-2. In its response to **RAI 512-3893, Question 03.11-34**, dated January 28 2010, the applicant committed to revising Tables 5-4 and 5-5 in MUAP-08015, Revision 1 and DCD Tier 2, Table 3D-2. The applicant submitted an amended response to **RAI 512-3893, Question 03.11-34**, dated September 5, 2012, which stated that the 60 year design expected doses are not assumed to determine the radiation dose for certain electrical equipment whose replacement is expected during the life of the plant (i.e., electrical equipment located in Zone 13-3). The assumed operating period is identified in Table 5-5 of MUAP-08015. The stated dose rate for Zone 13-3 listed in MUAP-08015, Table 5-4, was revised to correct a typographical error that overstated the assumed operational dose rate in the R/B passage ways. MUAP-08015, Table 5-5, (Sheet 7 of 7) was revised to show the corrected Normal Operation Cumulative Dose, based on the correction to Table 5-4. In addition, footnote 3 was added to MUAP-08015, Table 5-5, to show that the Radiation Condition for some equipment could be classified as "Mild" if the service life were limited to 35 years instead of 60 years. The applicant also committed to changing DCD Tier 2, Table 3D-2, to reflect the Radiation Condition of equipment impacted by the change to MUAP-08015, Tables 5-4 and 5-5. Because the response conforms to the guidance in RG 1.206, CI.3.11.1 for describing the location of equipment, the staff finds the response acceptable. The information provided in the response sufficiently describes how the ventilation system, and other restrictions, controls TID to equipment. The staff will confirm that this information is included in future revisions of MUAP-08015 and DCD Tier 2, Appendix 3D. Accordingly, **RAI 512-3893, Question 03.11-34, is being tracked as a Confirmatory Item.**

In addition, in **RAI 358-2642, Question 03.11-2, Item 3**, the staff requested additional information regarding the time base units for TID determination. The applicant referenced the response to **RAI 358-2642, Question 03.11-2, Item 4**. While the response to that question contained the information asked for in **RAI 358-2642, Question 03.11-2, Item 3**, the DCD Impact statement did not indicate how this information would be presented in DCD Tier 2, Section 3.11 or MUAP-08015. Therefore, staff closed as unresolved **RAI 358-2642, Question 03.11-2, Item 3** and in follow up **RAI 512-3893, Question 03.11-35**, the staff requested the applicant to revise and update DCD Tier 2, Section 3.11 or MUAP-08015 to include the TID time base information provided in the response to **RAI 358-2642, Question 03.11-2, Item 4**. In its response to **RAI 512-3893, Question 03.11-35**, dated January 28 2010, the applicant stated that MUAP-08015, Revision 1 includes the TID time base information provided in **RAI 358-2642, Question 03.11-2, Item 4**. The staff confirmed that this information has been incorporated into MUAP-08015, Revision 1. Accordingly, **RAI 512-3893, Question 03.11-35, is resolved**.

RG 1.206, Section C.I.3.11.5, “Estimated Chemical and Radiation Environment,” states that the applicant should provide the environmental conditions for each piece of equipment, and specifically mentions listing each type of radiation. DCD Tier 2, Section 3.11.5.2, states that radiation dose rates and integrated doses of neutrons, beta, and gamma radiation for harsh environmental conditions for various plant areas and systems, are presented in DCD Tier 2, Appendix 3D. DCD Tier 2, Chapter 12 only provides a description of neutron exposure associated with the RV. Therefore, in **RAI 358-2642, Question 03.11-2, Item 4**, the staff requested the applicant to provide neutron and beta exposure data in Appendix 3D. In its response to **RAI 358-2642, Question 03.11-2**, **Item 4** dated July 10, 2009, the applicant stated that MUAP-08015, Revision 0, Table 5-4, "Radiation Environments after LOCA Accident," will be replaced with a new Table 5-4, "Radiation Environment (Normal Operation)," indicating the dose rate for each zone including the contribution of gamma rays, neutrons and beta rays (airborne). The old Table 5-4 will be changed to new Table 5-5 "Total integrated dose for zone" which indicates Total Integrated Dose (normal operation for 60 years + accident (LOCA)) for each zone. Indications for "Mild" or "Harsh" will also be included in the Table 5-5 as a radiation condition for the Total Integrated Dose in each zone. Because the response conforms to the guidance in RG 1.206, Section CI.3.11.5, for describing the types of radiation exposure to equipment, the staff finds the response acceptable. The staff confirmed that MUAP-08015, Revision 1, contains the changes committed to in the RAI response. Accordingly, **RAI 358-2642 Question 03.11-2, Item 4, is resolved**.

RG 1.206, Section C.I.3.11.5, states that the applicant should provide the environmental conditions for each piece of equipment, and specifically mentions listing the assumptions used in the calculations. MUAP-08015, Revision 0, Table 5-4 contains integrated gamma and beta dose information, for a number of time intervals, including four months and one year, without providing the information supporting those values. Therefore, in **RAI 358-2642, Question 03.11-2, Item 5**, the staff requested the applicant to provide the supporting data in DCD Tier 2, Section 3.11, or DCD Tier 2, Chapter 12. In its response to **RAI 358-2642, Question 03.11-2**, **Item 5**, dated July 10, 2009, the applicant stated that for any zones not inside the CV, the annulus area, or the MS piping area, the LOCA cumulative doses listed in MUAP-08015, Table 5-4, were calculated based on the dose rate for each zone (the highest dose rate of the upper limits of the areas within the same zone) shown in DCD Tier 2, Figures 12.3-3 to 12.3-7, “Equipment Specification Limits for Cobalt Impurity Levels.” The dose rates in DCD Tier 2, Figures 12.3-3 to 12.3-7, are set based on the dose rates (those for gamma rays) from radioactivity in the CV at the time of the accident (LOCA). However, based on the information provided by the applicant, the staff was unable to determine the event duration, and the basis for the selection of that duration, and if any equipment listed in DCD Tier 2, Table 3D-2 is expected to need replacement, recalibration or repair for the duration of the event.

Therefore, staff closed, as unresolved, **RAI 358-2642, Question 03.11-2, Item 5** and in follow up **RAI 589-4536, Question 03.11-38**, the staff requested the applicant to include information regarding the event duration and methods, basis and assumptions used to determine that interval, especially as it relates to areas which will have radiologically harsh environments. The applicant was also requested to clearly describe in DCD Tier 2, Table 3D-1, "Equipment Post-Accident Operability Times," those pieces of equipment located in the radiologically controlled vital areas of the plant that will require replacement, calibration, or repair for the duration of the event, and to provide mission doses for the identified pieces of equipment in DCD Tier 2, Chapter 12.4, "Dose Assessment." Also, in light of the high dose rates experienced inside the containment building following the accident at TMI, Unit 2, and the resultant effort required to reenter the containment building, the staff requested the applicant to provide additional information describing how accessing equipment inside containment for calibration, repair or replacement, within the four months service time noted in DCD Tier 2, Table 3D-1, meets the requirements of 10 CFR 20.1101 (b) for maintaining Operational Radiation Exposure as low as reasonably achievable (ALARA). In its response to **RAI 589-4536 Question 03.11-38**, dated July 8 2010, the applicant committed to revising DCD Tier 2, Tables 3D-1 and 3D-2, and to include mission doses in new DCD Tier 2, Table 12.3-8, "Mission Dose for the Access Areas access route 1 week after an Accident," and the new DCD Tier 2, Figure 12.3-11, "Post Accident Radiation Zone MAP: 1week After Accident." Because the information provided by the applicant contains inconsistencies, this issue is open and not resolved. Accordingly, **RAI 589-4536, Question 03.11-38, is being tracked as an Open Item**.

RG 1.206, CI.3.11.1, indicates that for equipment inside containment, the applicant should specify whether the equipment is inside or outside the missile shield. Due to the large operational dose rate differences between the inside and outside shield wall locations the TID of the equipment could have significant variation. However, DCD Tier 2, Table 3D-2 did not contain this information. Therefore, in **RAI 358-2642, Question 03.11-3, Item 1**, the staff requested the applicant to identify in DCD Tier 2, Table 3D-2 which equipment is located inside of the missile shield. In its response to **RAI 358-2642, Question 03.11-3, Item 1**, dated July 10, 2009, the applicant committed to revising DCD Tier 2, Table 3D-2, to indicate which equipment was located inside of the secondary shield. The staff finds the response acceptable since the proposed revision to DCD Tier 2, Table 3D-2 contains the required information. The staff confirmed that DCD Tier 2 Revision 2 Table 3D-2 incorporates the change. Accordingly, **RAI 358-2642 Question 03.11-3, Item 1, is resolved**.

RG 1.183, Appendix I, "Assumptions for Evaluating Radiation Doses for Equipment Qualifications," notes that the radiation environment resulting from normal operations should be based on the conservative source term estimates reported in the facility's SAR or should be consistent with the primary coolant specific activity limits contained in the facility's TS. MUAP-08015, Revision 0, Section 5.1.2 stated that the TID from the normally expected radiation environment was derived from the radiation zones depicted in DCD Tier 2, Chapter 12. DCD Tier 2, Table 3D-2, identified a number of pieces of equipment as located in a mild environment. However, based on the Normal Operation zone maps presented in DCD Tier 2, Section 12.3, "Radiation Protection Design Features," some of these classifications may be non-conservative for operation with coolant activity values associated with design-basis cladding defects.

Therefore in **RAI 358-2642, Question 03.11-3, Item 2**, the staff requested the applicant to revise DCD Tier 2, Table 3D-2 to provide the location and dose rate information that supports the stated environmental environment type noted in DCD Tier 2, Table 3D-2. In its response to **RAI 358-2642 Question 03.11-3, Item 2**, dated July 10, 2009, the applicant stated that the response to this question was addressed by **RAI 358-2642, Question 03.11-2, Items 3 and Item 4**. Because of the link to **RAI 358-2642, Question 03.11-2, Item 3, RAI 358-2642 Question 03.11-3, Item 2** is considered closed and unresolved and addressed by follow up **RAI 512-3893, Question 03.11-34**, which is discussed above.

DCD Tier 2, Section 3.11.5.2, notes that radiation dose rates and integrated doses of neutrons, beta, and gamma radiation harsh environmental conditions for various plant areas and systems, are presented in DCD Tier 2, Appendix 3D and that the parameters are presented in time-based units, wherever applicable. However, the information contained in DCD Tier 2, Appendix 3D was incomplete. Therefore in **RAI 358-2642, Question 03.11-3, Item 3**, the staff requested the applicant to revise DCD Tier 2, Table 3D-2 to provide the beta, gamma and neutron dose rates, the time bases and the TID for neutron and beta radiation. In its response to **RAI 358-2642 Question 03.11-3, Item 3**, dated July 10, 2009, the applicant stated that the response to this question was addressed by **RAI 358-2642, Question 03.11-2, Items 3 and Item 4**. Because of the link to **RAI 358-2642, Question 03.11-2, Item 3, RAI 358-2642, Question 03.11-3, Item 3** is considered closed and unresolved and addressed by follow up **RAI 512-3893, Question 03.11-34**, which is discussed above.

SRP Section 3.11 and RG 1.206 note that the applicant should identify the radiation environment is based on the TID effects of the normally expected radiation environment over the equipment's installed life, plus the effects associated with the most severe DBE during or following which the equipment is required to remain functional. However, the information contained in DCD Tier 2, Table 3D-2 did not differentiate between radiation harsh environments and harsh environments due to non-radiological conditions. Therefore in **RAI 358-2642, Question 03.11-3, Item 4**, the staff requested the applicant to revise DCD Tier 2, Table 3D-2 to indicate in DCD Tier 2, Table 3D-2, which pieces of equipment will be in a harsh environment (i.e. TID exceeding 1 E+3 rads for electronic equipment or 1 E+4 rads for mechanical equipment) solely as a result of radiation exposure. In its response to **RAI 358-2642 Question 03.11-3, Item 4**, dated July 10, 2009, the applicant stated that the response to this question was addressed by **RAI 358-2642, Question 03.11-2, Items 3 and 4**. Because of the link to **RAI 358-2642, Question 03.11-2, Item 3, RAI 358-2642, Question 03.11-3, Item 4** is considered closed and unresolved and addressed by follow up **RAI 512-3893 Question 03.11-34**, which is discussed above.

SRP Section 3.11 states that the applicant should provide ITAAC to ensure that all required SSCs, are identified. DCD Tier 2, Table 3D-2, indicated that the area in the R/B containing the MS Safety Valves, and the MS Depressurization Valves was identified as a harsh environment. However, DCD Tier 1, Table 2.7.6.6-1, "Process Effluent Radiation Monitoring and Sampling System Equipment Characteristics," noted that the N-16 and MS Line RMS monitors were not located in a harsh environment. Therefore, in **RAI 358-2642, Question 03.11-4**, the staff requested the applicant to clarify the environmental conditions in the location of these radiation monitors. In its response to **RAI 358-2642, Question 03.11-4**, dated July 10, 2009, the applicant stated that N-16 and MS Line RMS monitors are located in a harsh environment. However both monitors are not classified as Class 1 E equipment and their operation is not required in the post-accident harsh environmental conditions. Thus the N-16 and MS Line RMS monitors do not need to be addressed in the DCD Tier 2, Table 3D-2. The staff finds the response acceptable since it is consistent with the guidance contained in SRP Section 3.11. Accordingly, **RAI 358-2642, Question 03.11-4, is resolved**.

SRP Section 3.11 and RG 1.206 state that the applicant should identify required operating time for equipment. DCD Tier 2, Table 3D-1, stated that some equipment located inside containment has a four-month operability requirement based on the acceptable time to replace, recalibrate or obtain equivalent indication. Because it was not clear how this equipment would be accessed to perform these repairs or calibrations and because DCD Tier 2, Table 3D-1 failed to note any limitations, due to dose rate, for equipment located outside containment, even though the post-accident dose rate maps provided in DCD Tier 2, Figure 12.3-6, "Post Accident Radiation Zone MAP: 1month After Accident," indicated that very high dose rates would be present for an extended period in a number of areas containing equipment, in **RAI 358-2642, Question 03.11-5**, the staff requested the applicant to revise DCD Tier 2, Table 3D-1 to provide information consistent with the expected radiological conditions as stated in the DCD Tier 2, Chapter 12. In its response to **RAI 358-2642, Question 03.11-5**, dated July 10, 2009, the applicant stated that equipment located inside containment that is inaccessible and is required for PAM has a four-month operability requirement based on the acceptable time to replace, recalibrate or obtain equivalent indication. When PAM instrumentation is located inside the CV or an inaccessible area and the PAM function is required over four months, the required operability time may be increased to one year. Additionally, when the PAM operability is four months and the PAM function is required over four months, the PAM function is required to have an equivalent indication with required operability for one year. However, four months is enough for all PAM instruments for the US-APWR and there are no PAM instruments that require one year duration. Thus, repairing, replacement, and recalibration are not required for the PAM equipment located inside the CV or inaccessible areas with operability of four months, with access inside the CV required after four months (post-accident). Currently, all operability is one year for PAM equipment located inside the CV as listed in DCD Tier 2, Table 3D-2. However, the applicant's response did not address how some components located within harsh environments, such as the CV, would be addressed after four months. The staff closed, as unresolved, **RAI 358-2642, Question 03.11-5,** and in follow up **RAI 589-4536, Question 03.11-38**, the staff requested the applicant to include information regarding the event duration and methods, basis and assumptions used to determine that interval, especially as it relates to areas which will have radiologically harsh environments. The evaluation of **RAI 589-4536, Question 03.11-38,** is discussed above.

3.11.4.1.1.2 Documentation of Equipment Qualification Program

The staff has reviewed the US-APWR EQ program that is described in DCD Tier 2, Section 3.11, and DCD Tier 2, Appendix 3D and the equipment qualification program in MUAP-08015, Revision 1. The staff finds that the US-APWR EQ program discusses the requirements of 10 CFR 50.49, which includes: (1) a list of equipment required (10 CFR 50.49(d)) to be environmentally qualified; (2) types (harsh and mild) of environmental conditions; equipment locations associated with environmental conditions; equipment operating times based on qualification methods; (3) a summary of tests and verification methods (10 CFR 50.49(f)) for EQ program that includes supporting analysis for the expected pressure, temperature, humidity, chemical, radiation, aging, submergence, and synergistic effects (10 CFR 50.49(e)); and (4) establishment of qualification documentation (10 CFR 50.49(j)).

For 10 CFR 50.49(j) requirements, the record of EQ qualification documentation must be maintained in auditable form for the entire period while the equipment is installed in the nuclear power plant. MUAP-08015, Revision 1, Section 9.9, "EQ Data Package or Data Sheets," refers the equipment qualification data for each SSC will be compiled and organized for easy access by the licensee and other users. The staff reviewed whether the applicant's equipment qualification data package contains sufficient EQ information to ensure that: the electrical, I&C, and mechanical equipment important to safety can withstand a harsh environmental condition and adequate EQ information for evaluating ITAACs listed in Table 3.11-1 of this report.

In **RAI 880-6142, Question 03.11-42**, the staff requested the applicant to provide the information that will be contained in the EQ information in the equipment qualification data package. In its response to **RAI 880-6142, Question 03.11-42**, dated February 23, 2012, the applicant provided brief contents in the data package, but provided no specific information. In its amended response to **RAI 880-6142, Question 03.11-42**, dated February 15, 2013, the applicant provided an equipment data package template, as Attachment D, "Description of Equipment Qualification Data Package Template for US-APWR equipment qualification program" to MUAP-08015.

The staff has not completed the review of equipment qualification template, whether the content of template contains sufficient information (scope, qualification methods, verification, and requirements). The staff is verifying that the template includes the information needed to ensure the SSCs important to safety for electrical, I&C, and mechanical equipment can withstand a harsh environmental condition and remain functional during and following DBEs. Therefore, **RAI 880-6142, Question 03.11-42, is being tracked as an Open Item.**

During the staff review of the EQ related ITAAC, the staff noticed that the ITAAC design commitment included Class 1E (i.e., safety-related) equipment required to be qualified per 10 CFR 50.49(b)(1), but no mention of qualifying nonsafety-related equipment per 10 CFR 50.49(b)(2). 10 CFR 50.49 includes both equipment to perform important to safety function under normal, operational occurrences, and accident environmental conditions as certain failure of nonsafety-related equipment, if any, can fail in a manner adverse to safety, or provide misleading information to the operator.

In **RAI 511-3739, Question 03.11-28**, the staff requested that the applicant revise its ITAAC to include applicable 10 CFR 50.49 (b)(2) equipment as 10 CFR 50.49 requires to be qualified, or explain the basis for the determination, that there is no 10 CFR 50.49(b)(2) equipment that needs to be environmentally qualified among other 10 CFR 50.49 (b)(1) SSCs.

In its response to **RAI 511-3739, Question 03.11-28,** dated February 2, 2010, the applicant explained that based on its review of the EQ equipment list in DCD Tier 2, Revision 2, Table 3D-2, there is no nonsafety-related electrical equipment (10 CFR 50.49(b)(2)) in the EQ program that requires to be qualified for a harsh environment. Therefore, there is no need to address 10 CFR 50.49(b)(2) in specific ITAAC for the US-APWR EQ program.

In its supplemental response to **RAI 511-3739, Question 03.11-28**, dated June 25, 2010, the applicant further elaborated that "Existing administrative and quality controls, specific SSC programs (e.g., IEEE 344 per RG 1.89) and design review procedures provide mechanisms to properly identify and document in that (b)(2) items are environmentally and seismically qualified." Thus, the response concluded that it is not necessary to establish a broad ITAAC item for 10 CFR 50.49(b)(2) equipment.

In its response to **RAI 688-5273, Question 07.07-32,** dated May 31, 2011, the applicant indicated that for the I&C system MUAP-08015 is used to qualify safety-related equipment for

10 CFR 50.49(b)(1) and industry standards [or a QAP based on 10 CFR Part 50, Appendix B] are used for addressing the nonsafety-related equipment for 10 CFR 50.49(b)(2).

Based on the above inconsistencies in the applicant's responses and the application of Appendix B into 10 CFR 50.49 requirements, the staff closed, as unresolved, **RAI 511-3739, Question 03.11-28**, and in follow-up **RAI 805-5915, Question 03.11-41, and RAI 880-6142, Question 03.11-43**, the staff requested the applicant to provide additional information how not considering 10 CFR 50.49(b)(2) equipment is going to satisfy the requirements of 10 CFR 50.49.

In its response to **RAI 805-5915, Questions 03.11-41, and RAI 880-6142, Question 03.11-43**, dated September 10, 2012, the applicant proposed to add a statement in DCD Tier 2, Section 3.11, that "In the US-APWR design, there is no nonsafety-related electrical equipment located in a harsh environment whose failure under postulated environmental conditions could prevent satisfactory accomplishment of safety functions." The applicant also proposed the following additional changes to DCD Tier 2, Section 3.11, DCD Tier 2, Table 3D-2, and MUAP-08015 by: (1) modifying DCD Tier 2, Table 3D-2 by removing and adding more mechanical and electrical equipment, (2) revising DCD Tier 2, Section 3.11 to include programs for EQ, seismic qualification, and functional qualification into the US-APWR equipment qualification program (as described in MUAP-08015), (3) adding a new DCD Tier 2, Table 3D-4, "List of Electrical Equipment with Special Seismic Qualification Requirements," and (4) deleting any reference to "important to safety" words throughout EQ documents (i.e., DCD Tier 2, Section 3.11, DCD Tier 2, Appendix 3D, and MUAP-08015).

EQ requirements for electrical equipment in 10 CFR 50.49 address important-to-safety equipment, which includes nonsafety-related equipment (10 CFR 50.49(b)(2)) located in a harsh environment, whose failure under postulated environmental condition could prevent satisfactory accomplishment of safety functions. With many similar requirements between the EQ program and the broader equipment qualification program, which includes seismic qualification (see Section 3.10 of this report), the EQ program could be consolidated with the MHI equipment qualification (i.e., MUAP-08015) program, but they are not addressing the same requirements. Thus, the respective provisions that are applicable to each program should be clearly identified and documented. As for deleting "important to safety" words throughout EQ- related documents, this is contrary to requirements of 10 CFR 50.49 and GDCs 1, 2, 4, and 23, as all SSCs to which those regulations are applicable are designated "important to safety."

The issues currently under staff review with these RAIs are: (1) consolidating the EQ program into the applicant's equipment qualification program under MUAP-08015, (2) deleting words "important to safety," and (3) changes made to the equipment list in Table 3D-2. Therefore, **RAI 805-5915, Question 03.11-41 and RAI 880-6142, Question 03.11-43 are being tracked as Open Items** for satisfying 10 CFR 50.49 requirements.

Based on the above, the staff concludes that the DCD Tier 2, Section 3.11 EQ program does not comply with 10 CFR 50.49 requirements.

3.11.4.1.2 Compliance with 10 CFR Part 50, Appendices A and B

For compliance with 10 CFR Part 50 Appendix A, the applicant elaborated on the requirements of GDC 1, 2, 4, and 23 as follows.

GDC 1 requires that for components important to safety a QAP be established and appropriate records of the design, fabrication, erection, and testing of equipment to be maintained by and under the control of the nuclear power unit licensee throughout the life of the unit.

The applicant elaborated in MUAP-08015, Revision 1 that a QAP will be established and implemented in order to provide adequate assurance that these SSCs shall satisfactorily perform their safety functions. In addition, appropriate records of the design, fabrication, erection, and testing of structures, systems, and components important to safety will be maintained by or under the control of the nuclear power unit licensee throughout the life of the unit.

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

The applicant elaborated in MUAP-08015, Revision 1 that the design bases for these SSCs shall reflect: (1) appropriate consideration of the most severe of the natural phenomena that have been historically reported for the site and surrounding area, with sufficient margin for the limited accuracy, quantity, and period of time in which the historical data have been accumulated, (2) appropriate combinations of the effects of normal and accident conditions with the effects of the natural phenomena, and (3) the importance of the safety functions to be performed.

In **RAI 880-6142, Question 03.11-44**, the staff asked the applicant to correct the frequency value, or provide justification for using (+/-) 10 percent for the frequency margin in Table 4-2, “US-APWR EQP Margin Values,” of MUAP-08015. In its response to **RAI 880-6142, Question 03.11-44**, dated March 23, 2012, the applicant agreed to revise (+/-) 5 percent to conform with IEEE Std. 323-1974 and the staff finds the change acceptable.

The incorporation of the proposed revision to Table 4-2 will be verified upon review of the next revision to the MUAP-08015, Revision 1. Therefore, **RAI 880-6142, Question 03.11-44, is being tracked as a Confirmatory Item**.

GDC 4 requires that components important to safety be designed to accommodate the effects of and to be comparable with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including LOCA.

The applicant elaborated in MUAP-08015, Revision 1 that SSCs important to safety shall be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including LOCAs. These SSCs shall be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that may result from equipment failures and from events and conditions outside the nuclear power unit.

GDC 23 requires that protection systems be designed to fail in a safe state, or in a state demonstrated to be acceptable on some other defined basis, if conditions such as postulated adverse environments (e.g., extreme heat or cold, pressure, steam, water, or radiation) are experienced.

The applicant elaborated in MUAP-08015, Revision 1 that the protection system shall be designed to fail into a safe state or into a state demonstrated to be acceptable on some other defined basis if conditions such as disconnection of the system, loss of energy (e.g., electric power, instrument air), or postulated adverse environments (e.g., extreme heat or cold, fire, pressure, steam, water, and radiation) are experienced.

For compliance with 10 CFR Part 50 Appendix B, requirement of Criterion III (Design Control), Criterion XI (Test Control), and Criterion XVII (QA Records), the applicant referenced the qualification testing that is described in DCD Tier 2, Section 3.11.3, "Qualification Test Results." MUAP-08015, Revision 1, Section 3.8, "Quality Assurance," outlines the project QA requirement for quality standards and records, and establishing records concerning the qualification of equipment (i.e., 10 CFR Part 50, Appendix B, Criterion XVII, "Quality Assurance Records"). To satisfy 10 CFR Part 50, Appendix B, Criterion III (Design Control), the applicant stated in DCD Tier 2, Section 3.11.1.4, "Standard Review Plan Evaluation," that "the US-APWR EQ Program establishes procedures to assure the proper control during the design process to identify, document, and implement the specific EQ parameters for each piece of equipment designated in Appendix 3D."

The staff has reviewed compliance with the above 10 CFR Part 50 Appendices A and B for US-APWR EQ Program for electrical and mechanical equipment outlined in DCD Tier 2, Section 3.11, Appendix 3D, and MUAP-08015, Revision 1. The staff finds that pending resolution of the confirmatory item, the applicant's stated approach to qualification of electrical and mechanical equipment in 10 CFR Part 50 Appendices A and B is acceptable.

3.11.4.1.3 Conformance with Regulatory Guides 1.89 and 1.97

RG 1.89 is used as the principle guidance for implementing the requirements and criteria of 10 CFR 50.49 for EQ of electrical equipment that is important to safety and must perform its safety function or not fail in a manner adverse to safety in a DBA harsh environment. RG 1.89 endorses IEEE Std. 323-1974, "IEEE Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations," which provides guidance for demonstrating the qualification of Class 1E equipment. DCD Tier 2, Section 3.11.2.1, "Environmental Qualification of Electrical and Mechanical Equipment," states that the US-APWR EQ program follows the guidance provided in RG 1.89 for electrical components and I&C equipment. The equipment location, relative to flooding and spray is factored into the US-APWR EQ qualification process. The DCD further stated that electrical equipment identified in DCD Tier 2, Table 3D-2 will be environmentally qualified by type testing or combination of type testing and analysis using the guidance provided in IEEE Std. 323-1974.

DCD Tier 2, Section 3.11.2.1, states that the environmental conditions shown in DCD Tier 2, Appendix 3D reflect the worse-case scenario identified by analysis of a compendium of accident conditions for that location. Since the environmental parameters in 3.11.2.1 and Appendix 3D are the expected conditions, it includes no tolerance margin. In order to allow for tolerance, the applicant stated that the margins between the most severe specified service conditions of the plant and the condition used for the qualification are evaluated based on IEEE Std. 323-1974 margin requirements. A 60-year qualified life is used for the design-basis for electrical and mechanical equipment and equipment that does not have a 60-year qualified life is expected to be replaced or otherwise evaluated during the life of the plant on a scheduled basis.

DCD Tier 2, Section 3.11.2.1 states that PAM equipment will be environmentally qualified in accordance with RG 1.97, Rev 4. The equipment regulated by this RG is used to process and display signals from accident monitoring instrumentation of all variable types. The MUAP-

08015, Revision 1 indicates that the US-APWR EQ program encompasses the requirements specified in RG 1.97.

Based on the above information, the staff finds that the US-APWR EQ program is consistent with the guidance provided in RG 1.89 and RG 1.97.

3.11.4.2 Environmental Qualification-Related Inspection Testing Analysis and Acceptance Criteria Evaluations

The staff has reviewed the applicant’s proposed EQ-related ITAAC, discussed in DCD Tier 2, Section 14.3, “Inspections, Tests, Analyses, and Acceptance Criteria,” and delineated in DCD Tier 1, Sections 2.4, 2.5, 2.6, 2.7, and 2.11 as listed in Table 3.11-1, “US-APWR ITAAC for Environmental Qualification of Electrical and Mechanical Equipment,” of this report.

The staff noted for several EQ-related ITAACs that these items: (1) did not address field inspection of as-built installation of EQ equipment, (2) did not include electrical equipment required to be qualified for a harsh environment, and (3) permitted analysis alone as a qualification method described in “Inspection, Tests, and Analysis (ITA)” column.

In the ITA column of the ITAAC tables for the aforementioned DCD Tier 1, Sections 2.4, 2.5, 2.6, 2.7, and 2.11 for a given equipment that needs to be qualified for harsh environment states: "Type tests and/or analyses will be performed on the Class 1E equipment located in a harsh environment." The way it is written, it allows the use of analysis alone. Analysis alone is contrary to the requirements of 10 CFR 50.49(f) for harsh EQ of electrical equipment. The regulation allows for type testing, type testing in conjunction with analysis, analysis supported by test data, or actual operational experience data in conjunction with analysis.

In **RAI 511-3739, Question 03.11-21**, the staff requested that the applicant revise all the above applicable ITAAC tables that allow wording "Type tests and/or analysis” to satisfy 10 CFR 50.49(f).

In its response to **RAI 511-3739, Question 03.11-21**, dated February 2, 2010, the applicant opted to revise all affected EQ-related ITAAC tables in DCD Tier 1 to state, "Type tests, analyses, or a combination of type tests and analyses will be performed," and claimed the ITA column wording used is consistent with 10 CFR 50.49(f). The applicant also stated that this wording of ITA column is consistent with the ITA description of other design center applications.

In its supplemental response to **RAI 511-3739, Question 03.11-21**, dated June 25, 2010, the applicant stated the following:

> The qualification of environmental parameters for Class 1E equipment in harsh environments is controlled by 10 CFR 50.49. The seismic qualification of Class 1E equipment is governed by 10 CFR 50, Appendix A, GDC 2 and 4. Seismic qualification can rely on test or analysis or a combination of both. Mechanical components are often integral to electrical Class 1E components (i.e., motor operated valve) and these components are often analyzed as an assembly. Analysis in conjunction with various tests may be used to qualify these types of components. As written, the text in the DCD is in compliance with these approved qualification methods. This wording is consistent with other vendor's design certification applications and the wording is intended to address [10 CFR]

50.49(f)(3) for electrical components as well as seismic qualification methodologies.

With the applicant’s proposed revision, the staff determined that the wording as proposed still permits the use of analysis alone, which is contrary to 10 CFR 50.49(f). Therefore, the staff closed as unresolved, **RAI 511-3739, Question 03.11-21** and in follow-up **RAI 650-5093, Question 03.11-40** the staff requested that the applicant revise the above ITAAC tables to reflect that "Type tests or testing and analysis” in accordance with 10 CFR 50.49(f). In its response to **RAI 650-5093, Question 03.11-40**, dated February 17, 2011, the applicant agreed to revise the above ITA language in the ITAAC tables to "Type tests or a combination of tests and analyses” as the staff requested. These DCD changes will affect Tier 1 related text and ITAAC tables for all Class 1E equipment being qualified in a harsh environment in Sections 2.4, 2.5, 2.6, 2.7, and 2.11 and Item 6a of DCD Tier 2, Table 14.3-2, “Example of ITAAC Table.” The staff has reviewed the changes and finds them acceptable since they conform to 10 CFR 50.49(f).

The staff has confirmed that DCD Tier 1, Revision 3, was revised as committed in the RAI response. Therefore, the staff finds that the applicant has adequately addressed this issue. Accordingly, **RAI 650-5093, Question 03.11-40 is resolved.**

In the DCD Tier 1, Sections 2.4, 2.5, 2.6 and 2.7 ITAAC tables, the staff also noted that the ITA columns of each table for several systems had no field inspection specified. For example, DCD Tier 1, Table 2.4.1-2, “Reactor System Inspections, Tests, Analyses, and Acceptance Criteria,” for reactor systems, DCD Tier 1, Table 2.6.8-1, “Containment Electrical Penetration Assemblies Inspections, Tests, Analyses, and Acceptance Criteria,” for electrical penetration assemblies, and DCD Tier 1, Table 2.7.6.13-3, “Area Radiation and Airborne Radioactivity Monitoring Systems Inspections, Tests, Analyses, and Acceptance Criteria,” for containment high-range radiation monitors included no inspection requirements.

In **RAI 511-3739, Question 03.11-22**, the staff requested that the applicant revise all ITAAC tables as required to indicate that all as-built or as-installed equipment (including associated wiring, cables, connections, and terminations) is to be inspected to verify that it is installed properly and in a manner that is consistent with or enveloped by the configuration in which the EQ samples on which its EQ is based were qualified by type test or provide justification for not performing such inspections.

In its response to **RAI 511-3739, Question 03.11-22**, dated June 25, 2010, the applicant stated that it has reviewed all ITAACs for these discrepancies and indicated that inspections were added: (1) DCD Tier 1, Table 2.4.1-2, Item 10, in the response to **RAI 193-1842, Question 14.03.04-22**, (2) DCD Tier 1, Table 2.7.6.13-3, Item 3, in the response to **RAI 184-1912, Question 14.03.07-24**, (3) DCD Tier 1, Table 2.6.8-1, Item 7 in the response to **RAI 182-1888, Question 14.03.06-8**.

However, the above ITAAC Tables did not include qualification requirements for the associated components (wiring, sensors, cables and terminations) that are integral part of the Class 1E equipment. The description under “Harsh Environment” in DCD Tier 2, Section 3.11.1.2, “Definition of Environmental Condition,” indicates that the above associated components that needed to function as an integral part of equipment assembly must be also qualified for harsh environmental conditions. Thus, the qualification of those associated components does not need to be addressed separately in ITAACs.

The staff reviewed the ITAACs in DCD Tier 2, Revision 3, and confirmed that the revisions have been made as proposed. Accordingly, **RAI 511-3739, Question 03.11-22, is resolved.**

DCD Tier 1, Table 2.7.1.10-1, "Steam Generator Blowdown System (SGBS) Equipment Characteristics," lists certain steam generator blowdown isolation valves and sampling isolation valves. It indicates that they include Class 1E equipment that is to be qualified for a harsh environment. However, DCD Tier 1, Table 2.7.1.10-4, “Steam Generator Blowdown System Inspections, Tests, Analyses, and Acceptance Criteria,” does not contain any EQ-related ITAAC.

In **RAI 511-3739, Question 03.11-23**, the staff requested that the applicant revise DCD Tier 1, Table 2.7.1.10-4 to include complete "EQ-related ITAAC" components, or confirm that there are no other Sections in DCD Tier 1 with a similar discrepancy, and if any are found, correct them accordingly.

In its supplemental response to **RAI 511-3739, Question 03.11-23**, dated June 25, 2010, the applicant stated: EQ ITAAC (Item 12) has been added to DCD Tier 1, Table 2.7.1.10-4, Revision 2, per the response to **RAI 191-2048, Question 14.03.04-3**. This ITAAC item includes inspections of associated wiring, cables, and terminations. The applicant confirmed that there are no other Sections in DCD Tier 1 with a similar discrepancy.

The staff has reviewed the aforementioned DCD Tier 1, Revision 2, Table 2.7.1.10-4, Item 12 and finds this revision acceptable. Therefore, **RAI 511-3739, Question 03.11-23, is resolved.**

DCD Tier 1, Section 2.7.3.5, "Essential Chilled Water System," (ECWS) page 2.7-104, under "Equipment to be Qualified for Harsh Environments," refers to "Equipment identified in Table 2.7.3.5-2 as being qualified for a harsh environment.” However, DCD Tier 1, Table 2.7.3.5-2, “Essential Chilled Water System Equipment Characteristics,” indicates that ECWS is not required to be qualified for harsh environment. Accordingly, there is no EQ-related ITAAC is listed in DCD Tier 1, Table 2.7.3.5-5, “Essential Chilled Water System Inspections, Tests, Analyses, and Acceptance Criteria.” Similar problems are identified for SFPCS in DCD Tier 1, Subsection 2.7.6.3.1, “Design Description,” and area radiation airborne radioactivity monitoring system (ARARMS) in DCD Tier 1, Subsection 2.7.6.13.1, “Design Description.”

In **RAI 511-3739, Question 03.11-24**, the staff requested the applicant to clarify or confirm that there is no EQ requirement for ECWS, SFPCS (or any other system with a similar discrepancy), and ARARMS for a harsh environment.

In its response to **RAI 511-3739, Question 03.11-24**, dated February 2, 2010, the applicant reviewed the equipment characteristics tables to determine consistency between the EQ design descriptions and ITAAC. The applicant stated that the DCD Tier 1, Subsection 2.7.3.5.1, “Design Description,” for ECWS has a discrepancy between the ECWS equipment to be qualified for harsh environments and the equipment identified in DCD Tier 1, Table 2.7.3.5-2. The applicant revised the design description section "Equipment to be Qualified for Harsh Environments" to "Not applicable" in Revision 3. In Revision 3, the applicant revised ECWS equipment as a Class 1E equipment, but the equipment will not be located in a harsh environment. DCD Tier 1, Subsection 2.7.6.3.1 for the discrepancy between the design description in DCD Tier 1, Subsection 2.7.6.3.1 for SFPCS and equipment characteristics table in the SFPCS had been revised already in DCD Tier 1, Revision 2.

As for the ARARM system equipment in a harsh environment is identified in DCD Tier 1, Table 2.7.6.13-1, “Area Radiation Monitoring System Equipment Characteristics,” the applicant stated that design description and ITAAC are correctly stated in Subsection 2.7.6.13.1.1, “Area Radiation Monitoring System,” and shown in DCD Tier 1, Table 2.7.6.13-3, Item 3.

In addition, DCD Tier 1, Table 2.5.4-2, “Information Systems Important to Safety Inspections, Tests, Analyses, and Acceptance Criteria,” Item 3, addresses EQ of the harsh environment field instrumentation for PAM variables listed in DCD Tier 1, Table 2.5.4-1, “Post Accident Monitoring Variables.” This lists PAM variables, but the applicant stated that it does not include the same information as equipment characteristics tables. Thus, the applicant proposed to revise the design commitment and acceptance criteria in DCD Tier 1, Table 2.5.4-2, Item 3, to be consistent with DCD Tier 1, Table 2.5.4-1. In DCD Tier 1, Revision 3, the following changes to the PAM EQ ITAAC were implemented.

> Item 3 Design Commitment: “The field instrumentation for the PAM variables identified in Table 2.5.4-1 that is subjected to a harsh environment can withstand the environmental conditions that would exist before, during, and following a design-basis accident without loss of safety function for the time required to perform the safety function."
>
> Item 3i Acceptance Criteria: “Type tests or a combination of type tests and analyses using the design environmental conditions, or under the conditions which bound the design environmental conditions, will be performed on the field instrumentation for the PAM variables identified in Table 2.5.4-1 that is subjected to a harsh environment."
>
> Item 3ii Acceptance Criteria: “The as-built field instrumentation and the associated wiring, cables, and terminations for the PAM variables identified in Table 2.5.4-1 that are subjected to a harsh environment are bounded by type tests or a combination of type tests and analyses."

The staff has confirmed that DCD Tier 1, Revision 3, was revised as committed in the RAI response. Therefore, the staff finds the revision to be acceptable. Accordingly, **RAI 650-5093, Question 03.11-24, is resolved.** Based on the above, the staff finds the EQ-related ITAAC, delineated in DCD Tier 1, Sections 2.4, 2.5, 2.6, 2.7, and 2.11 acceptable, and therefore meet the requirements of 10 CFR 52.47(b)(1).

3.11.4.3 EQ Program for Mechanical Equipment (Non-Metallic Components)

DCD Tier 2, Chapter 3, “Design of Structures, Systems, Components, and Equipment,” describes the approach for the EQ of nonmetallic components of mechanical equipment (e.g., seals, gaskets, lubricants, fluids for hydraulic systems, and diaphragms) in the DCD Tier 2, Section 3.11; DCD Tier 2, Appendix 3D; and MUAP-08015, Revision 1.

DCD Tier 2, Section 3.11, states that the implementation of the US-APWR EQ Program is described in MUAP-08015, Revision 1. The purpose of MUAP-08015 is to describe the Equipment Qualification Program applicable to safety-related and important to safety SSCs used to construct a US-APWR nuclear power plant that require environmental, seismic, or functional qualification. Since EQ of mechanical equipment described in DCD Tier 2, Section 3.11 is specific to the nonmetallic components of mechanical equipment, the staff reviewed the portions of MUAP-08015 applicable to the EQ of nonmetallic components of mechanical equipment.

DCD Tier 2, Section 3.11, states that the equipment addressed by the EQ Program is identified in DCD Tier 2, Appendix 3D. DCD Tier 2, Appendix 3D lists equipment by system and component type, and identifies the location, zone, operational duration (accident and post-accident operating times), and environmental/radiation conditions. The environmental conditions shown in DCD Tier 2, Appendix 3D reflect the worst-case scenario identified by analysis of the compendium of accident conditions for that location.

DCD Tier 2, Appendix 3D identifies the external environmental conditions of the components. The staff notes that qualification of nonmetallic components of mechanical equipment must also consider the internal service conditions of the system.

MUAP-08015, Revision 1, Section 6.2.3, "Qualification of Mechanical Equipment," describes the US-APWR approach for the EQ of nonmetallic components of mechanical equipment. In **RAI 901-6257, Question 03.11-51,** the staff requested the applicant to provide additional information to further describe the methodology used for the EQ of nonmetallic components. In its response to **RAI 901-6257, Question 03.11-51**, dated April 10, 2012, the applicant stated that MUAP-08015, Section 6.2.3 will be revised, in part, to include the following:

> Non-metallic components of mechanical equipment located in harsh environment are qualified in accordance with ASME QME-1-2007, Appendix QR-B as endorsed by RG 1.100 Revision 3. Acceptance design of nonmetallics located in mild environment is demonstrated by the design/purchase specifications which contains a description of the functional requirements for all anticipated service conditions. Environmental design and qualification status of nonmetallic components in both harsh and mild environments is maintained during the operational life of the plant by maintenance and surveillance procedures as described in Section 11.0 of MUAP-08015.

The staff finds the applicant's response to **RAI 901-6257, Question 03.11-51**, to be acceptable based on the planned revision to MUAP-08015 specifying that nonmetallic components of mechanical equipment are qualified in accordance with ASME QME-1-2007, Appendix QR-B, "Guide for Qualification of Nonmetallic Parts," as endorsed by RG 1.100, Revision 3. Pending verification that MUAP-08015 has been revised to include the guidance mentioned above, **RAI 901-6257, Question 03.11-51, is being tracked as a Confirmatory Item.**

MUAP-08015, Revision 1, Section 6.2.2, "Substitution," states that substitution is acceptable if a comparison or analysis of the form, fit, and function supports the conclusion that the equipment performance is equal or better than the originally qualified equipment. The staff does not consider the methods described in Section 6.2.2 to be acceptable for substitution of nonmetallic components for mechanical equipment. One method the staff considers acceptable for qualification of substitute nonmetallic components for mechanical equipment is described in ASME QME-1-2007, Appendix QR-B, Section QR-B5300, "Selection of Qualification Methods." Therefore, in **RAI 901-6257, Question 03.11-50,** the staff requested the applicant to further describe the EQ methods for substitute or replacement nonmetallic parts of mechanical equipment. In its response to **RAI 901-6257, Question 03.11-50**, dated April 10, 2012, the applicant stated that MUAP-08015, Section 6.2.2, will be revised to clarify that qualification of substitute or replacement nonmetallic parts of mechanical equipment shall be performed in accordance with ASME QME-1-2007, Appendix QR-B. The staff finds the applicant's response to **RAI 901-6257, Question 03.11-50** acceptable based on the planned revision to MUAP-08015, Section 6.2.2 specifying that substitute or replacement nonmetallic parts of mechanical equipment shall be performed in accordance with ASME QME-1-2007, Appendix QR-B. Since the applicant has identified technical report changes, **RAI 901-6257, Question 03.11-50, is being tracked as a Confirmatory Item.**

MUAP-08015, Revision 1, Section 7.7, "Development of Aging Program, and Spare or Replacement Parts," addresses commercial-grade dedication for spare or replacements parts. The staff notes that commercial-grade dedication is a process where a commercial-grade item is deemed equivalent to an item designated and manufactured under a 10 CFR 50, Appendix B, QAP. Commercial-grade dedication is a separate process from EQ. When the commercial-grade dedication program is used to dedicate commercial-grade parts, the spare or replacements parts shall meet all EQ requirements specified in the US-APWR EQ Program. For example, if a nonmetallic part of mechanical equipment located in a harsh environmental zone is dedicated by the commercial-grade program, then that item shall undergo EQ in accordance with ASME QME-1-2007, Appendix QR-B, as endorsed by RG 1.100, Revision 3. The staff also notes that the Arrhenius Model as described in ASME QME-1-2007, Appendix QR-B, Section QR-B6200, "Arrhenius Model," is considered an analysis for characterizing accelerated thermal aging effects and is not a substitution for qualification testing.

MUAP-08015, Revision 1, Section 11.0, "General Description of Utility (Licensee) Operating Equipment Qualification Program," provides a brief description of the Licensee Operating Equipment Qualification Program (OEQP). As discussed in RG 1.206 and Commission Paper SECY-05-0197, COL applicants must fully describe their operational programs to avoid the need for ITAAC regarding those programs. There are specific aspects the staff considers necessary to fully describe an EQ operational program. Therefore, in **RAI 901-6257, Question 03.11-54,** the staff requested the applicant to address specific aspects for the EQ Program in Section 11.0 of MUAP-08015. In its response to **RAI 901-6257, Question 03.11-54**, dated April 10, 2012, the applicant stated that MUAP-08015, Section 11.0 would be revised to include the following aspects to describe a COL applicant's EQ operational program: (1) evaluation of EQ results for design life to establish activities to support continued EQ; (2) determination of surveillance and preventive maintenance activities based on EQ results; (3) consideration of EQ maintenance recommendations from equipment vendors; (4) evaluation of operating experience in developing surveillance and preventive maintenance activities for specific equipment; (5) development of plant procedures that specify individual equipment identification, appropriate references, installation requirements, surveillance and maintenance requirements, post-maintenance testing requirements, condition monitoring requirements, replacement part identification, and applicable design changes and modifications; (6) development of plant procedures for reviewing equipment performance and EQ operational activities, and for trending the results to incorporate lessons learned through appropriate modifications to the EQ operational program; and (7) development of plant procedures for the control and maintenance of EQ records. The staff finds the applicant's response to **RAI 901-6257, Question 03.11-54** acceptable based on the planned revision to MUAP-08015, Section 11.0 include aspects to describe a COL applicant's EQ operational program. Pending verification that MUAP-08015 has been revised to include information discussed above, **RAI 901-6257, Question 03.11-54, is being tracked as a Confirmatory Item.**

DCD Tier 2, Section 3.11.6, "Qualification of Mechanical Equipment," specifies that the COL Applicant is to provide the site-specific mechanical equipment requirements, COL Information Item 3.11(8). This equipment is to be qualified using a qualification process that is equivalent to that delineated for the US-APWR Standard Plant, as described in MUAP-08015, Revision 1.

MUAP-08015, Revision 1, Section 4.1.1, “Mild Environment,” states that for electrical and mechanical equipment located in a mild environment, acceptable environmental design can be demonstrated by the "design/purchase" specification process for the equipment. The "design/purchase” specification contains a description of the functional requirements for a specific environmental zone during normal environmental conditions and AOOs.

MUAP-08015, Revision 1, Section 4.1.1, states “the maintenance/surveillance program, in conjunction with the preventive maintenance program, provides assurance that equipment meeting the design/purchase specifications is qualified for the designed life of the component. Compliance by the Licensee (owner) with 10 CFR 50.65 and associated guidance in RG 1.160, [“Monitoring the Effectiveness of Maintenance at Nuclear Power Plants,” Revision 2, issued March 1997,] are considered sufficient to provide reasonable assurance that environmental considerations established during design are reviewed every refueling outage and maintained on a continuing basis to ensure that the qualified design life has not been reduced by thermal, radiation, and/or cyclic degradation resulting from unanticipated operational occurrences or service conditions.” The staff does not consider the Maintenance Rule (10 CFR 50.65) and RG 1.160 to provide sufficient detail to maintain the environmental design and qualification status of safety-related mechanical and electrical equipment. In order to provide assurance that the environmental design and qualification status of equipment will be maintained during the operational life of the plant, the staff considers it necessary for this equipment to be included in the Licensee OEQP as described in MUAP-08015, Revision 1, Section 11.0. In **RAI 901-6257, Question 03.11-49,** the staff requested the applicant to clarify that the environmental design and qualification status of components in both mild and harsh environments will be maintained by the Licensee OEQP as described in MUAP-08015, Revision 1, Section 11.0. In its response to **RAI 901-6257, Question 03.11-49**, dated April 10, 2012, the applicant stated that MUAP-08015, Revision 1, Section 4.1.1 will be revised to clarify that the environmental design and qualification status of components in both mild and harsh environments will be maintained by the Licensee OEQP as described in Section 11.0 of MUAP-08015, Revision 1. The staff finds the applicant’s response to **RAI 901-6257, Question 03.11-49** acceptable based on the planned revision to MUAP-08015, Section 4.1.1 stating that the environmental design and qualification status of components in both mild and harsh environments will be maintained by the Licensee OEQP as described in Section 11.0 of MUAP-08015, Revision 1. Pending verification that MUAP-08015 has been revised to include information discussed above, **RAI 901-6257, Question 03.11-49, is being tracked as a Confirmatory Item**.

MUAP-08015, Revision 1, Section 8.0, “MHI US-APWR EQP,” and Section 9.0, “Equipment Qualification Implementation,” describe the generic and project specific qualification programs to be developed based on the provisions in the DCD. These MUAP-08015 sections reference procedure “Pro-20, US-APWR EQ Program Qualification by Analysis.” However, the staff does not consider analysis alone an acceptable method for EQ of mechanical or electrical equipment. The staff approved methods for EQ of mechanical equipment including testing or a combination of testing and analysis as described in ASME QME-1-2007. Therefore, in **RAI 901-6257, Question 03.11-52,** the staff requested the applicant to further describe procedure “Pro-20” in regard to EQ by analysis of mechanical equipment. In its response to **RAI 901-6257, Question 03.11-52**, dated April 16, 2012, the applicant stated the following regarding the EQ of mechanical equipment:

> EQ by analysis for active mechanical equipment is performed in accordance with Section QR-7320 of ASME QME-1-2007, as endorsed in RG 1.100, Revision 3. This section allows analysis to be used with data (e.g., partial test data) to support the

> analytical assumptions and conclusions. EQ by analysis with supporting data is used when testing is impractical.

The staff considers that the applicant's response acceptable based on qualification by analysis for active mechanical equipment is performed in accordance with Section QR-7320 of ASME QME-1-2007, as endorsed in RG 1.100, Revision 3. Accordingly, **RAI 901-6257, Question 03.11-52, is resolved.**

MUAP-08015, Revision 1, identifies the calculated EQ parameters for US-APWR plants. These environmental parameters are considered in the qualification of safety-related and important to safety equipment. However, the DCD does not clarify that the functional design and qualification of active mechanical equipment such as pumps, valves, and dynamic restraints addressed in other DCD sections shall include the potential impact of the adverse environmental conditions described in the US-APWR EQ Program. For example, electric motors might produce less torque under high temperature conditions identified in MUAP-08015, Revision 1, than under ambient conditions, which could impact their capability to operate their individual pumps or valves. Therefore, in **RAI 901-6257, Question 03.11-60**, the staff requested the applicant to clarify that the functional design and qualification of active mechanical equipment such as pumps and valves addressed in other DCD sections shall include the potential impact of the adverse environmental conditions described in MUAP-08015. In its response to **RAI 901-6257, Question 03.11-60**, dated April 16, 2012, the applicant stated in part:

> Electric motors located in harsh environments are qualified to the maximum expected harsh conditions identified in the analysis (these values are shown in Technical Report MUAP-08015, Chapter 5). IEEE 323-1974, as endorsed by RG 1.89, and IEEE 323-2003, as endorsed by RG 1.209, are the basis for the EQ for electric motors in harsh environments as described in Technical Report MUAP-08015. Sections 5 and 6 of IEEE 323-2003 describe the EQ process, including the application of margin and aging to establish qualified life. Section 6.1 of IEEE 323-1974 and Section 6.1.5.1 of IEEE 323-2003 specifically require the equipment to be environmentally qualified for its expected service condition for both nominal and extreme values and their expected durations. Section 6.2 of IEEE 323-1974 and Section 6.1.5.2 of IEEE 323-2003 require the EQ to address DBEs for the duration of the operational performance required during a DBE.
>
> DCD Subsection 3.11.2.1 is revised to state that the functional design and qualification of electrical and active mechanical components includes impacts from environmental parameters identified in Chapter 5, "Normal, Abnormal and Design-Basis Accident Conditions," of Technical Report MUAP-08015.

The applicant also stated that electric motors are qualified to the maximum expected harsh conditions and that DCD Tier 2, Section 3.11.2.1 would be revised to state that the functional design and qualification of electrical and active mechanical components includes impacts from environmental parameters identified in Chapter 5.0, "Normal, Abnormal and Design-Basis Accident Conditions," of MUAP-08015. The staff finds the applicant's response to **RAI 901-6257, Question 03.11-60** acceptable based on the planned revision to DCD Tier 2, Section 3.11.2.1 stating that the functional design and qualification of electrical and active mechanical components includes impacts from environmental parameters identified in Chapter 5 of MUAP-08015. Since the applicant has identified DCD changes, **RAI 901-6257, Question 03.11-60, is being tracked as a Confirmatory Item.**

The staff finds that the ITAAC for the EQ of US-APWR components listed in DCD Tier 1 are not consistent with the EQ requirements specified in DCD Tier 2. Also, the ITAAC specific to Class 1E equipment do not include nonmetallic parts of mechanical equipment. For example, the Design Commitment for ITAAC 9.a in DCD Tier 1, Table 2.4.2-5, "Reactor Coolant System Inspections, Tests, Analyses, and Acceptance Criteria," states "Class 1E equipment identified in Table 2.4.2-2, as being qualified for a harsh environment can withstand the environmental conditions that would exist before, during, and following a DBA without loss of safety function for the time required to perform the safety function." However, DCD Tier 2, Section 3.11, states that equipment designated as safety-related or important to safety is addressed in the EQ Program to verify it is capable of performing design safety function under all anticipated service conditions including normal operation, AOOs, and DBAs for the time required to perform the safety function. Therefore, in **RAI 901-6257, Question 03.11-55**, the staff requested the applicant to revise the DCD Tier 1 ITAAC to include EQ for nonmetallic part of mechanical equipment. In its response to **RAI 901-6257, Question 03.11-55**, dated April 10, 2012, the applicant described existing ITAAC for structural integrity and functional qualification of mechanical equipment and, in summary, stated that no ITAAC changes are necessary for EQ ITAAC. The staff does not agree that existing ITAAC include demonstration of EQ of nonmetallic parts of mechanical equipment. EQ of nonmetallic parts of mechanical equipment as described in DCD Tier 2, Section 3.11 demonstrates that nonmetallic parts (e.g., seals, gaskets, lubricants, fluids for hydraulic systems, and diaphragms) remain functional during the qualified life. Therefore, the staff does not consider the applicant's response to resolve **RAI 901-6257, Question 03.11-55**. **RAI 901-6257, Question 03.11-55, is being tracked as an Open Item**.

DCD Tier 2, Table 3D-2, lists mechanical equipment that is qualified in accordance with the US-APWR equipment qualification program. However, DCD Tier 2, Table 3D-2 does not specifically identify the nonmetallic subcomponents of mechanical equipment. In **RAI 901-6257, Question 03.11-56**, the staff requested the applicant to describe how nonmetallic subcomponents for mechanical equipment are identified for EQ. In its response to **RAI 901-6257, Question 03.11-56**, dated April 16, 2012, the applicant stated the following:

> Subcomponents that are to be environmentally qualified to the same environmental conditions as each main component are identified through the design-basis requirements on each component. The criterion for environmental qualification is that the property of the subcomponents with regard to its application is not degraded during the specified qualified life to the point that the main component is unable to perform its intended safety function. Examples of nonmetallic items that are subject to the qualification process include gaskets and O-rings, diaphragms, diaphragm support sheets, lubricants, hoses, nonmetallic tubing, nonmetallic valve seats or seals, shock mounting supports and various nonmetallic housings.

The staff finds the applicant's response to **RAI 901-6257, Question 03.11-56**, acceptable based on the statement that subcomponents that are to be environmentally qualified to the same environmental conditions as each main component. Examples of nonmetallic items that are subject to the qualification process include gaskets and O-rings, diaphragms, diaphragm support sheets, lubricants, hoses, nonmetallic tubing, nonmetallic valve seats or seals, shock mounting supports, and various nonmetallic housings. Accordingly, **RAI 901-6257, Question 03.11-56, is resolved**.

MUAP-08015, Revision 1, Section 9.0, “Equipment Qualification Implementation,” describes the generic equipment qualification program and its implementation for a specific US-APWR project. Section 9.9, “EQ Data Package or Data Sheets,” states that EQ data packages are developed in accordance with industry standard practices and are implemented by the following EQ program procedures:

- Pro-07, “US-APWR EQ Program Quality Assurance Program.”
- Pro-09, “US-APWR EQ Program Application During Procurement.”
- Pro-16 “US-APWR EQ Program Preparation of EQ Packages for Structures, Systems and Components (SSC).”

The applicant references the implementation procedures for development of EQ data packages but does not describe the type of EQ documentation that will be contained in the EQ data packages. Detailed documentation in the EQ data packages is needed by the staff to evaluate closure of EQ ITAAC. Therefore, in **RAI 880-6142, Question 03.11-42**, the staff requested the applicant to provide documentation that will be contained in the EQ data packages. In its response to **RAI 880-6142, Question 03.11-42**, dated March 23, 2012, the applicant discussed the documentation pertinent to the qualification of mechanical and electrical equipment and provided an example of the information in a typical EQ data package. In its amended response to **RAI 880-6142, Question 03.11-42**, dated February 15, 2013, the applicant provided an equipment data package template, as Attachment D, to MUAP-08015. The amended response to **RAI 880-6142, Question 03.11-42**, is under review. Therefore, **RAI 880-6142, Question 03.11-42, is being tracked as an Open Item.**

The staff found that MUAP-08015, Revision 1, Sections 3.0, 10.0, and Appendix D did not clearly identify a staff approved methodology for the qualification of mechanical components, such as valves, pumps and nonmetallic components. One staff approved methodology is that the qualification be performed in accordance with ASME QME-1-2007, as endorsed by RG 1.100, Revision 3. Therefore, in **RAI 901-6257, Questions 03.11-48, 03.11-53 and 03.11-57,** the staff requested the applicant to clarify that mechanical equipment be qualified in accordance with ASME QME-1-2007, as endorsed by RG 1.100, Revision 3. In its response to **RAI 901-6257, Questions 03.11-48, 03.11-53 and 03.11-57**, dated April 10, 2012, the applicant stated that MUAP-08015 Sections 3.0, 10.0, and Appendix D will be revised to state that mechanical equipment are qualified in accordance with ASME QME-1-2007, as endorsed by RG 1.100, Revision 3. The staff finds the applicant’s response acceptable based on the planned revision to MUAP-08015 Sections 3.0, 10.0, and Appendix D to state that mechanical equipment are qualified in accordance with ASME QME-1-2007, as endorsed by RG 1.100, Revision 3. Pending verification that MUAP-08015 has been revised to include information discussed above, **RAI 901-6257, Questions 03.11-48, 03.11-53 and 03.11-57, are being tracked as Confirmatory Items**.

As part of a COLA review, the NRC staff will evaluate the full description of the operational program for the environmental design and qualification of electrical and mechanical equipment provided by the COL applicant to supplement the general program description outlined in the DCD. For example, the COL applicant will need to confirm the program scope as part of its development of a full description of the EQ program on a plant-specific basis.

The staff reviewed the US-APWR DC application for compliance with the NRC regulations for the EQ of nonmetallic components of mechanical equipment. As a result of the two open items

(**RAI 880-6142, Question 03.11-42** and **RAI 901-6257, Question 03.11-55**) the staff is unable to finalize its conclusions related to EQ of nonmetallic components of mechanical equipment.

3.11.5 Combined License Information Items

The following is a list of COL item numbers and descriptions from Table 1.8-2 of the DCD related to EQ of electrical and mechanical equipment.

Table 3.11-2
US-APWR Combined License Information Items

Item No.	Description	Section
3.11(1)	The COL Applicant is responsible for assembling and maintaining the environmental qualification document, which summarizes the qualification results for all equipment identified in Appendix 3D, for the life of the plant.	3.11.7
3.11(2)	The COL Applicant is to describe how the results of the qualification tests are to be recorded in an auditable file in accordance with requirements of 10 CFR 50.49 (j).	3.11.7
3.11(3)	The COL Applicant is to provide a schedule showing the EQ Program proposed implementation milestones.	3.11.7
3.11(4)	The COL Applicant is to describe periodic tests, calibrations, and inspections to be performed during the life of the plant, which verify the identified equipment remains capable of fulfilling its intended function.	3.11.7
3.11(5)	The COL Applicant is to identify the site-specific equipment to be addressed in the EQ Program, Including locations and environmental conditions.	3.11.7
3.11(6)	The COL Applicant is to qualify site-specific electrical and mechanical equipment (including instrumentation and control, and certain accident monitoring equipment) using an equivalent qualification process to that delineated for the US-APWR Standard Plant.	3.11.7
3.11(7)	The COL Applicant is to identify chemical and radiation environmental requirements for site-specific qualification of electrical and mechanical equipment (including instrumentation and control, and certain accident monitoring Equipment).	3.11.7
3.11(8)	The COL Applicant is to provide the site-specific mechanical equipment requirements.	3.11.7
3.11(9)	Optionally, the COL Applicant may revise the parameters based on site-specific considerations.	3.11.7

In the above COL Information item 3.11(4), the COL applicant is to describe periodic tests, calibrations, and inspections to be performed during the life of the plant, which verify the identified equipment remains capable of fulfilling its intended function. In the discussion of EQ-related COL information items in DCD Tier 2, Section 3.11.2, the applicant requires a rigorous, periodic inspection, test and calibration program during the life of the plant to be implemented to verify that systems and components remain operational.

MUAP-08015, Revision 1, Section 4.1.1, “Mild Environment,” states: Compliance by the licensee (owner) with 10 CFR 50.65, "Requirements for monitoring the effectiveness of maintenance at nuclear power plants," and associated guidance in RG 1.160 are considered sufficient to provide reasonable assurance.

In **RAI 511-3739, Question 03.11-26**, the staff requested that the applicant revise MUAP-08015, Revision 1, Section 4.1.1, to state how specific maintenance requirements provided by vendors and determined by engineering judgment (periodic tests, calibrations, and inspections) for EQ condition monitoring and preventive maintenance activities should provide reasonable assurance that the qualified design life has not been reduced and remains capable of fulfilling its intended function.

In its supplemental response to **RAI 511-3739, Question 03.11-26,** dated June 25, 2010, the applicant pointed out that the wording of Section 4.1.1 of MUAP-08015, Revision 1, for mild environment is intended to indicate that the on-going verification of continued qualification of qualified components is an operational concern and will be addressed in the COL applicant's operational EQ program in Chapter 13.4, "Operational Program Implementation." In addition, the applicant pointed out that the wording in Section 4.1.1 of MUAP-08015, Revision 1, concerning 10 CFR 50.65 and RG 1.160 are consistent with SRP Section 3.11, SRP Acceptance Criterion 15. The staff has reviewed the applicant's response and concurs with the applicant that this issue should be addressed under their operational program implementation. Accordingly, **RAI 511-3739, Question 03.11-26, is resolved.**

In **RAI 880-6142, Question 03.11-46,** the staff requested the applicant to revise MUAP-08015, Section 11 to refer to an "Operational," equipment qualification program instead of an "Operating" EQ program and to make corresponding changes throughout the report.

In its response to **RAI 880-6142, Question 03.11-46**, dated March 23, 2012, the applicant agreed to revise in the next revision to MUAP-08015, Revision 1. Pending verification that MUAP-08015 has been revised to include the changes discussed above, **RAI 880-6142, Question 03.11-46, is being tracked as a Confirmatory Item.**

In COL Information Item 3.11(6) listed above, the COL applicant is to qualify site-specific electrical and mechanical equipment (including instrumentation and control, and certain accident monitoring equipment) using an "equivalent qualification process to that delineated for the US-APWR Standard Plant."

In DCD Tier 2, Section 3.11, an equivalent qualification process has been used for qualifying equipment subject to DCD Tier 2, Sections 3.11.4, 3.11.5, "Estimated Chemical and Radiation Environment," and 3.11.6. All site-specific equipment will be qualified by using the equivalent qualification process "to that delineated for the US-APWR standard plant." By contrast, equipment subject to DCD Tier 2, Sections 3.11.5.1, "Chemical Environment," and 3.11.5.2, are to be qualified "pursuant to the implementation of the US-APWR EQ program."

In **RAI 511-3739, Question 03.11-27**, the staff requested that the applicant explain the difference in equipment qualification that is performed by the so-called "equivalent qualification process" or "the US-APWR EQ program," and asked to provide details of what parameters are used to establish the equivalency in the process and to identify where the equivalent qualification process is defined or explained.

In its response to **RAI 511-3739, Question 03.11-27**, dated June 25, 2010, the applicant stated that the expression "equivalent" qualification process used in this context does not imply a different US-APWR EQ process, instead revised the equivalent qualification process to denote using a qualification process that is “equivalent to that delineated for US-APWR Standard plant as described in MUAP-08015, Revision 1,” that complies with NRC regulatory requirements and guidance and all applicable industry guidance. The staff has confirmed that DCD Tier 2, Revision 3, was revised as committed in the above RAI response. Accordingly, the staff finds that the applicant has adequately addressed this issue and, therefore, the staff considers **RAI 511-3739, Question 03.11-27, to be resolved.**

Pending the resolution of the confirmatory items in this section, the staff finds the above listing of COL information Items to be complete. Also, the list adequately describes actions necessary for the COL applicant or licensee. No additional COL information items were identified that need to be included in DCD Tier 2 Table 1.8-2 regarding equipment qualification.

3.11.6 Conclusions

As a result of the eight open items for **RAI 589-4536 Question 03.11-36, 03.11-37, and 03.11-38**; **RAI 650-5093, Question 03.11-39; RAI 805-5915, Question 03.11-41**; and **RAI 880-6142, Questions 03.11-42 and 03.11-43**, **RAI 901-6257, Question 03.11-55)** the staff is unable to finalize its conclusions on Section 3.11 related to EQ of electrical and mechanical equipment, in accordance with NRC regulations.

3.12 Piping Design Review (Related to NUREG-0800, Section 3.12, “ASME Code Class 1, 2 and 3 Piping Systems, Piping Components and Their Associated Supports”)

3.12.1 Introduction

This report provides the staff’s evaluation of the applicant’s design of the piping system for the US-APWR DC. The staff’s evaluation considered the adequacy of the structural integrity as well as the functional capability of piping systems. The review was not limited to only the ASME B&PV Code Class 1, 2, and 3 piping and supports but also included buried piping, instrumentation lines, and the interaction of non-seismic Category I piping with seismic Category I piping.

The applicant is using design acceptance criteria (DAC) for the piping design. The plan for resolving DAC closure is addressed by ITAAC which are provided in DCD Tier 1, Section 2.3, “Piping Systems and Components.” The staff reviewed the plan and ITAAC to ensure the proposed ITAAC are sufficient to provide reasonable assurance that the plant will be built and operated in accordance with the DC. The evaluation of piping ITAAC and its acceptance is documented in Section 14.3.3 of this report.

3.12.2 Summary of Application

DCD Tier 1: Tier 1 information associated with this section is found in DCD Tier 1, Section 2.3

DCD Tier 2: The applicant has provided a DCD Tier 2 description in Section 3.12, “Piping Design Review,” summarized here in part, as follows:

1. Codes and Standards.

2. Piping Analysis Methods.

3. Piping Modeling Techniques.

4. Piping Stress Analysis Criteria.

5. Piping Support Design Criteria.

ITAAC: The ITAAC associated with DCD Tier 2, Section 3.12 are given in DCD Tier 1, Section 2.3, Items 1, 2, 3, and 4 in Table 2.3-2, “Piping Systems and Components Inspections, Tests, Analyses, and Acceptance Criteria,” which indicates that inspections and analyses will be performed concerning piping systems.

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 for this area of review.

Cross-cutting Requirements (TMI, USI/GSI, Op Ex): There are no cross-cutting requirements for this area of review.

US-APWR Interface Issues Identified in the DCD: See DCD Tier 2, Table 1.8-2, “Compilation of All Combined License Applicant Items for Chapters 1-19.”

Site Interface Issues Identified in the DCD: See DCD Tier 2, Table 1.8-1, “Significant Site Specific Interfaces with the Standard US-APWR Design.”

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

3.12.3 Regulatory Basis

The relevant requirements of the Commission’s regulations for this area of review, and the associated acceptance criteria, are given in Section 3.12, “ASME Code Class 1, 2, and 3 Piping Systems, Piping Components and Their Associated Supports,” Revision 0, issued March 2007, of NUREG-0800, the SRP, and are summarized below. Review interfaces with other SRP sections also can be found in Section 3.12 of NUREG-0800.

1. 10 CFR 50.55a and GDC 1, as they relate to piping systems, pipe supports, and components being designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with the importance of the safety function to be performed.

2. GDC 2 and 10 CFR Part 50, Appendix S, with regard to design transients and resulting load combinations for piping and pipe supports necessary to withstand the effects of earthquakes combined with the effects of normal or accident conditions.

3. GDC 4, with regard to piping systems and pipe supports important to safety, being designed to accommodate the effects of, and to be compatible with, the

environmental conditions of normal operation, maintenance, testing, and postulated accidents, including LOCAs. These SSCs must also be appropriately protected against dynamic effects.

4. GDC 14, with regard to the RCPB of the primary piping systems being designed, fabricated, constructed, and tested to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture.

5. GDC 15, with regard to the RCSs and associated auxiliary, control, and protection systems being designed with sufficient margin to assure that the design conditions of the RCPB are not exceeded during any condition of normal operation, including AOOs.

6. 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 has been constructed and will operate in accordance with the DC, the provisions of the Atomic Energy Act of 1954, and the NRC's regulations.

The NRC established requirements in 10 CFR Part 50 to ensure the pressure boundary leakage integrity of the piping components and structural integrity of the pipe supports in nuclear power plants. Detailed acceptance criteria are given in SRP Section 3.12.

3.12.4 Technical Evaluation

The staff used SRP Section 3.7.2, "Seismic System Analysis," Revision 3, issued March 2007; SRP Section 3.7.3, "Seismic Subsystem Analysis," Revision 3, issued March 2007; SRP Section 3.9.1, "Special Topics for Mechanical Components," Revision 3, issued March 2007; SRP Section 3.9.2, "Dynamic Testing and Analysis of Systems, Structures, and Components," Revision 3, issued March 2007; SRP Section 3.9.3, "ASME Code Class 1, 2, and 3 Components and Component Supports, and Core Support Structures," Revision 2, issued March 2007; and SRP Section 3.12 to evaluate the piping and pipe support design information in the DCD. The staff evaluated the design, materials, fabrication, erection, inspection, testing, and inservice surveillance of piping and pipe supports using the industry codes and standards, RGs, and staff technical reports listed in the SRP. The staff also considers industrial practice and programs during the review process.

3.12.4.1 Codes and Standards

DCD Tier 2, Section 3.12.2, "Codes and Standards," states that the Codes and Standards to be used in the design of piping and pipe supports satisfy the requirements of GDC 1, 2, 4, 14, and 15 as described in DCD Tier 2, Section 3.1, "Conformance with NRC General Design Criteria," and 10 CFR Part 50, Appendix S. The staff evaluated the codes and standards and documented the results of its review in the following sections.

3.12.4.1.1 ASME B&PV Code

DCD Tier 2, Section 3.12.2.1, “American Society of Mechanical Engineers Boiler and Pressure Vessel Code,” states that piping design for the US-APWR uses the 1992 Edition, including 1992 addenda, of the ASME Code, Section III, Division 1, Subsections NB, NC, and ND.

The staff noted that 10 CFR 50.55a(b)(1)(iii) provides that applicants and licensees may use Subarticles NB-3200, NB-3600, NC-3600 and ND-3600 for seismic design of piping, up to and including 1993 Addenda, subject to the limitation specified in paragraph (b)(1)(ii). This limitation in 10 CFR 50.55a(b)(1)(ii) applies to welds with leg size less than 1.09 t_n. In these cases, applicants or licensees , applying the 1989 Addenda through the latest edition and addenda incorporated by reference in 10 CFR 50.55a(b)(1), may not apply subparagraphs NB-3683.4(c)(1) and NB-3683.4(c)(2), or specific footnotes:

- For the 1989 Addenda through the 2003 Addenda, Footnote 11 to Figures NC-3673.2(b)-1 and ND-3673.2(b)-1
- For the 2004 Edition through the 2008 Addenda, Footnote 13 to Figures NC-3673.2(b)-1 and ND-3673.2(b)-1

The staff determined that the applicant’s position to use the 1992 Edition including 1992 Addendum for piping design without specifying this limitation does not comply with 10 CFR 50.55a(b)(1)(ii). The staff also noted that the DCD Tier 2, Section 3.12.2 position is not consistent with the position stated in DCD Tier 2, Table 5.2.1-1, “Applicable Code Addenda for RCS Class 1 Components.” However, the applicant’s position described in DCD Tier 2, Table 5.2.1-1 is acceptable as discussed in Section 5.2.1 of this report. In **RAI 804-5938, Question 03.12-27**, the staff requested the applicant to clarify the difference to assure the piping design is consistent with 10 CFR 50.55a and also address the limitations and modifications described in 10 CFR 50.55a(b)(1).

In its response to **RAI 804-5938, Question 03.12-27**, dated November 11, 2011, the applicant stated that MHI will revise the DCD and provided the DCD mark-up to address 10 CFR 50.55a consistency and limitations and modifications described in 10 CFR 50.55a(b) for the piping system design. The staff compared the DCD mark-up to the limitations and modifications described in 10 CFR 50.55a(b) and determined that the proposed changes are consistent with the requirements of 10 CFR 50.55a. Therefore, the staff finds the response is acceptable. Since the applicant has proposed DCD changes, **RAI 804-5938, Question 03.12-27, is being tracked as a Confirmatory Item**.

3.12.4.1.2 ASME Code Cases

DCD Tier 2, Section 3.12.2.2, “American Society of Mechanical Engineers Code Cases,” states that ASME Code Cases N-122-2, N-318-5, N-391-2, N-392-3, and N-319-3 are applicable for the design of the piping system and the piping supports for the US-APWR and these code cases are acceptable on the basis of RG 1.84, “Design, Fabrication, and Materials Code Case Acceptability, ASME Section III,” Revision 34, issued October 2007. The applicant also states that other ASME code cases may be used in the DC if they are either conditionally or unconditionally approved in RG 1.84. Code cases listed in RG 1.193, “ASME Code Cases Not Approved for Use,” are not to be utilized without the approval of the NRC.

In its response to **RAI 804-5938, Question 03.12-27**, dated November 11, 2011, the applicant also indicated that ASME Code Case N-782 would be added to DCD Tier 2, Table 5.2.1-2, “ASME Code Cases,” as described in the response to **RAI 575-4422, Question 05.02.02.02-7**, dated May 7, 2010, and in its amended response, dated April 26, 2012. In addition, the applicant stated that the following statement would be added to the end of DCD Tier 2, Section 3.12.2.2:

> ASME Code Case N-782 is applicable for the design of piping system and piping support for the US-APWR on the basis of 10 CFR 50.55a(a)(3)(i) as discussed in Table 5.2.1-2 Note(1).

The staff accepted Code Case N-782 and documented its evaluation in Section 5.2.1 of this report. Therefore, the staff finds this portion of the response to **RAI 804-5938, Question 03.12-27**, to be acceptable. Since the applicant has proposed DCD changes, **RAI 804-5938, Question 03.12-27, is being tracked as a Confirmatory Item.**

In Revision 35 of RG 1.84, dated October 2011, the staff endorsed ASME Code Cases N-122-2, N-318-5, N-391-2, N-392-3, and N-319-3. The staff also agrees that other Code cases can be used in the design if they are approved in RG 1.84. Because RG 1.84 is endorsed by the staff as identified in 10 CFR 50.55a, the staff finds that the ASME Code Cases proposed by the applicant for the US-APWR design are acceptable.

3.12.4.1.3 Design Specifications

Section III of the ASME Code requires that a design specification be prepared for Class 1, 2, and 3 components such as pumps, valves, and piping systems. The design specification is intended to become a principal document governing the design and construction of these components and should specify loading combinations, design data, and other design inputs. The Code also requires a design report for ASME Code Class 1, 2, and 3 piping and components.

DCD Tier 2, Section 3.12.2.3, "Design Specification," states that the design specifications and the design reports are to be developed in accordance with the ASME Code, Section III, Division 1. The staff reviewed three piping system design specifications during an audit on August 22 - 30, 2011, in order to verify that the applicant is implementing the design in accordance with the ASME Code requirements. The staff identified areas in which the specifications were inconsistent with commitments made in the DCD as well as areas where the DCD needed to be modified to reflect the design specifications, as discussed below in the review of thermal stratification. The staff communicated these inconsistencies to the applicant for integration at the time the piping design specifications are finalized. The applicant addressed the staff's concerns in the response to **RAI 804-5938, Question 03.12-29**, which is being tracked as a Confirmatory Item and is discussed in Section 3.12.4.4.8 of this report. The staff recognizes that the piping design reports and piping design specifications will not be completed until the piping design is finalized and the plant is constructed, and ITAAC are included in the DCD to address verification of this process. A summary of the staff's audit is available in, "Report of the August 22 - 30, 2011, Audit Regarding the United States - Advanced Pressurized Water Reactor Computer Programs and Piping Described in Design Control Document Section 3.9.1 and Section 3.12," dated November 15, 2012.

On the basis of the applicant's commitment that design specifications and design reports are to be developed in accordance with the ASME Code, Section III, the staff finds that the process being used by the applicant to prepare design specifications to meet the ASME Code requirements acceptable.

3.12.4.1.4 Conclusions Regarding Codes and Standards

On the basis of the foregoing evaluation of DCD Tier 2, Section 3.12.2, pending successful resolution of the confirmatory item, the staff concludes that the piping systems important to safety are designed to quality standards commensurate with their importance to safety. The staff's conclusion is based on the following:

- The applicant satisfied the requirements of GDC 1 and 10 CFR 50.55a by specifying appropriate codes and standards for the design and construction of safety-related piping and pipe supports.

- The applicant identified ASME Codes and Code cases that may be applied to ASME Code Class 1, 2, and 3 piping and pipe supports and which are acceptable to the staff.

3.12.4.2 Piping Analysis Methods

DCD Tier 2, Section 3.12.3, "Piping Analysis Methods," states that seismic analysis for all seismic Category I and non-seismic Category I (seismic Category II and non-seismic) piping systems use methods in accordance with SRP Section 3.7.3, Revision 3. These methods, as discussed below, include the response spectrum method, time history method, or where applicable, the equivalent static load method. Because the seismic analysis methods meet the SRP recommendations, the staff finds this acceptable.

3.12.4.2.1 Experimental Stress Analysis Methods

DCD Tier 2, Section 3.12.3.1, "Experimental Stress Analyses," states that experimental stress analysis methods are not used for the design of piping and supports. The staff finds this acceptable as new, previously unreviewed methods will not be used.

3.12.4.2.2 Modal Response Spectrum Method

DCD Tier 2, Section 3.12.3.2, "Modal Response Spectrum Method," states that the modal response spectrum method consists of either the uniform support motion (USM) or the ISM techniques. To account for the uncertainty in the seismic response spectrum, either the peak broadening method or the peak shifting method is used for the design and analysis of piping systems. The staff evaluated the modal response spectrum method and documented its results in the following sections.

3.12.4.2.2.1 Peak Broadening Method

DCD Tier 2, Subsection 3.12.3.2.1, "Peak Broadening Method," states that the peak broadened ISRS are generated according to RG 1.122, "Development of Floor Design Response Spectra for Seismic Design of Floor-Supported Equipment or Components," Revision 1, issued February 1978, and that the design ISRS are peak broadened by a minimum of ±15 percent.

SRP Section 3.7.2, Paragraph II.5.C.(3a), provides the acceptance criteria for development of ISRS and states that the ISRS are smoothed and broadened in accordance with the provisions of RG 1.122 to account for uncertainty.

Because the applicant's peak broadening method is in accordance with the SRP recommendation, the staff finds this acceptable.

3.12.4.2.2.2 Peak Shifting Method

DCD Tier 2, Subsection 3.12.3.2.2, "Peak Shifting Method," states that the peak broadening may become an overly conservative method in piping design. As an alternative, peak shifting may be considered. Peak shifting considers a minimum of ±15 percent uncertainty in the peak structural frequencies. In this section, the applicant also provided its procedure. The staff reviewed the applicant's peak shifting method. The staff finds that the applicant's alternative method follows the alternative method of ASME Code, Section III, Appendix N, Subsection N-1226.3. The staff approved this peak shifting method previously as discussed in RG 1.84, Revision 34. On the basis of past precedent, the staff finds this acceptable.

3.12.4.2.3 Uniform Support Motion Method

Revision 0 of DCD Tier 2, Subsection 3.12.3.2.4, "Modal Combination," stated that the response from high frequency modes (i.e., modes with frequencies greater than the ZPA cutoff frequency) must be included in the response of the piping system, if it results in an increase in the dynamic response of more than 10 percent. The staff noted that RG 1.92, "Combining Modal Responses and Spatial Components in Seismic Response Analysis," Revision 2, issued July 2006, Regulatory Position C1.4.1 states that the 10 percent criteria for increase in the dynamic response is non-conservative and should not be used. In **RAI 465-3382, Question 03.12-20**, the staff requested the applicant to demonstrate that this criterion is adequate or to modify this criterion.

In its response to **RAI 465-3382, Question 03.12-20**, dated November 18, 2009, the applicant deleted the qualifying statement "if it results in an increase in the dynamic response of more than 10 percent" and replaced the language in the DCD with "The response from high frequency modes must be included in the response of the piping system." The staff found that the applicant's revised position is in conformance with Regulatory Position C.1.4.1 of R.G. 1.92. On this basis, the staff finds the response acceptable. The staff confirmed the response was incorporated into DCD Revision 3. Accordingly, **RAI 465-3382, Question 03.12-20, is resolved**. Therefore, the staff finds the applicant modal combination approach acceptable.

DCD Tier 2, Subsection 3.12.3.2.3, "Uniform Support Motion," states that piping systems supported by structures located at multiple elevations within one or more buildings may be analyzed using USM. This analysis method applies a single set of spectra at all support locations, which envelopes all of the individual response spectra for these locations. The enveloped response spectrum is developed and applied in the two mutually perpendicular horizontal directions and the vertical direction. SRP Section 3.7.3, Paragraph II.9 indicates that the USM method is a conservative and acceptable approach for analyzing component items supported at two or more locations to calculate the maximum inertial response of the component. Therefore, the staff finds the USM method is an acceptable method.

DCD Tier 2, Subsection 3.12.3.2.4 states that guidance on combining the individual modal results due to each response spectrum in a dynamic analysis is provided in RG 1.92, Revision 1. DCD Tier 2, Subsection 3.9.2.2.5, "Three Components of Earthquake Motion," also states that for piping analysis, the three sets of mutually orthogonal components of earthquake motion are combined by the SRSS method per RG 1.92, Revision 1. DCD Tier 2, Subsection 3.12.3.2.5, "Directional Combination," states that the responses due to each of the three spatial input components of motion are combined using the SRSS method as provided in Regulatory Position C2.1 of RG 1.92, Revision 1. Because the applicant's position is in conformance with staff's recommendation in RG 1.92, the staff finds this acceptable.

DCD Tier 2, Subsection 3.12.3.2.4, states that the PIPESTRESS computer program is used for analyzing most of the piping systems. This program uses the left-out-force (LOF) method in order to calculate the effect of the high frequency rigid modes. The LOF method of the PIPESTRESS computer program has been approved by the NRC staff and the evaluation of the acceptance is documented in Section 3.12.5.6 of NUREG-1793, "Final Safety Evaluation Report Related to Certification of the AP1000 Standard Design," issued September 2004. On this basis, the staff finds this acceptable.

3.12.4.2.3.1 Seismic Anchor Motions

SRP Section 3.9.2, Subsection II.2.G identifies acceptable methods to obtain the maximum relative support displacements for SAMs. The staff noted that Revision 0 of DCD Tier 2, Subsection 3.12.3.2.6, "Seismic Anchor Motions," did not provide the method in determining the maximum relative movements for SAMs. In **RAI 260-2023, Question 03.12-6**, the staff requested the applicant to describe the method for obtaining the maximum relative support displacements.

In its response to **RAI 260-2023, Question 03.12-6**, dated April 17, 2009, the applicant responded by revising DCD Tier 2, Section 3.12.3.2.6 to state that for piping supported by a single concrete building, the SAMs at all elevations above the foundation basemat are considered to be in phase and that support movements relative to the foundation basemat are used in the analysis. The applicant provided alternatives for when this method cannot be justified or when supports are attached to different structures or a method where data can be used. The staff reviewed the revision against the acceptance criteria provided in SRP Section 3.9.2 and concludes that the revision is in conformance with SRP Section 3.9.2, Subsection II.2.G. The staff also verified that the revision was included in the DCD. Therefore, the staff finds the response acceptable. Accordingly, **RAI 260-2023, Question 03.12-6, is resolved**.

3.12.4.2.4 Individual Support Motion Method

Revision 0 of DCD Tier 2, Section 3.12.3.3, "Response Spectra Method (or Independent Support Motion Method)," stated that the responses caused by each support group are combined by the SRSS method and the modal and directional responses are then combined as discussed in DCD Tier 2, Section 3.12.3.2. DCD Tier 2, Subsection 3.12.3.2.4 stated that a 10 percent grouping method is used for combining the responses of closely spaced modes as delineated in Regulatory Position C1.1 of RG 1.92, Revision 1. The staff noted that the modal and spatial combination methods described in RG 1.92 apply only when using the USM method for response spectrum analysis of multi-supported systems and cannot be used for the ISM method. In **RAI 260-2023, Question 03.12-5**, the staff requested the applicant to provide technical justification of using RG 1.92 for its ISM method.

In response to **RAI 260-2023, Question 03.12-5**, dated April 27, 2009, the applicant provided revisions to DCD Tier 2, Section 3.12.3.3 and Subsection 3.12.3.2.6 and deleted the references to RG 1.92. The staff's evaluation of the revised information is in the following paragraphs.

The revised DCD Tier 2, Section 3.12.3.3 states “ISM may be used when piping systems are supported by multiple support structures or at multiple levels within a structure. The supports are divided into support groups. Each support group is made up of supports that have similar time-history input. Each support group is considered to be in a random-phase relationship to the other support groups. The responses caused by each support group are combined by the absolute summation method. The modal and directional responses are then combined as discussed in Section 2 of NUREG-1061, Volume 4.”

SRP Section 3.12, Subsection II.B.xiii states that “[t]he acceptance criteria provided in NUREG-1061, Volume 4 are applicable for independent support motion [ISM] analysis.” SRP Section 3.7.3, Subsection II.9 states that, “[g]uidance and criteria for the use of the ISM method is given in NUREG-1061, Section 2, Volume 4. If the ISM method is utilized, all of the criteria presented in NUREG-1061 related to the ISM method must be followed.” NUREG-1061, "Report of the U.S. Nuclear Regulatory Commission Piping Review Committee; Volume 4: Evaluation of Other Loads and Load Combinations," issued December 1984, states that group responses for each direction should be combined by the absolute sum method and modal and directional response should be combined by the SRSS method without considering closely spaced frequencies.

The revised DCD Tier 2, Section 3.12.3.2.6, states that “[t]he results of ISM floor response spectra analysis and SAM analysis are combined by the SRSS rule for use in various load combinations in the design and/or analysis of pipe support and piping. The relative displacements between groups are assumed to move 180 degrees out-of-phase as specified in NUREG-1061.” NUREG-1061 indicates that ISM dynamic inertia response and SAM response should be combined by the SRSS method.

The staff reviewed the revised ISM method against the acceptance criteria provided in the SRP Section 3.7.3, Subsection II.9. Because the applicant’s ISM method is in conformance with NUREG-1061, Section 2, Volume 4 as recommended by the SRP, the staff finds that the applicant’s ISM method is acceptable. Therefore, the staff finds the RAI response acceptable. Accordingly, **RAI 260-2023, Question 03.12-5, is resolved**.

3.12.4.2.5 Time History Method

DCD Tier 2, Section 3.12.3.4, “Time-History Method,” states that seismic analysis of piping systems is not performed using the time-history method. However, DCD Tier 2, Appendix 3C, “Reactor Coolant Loop Analysis Methods,” indicates that RCL piping dynamic analysis is performed using time-history direct integration, the time-history modal, or response spectra methods. In **RAI 804-5938, Question 03.12-26**, the staff requested the applicant to clarify the difference by revising DCD Tier 2, Section 3.12.3.4. In its response to **RAI 804-5938, Question 03.12-26**, dated September 19, 2012, the applicant responded that “DCD Tier 2, Subsection 3.12.3.4, applies to piping other than reactor coolant loop (RCL) piping, as described in Subsection 3.12.1 which states: ‘The design of the RCL piping is covered by Subsection 3.9.3.’ Thus, there is no inconsistency to state in Subsection 3.9.3.1.1 that RCL piping dynamic analysis is performed using time-history direct integration, the time-history modal, or response spectra methods.” The staff determined that the part of the response regarding DCD Tier 2, Section 3.12.3.4 was acceptable since the applicant clarified that the different methods are being applied to different piping.

DCD Tier 2, Figure 3.8.3-2, “SG Support System,” showed the SG lower support structure drawing. Two supports (parallel to hot leg) of SG lower lateral support structure functioned as non-linear one-directional supports. In general, non-linear dynamic analysis is analyzed using time-history direct integration method. In general, the response spectra method or time-history modal method is used for linear elastic dynamic analysis. In **RAI 804-5938, Question 03.12-26**, the staff requested the applicant to provide the basis for using time-history modal, or response spectra methods in the non-linear stiffness system. The staff also requested the applicant to clarify which method is used for the RCL piping analysis and identify the correct method in the DCD. In its response to **RAI 804-5938, Question 03.12-26**, the applicant described both coupled and decoupled models. The structural dynamic analysis using methods described in DCD Tier 2, Section 3.7.2 is a dynamic analysis of a coupled model between building structure and the RCL. A separate RCL dynamic analysis is based on a model with the RCL decoupled from the reactor building and is performed using either the independent support motion method or time-history direct integration analysis method described in DCD Tier 2, Section 3.7.3. Therefore, the time-history modal analysis is not currently being used.

The applicant's response also stated that the seismic and accident design analysis method for RCL system components (i.e. the RCL dynamic analysis) is presented in MUAP-09002, "Summary of Seismic and Accident Load Conditions of Primary Components and Piping," Revision 2, issued December 2010. The applicant also stated that the seismic input is obtained from the coupled model of the RCL, R/B, PCCV, and CIS (i.e. the structural dynamic analysis), considering the dynamic effects of an SSE and the effects of SSI is also considered for the subgrade profiles described in Subsection 3.7.1. The response does provide sufficient clarification to address the staff's concern since it clarifies the methods to be applied for the structural dynamic analysis and RCL dynamic analysis. However, the use of the response spectra analysis method to evaluate non-linear SG supports may require additional iterations by considering one case with SG support restrained in both directions and one case without SG support at all to be conservative. In the applicant's letter, "Revised Design Completion Plan for US-APWR Piping Systems and Components," dated December 7, 2012, the applicant stated that the design input to the piping will require a revision to incorporate a seismic analysis methodology update and design enhancements. The revised design loads for the piping will be presented in a revised MUAP-09002. Based on the above, the staff has determined that this RAI cannot be resolved until the revision to MUAP-09002 is submitted and reviewed along with corresponding changes to DCD Tier 2, Sections 3.7.1, 3.7.2, and 3.7.3. Therefore, **RAI 804-5938, Question 03.12-26, is being tracked as an Open Item**.

DCD Tier 2, Section 3.7.3.1, "Seismic Analysis Methods," provides descriptions for the seismic analysis methods. The staff evaluated the time-history seismic analysis methods and documented their acceptance in Section 3.7.3 of this report.

3.12.4.2.6 Inelastic Analysis Methods

DCD Tier 2, Section 3.12.3.5, "Inelastic Analyses Methods," states that inelastic analyses methods are not used to qualify piping for the US-APWR design. If inelastic analysis methods are used for the piping design, the applicant has to submit sufficient detail for staff approval. Because inelastic analysis methods are not used, this is acceptable to the staff.

3.12.4.2.7 Small Bore Piping Methods

DCD Tier 2, Section 3.12.3.6, "Small-Bore Piping Method," states that for small-bore piping, either the equivalent static load method or the modal response spectrum method is used. The modal response spectrum method is evaluated in Section 3.12.4.2.2 above. The applicant states that the equivalent static load method is consistent with the guidelines of SRP Section 3.9.2, Subsection II.2.A(ii) and addressed the description of the method in the SRP. The acceptance criteria provided in SRP Section 3.9.2, Subsection II.2.A are applicable for seismic Category I seismic analysis methods. Because the modal response spectrum method is consistent with the method recommended in SRP Section 3.9.2, Subsection II.2.A.(i) and the equivalent static load method is consistent with the method recommended in SRP Section 3.9.2, Subsection II.2.A.(ii), the staff finds that the small bore piping methods are acceptable.

3.12.4.2.8 Non-seismic/Seismic Interaction

DCD Tier 2, Section 3.12.3.7, "Non-seismic/Seismic Interaction (II/I)," states in the design of the US-APWR, the primary method of protection for seismic Category I piping is its isolation from piping that is not required to be designed to seismic Category I requirements. In cases where it is not possible or practical to isolate the seismic Category I piping, adjacent non-seismic piping is classified as seismic Category II and analyzed and supported such that a SSE event does not cause an unacceptable interaction with the seismic Category I piping. The staff reviewed the applicant's position on the seismic/non-seismic interaction against the acceptance criteria provided in SRP Section 3.9.2. SRP Section 3.9.2, Subsection II.2.K states, "If isolation of the Category I piping system is not feasible or practical, adjacent non-Category I piping should be analyzed according to the same seismic criteria applicable to the Category I piping system." For piping design, the seismic criteria for seismic Category I and seismic Category II are the same. Because the applicant's position is consistent with the staff's recommendation as described in SRP Section 3.9.2, Subsection II.2.K, the staff finds the applicant's position acceptable.

3.12.4.2.9 Category I Buried Piping

DCD Tier 2, Section 3.12.3.8, "Seismic Category I Buried Piping," states that in the design of the US-APWR, there are no seismic Category I buried piping systems. Therefore, the review of this item is not applicable to the US-APWR.

3.12.4.2.10 Conclusions Regarding Piping Analysis Methods

As a result of the open item for **RAI 804-5938, Question 03.12-26**, the staff is unable to finalize its conclusions on Section 3.12 related to piping analysis methods, in accordance with NRC regulations.

3.12.4.3 Piping Modeling Technique

3.12.4.3.1 Computer Codes

DCD Tier 2, Section 3.12.4.1, "Computer Codes," lists the computer programs to be used in the analysis of safety-related piping systems. Piping related computer programs include PIPESTRESS, ABAQUS, ANSYS, RELAP-5, E/PD STRUDL, and STAAD.Pro. DCD Tier 2, Section 3.9.1.2.2, "Program Validations," states that the verification and validation of computer programs is performed in compliance with the established QAP. Error reporting and resolution of the errors are tracked as described in the QAP. The DCD also states that the computer programs are validated using one of the methods, such as hand calculations, known solutions for similar or standard problems, acceptable experimental test results, published analytical results, or results from other similar verified programs. Verification tests demonstrate the capability of a computer program to produce valid results for the test problems encompassing the range of permitted usages defined by the program documentation. The staff finds that the methods used by the applicant for verification and validation of the computer programs are consistent with the guidance of SRP Section 3.9.1. The acceptance of the applicant's QAP is evaluated and documented in Chapter 17 of this report. The staff's review of those computer codes identified in DCD tier 2, Section 3.12.4.1 is documented in Section 3.9.1 of this report. The staff also performed an audit to confirm the acceptability of the computer codes during the site audit on August 22 - 30, 2011. This audit and its conclusions are also discussed further in Section 3.9.1 of this report.

3.12.4.3.2 Dynamic Piping Model

In DCD Tier 2, Section 3.12.4.2, the applicant provided its description for analytical modeling of the piping system analysis. The applicant states that in the dynamic mathematical model, the distributed mass of the system, including pipe, contents, and insulation weight, is represented as lumped masses located at each node, which is designated as a mass point. DCD Tier 2, Section 3.12.4.2, states that the minimum number of degrees of freedom in the model is to be equal to twice the number of modes with frequencies below a pre-selected cut-off-frequency. The staff reviewed the applicant's position against acceptance criteria provided in SRP Section 3.9.2, Subsection II.2. Because the applicant's position meets the acceptance criteria provided in SRP Section 3.9.2, Subsection II.2.A.i.(2), the staff finds this acceptable.

The applicant also provided the formula used to determine the maximum spacing between two successive mass points. The formula is based on a simply supported beam that would produce a natural frequency equal to a preselected cut-off-frequency. The staff confirmed that the formula is derived from a simply supported beam dynamic evaluation. However, the PIPESTRESS User's Manual states that if the distance between mass points exceeds one half of the span length for the simply supported beam with a preselected cut-off-frequency, then additional mass points are generated. The staff noted that there is difference between the DCD and the PIPESTRESS mass spacing criterion. In **RAI 846-6076, Question 03.12-30**, the staff requested the applicant to clarify the criterion for spacing of mass points in the piping model. In its response to **RAI 846-6076, Question 03.12-30**, dated November 18, 2011, the applicant stated that the fifth paragraph of DCD Tier 2 Subsection 3.12.4.2 will be modified to make the spacing between two successive mass points to be less than or equal to one half of the span length for the simply supported beam with a pre-selected cut-off-frequency and provided the mark-up of DCD Tier 2 for the changes. On the basis of the changes doubling the number of degrees of freedom, the applicant's position is consistent with the position recommended in SRP Section 3.9.2, Subsection II.2.A.i.(2). Therefore, the staff finds this acceptable. Since the applicant has proposed DCD changes, **RAI 846-6076, Question 03.12-30, is being tracked as a Confirmatory Item**.

The applicant states that the mass contributed by the support is included in the analysis when it is greater than 10 percent of the total mass of the adjacent pipe span. The staff reviewed the above position and determined that the mass effect position meets the position recommended in SRP Section 3.7.2, Subsection II.3.B.iii. Therefore, the staff finds this acceptable.

3.12.4.3.3 Piping Benchmark Program

DCD Tier 2, Section 3.12.4.3, "Piping Benchmark Program," states that piping benchmark problems included in NUREG/CR-1677, Volume 1, "Piping Benchmark Problems," issued August 1980, and NUREG/CR-1677, Volume 2, "Piping Benchmark Problems," issued August 1985, are used to validate the PIPESTRESS computer code used in piping stress analysis. In addition, three piping benchmark problems from NUREG/CR-6414, "Piping Benchmark Problems for the Westinghouse AP600 Standardized Plant," issued August 1996, are also used to validate the PIPESTRESS computer code. SRP Section 3.12, Subsection II.B.iii provides that the computer programs should be benchmarked with the appropriate NRC benchmarks. The staff reviewed the computer code PIPESTRESS validation documents during the previously mentioned August 22 - 30, 2011, onsite audit. The staff confirmed that the applicant performed 11 piping benchmark problems included in NUREG/CR-1677 and three piping benchmark problems included in NUREG/CR-6414. Because the piping benchmark programs are in conformance with SRP Section 3.12, Subsection II.B.iii, the staff finds this acceptable.

3.12.4.3.4 Decoupling Criteria

DCD Tier 2, Section 3.12.4.4, "Decoupling Criteria," states that branch lines and instrument connections may be decoupled from the analysis model of a larger run of piping provided that either the ratio of the branch pipe mean diameter to the pipe run mean diameter (*Db/Dr*) is less than or equal to 1/3, or the ratio of the moments of inertia of the two lines (*Ib/Ir)* is less than or equal to 1/25. The pipe size limitations used for decoupling is widely adopted by the industry and has been approved in many nuclear plants. Therefore, the staff finds the pipe size limitations used for decoupling is acceptable.

DCD Tier 2, Section 3.12.4.4 also states that the pipe run seismic analysis is performed without the decoupled branch. However, the mass effect is considered when the mass of half the span of the branch pipe is greater than 10 percent of the mass of the pipe run span. The staff reviewed the above position and determined that the mass effect position is consistent with the position recommended in SRP Section 3.7.2, Paragraph II.3.B.iii.

Revision 0 of DCD Tier 2, Section 3.12.4.4, stated that if the amplified response spectra at the branch connection point cannot be developed, the response spectra applicable to the connection point or the nearest restraints on the pipe run are used. In **RAI 260-2023, Question 03.12-7**, the staff requested the applicant to justify why non-amplified response spectra can be used for the branch line analysis. In its response to **RAI 260-2023, Question 03.12-7**, dated April 17, 2009, the applicant stated that the response at the connection point is influenced by the input floor response spectra used for the pipe run analysis and will not differ significantly from the floor response spectra for the elevation in the near vicinity. The staff determined that the applicant did not address the amplified response spectra issue.

The staff closed, as unresolved, **RAI 260-2023, Question 03.12-7** and in follow-up **RAI 465-3382, Question 03.12-23,** the staff requested the applicant to explain why the response spectra will not differ significantly or will not be amplified by the run piping. In its response to **RAI 465-3382, Question 03.12-23**, dated December 2, 2009, the applicant proposed to revise DCD Tier 2, Section 3.12.4.4 to reflect that if amplified response spectra at the joint point cannot be developed, modeling of the branch pipe will be included in the main run analysis. The applicant also stated that the portion of the branch pipe included in the analysis ends at either (a) the first anchor (including equipment nozzle or containment penetration) or (b) four seismic supports in each of the three perpendicular directions. In case of option (b), the overlapping method of NUREG/CR-1980, "Dynamic Analysis of Piping Using the Structural Overlap Method," issued March 1981, is used. The staff reviewed the applicant's position and confirmed the DCD revision. Because the applicant's position is in conformance with the staff's recommended position in NUREG/CR-1980, the staff finds this acceptable. Therefore, the staff finds the RAI response acceptable. Accordingly, **RAI 465-3382, Question 03.12-23, is resolved**.

3.12.4.3.5 Conclusions Regarding Piping Modeling Technique

The staff concludes that the applicant has met GDC 1 by providing information that demonstrates the applicability and validity of the design methods and computer programs used for the design and analysis of seismic Category I piping designated as ASME Code Class 1, 2, and 3, and by having design control measures that are acceptable for ensuring the quality of its computer programs.

3.12.4.4 Piping Stress Analysis Criteria

3.12.4.4.1 Seismic Input

DCD Tier 2, Section 3.12.5.1, "Seismic Input Envelope vs. Site-Specific Spectra." states that the development of floor response spectra for the US-APWR design is described in DCD Tier 2, Section 3.7.2.5, "Development of Floor Response Spectra." Per COL Information Item 3.12(2), if any piping is routed in tunnels or trenches in the yard, the COL Applicant is to generate site-specific seismic response spectra, which may be used for the design of these piping systems. SRP Section 3.7.2, Subsection II.5 states that RG 1.122 describes the methods generally acceptable to the staff for developing the two horizontal and the vertical in-structure response spectra from the time history motions resulting from the dynamic analysis of the supporting structure. DCD Tier 2, Section 3.7.2.5 indicated that the response spectra are generated for the three components of an earthquake by SRSS, following the general guidance of RG 1.122 for frequency up to 100 Hz. The acceptance of the development of floor response spectra is evaluated in Section 3.7.2 of this report.

3.12.4.4.2 Design Transients

DCD Tier 2, Section 3.12.5.2, "Design Transients," states that ASME Code, Section III, Class 1 piping system and support component experience the RCS transients identified in DCD Tier 2, Table 3.9-1, "RCS Design Transients." On the other hand, Class 1 piping experiences the specific transients caused by the flow injection or discharge through this piping. These transients are listed in DCD Tier 2, Table 3.12-6, "Design Transients of Reactor Coolant Piping Branch Connections." The staff reviewed the RCS transients identified in DCD Tier 2, Table 3.9-1 and documented its evaluation and acceptance in Section 3.9.1 of this report. The staff also reviewed the transients listed in DCD Tier 2, Table 3.12-6 against the acceptance criteria provided in SRP Section 3.9.1. The staff reviewed the list of transients and the number of events for each transient provided by the applicant and compared this to the same information on similar and previously licensed applications. Based on the comparison with the previously licensed PWR plants, the staff has determined that the list of transients is adequately defined. The staff also compared the number of events with the number of events of the RCS transients and concluded that the number of events is properly defined. On this basis, the staff determined that design transients defined in DCD Tier 2, Table 3.12-6 are acceptable.

3.12.4.4.3 Loadings and Load Combinations

DCD Tier 2, Section 3.12.5.3, "Loadings and Load Combination," provides loading conditions and load combinations. The applicant states that using the ASME Code, Section III, NB/NC/ND-3600 stress equations, piping system stresses is calculated for various load combinations in DCD Tier 2, Table 3.12-2, "Loading Combinations for ASME Code, Section III, Class 1 Piping," and DCD Tier 2, Table 3.12-3, "Loading Combinations for ASME Code, Section III, Class 2 and 3 Piping." The staff reviewed DCD Tier 2, Section 3.12.5.3 against the acceptance criteria provided in SRP Section 3.9.3, Subsection II.1, which states that the acceptability of the load combination is judged by comparison with positions stated in SRP

Section 3.9.3, Appendix A, and with appropriate standards acceptable to the staff, developed by professional societies and standards organizations. DCD Tier 2, Table 3.12-2, Note 1 states that dynamic loads are combined considering timing and causal relationships and SSE and design-basis pipe break (DBPB) is combined using the SRSS. Because the applicant's dynamic loads combination method conforms to the recommendation provided in NUREG-0484, "Methodology for Combining Dynamic Responses," Revision 1, issued May 1980, the staff finds that the applicant's dynamic loads combination method is acceptable.

DCD Tier 2, Table 3.12-2 Note 2 states that the ASME Code, Section III, NB-3653.6, Equations 12 and 13 need only be calculated for those load sets that do not meet the primary plus secondary stress intensity range (equation 10) allowable. The staff finds this acceptable because the applicant's position is in compliance with the position described in ASME Code, Section III, NB-3653.6.

DCD Tier 2, Table 3.12-2 Note 3 states that the earthquake loads used in Level B stress intensity range and alternating stress calculations is taken as 1/3 of the peak SSE loads. The applicant also states that the number of cycles for the earthquake is 300 if the earthquake loads are taken as 1/3 of the peak SSE loads; but the number of cycles is 20 if the earthquake loads are taken as peak SSE loads. The staff reviewed the applicant's position for the determination of the number of earthquake cycles against the acceptance criteria for the determination of the number of earthquake cycles provided in SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Design," issued April 2, 1993, and the associated SRM, issued July 21, 1983. Because the applicant's position conforms to the recommendation provided in SECY-93-087, the staff finds this acceptable.

DCD Tier 2, Table 3.12-2 Note 4 states that the DCD Tier 2, Table 3.12-2 stress limit follows the requirements of ASME Code, Section III. The staff confirmed that the stress limit is in compliance with ASME Code, Section III.

DCD Tier 2, Table 3.12-2 Note 5 provides the design stress limit for the SAM condition. Because the stress limit for the SAM condition has been included in NB-3656 in the 2002 Addendum of ASME Code, Section III, which has been incorporated by reference in 10 CFR 50.55a, the staff finds this acceptable.

The staff also compared the listed condition loadings, equations and stress limits of the DCD Tier 2, Tables 3.12-2 and 3.12-3 with the condition loadings, equations and stress limits of ASME Code, Section III and concluded that the applicant's position is in compliance with the requirements of ASME Code, Section III. On this basis, the staff finds this acceptable.

Per COL Information Item 3.12(3), if the COL Applicant finds it necessary to lay ASME Code, Section III, Class 2 or 3 piping exposed to wind or tornado loads, then such piping must be designed to the plant design-basis loads. Because this piping is designed to the plant design-basis loads, the staff finds this acceptable.

3.12.4.4.4 Damping Values

DCD Tier 2, Section 3.12.5.4, "Damping Values," states that the damping values used in the seismic analysis of the piping systems is dependent upon the method of seismic analysis used. The damping value used for the SSE is four percent, which is consistent with Table 3 of the RG 1.61, "Damping Values for Seismic Design of Nuclear Power Plants," Revision 1, issued March 2007. The frequency–dependent damping values of Figure 1 of RG 1.61, Revision 1, are used only for the USM technique. Because the applicant defined damping values are consistent with the damping values provided in RG 1.61 Revision 1, the staff concludes that the damping values are acceptable.

3.12.4.4.5 High Frequency Modes

DCD Tier 2, Section 3.12.5.6, “High-Frequency Modes,” states that the PIPESTRESS computer program is used for analyzing most of the piping systems. This program uses the LOF method to calculate the effect of the high frequency rigid modes. The results obtained are treated as an additional modal result from a non-closely spaced last mode, and are combined with other modal responses by the methods described in DCD Tier 2, Section 3.12.5.5, “Combination of Modal Responses.” The staff has reviewed and endorsed the LOF method of the PIPESTRESS computer program. The evaluation of the acceptance is documented in Section 3.12.5.6 of NUREG-1793. On this basis, the staff finds the use of the LOF method acceptable.

The applicant states that site-specific high frequency exceedances of piping systems that are not sensitive to high frequency exceedances in the 25 Hz to 50 Hz range are acceptable as described in DCD Tier 2, Section 3.7, “Seismic Design,” and per COL Information Item 12(4) the COL applicant will screen piping systems that are sensitive to high frequency modes for further evaluation. The staff reviewed the applicant’s position for the high frequency exceedances against the position in COL/DC-ISG-1, “Interim Staff Guidance on Seismic Issues Associated with High Frequency Ground Motion in Design Certification and Combined License Applications,” issued May 19, 2008. Because the applicant’s position follows the recommendation of COL/DC-ISG-1, the staff finds this acceptable.

3.12.4.4.6 Fatigue Evaluation for ASME Code Class 1 Piping

DCD Tier 2, Section 3.12.5.7, “Fatigue Evaluation of ASME Code Class 1 Piping,” states that ASME Code, Section III, Class 1 piping is evaluated for the effects of fatigue caused by various thermal and pressure transients and other cyclic events including earthquakes and thermal stratification. The criteria described in ASME Code, Section III, Class 1, Subsection NB-3653 are used for all piping greater than 1 in. (25 mm) nominal pipe size (NPS) designated as ASME Code, Section III, Class 1. 1 in. (25 mm) NPS and smaller Class 1 piping is analyzed by the rules of Subsection NC. SRP Section 3.12, Subsection II.C.vii states that the acceptance criteria in Section III of the ASME Code are applicable to fatigue evaluation for ASME Code Class 1 piping.

The staff confirmed that ASME Code, Section III, Subsection NB-3630(d)(1) provides that 1 in. NPS or less Class 1 piping is designed in accordance with the design requirements of Subsection NC. The US-APWR piping fatigue evaluation uses the criteria described in ASME Code, which is consistent with NRC regulations and the SRP’s recommendations. Therefore, the staff finds this acceptable.

3.12.4.4.7 Fatigue Evaluation for ASME Code Class 2 and 3 Piping

DCD Tier 2, Section 3.12.5.8, “Fatigue Evaluation of ASME Code Class 2 and 3 Piping,” states that ASME Code, Section III, Class 2 and 3 piping are not explicitly analyzed for calculation of cumulative usage factors in a manner similar to the ASME Code, Section III, Class 1 piping. ASME Code, Section III, Class 2 and 3 piping are evaluated using the requirements of Subsections NC/ND-3611.2, which allow the reduction of allowable stress for thermal expansion stress ranges calculated using the requirements of Subsections NC/ND-3653.2(a) by a stress range reduction factor (*f*) from Table NC/ND- 3611.2(e)-1. The DCD also states that the environmental impact on fatigue of Class 2 and 3 piping follows guidelines established by the NRC at the time of actual analysis. SRP Section 3.12, Subsection II.C.viii states that the acceptance criteria in Section III of the ASME Code are applicable to fatigue evaluation for ASME Code Class 2 and 3 piping. US-APWR piping fatigue evaluation using the criteria described in ASME Code is consistent with NRC regulations and the SRP's recommendations. Therefore, the staff finds this acceptable.

3.12.4.4.8 Thermal Oscillations in Piping Connected to the Reactor Coolant System

Thermal stratification, cycling, and striping (TASCS) are thermal mechanisms that have caused significant damage to power plant pressure boundary components; most commonly fatigue cracking of piping. NRC Bulletin 88-08, "Thermal Stresses in Piping Connected to Reactor Cooling Systems," issued June 22, 1988, requests licensees to identify and evaluate the piping system connected to the RCS susceptible to TASCS to ensure that the piping will not be subjected to unacceptable thermal stresses.

DCD Tier 2, Section 3.12.5.9, "Thermal Oscillations in Piping Connected to the Reactor Coolant System," states that the following US-APWR design approach to address valve leakage is provided as assurance against thermal stratification and thermal oscillation.

1. A double isolation valve configuration successfully restricts leakage, preventing fatigue failure from thermal stratification or oscillation.

2. For a single valve configuration, leakage can be detected by measuring the downstream temperature. Monitoring of downstream temperature is utilized to detect valve leakage. As a result of leakage detection, valve repair can be scheduled, thereby preventing fatigue failure from thermal stratification or oscillation.

3. In the case of a gate valve configuration, HCF could be caused by repeated leaks from the valve gland. Leaks would occur even when double isolation valves are installed in series. By permitting continuous leakage through the valve gland packing by valve disk position adjustment, valve disk expansion and contraction cycle is prevented and cyclic fatigue failure caused by thermal stratification or thermal oscillation is eliminated.

The staff reviewed the applicant's position and issued **RAI 260-2023, Question 03.12-10** to request the applicant to explain how each of these three approaches would ensure mitigating the effects of thermal oscillations in un-isolable piping connected to the RCS.

In its response to **RAI 260-2023, Question 03.12-10**, dated April 17, 2009, the applicant responded that the three approaches to the mitigation of the effects of thermal oscillations induced by leaking valves are outlined and discussed individually as follows:

> Installation of double isolation valves - This approach decreases the possibility of leakage to the downstream of the isolation valves, which results in prevention of thermal stratification or oscillation.
>
> Leakage detection by measuring the downstream temperature for a single valve configuration - In the case of a single isolation valve configuration, leakage detection is

performed by measuring the downstream (from RCS) temperature of the isolation valve. If leakage occurs, the valve can be repaired before occurrence of fatigue failure caused by thermal stratification or oscillation resulting from valve leakage.

> Permitting continuous leakage through the valve gland packing in a gate valve configuration - NRC Bulletin No. 88-08 Supplement 3 describes this approach. If a gland leak occurs at the gate valve closure and the disk position is not appropriate, there is a possibility of cyclic fatigue failure. By permitting continuous gland leakage, the cyclic fatigue phenomenon is eliminated since cyclic expansion and contraction of valve disk does not occur.

The applicant intends to address thermal cycling by proposing measures addressing valve leakage. However, the staff noted that thermal cycling can occur in a bottom-connected line even without valve inleakage, where thermal cycling occurs due to the cyclic penetration and retreat of the swirl flow in the branch line. Therefore, the staff closed as unresolved **RAI 260-2023, Question 03.12-10** and issued follow-up **RAI 465-3381, Question 03.12-17**, requesting the applicant to identify and address thermal cycling for bottom-connected lines.

In its response to **RAI 465-3381, Question 03.12-17**, dated December 2, 2009, the applicant stated that when the stratified surface boundary (end of swirl penetration/cavity flow) occurs at horizontal pipe bends and elbows, there is a possibility of significant thermal fluctuation inducing high-cycle thermal fatigue. As a result, the US-APWR pipe routing is made such that the stratified surface boundaries are not located at horizontal pipe bends and elbows along isolated branch pipes connected to the RCS. The applicant also states that the prediction method for the position of the stratification surface became publicly known at the 10th and 11th International Conference on Nuclear Engineering (refer to ICONE10-23340 and ICONE11-36214). To confirm that there is no stratification surface boundary (top of cavity flow) at pipe bends and elbows migrating horizontally from vertical risers, the licensee will perform temperature measurement on actual installed equipment of isolated branch pipes around RCS during the initial startup test.

Additionally, the applicant identifies the safety-related parts that apply double isolation valves from the RCS and the detection of leakage for these safety-related parts. In its response to **RAI 465-3381, Question 03.12-17**, the applicant also provided a DCD Tier 2, Section 3.12 mark-up by adding the following as the last paragraph in Section 3.12.5.9:

> Furthermore, high temperature water that flows from the RCS pipe run into isolated branch pipes occurs as swirling water, referred to as cavity flow induced by the high temperature pipe. In addition, there is a phenomenon in which stratification occurs at the surface boundary between the high temperature water entering the branch pipe and the low temperature water already in the branch. When the stratified surface boundary (top of cavity flow) occurs at horizontal pipe bends and elbows, there is a possibility of significant thermal fluctuation inducing high cycle thermal fatigue at this point. Therefore, the piping is routed such that a stratified surface boundary does not occur at horizontal pipe bends and elbows.

The staff noted that the applicant stated that the piping is routed such that a stratified surface boundary does not occur at horizontal pipe bends and elbows. However, the applicant did not provide discussion of the piping routing methodology in the DCD mark-up. Further, in its **RAI 465-3381, Question 03.12-17** response, the applicant stated that to verify that there is no stratification surface boundary (top of cavity flow) at pipe bends and elbows migrating horizontally from vertical risers, the licensee will perform temperature measurement on actual equipment of isolated branch pipes around RCS during the initial startup test. However, the staff reviewed DCD Tier 2, Chapter 14, "Verification Programs," but did not find any that mention an activity related to the stratification surface boundary verification.

The staff closed, as unresolved, **RAI 465-3381, Question 03.12-17** and in follow-up **RAI 804-5938, Question 03.12-29**, the staff requested the applicant to provide its piping stratification prediction method. In general, U.S. utilities followed EPRI Material Reliability Program (MRP) guidance MRP-146, **"**Material Reliability Program: *Management of Thermal Fatigue in Normally Stagnant Non-Isolable Reactor Coolant System Branch Lines,"* issued June 2005, which provides a model for predicting and evaluating thermal cycling for PWR stagnant lines. The applicant provided research results from ICONE 10-23340, "Study on Relationship Between Thermal Stratification and Cavity Flow in a Branch Pipe With a Closed End", and ICONE 11-36214, "Study on Relationship Between Thermal Stratification and Cavity Flow in a Branch Pipe With a Closed End (Development of Evaluation Method for Penetration Length of Cavity Flow in Branch Pipe Consisted of Vertical and Horizontal Pipe)". However, ICONE 11-36214 identified that the predictive accuracy of the evaluation method is about $\pm$ 20 percent from the limited experiment data. The staff further stated in RAI 804-5938, Question 03.12-29 that if the applicant's model is not consistent with the MRP-146 model, which has been shown through benchmarking results to be effective in predicting the location of thermal cycling in a branch line attached to the RCL, the applicant should identify the difference and provide justification. Further, the staff requested the applicant to provide additional information for the test abstract including stating the standard operating conditions in DCD Tier 2, Chapter 14 that identifies the objective, prerequisites, test method, data required, and acceptance criteria for un-isolable piping connected to the RCS to address NRC Bulletin 88-08. In general, stratification verification monitoring activity shall be the COL applicant's responsibility. However, this activity has not been listed as a COL action item in the DCD. The staff requested the applicant to clarify the responsibility. If this activity is to be completed by during the COL review, the DCD should be modified to add this activity as a COL item. The staff also requested the applicant to add a table to address detection of leakage of the safety-related valves in DCD Tier 2.

In its response to **RAI 804-5938, Question 03.12-29**, dated November 25, 2011, the applicant provided its justification to prevent the top of the cavity flow from being located at a pipe bend or an elbow. The applicant stated that the ICONE-10-23340 and ICONE-11-36214 essentially explain the empirical knowledge contained in the Japan Society of Mechanical Engineers (JSME) S 017-2003 "Guideline for Evaluation of High-Cycle Thermal fatigue of a Pipe." Because ICONE papers and MRP-146 all use the empirical results to determine the top of the cavity flow, the staff finds that the applicant's pipe routing design based on the JSME guideline to avoid thermal stratification occurrence is acceptable. In this response, the applicant also provided a description of the routing approach and the prediction accuracy of the penetration depth of the cavity flow consideration in detail to ensure that significant thermal cycling does not occur. Because the pipe routing ensures the top of the cavity flow cannot occur at a pipe bend or an elbow, the staff finds this acceptable.

In its response, the applicant also stated that the description of the test method for "confirmation of the top location of the cavity flow" listed in the revised Table 14.2-1 is not a COL action item and would be addressed by the applicant as part the DCD review. The staff evaluated the applicant's description of this test method and documented its acceptance in Section 14.2 of this report.

During the on-site audit performed August 22 - 30, 2011, the staff reviewed the applicant's basis documents and identified that RHR lines do not provide double isolation valves as stated in the DCD. The staff requested the applicant to take action to address the difference. The applicant addressed the staff's concern in the response to **RAI 804-5938, Question 03.12-29**. In its response, the applicant stated that the double isolation valves, assuming leakage from RCS with high temperature to downstream of the valves, do not need to be considered since there is a low probability of thermal stratification. The staff determined that this is acceptable since the RCS pressure is always higher than the pressure in the branch lines during operation except for the out-of-service charging line and auxiliary pressurizer spray line. The US-APWR system design does not have out-of-service charging. Therefore, there is no concern regarding thermal stratification and thermal oscillation. The staff confirmed that the US-APWR does not have any alternate charging lines. Therefore, the staff finds the applicant's response regarding out-of-service charging acceptable. With respect to the auxiliary pressurizer spray line, the applicant stated that the velocities in the spray lines are not sufficient to produce thermal fatigue cycling due to insufficient swirl penetration into the auxiliary spray line. Also, the temperature difference between spray line flow and valve seating leakage is small. Therefore, the auxiliary spray line can be screened out for normal operating conditions. The staff verified that the pipe size of the auxiliary spray and the velocities are not sufficient to produce thermal fatigue cycling and concluded that the applicant's position is acceptable.

In its response, the applicant also provided its mark-up DCD revision to address the staff's concern during the on-site audit. In DCD Tier 2, Section 3.12.5.9, the applicant removed its crediting of a double isolation valve configuration to restrict leakage to prevent fatigue failure from thermal stratification or oscillation. The staff noted that NRC Bulletin 88-08, "Thermal Stresses in Piping Connected to Reactor Coolant System", identified that ECCS cold water leakage into the reactor coolant system would cause thermal stratification. This situation occurred due to the design of the Joseph M. Farley Nuclear Plant sharing the same pump for both charging system and high pressure safety injection system. The charging system pressure is higher than the RCS pressure and the charging system temperature is cooler than the RCS temperature. Therefore, the high pressure safety injection line has a pressure higher than the RCS and the bypass valve leakage caused thermal stratification. The staff reviewed the applicant's system design and determined that the applicant does not use this type of design and the ECCS system pressure is not higher than RCS pressure during operation. Therefore, this situation cannot occur. The staff notes that a double isolation valve configuration cannot prevent fatigue failure from thermal stratification or oscillation caused by swirl penetration. A double isolation valve configuration does not have to be credited for the US-APWR design for valve in-leakage since the valve downstream pressure is less than RCS pressure. Therefore, the staff agrees with the removal of the double isolation valve configuration approach.

In the response in DCD Tier 2, Section 3.12.5.9, the applicant also removed its crediting of valve leakage detection and valve repair to prevent fatigue failure from thermal stratification or oscillation since the applicant was no longer using this approach. In an amended response to **RAI 804-5938, Question 03.12-29**, dated May 8, 2013, the applicant revised the content in DCD Tier 2 Table 3.12-7, "Evaluation from Viewpoint of Valve Seat Leakage," to reflect that it was no longer using double isolation valves or leak detection to prevent fatigue failure. With the exception of the pressurizer auxiliary spray line, the applicant removed the remarks for the evaluation of each piping line since the pressure downstream from a closed valve in the related line is lower than RCS pressure. The staff determined that there is no valve in-leakage due to pressure difference for those lines. Regarding the pressurizer auxiliary spray line, the temperature difference between spray line flow and valve seat leakage flow is small for the pressurizer auxiliary spray line. On this basis, the staff concluded that fatigue failure by valve seat leakage and resulting thermal stratification or thermal oscillation is not a concern. Therefore, the staff agrees that valve leakage detection and valve repair is not needed to prevent fatigue failure and the staff finds the proposed changes to DCD Tier 2, Section 3.12.5.9 and Table 3.12-7 acceptable. Therefore, the amended RAI response is acceptable. Since the applicant has proposed DCD changes, **RAI 804-5938, Question 03.12-29 is being tracked as a Confirmatory Item**.

3.12.4.4.9 Thermal Stratification

DCD Tier 2, Section 3.12.5.10, "Thermal Stratification," states that the structural integrity of the pressurizer surge line of the US-APWR plant is to be assured by performing monitoring activities for the first US-APWR plant. The staff noted that in order to use the first US- APWR initial plant operation to verify that the design transients for the surge line are representative; the applicant has to assure that all US-APWR plants use the same heatup and cooldown procedures and methods. Currently, most of the US plants heatup and cooldown procedures are not the same as the heatup and cooldown procedures used by many Japanese units. In **RAI 742-5703, Question 03.12-25**, the staff requested that the applicant describe how it would ensure that all US-APWR plants will use the same heatup and cooldown procedures and methods.

In its response to **RAI 742-5703, Question 03.12-25**, dated July 8, 2011, the applicant provided its response to the staff's questions. However, the staff reviewed this response to **RAI 742-5703, Question 03.12-25** and determined that the response is not sufficient to resolve this issue for the following reasons:

1. The applicant stated that the pressurizer (PZR) surge line monitoring will be monitored during hot functional testing (HFT). The staff has reviewed DCD Tier 2, Chapter 14 and found no mention of any activity related to PZR surge line stratification monitoring. The applicant should provide additional information including a test abstract including stating the standard operating conditions in DCD Tier 2, Chapter 14 that identifies the objective, prerequisites, test method, data required, and acceptance criteria for surge line thermal monitoring that comports with NRC Bulletin 88-11, "Pressurizer Surge Line Thermal Stratification," issued December 20, 1988.

2. In general, the surge line monitoring activity will be the COL applicant's responsibility. However, this activity has not been listed as a COL action item in the DCD. The staff requested the applicant to clarify the responsibility. If this activity is to be completed by the COL applicant, the DCD should be modified to add this activity as COL item.

3. The applicant stated that it is normal practice for the RCS heatup/cooldown rate to be limited in the plant licensing documents (TS). This in turn is subsequently made part of the generic US-APWR operating procedures as well as plant specific operating procedures. The staff noted that limiting the plants to the same heatup/cooldown rate does not provide a sufficient basis for stating the pressurizer surge line is subject to the same transients/cycles. Surge line stratification cycles/transients are controlled by the RCS and PZR temperature difference, RCP status, and other pertinent parameters. The response did not provide sufficient basis to explain how monitoring the first US-APWR plant would represent all US-APWR plants. There are different ways to heat up and cool

down the RCS with different RCPs and PZR heater operation which significantly impact the system temperature differential and cycles.

4. The markup for the change did not provide enough detail to address this design issue. Specifically, the applicant stated that if the fatigue evaluation result yields non-compliance with the ASME Code, Section III, Items 2 through 4 are to be performed. Items 2 through 4 showed that a program for continuously monitoring for the life of the plant will be used to ensure ASME Code, Section III compliance. However, the applicant did not identify a corresponding COL item. This statement also indicated that the initial fatigue evaluation will not be qualified per the ASME Code. This is not an acceptable position to the staff.

5. The mark-up of the DCD stated, "The outlines of the heatup and cooldown operations are described in the Subsection 5.2.2.2.2.1 and 5.2.2.2.2.2." The staff concluded that the description of the heatup and cooldown operations do not provide information to address the original RAI.

In its amended response to **RAI 742-5703, Question 03.12-25**, dated July 6, 2012, the applicant addressed the staff's comments listed above and provided a mark-up of the DCD.

The applicant's response to comment 1 stated that a test to evaluate the temperature and deflection of the pressurizer surge line will be part of the HFT program as outlined in Chapter 14 of the DCD. Because the applicant's response addresses the issues described in NRC Bulletin 88-08, the staff finds this acceptable. The staff evaluated the test program and documented its acceptance in Section 14.2 of this report.

The applicant's response to comment 2 stated that monitoring will be performed to demonstrate the satisfactory response of the pressurizer surge line during the first plant HFT and will be continued during the first cycle of operation, monitoring during HFT will be added to Chapter 14 as part of the HFT program, and a COL item for the HFT of the first plant is described in COL 14.2(11) of the DCD. Monitoring during the first cycle operation will be a COL action item for the first US-APWR plant. Because the effect of thermal stratification and thermal striping is considered in the stress and fatigue evaluations and the monitoring program is used to verify the surge line operating characteristics meet the design-basis as described in the DCD, the staff finds this acceptable.

The applicant's response to comment 3 stated that a statement describing limits on pressurizer surge line temperature differences during heatup and cooldown will be added to DCD Subsection 3.12.5.10. The response stated that the temperature difference between the two ends of the pressurizer surge line during heatup and cooldown is 145 °F (80.6 °C). The measurement of thermal stratification temperatures from the HFT surge line performance test will be compared with the value used in the design analysis to confirm the design margin. Because the surge line design analysis can be verified with the test data to ensure the surge line structural integrity, the staff finds this response acceptable.

The applicant's response for comment 4 stated that the monitoring will be continued during the first cycle of operation for the first US-APWR plant. The applicant stated this is COL action item 3.12(5). The applicant proposed adding COL Information Item 3.12(5) as follows:

> The COL holder for the first plant is to perform the pressurizer surge line monitoring subsequent to the COL item 14.2(11).

Because the surge line design analysis can be verified with the data from actual operating conditions to ensure the surge line structural integrity, the staff finds this acceptable.

The applicant's response to comment 5 stated that the DCD statement in question has been replaced by a description of a limit on the pressurizer surge line temperature difference during heatup and cooldown. The staff verified that the applicant clearly defined the temperature difference limit for the heatup and cooldown conditions to provide a design and operating basis to be used in the design analysis. The data from HFT and monitoring operating conditions can also be used to verify the design analysis. On this basis, the staff finds the applicant's approach acceptable.

Therefore, the staff concluded that the RAI response is acceptable. Since the applicant has proposed DCD changes, **RAI 742-5703, Question 03.12-25, is being tracked as a Confirmatory Item**.

3.12.4.4.10 Safety Relief Valve Design, Installation, and Testing

DCD Tier 2, Section 3.12.5.11, "Safety Relief Valve Design, Installation, and Testing," states that the requirements of "Rules for the Design of Safety Valve Installations," ASME Code, Section III Appendix O, are followed in the design and installation of safety and relief valves for overpressure protection. In DCD Tier 2, Section 3.9.3.2, "Design and Installation of Pressure-Relief Devices," the applicant describes analysis requirements for pressure relieving devices when the discharge is directly to the atmosphere (open discharge) and to headers or tanks (closed discharge). Static methods using a conservative dynamic load factor are used for open discharge directly to the atmosphere to evaluate stresses and restraint/support design loads. A static or a time-history analysis is performed on the piping system to evaluate resulting stresses and support/restraint design loads for discharges to vessels or headers. If several relief or safety valves are placed on a common header, the most adverse sequence of valve discharges is used to calculate piping stresses and support design loads. The applicant also states that the reaction forces and moments are based on a dynamic load factor (DLF) of 2.0 unless a dynamic structural analysis is performed to calculate these forces and moments.

SRP Section 3.9.3, Subsection II.2 provides pressure relief device design and installation acceptance criteria which state that the applicant should use design criteria for pressure relief installations specified in Appendix O, ASME Code, Section III, Division 1, "Rules for the Design of Safety Valve Installations," and additional considerations to ensure design conservatism. The staff reviewed the applicant's description in DCD Tier 2, Sections 3.12.5.11 and 3.9.3.2 against the criteria in SRP Section 3.9.3. Because the applicant's design for safety relief valve installation is in conformance with the staff's recommendation in SRP Section 3.9.3, the staff finds the applicant's approach acceptable. The testing of safety relief valves is reviewed and documented in Section 3.9.6 of this report.

3.12.4.4.11 Functional Capability

DCD Tier 2, Section 3.12.5.12, "Functional Capability," states that the functional capability requirements for ASME piping systems that must maintain an adequate fluid flow path to mitigate ASME Code, Section III Level C or D service conditions are shown in DCD Tier 2, Table 3.12-5, "Piping Functional Capability – ASME Code, Section III, Class 1, 2, and 3." These requirements are based on NUREG-1367, "Functional Capability of Piping Systems," issued November 1992. The staff reviewed DCD Tier 2, Section 3.12.5.12 and DCD Tier 2, Table

3.12-5 and confirmed that the applicant's position is consistent with staff's recommendation as described in NUREG-1367. Therefore, the staff finds that US-APWR design requirements are sufficient to ensure maintenance of functional capability.

3.12.4.4.12 Combination of Inertial and Seismic Anchor Motion Effects

DCD Tier 2, Section 3.12.5.13, "Combination of Inertial and Seismic Anchor Motion Effects," states that the inertial effects and anchor movement effects due to an earthquake are analyzed separately. The results from these two separate analyses are combined by the absolute summation method for support design loads and for the fatigue analysis of ASME Code, Section III, Class 1 piping systems. SRP Section 3.9.2, Subsection II.2.G states that the response due to the inertia effect and relative displacement should be combined by the absolute sum method. The staff concluded that the applicant's position is in conformance with staff's recommendation as described in SRP Section 3.9.2. On this basis, the staff finds that the applicant's absolute summation method is acceptable.

3.12.4.4.13 Operating-Basis Earthquake as a Design Load

DCD Tier 2, Section 3.12.5.14, "Operating-Basis Earthquake as a Design Load," states that for the US-APWR piping design, the main earthquake load used is defined in DCD Tier 2, Section 3.7. By virtue of the design criteria used for piping components and supports, this design-basis criterion assures that the SSE controls the seismic design of systems and components. DCD Tier 2, Section 3.2.1, "Seismic Classification," states that the site-independent seismic design of the US-APWR sets the OBE ground motion at 1/3 of the SSE as discussed in DCD Tier 2, Section 3.7.1.1, "Design Ground Motion," which eliminates the requirement for performing explicit design analysis for OBE loads. In accordance with Appendix S to 10 CFR Part 50, when subjected to the OBE ground motion in combination with normal operating loads, SSCs necessary for continued operation without undue risk to the health and safety of the public must remain functional and within applicable stress, strain, and deformation limits.

DCD Tier 2, Section 3.7.1.1 indicates that for fatigue evaluations, based on the OBE defined as less than 1/3 of the SSE, the guidance for determining the number of earthquake cycles for use in fatigue calculations is the same as the guidance provided in the SRM for SECY-93-087 for piping systems. Because the applicant's position is in compliance with Appendix S to 10 CFR Part 50, the staff finds the applicant's position for not considering the OBE load is acceptable. For fatigue evaluation, the applicant's position is in conformance with the SRM for SECY-93-087, which the staff finds acceptable.

3.12.4.4.14 Welded Attachments

DCD Tier 2, Section 3.12.5.15, "Welded Attachments," states that where integral welded attachments to piping are used in restraint design, standard industry practices and ASME Code Cases identified in DCD Tier 2, Section 3.12.2.2 are used. SRP Section 3.12, Subsection II.C.xv states, "Support members, connections, or attachments welded to piping should be designed such that their failure under unanticipated loads does not cause failure at the pipe pressure boundary. The applicant may use code cases for the design of the welded attachments. Acceptable code cases are listed in RG 1.84." The staff confirmed that in RG 1.84, ASME Code Cases N-122-2, N391-2, N-392-3, N-318-5 were endorsed by the staff. Therefore, the staff finds this acceptable.

In **RAI 804-5938, Question 03.12-28**, the staff requested that the applicant explain what kind of standard industry practices were used for integral welded attachments for piping and update the DCD by adding the description of these standard industry practices. In its response to **RAI 804-5938, Question 03.12-28**, dated November 10, 2011, the applicant provided the standard industry practices and mark-up of the DCD Tier 2, Subsection 3.12.5.15.

The response stated that where integral weld attachments to piping are used in restraint design, ASME Code Cases identified in Subsection 3.12.2.2 are used, and for attachments that are outside the range of parameters of the above method, Welding Research Council (WRC) Bulletin 107, August 1965, revised in March 1979, is used when applicable. The response also stated that for cases where the above methods are not applicable, a finite element analysis (FEA) will be used. The staff noted that the integral weld attachments to piping are used for restraint design in ASME Code Cases N-122-2, N-318-5, N-391-2 and N-392-3. The staff confirmed that all these ASME Code Cases are identified in Subsection 3.12.2.2 and all are endorsed by the staff as documented in RG 1.84. The WRC Bulletin 107 method is acceptable to the staff as previously approved. The FEA method to be used for the design analysis is always considered acceptable by the staff. On the basis mentioned above, the staff finds this acceptable. Therefore, the staff concluded that the applicant's response is acceptable. Since the applicant has proposed DCD changes, **RAI 804-5938, Question 03.12-28, is being tracked as a Confirmatory Item**.

3.12.4.4.15 Modal Damping for Composite Structures

DCD Tier 2, Section 3.12.5.16, "Modal Damping for Composite Structures," states that modal damping for composite structures is described in DCD Tier 2, Section 3.7.3.3, "Analysis Procedure for Damping." The acceptance of modal damping for composite structures is evaluated and documented in Section 3.7.3 of this report.

3.12.4.4.16 Temperature for Thermal Analyses

DCD Tier 2, Section 3.12.5.17, "Minimum Temperature for Thermal Analyses," states that the stress-free state of a piping system is defined as the state of the piping at a temperature of 70 °F (21 °C). Piping systems subjected to operating temperatures greater than 150 °F (65.6 °C) are analyzed for the effects due to thermal expansion. DCD Tier 2, Section 3.12.5.3.3, "Thermal Expansion," states that if the piping system does not contain at least one 90-degree bend, then an analysis of thermal expansion is needed. Straight pipe layout without at least one 90-degree bend is avoided when practical. SRP Section 3.12, Subsection II.C.xvii states that the applicant should perform thermal expansion analyses for piping systems that operate at temperatures above or below the stress-free reference temperature. The stress free reference temperature for a piping system is typically defined as a temperature of 70 °F (21 °C). The SRP further states that the applicant should provide justification if thermal expansion analyses are not performed. The staff reviewed the applicant's position for considering 150 °F (65.6 °C) as the dividing line for cold vs. hot piping. The hot piping is analyzed for thermal expansion. The staff acknowledged that this position has been approved and used in many nuclear power plants. Based on these precedents, the staff finds this acceptable.

3.12.4.4.17 Intersystem Loss-of-Coolant Accident

DCD Tier 2, Section 3.12.5.18, "Intersystem Loss-of-Coolant Accident," states that piping systems designed to operate under low pressure conditions may interface with the RCPB. For such piping systems, the minimum wall thickness is calculated per the requirements of ASME

Code, Section III, NB-3640, and stress analysis is performed in accordance with ASME NB/NC/ND-3650 equations using the appropriate RCPB pressure. Because the high RCPB pressure is used in the stress analysis to qualify the low pressure piping system for conservatism, the staff finds this acceptable.

3.12.4.4.18 Effects of Environment on Fatigue Design

DCD Tier 2, Section 3.12.5.7 states that the environmental impact on fatigue of ASME Code, Section III, Class 1 piping follow the guidance delineated in RG 1.207, “Guidelines for Evaluating Fatigue Analyses Incorporating the Life Reduction of Metal Components Due to the Effects of the Light-Water Reactor Environment for New Reactors,” issued March 2007. SRP Section 3.12, Subsection II.C.xix states that the guidance provided in RG 1.207 is applicable to the effects of environment on fatigue design. Because the US-APWR design addresses the effects of environment on fatigue design in conformance with the guidance in RG 1.207, the staff finds this acceptable.

3.12.4.4.19 Conclusions Regarding Piping Stress Analysis Criteria

The staff finds that the US-APWR DCD adequately addresses piping stress analysis criteria and reflects the staff’s position as indicated. Therefore, the staff concludes that the applicant has met the following requirements with regard to piping stress analysis criteria pending successful resolution of the confirmatory items:

- GDC 1 and 10 CFR 50.55a, with regard to piping systems being designed, fabricated, constructed, tested, and inspected to quality standards commensurate with the importance of the safety functions to be performed and with appropriate quality control.

- GDC 2 and 10 CFR Part 50, Appendix S, with regard to design transients and resulting load combinations for piping and pipe supports to withstand the effects of earthquakes combined with the effects of normal or accident conditions.

- GDC 4, with regard to piping systems important to safety being designed to accommodate the effects of, and to be compatible with, the environmental conditions of normal and accident conditions.

- GDC 14, with regard to the RCPB of the primary piping systems being designed, fabricated, constructed, and tested to have an extremely low probability of abnormal leakage, of rapid propagating failure, and of gross rupture.

- GDC 15, with regard to the reactor coolant piping systems being designed with specific design and service limits to assure sufficient margin that the design conditions are not exceeded.

3.12.4.5 Piping Support Design Criteria

3.12.4.5.1 Applicable Codes

DCD Tier 2, Section 3.12.6.1, “Applicable Codes.” provides the piping support design criteria requirements which states that seismic Category I and seismic Category II pipe supports are designed in accordance with Subsection NF of the ASME Code, Section III, 2001 Edition. However, the acceptable stress limits permitted by the non-mandatory Appendix F of Section III of the ASME Code, 2001 Edition are utilized for Service Level D supports. The welding requirements for A500, grade tube steel from AWS D1.1 are utilized. The DCD also provides non-seismic Category design criteria. The staff reviewed the applicant's applicable codes for piping support design against the acceptance criteria provided in SRP Section 3.12, Subsection II.D.i, which states that the design of ASME Code, Section III, Class 1, 2, and 3 piping supports should comply with the design criteria requirements of ASME Code, Section III, Subsection NF. ASME Code, Section III, Subsection NF states that Appendix F is used for the stress limit factors for Service Level D. Because the applicant's support design codes are in conformance with the SRP recommendation, the staff finds this acceptable.

DCD Tier 2, Section 3.12.6.1, states that the ASME Code, Section III, Class 1 linear-type piping supports (excluding snubbers) follow the rule of ASME Code, Section III, Subsection NF as supplemented by the stipulations of Section C (Regulatory Position) of RG 1.124, "Service Limits and Loading Combinations for Class 1 Linear-Type Supports," Revision 2, issued February 2007. DCD Tier 2, Section 3.12.6.1, also states that Class 1 plate and shell-type piping supports (excluding snubbers) follow the rule of ASME Code, Section III, Subsection NF as supplemented by the stipulations of Section C (Regulatory Position) of RG 1.130, "Service Limits and Loading Combinations for Class 1 Plate-and-Shell-Type Component Supports," Revision 2, issued March 2007. The staff reviewed the applicant's position for Class 1 linear-type and plate to shell-type piping supports construction. Because the applicant's positions comply with the ASME Code and meet the regulatory positions recommended by the staff in RG 1.124, Revision 2, and RG 1.130, Revision 2, the staff finds this acceptable.

DCD Tier 2, Section 3.12.6.1, also states that non-seismic category piping supports are designed in accordance with the requirements of ASME/ANSI B31.1 Power Piping Code, Paragraph 120 for loads on pipe supporting elements and Paragraph 121 for design of pipe supporting elements. The staff recognized that ASME/ANSI B31.1 Power Piping Code provides reasonable assurance for the standard piping design and has been used in piping and piping support design in current nuclear plants. Based on past precedents, the staff finds that the non-seismic category piping supports design criteria included in the DCD are acceptable.

DCD Tier 2, Section 3.12.6.1 states that expansion anchors and other steel embedments are designed in accordance with "Code Requirements for Nuclear Safety-Related Concrete Structures," ACI 349. The design of steel embedments in accordance with ACI 349 is acceptable to the staff as endorsed in RG 1.142, "Safety-Related Concrete Structures for Nuclear Power Plants (Other than Reactor Vessels and Containments)," Revision 2, issued November 2001. The design of expansion anchors in accordance with ACI 349 is acceptable to the staff as endorsed in RG 1.199, "Anchoring Components and Structural Supports in Concrete," issued November 2003. Therefore, the staff finds the applicant's approach acceptable.

DCD Tier 2, Section 3.12.6.1 also states that pipe support catalog items, which are fabricated/manufactured to later code editions than identified in the above paragraphs, may be used on a site-specific basis. This use of the later codes shall be reconciled to the code of record identified in the above paragraphs. The staff reviewed the applicant's position and determined that the use on a site-specific basis of later codes that are reconciled to the code of record does meet the intent of the code. On this basis, the staff finds the applicant's approach acceptable.

3.12.4.5.2 Jurisdictional Boundaries

DCD Tier 2, Section 3.12.6.2.1, “Pipe Supports and Attachment Points,” states that the jurisdictional boundary between a pipe and its support structure follows the requirements of ASME Code, Section III, Subsections NB-1132, NC-1132, or ND-1132. For piping analyzed per the requirements of ASME B31.1, the jurisdictional boundary requirements of ASME Code, Section III, Subsection ND-1132 is followed. The staff recognized that Subsections NB/NC/ND-1132 provides the jurisdictional boundaries between piping and attachments which may be part of the piping support. Because the applicant position follows the ASME code position which is endorsed by the NRC, the staff finds this acceptable.

DCD Tier 2, Section 3.12.6.2.2, “Subsection NF Boundaries,” states that the jurisdictional boundaries between the pipe support and the building structure follow the requirements of Subsection NF-1130 of the ASME Code, Section III. Because the jurisdictional boundaries between piping supports and interface attachment points is in conformance with the criteria recommended in SRP Section 3.12, Paragraph II.D.ii, the staff finds the applicant’s approach acceptable.

3.12.4.5.3 Loads and Load Combinations

DCD Tier 2, Section 3.12.6.3, “Loads and Load Combinations,” provides the loading conditions and combinations for the design of piping support in DCD Tier 2, Table 3.12-4, “Loading Combinations for Piping Supports.” SRP Section 3.9.3, Subsection II.1 provides acceptance criteria for component and component support design. The SRP states that the design and service loading combinations should be sufficiently defined to provide the basis for the design of Code Class 1, 2, and 3 components and component supports for all conditions. The SRP also states that the acceptability of the combination of design and service loadings applicable to the design of Class 1, 2, and 3 components and component supports is judged by comparison with positions stated in Appendix A of SRP Section 3.9.3.

The staff reviewed the loads and loading combinations for piping supports against the criteria provided in SRP Section 3.9.3. The staff compared the loading combinations in DCD Tier 2, Table 3.12-4 with those in SRP Section 3.9.3, Appendix A, Table I. DCD Tier 2, Table 3.12-4 showed that only DBPB and SSE are combined with SRSS method and all other ASME Code, Section III Level D dynamic loadings are not combined using SRSS. This is consistent with staff recommendations as described in NUREG-0484, Revision 1. The staff finds that DCD Tier 2, Table 3.12-4 meets the criteria of SRP Section 3.9.3. Therefore, the staff determined that the applicant’s load conditions and load combinations are acceptable.

3.12.4.5.4 Pipe Support Baseplate and Anchor Bolt Design

DCD Tier 2, Section 3.12.6.4, “Pipe Support Baseplate and Anchor Bolt Design,” states that while every effort is made to minimize the use of baseplates with expansion anchors, there may be some instances where baseplate designs are used. In such cases, an evaluation of concrete is performed using ACI-349, Appendix B, considering the limitations of RG 1.199. All aspects of the anchor bolt design, baseplate flexibility and factors of safety are utilized as identified in NRC Bulletin 79-02, “Pipe Support Base Plate Designs Using Concrete Expansion Anchor Bolts,” Revision 2, issued November 8, 1979. The staff reviewed DCD Tier 2, Section 3.12.6.4, against the criteria provided in SRP Section 3.12, Subsection II.D.iv, which states that the design of the pipe support baseplates and anchor bolts should comply with guidance provided in NRC Bulletin 79-02, Revision 2. Because the US-APWR anchor bolt design, baseplate flexibility and factors of safety are in conformance with NRC Bulletin 79-02, Revision 2 per the SRP recommendations, the staff finds the applicant's approach acceptable.

3.12.4.5.5 Use of Energy Absorber and Limit Stops

DCD Tier 2, Section 3.12.6.5, "Use of Energy Absorber and Limit Stops," states that energy absorbers and limit stops are not used as piping supports in the US-APWR design. Because energy absorber and limit stops are not part of the US-APWR design, the staff finds that this portion of the SRP is not applicable to the US-APWR.

3.12.4.5.6 Use of Snubbers

DCD Tier 2, Section 3.12.6.6, "Use of Snubbers," states that snubbers are used to provide dynamic restraint at piping locations with high thermal movement in the restrained direction. The main consideration in the design of any snubber is the ability of the snubber to properly activate and restrain movement for a given design load. The applicant also states that care is taken so that the thermal growth does not exceed the snubber lock-up velocity. DCD Tier 2, Section 3.12.6.6 also states that the specification provided to the supplier (manufacturer) contains the following information:

- Applicable codes (e.g., ASME Code, Section III, Subsection NF and standards).
- Functional requirements.
- Operating environment (normal as well as post-accident).
- Materials to be used in manufacturing/fabrication.
- Functional testing and certification.

The applicant also states that the proper installation and operation of snubbers by the use of visual inspection, operational and installation measurements, and observation of thermal movements during plant startup are to be verified. The staff reviewed the functional requirements and testing for operational readiness of snubbers and documented the staff's acceptance of the applicant's approach in Section 3.9.6 of this report.

The staff reviewed DCD Tier 2, Section 3.12.6.6, against the snubber functionality assurance criteria in SRP Section 3.9.3 Subsection II.3.B. The staff finds that the requirements described in the DCD are consistent with applicable portions of SRP Section 3.9.3 and are acceptable for the US-APWR.

3.12.4.5.7 Pipe Support Stiffness

DCD Tier 2, Section 3.12.6.7, "Pipe Support Stiffness," states that in the de-coupled piping stress analysis (where support is modeled as a restraint versus the integrated analysis), supports are modeled with either calculated actual stiffness of the support structure or with an arbitrarily chosen rigid stiffness. In general, rigid support stiffness is used for all ASME Code, Section III, Classes 2 and 3, and ASME B31.1 piping, with a check on support deflection in the restrained directions to verify the assumed rigidity. The actual stiffness is used for all variable support hangers (springs). For all ASME Code, Section III, Class 1 piping, during the final design fatigue analyses the actual support stiffness is used. If the actual support stiffness is used for any support excluding variable support hangers, the actual stiffness for all supports within the piping analytical problem boundary model is used.

Also, the applicant discusses two deflection checks that will be performed for each support modeled as rigid in the piping analysis. The first check will compare the deflection in the restrained direction(s) to a maximum of 1/16 in. (1.6 mm) for SSE loadings or the minimum support design loadings. The second check will compare the deflection in the restrained direction(s) to a maximum of 1/8 in. (3.2 mm) for the worst case deflection for any load case combination. In the development of the support deflections, dynamically flexible building elements beyond the support jurisdictional boundaries will also be considered.

The staff reviewed DCD Tier 2, Section 3.12.6.7, as well as the deflection check standards described above, against the criteria in SRP Section 3.9.3 Subsection II.3 and determined that they are reasonable and consistent with industry practices for rigid pipe supports as presented in WRC Bulletin 353. On this basis, the staff concludes that these criteria are acceptable for the US-APWR.

3.12.4.5.8 Seismic Self-Weight Excitation

DCD Tier 2, Section 3.12.6.8, "Seismic Self-Weight Excitation," states that in the de-coupled piping stress analysis where a support is modeled as a restraint (as against integrated analysis), the response of the support structure to SSE loadings is included in the pipe support analysis. In general, the inertial response of the support structure mass is evaluated. The damping values of RG 1.61, Revision 1 are used in such analysis. The support self-weight response to the SSE is combined with the piping response to the SSE (inertial loads and anchor movements) by absolute summation. The staff reviewed DCD Tier 2, Section 3.12.6.8, against the criteria in SRP Section 3.9.3. Because the support self-weight excitation response has been evaluated for the load combinations, the staff finds the applicant's approach acceptable.

3.12.4.5.9 Design of Supplementary Steel

DCD Tier 2, Section 3.12.6.9, "Design of Supplementary Steel," states that seismic category I and II piping supports are designed per the rules of ASME Code, Section III, Subsection NF. Any supplemental steel required to connect the piping support structure to the building structure is also designed to the rules of Subsection NF, following the requirements on jurisdictional boundaries. The staff reviewed DCD Tier 2, Section 3.12.6.9, against the criteria in SRP Section 3.12, Subsection II.D.ix. Because the seismic Category I and II piping supports design is in compliance with ASME Code, Section III, Subsection NF, which is also recommended by the SRP, the staff finds the applicant's approach acceptable.

3.12.4.5.10 Consideration of Friction Forces

DCD Tier 2, Section 3.12.6.10, "Consideration of Friction Forces," states that friction forces are calculated using the loads due to thermal expansion and deadweight that are normal to the applicable support member. The friction force is calculated only when the movement in the unrestrained direction due to thermal expansion exceeds 1/16th in. (1.6 mm). The friction force is calculated using μ x N, where μ is the appropriate coefficient of friction and N is the total force normal to the movement. The coefficient of friction will be taken as 0.3 for steel-to-steel conditions and 0.1 for low friction slide/bearing plates. The staff reviewed DCD Tier 2, Section 3.12.6.10, against the criteria in SRP Section 3.12 Subsection II.D.x. Because the friction loads induced by the pipe on the support was evaluated and the friction coefficients used for this evaluation are reasonable, the staff finds the applicant's approach acceptable.

3.12.4.5.11 Pipe Support Gaps and Clearances

DCD Tier 2, Section 3.12.6.11, “Pipe Support Gaps and Clearances,” states that all rigid supports have a cold condition gap of 1/16 in. (1.6 mm) all around the pipe surface in the restrained direction. These small gaps allow the rotation of the pipe and also allow for radial thermal expansion of the pipe. However, in the case of vertical restraints during the cold condition the pipe surface will be in contact with the support in the direction of gravity. An 1/8 in. (3.2 mm) gap will be maintained above the pipe surface in the vertical upward direction. In the unrestrained direction, the gaps are greater than the expected maximum movement of the pipe. The staff recognized that a thorough discussion of pipe support gaps is provided in WRC-353 to support the industry practice for the analysis of 1/16 in. (1.6 mm) per side or 1/8 in. (3.2 mm) total clearance. The staff reviewed DCD Tier 2, Section 3.12.6.11, against the criteria in SRP Section 3.12 Subsection II.D.xi regarding providing sufficient pipe support gaps. Because the 1/8 in. (3.2 mm) total gap between the pipe and the support is sufficient to allow for radial thermal expansion of the pipe and for pipe rotation, the staff finds the applicant’s approach acceptable.

3.12.4.5.12 Instrumentation Line Support Criteria

DCD Tier 2, Section 3.12.6.12, “Instrumentation Line Support Criteria,” states that the acceptance criteria for instrumentation line supports for seismic Category I and II instrumentation lines are from ASME Code, Section III, Subsection NF. Non-seismic instrumentation lines are designed per the rules of “Manual of Steel Construction, 9th Edition,” AISC. The applicable loading combinations for these supports are those used for normal and faulted conditions in DCD Tier 2, Table 3.12-4. The staff reviewed DCD Tier 2, Section 3.12.6.12, against the criteria in SRP Section 3.12 Subsection II.D.xii, which states that the acceptance criteria provided in ASME Code, Section III, Subsection NF are applicable. Because the applicant’s criteria for seismic category I and II instrumentation line supports are in conformance with the ASME Code and the recommendations of SRP Section 3.12 Subsection II.D.xii, the staff finds the applicant’s approach acceptable. Because the non-seismic instrumentation lines are designed in accordance with a standard industry AISC manual, the staff finds the applicant’s approach acceptable.

3.12.4.5.13 Pipe Deflection Limit

DCD Tier 2, Section 3.12.6.13, “Pipe Deflection Limit,” states that manufacturer’s recommendations for the limitations in its hardware are followed for those piping supports that utilize standard manufactured components. Such limitations include travel limits for variable and constant support spring hangers, swing angles for rod hangers, struts, and snubbers. The variability check of variable support spring hangers is performed per applicable Codes. The staff reviewed DCD Tier 2, Section 3.12.6.13, against the criteria in SRP Section 3.12 Subsection II.D.xiii. Because the applicant used manufacture’s recommendations to limit pipe deflection to ensure that the pipe deflection does not cause the failure of the supports, the staff finds the applicant’s approach acceptable.

3.12.4.5.14 Conclusions Regarding Piping Support Design Criteria

The staff concludes that the applicant satisfies the following regarding piping support design criteria:

- The requirements of GDC 1 and 10 CFR 50.55a by specifying methods and procedures for the design and construction of safety-related pipe supports in conformance with these requirements and general engineering practice.

- The requirements of GDC 2 and 4 by designing and constructing the safety-related pipe supports to withstand the effects of normal operation, as well as postulated accidents such as LOCAs and effects resulting from the SSE.

- The requirements of GDC 1, 2, 4, 14, and 15 and 10 CFR Part 50, Appendix S, by identifying applicable codes and standards, design and analysis methods, design transients and load combinations, and design limits and service conditions to assure adequate design of all safety-related piping and pipe supports in the US-APWR for their safety functions.

- The requirements of 10 CFR Part 50, Appendix S, by providing reasonable assurance that the safety-related piping systems are designed to withstand the effects of earthquakes with an appropriate combination of other loads of normal operation and postulated accidents with an adequate margin for ensuring their safety functions.

3.12.5 Combined License Information Items

The following is a list of COL item numbers and descriptions from Table 1.8-2 of the DCD related to ASME Code Class 1, 2 and 3 Piping Systems, Piping Components and Their Associated Supports:

Table 3.12-1
US-APWR Combined License Information Items

Item No.	Description	Section
3.12(2)	If any piping is routed in tunnels or trenches in the yard, the COL Applicant is to generate site-specific seismic response spectra, which may be used for the design of these piping systems.	3.12.5.1
3.12(3)	If the COL Applicant finds it necessary to lay ASME Code, Section III, Class 2 or 3, piping exposed to wind or tornado loads, then such piping must be designed to the plant design-basis loads.	3.12.5.3.6
3.12(4)	The COL Applicant will screen piping systems that are sensitive to high frequency modes for further evaluation.	3.12.5.6
3.12(5)	The COL holder for the first plant is to perform the pressurizer surge line monitoring subsequent to the COL item 14.2(11).	3.12.7

The staff finds the above listing to be complete. Also, the list adequately describes actions necessary for the COL applicant. No additional COL information items were identified that need to be included.

Note that the applicant added COL Information Item 3.12(5) in the amended response to **RAI 742-5703, Question 03.12-25.** As discussed in Section 3.12.4.4.9 of this report above, **RAI 742-5703, Question 03.12-25, is being tracked as a Confirmatory Item.**

3.12.6 Conclusions

As a result of the open item for **RAI 804-5938, Question 03.12-26**, the staff is unable to finalize its conclusions on Section 3.12 related to ASME Code Class 1, 2 and 3 piping systems, piping components and their associated supports, in accordance with NRC regulations.

3.13 Threaded Fasteners (ASME Code Class 1, 2, and 3)

3.13.1 Introduction

The purpose of this section is to review and evaluate the adequacy of the applicant's criteria with regard to selection of materials, design, inspection and testing of its threaded fasteners (i.e., threaded bolts, studs, etc.) prior to initial service and during service. The scope of this review is limited to threaded fasteners in ASME B&PV Code Class 1, 2 or 3 systems.

3.13.2 Summary of Application

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

DCD Tier 2: The applicant has provided a DCD Tier 2 description in Section 3.13, "Threaded Fasteners (ASME Code Class 1, 2, and 3)," summarized here in part, as follows:

> The application describes the US-APWR standard plant and site-specific plant design and testing of threaded fasteners for ASME Code, Section III, Class 1, 2, and 3 systems, and applications from NUREG-0800, Section 3.13. ASME Code Class 1, 2, and 3 threaded fasteners used in US-APWR nuclear power plants comprise bolts, studs, nuts, and washers.
>
> DCD Tier 2, Section 3.13.1, "Design Considerations," describes design considerations which includes material selection, special materials fabrication processes and special controls, material test coupons and specimens for ferritic steel materials (tensile test criteria), test coupons requirements for bolting and stud materials, RV closure stud bolting, general corrosion and stress corrosion cracking of threaded fasteners, fastener thread lubricants and sealants, fracture toughness requirements for threaded fasteners made of ferritic materials, and certified material test reports. DCD Tier 2, Section 3.13.2, "Inservice Inspection Requirements," describes inservice testing requirements for threaded fasteners. DCD Tier 2, Section 3.13.3, "Combined License Information," describes COL information.

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

TS: 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 for this area of review.

Cross-cutting Requirements (TMI, USI/ GSI, Op Ex): There are no cross-cutting requirements for this area of review.

US-APWR Interface Issues Identified in the DCD: See DCD Tier 2, Table 1.8-2, "Compilation of All Combined License Applicant Items for Chapters 1-19."

Site Interface Issues Identified in the DCD: See DCD Tier 2, Table 1.8-1, "Significant Site Specific Interfaces with the Standard US-APWR Design."

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

3.13.3 Regulatory Basis

The relevant requirements of the Commission's regulations for this area of review, and the associated acceptance criteria, are given in Section 3.13, "Threaded Fasteners (ASME Code Class 1, 2, and 3)," issued March 2007, of NUREG-0800, the SRP, and are summarized below. Review interfaces with other SRP sections also can be found in Section 3.13 of NUREG-0800.

1. GDC 1 as it relates to the requirement that SSCs important to safety be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety function to be performed.

2. GDC 4, as it relates to the compatibility of components with environmental conditions.

3. GDC 14, as it relates to the requirement that the RCPB be designed, fabricated, erected, and tested in a manner that provides assurance of an extremely low probability of abnormal leakage, rapidly propagating failure, or gross rupture.

4. GDC 30, as it relates to the requirement that the RCPB shall be designed, fabricated, erected, and tested to the highest quality standards practical.

5. GDC 31, as it relates to the requirement that the RCPB be designed with sufficient margin to ensure that when stressed under operating, maintenance, testing, and postulated accident conditions, the boundary behaves in a non-brittle manner and the probability of rapidly propagating fracture is minimized.

6. 10 CFR Part 50, Appendix B, as it relates to controlling the cleaning of material and equipment to prevent damage or deterioration.

7. 10 CFR Part 50, Appendix G, as it relates to materials testing and acceptance criteria for fracture toughness of reactor pressure boundary components.

8. 10 CFR 50.55a incorporates by reference the design criteria of ASME Code, Section III, Class 1, 2, and 3 components. The selection of materials, design, testing, fabrication, installation and inspection of threaded fasteners and mechanical joints are acceptable if they meet the criteria of the ASME Code, Section III, Class 1, 2, and 3 components. However, 10 CFR 50.55a(b)(4) permits the use of Code cases that have been adopted by the staff in RG 1.84, "Design, Fabrication, and Materials Code Case Acceptability, ASME Section III," Revision 35, issued October 2010, in lieu of applicable criteria for ASME Code, Section III, Class 1, 2, and 3 components.

Acceptance criteria adequate to meet the above requirements include:

1. RG 1.37, “Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants,” Revision 1, issued March 2007, as it relates to QA criteria for cleaning fluid systems and associated components that comply with 10 CFR Part 50, Appendix B.

2. RG 1.65, “Materials and Inspections for Reactor Vessel Closure Studs,” Revision 0, issued October 1973, as it relates to selecting materials for and testing of RV closure studs.

3. RG 1.84, as it relates to ASME Code cases found by the staff to be acceptable for implementation.

4. NUREG-1339, “Resolution of Generic Safety Issue 29: Bolting Degradation or Failure in Nuclear Power Plants,” issued June 1990, as it relates to selecting lubricants for use with threaded fasteners.

3.13.4 Technical Evaluation

The staff reviewed the information in DCD Tier 2, Section 3.13 in accordance with the guidance provided in SRP Section 3.13. The review was performed in order to determine the adequacy of the applicant’s design criteria for threaded fasteners in ASME Class 1, 2, and 3 systems for material selection; special materials fabrication processes and controls (material test coupons, RV bolting, general corrosion and stress cracking corrosion, and lubricants and sealant); fracture toughness requirements for ferritic materials; fabrication inspection; certified material test reports; and preservice and ISI requirements. The staff evaluation for ASME Code Class 1, 2, and 3 threaded fasteners is stated below.

3.13.4.1 Material Selection

Per SRP Section 3.13, the selection of materials used for the design of threaded fasteners is acceptable if ASME Code, Section III criteria for material selection is appropriately specified for threaded fasteners in ASME Class 1, 2, and 3 systems.

DCD Tier 2, Section 3.13.1.1, “Materials Selection,” and Table 3.13-1, “ASME Code, Section III Criteria for Selection and Testing of Bolting Materials,” provide material selection criteria for threaded fasteners in ASME Class 1, 2, and 3 systems. Material selection is in accordance with ASME Code, Section III, NCA-1220 and NB-2128 for Class 1 fasteners, NCA-1220 and NC-2128 for Class 2 fasteners, and NCA-1220 and ND-2128 for Class 3 fasteners. ASME Code, Section III provides acceptance standards for selecting threaded fasteners in Class 1, 2 and 3 applications and the materials used for all threaded fasteners must be suitable for, and compatible with, the plant design temperatures, pressures, loads, stresses, and operating service conditions including corrosion and radiation exposures. In DCD Tier 2, Section 3.13.1.1, the applicant addressed code requirements and guidance for use of nuts and washers, fracture toughness, ultimate tensile strength, Charpy V-notch impact testing, and thread lubricants.

DCD Tier 2, Section 5.3.1.7, “Reactor Vessel Fasteners,” which is referenced in DCD Tier 2, Subsection 3.13.1.1.1, “Class 1 Applications,” addresses code requirements and guidance specific to RV fasteners. RV closure stud bolting specified by the applicant is SA-540, Grade 24 material approved by ASME Code, Section III, Subsection NB and conforms with the material recommendations of RG 1.65, Revision 0, issued October 1973. In DCD Tier 2, Section 5.3.1.7, the applicant referenced the guidance in RG.1.65 for RV fasteners; identified code requirements for fastener material testing; and addressed surface treatments to protect against corrosion and lubricants for ASME Class 1 fasteners.

The staff finds the material selection for fasteners in ASME Class 1, 2, and 3 systems, including RV closure studs, to be acceptable because the applicant complies with Section II and III of the ASME Code and conforms with RG 1.65.

3.13.4.2 Material Test Coupons

Per SRP Section 3.13, the heat treatment and tensile test coupon criteria for threaded fasteners are acceptable if the ASME Code, Section III criteria are appropriately specified for threaded fasteners in ASME Class 1, 2, and 3 systems.

DCD Tier 2, Sections 3.13.1.2, "Special Materials Fabrication Processes and Special Controls," 3.13.1.2.1, "Material Test Coupons and Specimens for Ferritic Steel Material (Tensile Test Criteria)," and 3.13.1.2.2, "Test Coupons Requirements for Bolting/Stud Materials," and Table 3.13-1 provide heat treatment and tensile test coupon preparation criteria for threaded fasteners fabricated from ferritic materials e.g., carbon steel, low-alloy steel, quenched and tempered steel). The applicant specified the appropriate test coupon preparation criteria for ASME Class 1, 2, and 3 threaded fasteners. DCD Tier 2, Table 3.13-1 identifies the appropriate sections of the ASME Code relative to heat treatment criteria and test coupon preparation criteria for ferritic steel materials. As stated in the ASME Code, the applicant should apply the criteria of ASME Code, Section III, Subparagraphs NB-2200, NC-2200, or ND-2200 rather than the criteria of the material specification applicable to the mechanical testing if there is a conflict between the two sets of criteria.

The staff finds the heat treatment and tensile test coupon preparation criteria for threaded fasteners in ASME Class 1, 2, and 3 systems to be acceptable because the applicant complies with Section III of the ASME Code.

3.13.4.3 Reactor Vessel Closure Stud Bolting (Special Processes and Controls)

Per SRP Section 3.13, the special processes for heat treatment, material toughness and corrosion protection for ASME Class 1 RV closure stud bolting are acceptable if the ASME Code, Section III criteria are appropriately specified.

Tier 2, Subsection 3.13.1.2.3, "Reactor Vessel Closure Stud Bolting," provides special fabrication processes and controls for RV closure stud bolting. The applicant discusses special processes for heat treatment, material toughness and corrosion protection of the bolting and references appropriate sections of the ASME Code and includes regulatory guidance contained in RG 1.65.

The staff finds the special fabrication processes and controls for RV closure stud bolting to be acceptable because the applicant complies with Section III of the ASME Code and conforms with RG 1.65.

3.13.4.4 General Corrosion and Stress Cracking Corrosion of Threaded Fasteners (Special Processes and Controls)

DCD Tier 2, Subsection 3.13.1.2.4, “General Corrosion and Stress Corrosion Cracking of Threaded Fasteners,” provides criteria for general corrosion, stress corrosion cracking, and galvanic corrosion of threaded fasteners. The applicant discusses causes, effects, and industry practice to minimize or preclude general corrosion, stress corrosion cracking, and galvanic corrosion for threaded fasteners contained in EPRI NP-5067,” Volume I, “Good Bolting Practices – A Reference Manual for Nuclear Power Plant Maintenance Personnel, Large Bolt Manual,” 1987, and EPRI NP-5067, Volume II, “Good Bolting Practices – A Reference Manual for Nuclear Power Plant Maintenance Personnel, Small Bolt Manual,” Volume II, 1990.

The staff finds the special fabrication processes and controls for general corrosion, stress corrosion cracking, and galvanic corrosion of threaded fasteners as discussed in DCD Tier 2, Subsection 3.13.1.2.4 to be acceptable because the applicant specifies criteria complying with industry guidance contained in EPRI NP-5067, which is widely used industry guidance which has been accepted for use by the NRC.

3.13.4.5 Lubricant and Sealants

DCD Tier 2, Section 3.13.1.2.5, “Fastener Thread Lubricants and Sealants,” provides criteria for thread lubricants and sealants for threaded fasteners. Per SRP Section 3.13, Subsection II.1.C, criteria for threaded fastener lubricants and sealants are acceptable if the regulatory guidance and industry experience are appropriately specified by the applicant.

The applicant provided the following information regarding lubricants and sealants:

> As stated in NUREG-1339, “Resolution of Generic Safety Issue 29: Bolting Degradation or Failure in Nuclear Power Plants,” issued June 1990, which cites RG 1.37, “Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants,” Revision 1, issued March 2007, MoS2 is known to promote corrosion in high strength low alloy fasteners and should not be used on any type of nuclear fastener regardless of material. Teflon, polytetrafluoroethylene (PTFE) has poor radiation resistance and, therefore, is not used in the primary containment area or any other high radiation zones. A better all-purpose, inorganic thread lubricant is graphite, which also has good lubricity (low friction coefficient) with high thermal stability and radiation resistance. One commercial product used in nuclear power plants is “Neolube,” a graphite-based sealant/lubricant. Nuclear grade Neolube #1260 has excellent thermal stability (to 1200 °F) (666.7 °C) and high radiation resistance (1.5 x 10^9 radiation absorbed dose). Neolube #1260 also seals up to 7,500 psi (51.71 MPa).
>
> This graphite flake paste is compatible with any ferritic or austenitic alloy bolting material. Therefore, graphite (e.g., Neolube #1260) is considered a generally acceptable thread lubricant for nuclear fasteners, while MoS2 is not to be used at all, and PTFE is unsuitable for primary containment applications because of low radiation resistance.
>
> EPRI NP-5067, Volume I discusses the use of and guidelines for leak sealants including bolt thread sealants. The application of leak sealants should be considered as temporary solutions, with leaking components repaired or replaced at the next available opportunity. Repairs should involve complete removal of the sealant and restoration of the component to its original condition or configuration.

DCD Tier 2, Revision 1, Subsection 3.13.1.2.5 references ANSI N45.2.1, "Cleaning of Fluid Systems and Associated Components During Construction Phase of Nuclear Power Plants," 1973, for cleaning and cleanliness controls of threaded fasteners. For cleaning and cleanliness controls of components (including threaded fasteners), RG 1.37, endorses ASME NQA-1-1994, Part II, Subpart 2.1. In **RAI 273-2239, Question 03.13-5**, the staff requested the applicant to use the guidance contained in RG 1.37 and ASME NQA-1-1994 for cleaning and cleanliness controls of threaded fasteners. In its response to **RAI 273-2239, Question 03.13-5**, dated April 9, 2009, the applicant stated that DCD Tier 2, Subsection 3.13.1.2.5 will be revised to delete the reference to ANSI N45.2.1 and add the endorsement by RG 1.37 of ASME NQA-1-1994, Part II, Subpart 2.1 as a reference for cleaning and cleanliness controls of threaded fasteners. The staff finds the response acceptable since the applicant agreed to follow staff guidance. The staff confirmed that DCD Tier 2, Subsection 3.13.1.2.5 was revised to reference the endorsement by RG 1.37 of ASME NQA-1-1994, Part II, Subpart 2.1, as stated above. Accordingly, **RAI 273-2239, Question 03.13-5, is resolved**.

Based on the above, the staff finds the applicant's lubricants, sealants, and cleaning processes for threaded fasteners in ASME Class 1, 2, and 3 systems to be acceptable because the applicant conforms with appropriate regulatory guidance and industry experience contained in NUREG-1339, RG 1.37, and EPRI NP-5067.

3.13.4.6 Fracture Toughness Requirements for Ferritic Materials

Per SRP Section 3.13, the fracture toughness of ferritic bolts, studs, and nuts (e.g., made from either low-alloy steel or carbon steel materials) is acceptable if the appropriate ASME Code, Section III criteria are specified by the applicant. Ferritic bolts, studs, and nuts used in RCPB applications are to meet the requirements contained in 10 CFR Part 50, Appendix G.

DCD Tier 2, Section 3.13.1.3, "Fracture Toughness Requirements for Threaded Fasteners Made of Ferritic Materials," and Table 3.13-1 provide criteria for fracture toughness of threaded fasteners made from ferritic materials. The applicant specifies ASME Code requirements for materials to be impact tested, types of impact tests, test coupons, acceptance standards, number of impact tests necessary, retesting, and calibration of test equipment. ASME Code Class 1 fasteners meet the fracture toughness requirements of ASME, Section III, NB-2300 and Appendix G of 10 CFR Part 50, and the RV studs and fasteners also conform with the fracture toughness recommendations of RG 1.65. The ASME Code Class 2 and 3 fasteners comply with the fracture toughness requirements of ASME, Section III, NC-2300 and ND-2300, respectively. The staff finds the applicant's fracture toughness requirements for ferritic threaded fasteners in ASME Class 1, 2, and 3 systems to be acceptable because the applicant complies with Section III of the ASME Code and 10 CFR Part 50, Appendix G, and conforms with RG 1.65.

3.13.4.7 Fabrication Inspection (Examination Criteria)

Per SRP Section 3.13, fabrication inspection criteria for threaded fasteners are acceptable if ASME Code, Section III criteria are appropriately specified for threaded fasteners in ASME Class 1, 2, and 3 systems.

DCD Tier 2, Section 3.13.1.3, "Fracture Toughness Requirements for Threaded Fasteners Made of Ferritic Materials," and Table 3.13-1 provide criteria for fabrication inspection for threaded fasteners. The fabrication inspection criteria includes visual, magnetic particle, liquid penetrant, and ultrasonic examination criteria for bolts, studs, and nuts. The applicant specifies the use of ASME Code, Section III, NB-2580 for Class 1 fasteners, NC-2580 for Class 2 fasteners, and ND-2580 for Class 3 fasteners.

The staff finds the fabrication inspection for threaded fasteners in ASME Class 1, 2, and 3 systems to be acceptable because the applicant complies with Section III of the ASME Code.

3.13.4.8 Quality Records (Certified Material Test Reports)

Quality records for threaded fasteners used in ASME Code Class 1, 2, and 3 systems are acceptable if the applicant commits to retaining the Certified Material Test Reports (CMTRs) in accordance with the requirements of 10 CFR 50.71. The applicant should commit to recording the results of material chemistry tests (i.e., alloying elements) and physical property tests in CMTRs.

DCD Tier 2, Section 3.13.1.5, “Certified Material Test Reports,” and Table 3.13-1 provide criteria for CMTRs. CMTRs will be retained in accordance with 10 CFR 50.71. The CMTR report criteria for Classes NB, NC, and ND are specified in ASME Code, Section III, NCA-3680. The applicant committed to retaining the results of its material chemistry analyses, fabrication, and mechanical property tests in the applicable CMTRs in accordance with ASME Code, Section III, Subsections NB, NC, and ND.

COL Information Item 3.13(3) states that the COL Applicant is to retain quality records including certified material test reports for all property test and analytical work performed on nuclear threaded fasteners in accordance with the requirements of 10 CFR 50.71. The staff reviewed the above COL Item and considers it appropriate for this item to be addressed by COL applicants referencing the US-AWPR DC.

The staff finds the quality records for threaded fasteners in ASME Class 1, 2, and 3 systems to be acceptable because the applicant complies with 10 CFR 50.71 and Section III of the ASME Code related to documentation of tests in CMTRs.

3.13.4.9 Preservice and Inservice Inspection Requirements for Threaded Fasteners

Preservice and ISI provisions for threaded fasteners used in ASME Code Class 1, 2, and 3 systems are acceptable if the applicant meets ISI requirements of 10 CFR 50.55a and ASME Code Section XI for threaded fasteners used in ASME Code Class 1, 2, and 3 systems.

DCD Tier 2, Section 3.13.2 and Table 3.13-2, “ASME Code, Section XI Examination Categories for Inservice Inspections of Mechanical Joints in ASME,” provide criteria for preservice and ISI requirements for threaded fasteners. These inspections include visual, surface, and volumetric examinations for threaded fasteners. The applicant specifies conformance to ASME Code, Section XI, IWB-2500 for Class 1 threaded fasteners; ASME Code, Section XI, IWC-2500 for Class 2 threaded fasteners. The staff notes that for Class 3 threaded fasteners there are no specific ASME Code inspection requirements.

DCD Tier 2, Section 3.13.2 and Table 3.13-2 provide criteria for ISI requirements of threaded fasteners during the performance of system pressure tests. For ISI requirements, the applicant specifies conformance to ASME Code, Section XI, IWB-2500 for Class 1 threaded fasteners; ASME Code, Section XI, IWC-2500 for Class 2 threaded fasteners; and ASME Code, Section XI, IWD-2500 for Class 3 threaded fasteners. COL Item 3.13(4) states that the COL Applicant is to address compliance with ISI requirements as summarized in DCD Tier 2, Section 3.13.2.

The staff reviewed the above COL Action Item and considers it appropriate for this item to be addressed by COL applicants referencing the US-APWR DC.

The DCD Tier 2, Section 3.13.2 cites the requirements of ASME Code, Section XI, IWA-5000; the requirements of 10 CFR 50.55a(b)(2)(xxvi) for pressure testing mechanical joints; and 10 CFR 50.55a(b)(2)(xxvii) for removal of insulation. COL Item 3.13(5) states that the COL Applicant must meet the requirements of ASME Code, Section XI, IWA-5000; the requirements of 10 CFR 50.55a(b)(2)(xxvi), “Pressure Testing Class 1, 2, and 3 Mechanical Joints;” and the requirements of 10 CFR 50.55a(b)(2)(xxvii), “Removal of Insulation.” The staff reviewed the above COL Action Item and considers it appropriate for this item to be addressed by COL applicants referencing the US-APWR DC.

The staff finds the preservice and ISIs for threaded fasteners in ASME Class 1, 2, and 3 systems to be acceptable because the applicant complies with the preservice and ISI requirements of 10 CFR 50.55a and Section XI of the ASME Code.

3.13.5 Combined License Information Items

The following is a list of COL item numbers and descriptions from Table 1.8-2 of the DCD related to threaded fasteners (ASME Code Class 1, 2, and 3):

Table 3.13-1
US-APWR Combined License Information Items

Item No.	Description	Section
3.13(3)	The COL Applicant is to retain quality records including certified material test reports for all property test and analytical work performed on nuclear threaded fasteners in accordance with the requirements of 10 CFR 50.71.	3.13.1.5
3.13(4)	The COL Applicant is to address compliance with ISI requirements as summarized in Subsection 3.13.2.	3.13.2
3.13(5)	The COL Applicant is to commit to complying with the requirements of ASME Code, Section XI, IWA-5000, and the requirements of 10 CFR 50.55a(b)(2)(xxvi), Pressure Testing Class 1, 2, and 3 Mechanical Joints, and Paragraph (xxvii) Removal of Insulation.	3.13.2

The staff reviewed the above COL Action Items in Section 3.13.4 of this report and considers them appropriate for COL applicants referencing the US-APWR DC. The staff finds the above listing to be complete. Also, the list adequately describes actions necessary for the COL applicant. No additional COL information items were identified that need to be included.

3.13.6 Conclusions

The staff reviewed the US-APWR DC application for compliance with the NRC regulations and the applicable edition and addenda of the ASME Code for threaded fasteners in ASME Class 1, 2, or 3 systems to be used in US-APWR nuclear power plants. Based on its review, the staff concludes that the selection of materials, design, inspection and testing prior to initial service and during service for threaded fasteners in ASME Class 1, 2, or 3 systems to be used in a US-APWR satisfies the requirements of GDC 1, 4, 14, 30, and 31; 10 CFR Part 50, Appendices B and G; 10 CFR 50.55a; and the ASME Code in an adequate manner for a DC application. Therefore, the staff concludes that the selection of materials, design, inspection and testing prior to initial service and during service for threaded fasteners in ASME Class 1, 2, or 3 systems for the US-APWR nuclear plant are acceptable.