5.0 REACTOR COOLANT SYSTEM AND CONNECTING SYSTEMS

Chapter 5.0 of this final safety evaluation report (FSER) presents the results of the United States Nuclear Regulatory Commission (NRC) staff's (the staff's) review of Chapter 5, "Reactor Coolant and Connecting Systems," of Mitsubishi Heavy Industries' (MHI's), the applicant's, United States-Advanced Pressurized-Water Reactor (US-APWR) Design Control Document (DCD), MUAP-DC005, Revision 2, for the purpose of design certification (DC) under Subpart B, "Standard Design Certifications," of Part 52, "Licenses, Certifications and Approvals for Nuclear Power Plants," of Title 10, "Energy," of the *Code of Federal Regulations* (10 CFR Part 52).

5.1 Summary Description

5.1.1 Introduction

The reactor coolant system (RCS) consists of the reactor vessel (RV), which contains the reactor core, and four reactor coolant loops connected to the reactor vessel. Each loop contains a steam generator (SG) and a reactor coolant pump (RCP). The RCS also includes a pressurizer, relief tank (PRT), and RCS pipes and valves. There are several systems connected to the RCS. They are covered in detail elsewhere in the DCD, except for the residual heat removal (RHR) system, which is covered in this chapter. All reactor systems, including the RCS, are located within the containment. The RV is also considered part of the reactor system it contains.

The reactor system includes the reactor internals, the fuel assembly, the control rods, the RV, and the control rod drive mechanisms (CRDMs). DCD Figure 2.4.1-1 illustrates the reactor general assembly, showing the arrangement of the reactor system components. Figure 2.4.1-2 and Figure 2.4.1-3 show the arrangement of the fuel assemblies and rod cluster control assemblies and the arrangement of the RV, respectively. The cylindrical RV measures approximately 435.1 inches from the bottom of its hemispherical bottom head dome to the top of the vessel flange mating surface, with an inside diameter of approximately 202.8 inches. Eight nozzles, located above the fuel, are attached to four coolant loops. No penetrations are located below the top of the reactor core. The RV is supported by eight steel support pads, which are integral with the nozzles. The closure head is held in place with 58 preloaded closure stud assemblies.

The primary function of the RCS is to transfer the heat generated in the reactor core to the secondary system via the steam generators to produce steam for the turbine. Design and operating parameters of the RCS are provided in Table 5.1-1. The RCS has two safety functions. First, the RCS, including connections to related auxiliary systems, constitutes the reactor coolant pressure boundary (RCPB), by which reactor coolant inventory is maintained and which allows coolant pressure to be maintained by the pressurizer as required for safe operation of the reactor. Secondly, the RCS constitutes the second of the three fission product barriers, the first being the fuel cladding and the third being the containment.

5.1.2 Summary of Application

DCD Tier 1: The Tier 1 information associated with this section is found in DCD Tier 1, Section 2.4.

DCD Tier 2: The applicant has provided a DCD Tier 2 system description in Section 5.0, summarized here in part, as follows:

- The RCS, including connections to related auxiliary systems, constitutes the RCPB. The performance and safety design bases of the RCS and its major components are interrelated. The design bases include:

  - The RCS has the capability to transfer heat to the main steam system during power operation and also when the reactor is subcritical, including the initial phase of plant cool-down.

  - The RCS has the capability to transfer residual and decay heat to the RHR system during the subsequent phase of plant cooldown and cold shutdown.

  - The RCS heat removal capability, under power operation and normal operational transients, including the transition from forced to natural circulation, ensures no fuel damage within the operating bounds permitted by the reactor control and protection systems.

  - The RCS provides the water used as the core neutron moderator and reflector for conserving thermal neutrons and improving neutron economy, and as a solvent for the neutron absorber used in chemical shim reactivity control.

  - The RCS maintains the homogeneity of the soluble neutron poison concentration and the rate of change of the coolant temperature so that uncontrolled reactivity changes do not occur.

  - The RCPB components are capable of accommodating the temperatures and pressures associated with operational transients.

  - The pressurizer maintains the system pressure during operation and limits pressure transients. During the reduction or increase of plant load, the pressurizer accommodates volume changes in the reactor coolant.

  - The RCPs produce the coolant flow necessary to remove heat from the reactor core and transfer it to the SGs.

  - The SGs provide high-quality (low moisture content) steam to the turbine. The tubes and tube sheet boundary are designed to prevent the transfer of radioactivity generated within the core to the secondary system. With the reactor subcritical, the SGs provide two independent means of cooling down the RCS from normal operating temperature and pressure to the temperature and pressure at which the RHR system can be operated to cool the RCS further.

- The RCS piping contains the coolant under operating temperature and pressure conditions and limits leakage (and radioactivity release) to the containment atmosphere. The RCS piping contains demineralized, borated water, which is circulated at the flow rate and temperature consistent with achieving the reactor core thermal and hydraulic performance.

- The RCS is monitored for loose parts. This is further described in detail in Section 4.4.6.

- The RCPB components are consistent with 10 CFR 50.2 (Ref. 5.1-1) and 10 CFR 50.55a (Ref. 5.1-2). Applicable codes and standards of RCS components are identified in Table 3.2-2.

- The RV is equipped with suitable provision for connecting the head vent and, thus, meets the requirements of 10 CFR 50.34(f)(2)(vi) (Ref. 5.1-3) (Three Mile Island [TMI] Action Item II.B.1).

- Each of the pressurizer spray lines interconnected to the RCS and the pressurizer surge line are instrumented with temperature detectors to detect low temperature. The low temperature indicates insufficient flow through the pressurizer spray bypass line, which may cause thermal shock at the spray nozzle when pressurizer spray valves open. Thermal shock is avoided by means of the pressurizer spray bypass line, through a normally open bypass valve that provides a constant small flow to maintain the pressurizer spray line temperature.

ITAAC: The inspections, tests, analyses and acceptance criteria (ITAAC) associated with DCD Tier 2, Section 5.0, are given in DCD Tier 1, Section 2.4. Item 3 in Table 2.4.1-2, "Reactor System Inspections, Tests, Analyses, and Acceptance Criteria," states that an inspection will be conducted to verify that the as-built RCS conforms to the functional arrangement shown in Figure 2.4.1-3. Additional ITAAC will be performed to verify the detailed Tier 1 mechanical, electrical, and instrumentation information associated with the RCS.

Technical Specifications: The Technical Specifications associated with DCD Tier 2, Section 5.0, are given in DCD Tier 2, Chapter 16, Sections 3.4 and B 3.4.

5.1.3 Regulatory Basis

The relevant requirements of the Commission's regulations for this area of review, and the associated acceptance criteria, are given in Section 5.0 of NUREG-0800, "Standard Review Plan [SRP] for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR [light-water reactor] Edition," and are summarized below. Review interfaces with other SRP sections can be found in Section 5.4 of NUREG-0800.

1. General Design Criterion (GDC) 1, "Quality Standards and Records," of Appendix A, "General Design Criteria for Nuclear Power Plants," to Part 50, "Domestic Licensing of Production and Utilization Facilities," of Title 10, "Energy," of the *Code of Federal Regulations* (10 CFR Part 50) and paragraph (a)(1) of Section 50.55a of 10 CFR Part 50 [10 CFR 50.55a(a)(1)], as they relate to the assurance of a quality product commensurate with the importance of the safety function to be performed.

2. GDC 2, as it relates to the RCS being designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches without loss of capability to perform its safety functions.

3. GDC 4, as it relates to the RCS 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. GDC 4 also requires that the RCS 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.

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 requires that the RCS and associated auxiliary control and protection systems be designed with sufficient margin to ensure that the design conditions of the RCPB are not exceeded during any condition of normal operation, including anticipated operational occurrences (AOOs).

6. GDC 30 requires, in part, that components that are part of the RCPB be designed, fabricated, erected, and tested to the highest quality standards practical.

7. GDC 31 requires, in part, 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, thereby minimizing the probability of rapidly propagating fracture.

8. 10 CFR 50.55a(c), 10 CFR 50.55a(d), and 10 CFR 50.55a(e) generally require certain grouping of components, including those comprising the pressure boundary, to meet the requirements of Section III of the Boiler and Pressure Vessel Code of the American Society of Mechanical Engineers (ASME Code).

9. 10 CFR 50.55a(f) and 10 CFR 50.55a(g), as they relate to operational programs, such as pre-service inspection, in-service inspection and in-service testing.

10. 10 CFR 52.47(b)(1), which requires that a DC application include the proposed ITAAC that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria

met, a plant that incorporates the DC is built and will operate in accordance with the DC, the provisions of the Atomic Energy Act, and the NRC's regulations;

11. Appendix B, "Quality Assurance Criteria for Nuclear Power Plants," to 10 CFR Part 50 applies to RCS design, procurement, operation, maintenance, and testing.

12. Appendix G, "Fracture Toughness Requirements," to 10 CFR Part 50 requires that RCPB pressure-retaining components that are made of ferritic materials meet ASME Code requirements for fracture toughness during system hydrostatic tests and any condition of normal operation, including AOOs.

Acceptance criteria adequate to meet the above requirements include:

1. Regulatory Guides (RGs) 1.31, "Control of Ferrite Content in Stainless Steel Weld Metal"; 1.34, "Control of Electroslag Weld Properties"; 1.43, "Control of Stainless Steel Weld Cladding of Low-Alloy Steel Components"; 1.50, "Control of Preheat Temperature for Welding of Low-Alloy Steel"; and 1.71, "Welder Qualification for Areas of Limited Accessibility"; as they relate to the welding of RCS components.

2. RG 1.44, "Control of the Use of Sensitized Stainless Steel," as it relates to the RCS water chemistry program.

3. RG 1.36, "Nonmetallic Thermal Insulation for Austenitic Stainless Steel," as it relates to the selection and use of thermal insulation.

5.1.4 Technical Evaluation

The detailed results of the staff's technical evaluation of the information in DCD Chapter 5 are presented in the individual sections of Chapter 5 of this FSER.

5.1.5 Combined License Information Items

There are no combined license (COL) information items that pertain to Section 5.1 in Table 1.8-2 of the DCD.

5.1.6 Conclusions

The staff's conclusions regarding the adequacy of the information in DCD Chapter 5 are presented in the individual sections of Chapter 5 of this FSER.

5.2 RCS Pressure Boundary Integrity

5.2.1 Compliance with Codes and Code Cases

GDC 1 requires that nuclear power plant structures, systems, and components (SSCs) important to safety be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety function to be performed. Where generally recognized codes and standards are used, they must be identified and evaluated to determine their adequacy and applicability. This requirement applies to both pressure-retaining and non-pressure-retaining SSCs that are part of the reactor coolant pressure boundary (RCPB), as well as to other systems important to safety.

5.2.1.1 Compliance with 10 CFR 50.55a

5.2.1.1.1 Introduction

This Safety Evaluation Report (SER) section discusses the use by the DC applicant of the ASME *Boiler and Pressure Vessel Code* (BPV Code) and the applicable editions of and addenda to the ASME BPV Code in accordance with 10 CFR 50.55a in the design, fabrication, and construction of RCS SSCs as described in DCD Tier 2, Section 5.2.1.1, "Compliance with 10 CFR 50.55a."

The review under this section is coordinated closely with the review described in Section 3.2.2 of NUREG-0800. More detailed review of compliance with ASME Code requirements for the component code class (e.g., component welds verified to meet requirements applicable for the Code class) is discussed under many other sections in NUREG-0800. The applicant's framework for compliance with 10 CFR 50.55a requirements for application of the codes during the inservice phase of the component life is also reviewed in accordance with many other sections of NUREG-0800.

5.2.1.1.2 Summary of Application

DCD Tier 1: The Tier 1 descriptions of several of the Nuclear Island systems, e.g., reactor coolant system (RCS), in-containment refueling water storage tank system (IRWSTS), safety injection system/residual heat removal system (SIS/RHRS), emergency feedwater system (EFWS), fuel pool cooling system (FPCS), chemical and volume control system (CVCS), extra borating system (EBS), fuel-handling system (FHS), indicate that components in these systems will be designed, constructed, and tested in accordance with Section III of the ASME Code. The system descriptions in DCD Tier 1 Sections 2.3, 2.4 and 2.7, also indicate that a fatigue analysis will be done for Class 1 piping and components, as required by Section III of the ASME Code.

DCD Tier 2: The applicant has provided a DCD Tier 2 description in Section 5.2.1.1, summarized here in part, as follows:

DCD Tier 2 (Revision 2), Section 5.2.1.1, states that RCPB components are designed and fabricated as Class 1 components in accordance with Section III, "Rules for Construction of Nuclear Facility Components," of the ASME BPV Code, except for components such as the isolation valves and the flow restricting device that meet the exclusion requirements of 10 CFR 50.55a(c), which are designed and fabricated as Quality Group B (Class 2) components. DCD Tier 2, Table 3.2-2, "Classification of Mechanical and Fluid Systems, Components, and Equipment," lists the RCPB components, including pressure vessels, piping, pumps, and valves, along with the applicable component codes. Other safety-related plant components are classified in accordance with RG 1.26, "Quality Group Classifications and Standards for Water, Steam, and Radioactive Waste-Containing Components of Nuclear Power Plants," and are specified in DCD Tier 2, Section 3.2, "Classification of Structures, Systems, and Components."

DCD Tier 2, Section 5.2.1.1, states that the applicable edition and addenda of the ASME Code applied in the design of each Class 1 component are listed in Table 5.2.1-1, "Applicable Code Addenda for RCS Class 1 Components." Table 5.2.1-1 refers to the 2001 Edition with the 2003 Addenda of the ASME BPV Code, Sections II, III, V, and XI, for various RCS Class 1 components. The code of record for the design of the US-APWR is the 2001 Edition of the ASME Code through and including the 2003 Addenda.

ITAAC: The ITAAC, as required by 10 CFR 52.47(b)(1), are given in DCD Tier 1, Section 2.4, based on the selection criteria of DCD Tier 2, Section 14.3.

Technical Specifications: There are no Technical Specifications associated with this area of review.

5.2.1.1.3 Regulatory Basis

The relevant requirements of the Commission's regulations for this area of review, and the associated acceptance criteria, are given in Section 5.2.1.1 of NUREG-0800, the SRP, and are summarized below. Review interfaces with other SRP sections can be found in Section 5.2.1.1 of NUREG-0800.

1. GDC 1, which requires that safety-related structures, systems, and components (SSCs) be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety function to be performed.

2. 10 CFR 50.55a, which establishes minimum quality standards for the design, fabrication, erection, construction, testing, and inspection of certain components of boiling- and pressurized-water reactor nuclear power plants by requiring conformance with appropriate editions of specified published industry codes and standards. Pursuant to 10 CFR 50.55a, components important to safety are designed, fabricated, and constructed to the following requirements:

    a. RCPB components must meet the requirements for Class 1 (Quality Group A) components specified in the ASME BPV Code, Section III, except for those components that meet the exclusion requirements of 10 CFR 50.55a(c)(2).

    b. Components classified as Quality Groups B and C must meet the requirements for Class 2 and 3 components, respectively, specified in ASME Code, Section III.

3. 10 CFR 52.47(b)(1), as it relates to the rule that establishes that a DC application include the proposed ITAAC that are necessary and sufficient to provide reasonable assurance that, if the ITAAC are met, a plant that incorporates the design certification is built 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:

RG 1.26 as it relates to determining quality standards acceptable to the staff for satisfying GDC 1 for other (i.e., non-reactor coolant pressure boundary) safety-related components containing water, steam, or radioactive material in light-water-cooled nuclear power plants. Other system-specific acceptance criteria are listed in NUREG-0800 Section 5.2.1.1.

5.2.1.1.4 Technical Evaluation

The staff discusses the classification of US-APWR SSCs in Section 3.2 of this SER. DCD Tier 2, Table 3.2-2, specifies the 2001 Edition with the 2003 Addenda of the ASME BPV Code for the US-APWR design for Class 1, 2, and 3 components. The 2001 Edition and 2003 Addenda are incorporated by reference in 10 CFR 50.55a . Based on its review, the staff finds that the DCD meets the requirements of 10 CFR 50.55a and GDC 1 for the construction of SSCs important to safety by ensuring that RCPB components, as defined in 10 CFR 50.55a, are classified properly in DCD Tier 2, Table 3.2-2, as ASME Code Section III, Class 1 (Quality Group A) components, except for those that meet the 10 CFR 50.55a(c) exclusion requirements. The staff also finds other US-APWR components to be properly categorized in Table 3.2-2 as ASME BPV Code, Section III, Class 2 (Quality Group B) or Class 3 (Quality Group C).

DCD Tier 2, Section 5.2.1.1, states that the applicable edition and addenda of the ASME Code applied in the design of each Class 1 component for the US-APWR are listed in Table 5.2.1-1, "Applicable Code Addenda for RCS Class 1 Components." Table 5.2.1-1 refers to the 2001 Edition with the 2003 Addenda of the ASME BPV Code, Sections II, III, V, and XI, for various RCS Class 1 components. The staff further finds that the use of these edition and addenda meets the requirements of 10 CFR 50.55a(b) and associated modifications in 10 CFR 50.55a(b)(1)(iii), and is therefore acceptable.

DCD Tier 2, Section 5.2.1.1, states that the use of Code editions and addenda issued and endorsed by the NRC subsequent to the US-APWR DC is permitted. However, the staff noted that any proposed change by the COL applicant in the use of the ASME Code editions or addenda specified as the Code of record for the US-APWR DC would require NRC approval prior to implementation. Therefore, in Request for Additional Information **(RAI) 264-2062, Question 05.02.01.01-01**, the staff requested that the applicant clarify the DCD regarding the need for the COL applicant to obtain NRC approval for the use of ASME Code editions or addenda other than the code of record specified in the US-APWR DCD. In its response to this RAI in a letter dated April 6, 2009, the applicant indicated its understanding of the need for the COL applicant to obtain NRC approval for the use of ASME Code editions or addenda other than the code of record in the DCD. The applicant stated that a provision would be included in the DCD to specify that any proposed changes by the COL applicant in the use of the ASME Code editions or addenda specified in the DCD would require NRC approval prior to implementation. Subsequently, the staff found that the modification to Section 5.2.1.1 in Revision 2 to DCD Tier 2 was not consistent with the RAI response and this was communicated to the applicant. In an amended response, dated December 15, 2009, to **RAI 264-2062, Question 05.02.01.01-01**, the applicant indicated that DCD Tier 2, Section 5.2.1.1, would be revised to state that any proposed changes by the COL applicant in the use of the ASME Code editions or addenda specified in the DCD would require NRC approval prior to implementation. The applicant also stated that DCD Tier

2, Section 5.2.6, would be revised to include a COL information item that would specify that the COL applicant address whether ASME Code editions or addenda other than those specified in Table 5.2.1-1 would be used. The NRC staff considers that the proposed changes to the DCD will clarify the need for a COL applicant to obtain approval to use ASME Code editions or addenda other than the code of record specified in the DCD. The staff will confirm that the proposed changes are properly incorporated in the next revision of the DCD. **This is Confirmatory Item (CI)-05.02.01.01-01.**

The staff notes that the ASME Code as the code of record for the US-APWR design is Tier 1 information, but the specific edition and addenda are designated Tier 2 because of the continually evolving design and construction practices (including inspection and examination techniques) of the Code. Establishing a specific edition and addenda during the design certification stage might result in inconsistencies between design and construction practices during the detailed design and construction stages. The ASME Code involves a consensus process to reflect the evolving design and construction practices of the industry. Although reference to a specific edition of the Code for the design of the ASME Code class components and their supports is necessary to reach a safety finding during the design certification stage, it is also important that the construction practices and examination methods of an updated Code be consistent with the design practices established at the design certification stage.

To avoid this potential inconsistency for US-APWR pressure-retaining components and their supports, proposed changes to the specific edition and addenda require NRC approval at the COL stage before implementation as discussed above. This provides the COL applicant with the option to revise or supplement the referenced Code edition with portions of later Code editions and addenda to ensure consistency between the design and construction practices. However, the staff acknowledges that a need might exist to establish certain design parameters from a specific Code edition or addenda during its design certification review, particularly when the information is important to developing a significant aspect of the design or is used by the staff to reach its safety determination. Various sections of this report reflect such considerations, as necessary. Therefore, all ASME Code Class 1, 2, and 3 pressure-retaining components and their supports shall be designed in accordance with the requirements of ASME Code, Section III, using the specific edition and addenda indicated in the DCD.

A COL applicant referencing the US-APWR design will be expected to ensure that the design is consistent with the construction practices (including inspection and examination methods) of the ASME Code edition and addenda in effect at the time of the COL application, as endorsed in 10 CFR 50.55a. If the ASME Code edition and addenda differ from those specified in the US-APWR DCD, as discussed above, the COL applicant must identify in its application the portions of the later code editions and addenda for NRC staff review and approval.

DCD Tier 2, Section 5.2.1.1, states that the RCPB component classification complies with the requirements of GDC 1 and 10 CFR 50.55a. Table 3.2-2, "Classification Summary," lists the RCPB components, including pressure vessels, piping, pumps, and valves, along with the applicable component codes. Other safety-related plant components are classified in accordance with RG 1.26, as specified in DCD Section 3.2.

DCD, Tier 2, Table 5.2.1-1, specifies the 2001 Edition with the 2003 Addenda of the ASME Code applicable for the design of Class1 components such as Reactor Vessel (RV), Steam Generators (SGs), Pressurizer, Control Rod Drive Mechanism (CRDM) housing, CRDM head adapter, Reactor Coolant Pumps (RCPs), Valves, except for seismic design of piping. Note (1) of Table 5.2.1-1 indicates that 1992 Edition and 1992 Addenda of ASME Code, Section III NB-3200, NB-3600, NC-3600 and ND-3600, will be used for the seismic design of US-APWR piping in accordance with the requirements of 10 CFR 50.55a(b)(1)(iii). In **RAI 290-2303, Question 05.02.01.01-02 (eRAI 2303 / Question 9592)**, the staff requested that the applicant discuss how the use of the baseline code of the 2001 Edition with the 2003 Addenda of the ASME Code and the use of the 1992 Edition and 1992 Addenda for seismic design piping would satisfy the requirements of 10 CFR 50.55a (b)(1)(ii). The staff noted that when applying the 1989 Addenda through the latest edition, and addenda incorporated by reference in paragraph (b)(1) of this section, applicants or licensees may not apply paragraph NB–3683.4(c), Footnote 11 to Figure NC–3673.2(b)–1, and Figure ND–3673.2(b)–1.

In its response to **RAI 290-2303, Question 05.02.01.01-02**, dated April 24, 2009, the applicant stated that it will continue to comply with the ASME Code requirements designated in 10 CFR 50.55a. The baseline Code of 2001 Edition with 2003 Addenda will be used as a Code of Record applicable for all components, except stress analysis for ASME Classes 1, 2 and 3 piping. The code criteria for stress analysis of the above mentioned piping systems will be based on 1992 Edition of ASME Code Section III through 1992 Addenda of Articles NB-3200, NB-3600, NC-3600 and ND-3600. However, for analyzing fillet welds, B1 and B2 stress indices for ASME Class 1 and stress intensification factor (SIF) for ASME Classes 2 and 3 piping analyses, the 1989 Edition (no addenda) will be used. The applicant indicated that DCD Section 5.2.1, Table 5.2.1-1, Note (1), will be changed to read: "For seismic design, 1992 Edition including 1992 Addenda will be used for ASME Section III NB-3200, NB-3600, NC-3600, ND-3600 analyses, in accordance with the requirements of 10 CFR 50.55a(b)(1)(iii). Except for analyzing fillet welds, B1 and B2 stress indices for ASME Class 1 piping analyses and stress intensification factor (SIF) for ASME Classes 2 and 3 piping analyses will use the 1989 Edition of ASME Code, Section III, Division 1, Subsections NB, NC and ND." The staff will confirm that the revised Note (1) statement is properly incorporated in the next DCD. **This is Confirmatory Item 05.02.01.01-02.**

DCD Tier 2, Section 5.2.1.1, states that the use of Code editions and addenda issued and endorsed by the NRC subsequent to the US-APWR is permitted. In **RAI 290-2303, Question 05.02.01.01-03 (eRAI 2303 / Question 9593)**, the staff requested the applicant to confirm the baseline code for the US-APWR design is the 2001 Edition with the 2003 Addenda of the ASME Code, incorporating the 1992 Edition including 1992 Addenda Code for US-APWR piping seismic design. The applicant was also asked to discuss how the COL applicant would justify when to apply ASME Code Edition and Addenda other than the code of record in the DCD as stated above. In its response to **RAI 2304, Question 05.02.01.01-03**, dated April 24, 2009, the applicant confirmed that the baseline code for the US-APWR design would be as specified in 10 CFR 50.55a and that the applicant would use the 1992 Edition and Addenda for seismic piping design. The applicant also stated that a COL application that incorporates by reference the US-APWR design certification may use subsequent ASME Code editions or addenda allowed by future revisions to 10 CFR 50.55a. When the COL applicant changes the code of record for construction, the deviation of the US-APWR design from construction practice based on the subsequent code editions and addenda identified in the COL application should be reconciled. The staff finds the applicant's response to be acceptable in accordance with requirements of ASME Section III, NCA-1140(b). **Therefore, RAI 2304, Question 05.02.01.01-03,** is resolved and closed.

5.2.1.1.5 Combined License Information Items

In its review of the US-APWR DCD, Tier 2, Section 5.2.1.1, the NRC noted that there are no applicable COL Information Items identified in Section 5.2.6. As noted in its amended response to **RAI 264-2062, Question 05.02.01.01-01**, the applicant plans to include a COL information item in DCD Tier 2, Section 5.2.6, that will specify that the COL applicant address whether the ASME Code editions or addenda other than those specified in Table 5.2.1-1 will be used. This DCD change will be reviewed as part of **CI-05.02.01.01-01**.

5.2.1.1.6 Conclusions

The NRC staff concludes that with the exception of the confirmatory items noted above, the information provided in the US-APWR DCD with respect to the use of codes and standards is sufficient to support compliance with the requirements of GDC 1, and 10 CFR 50.55a that nuclear power plant SSCs important to safety be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety function to be performed. Therefore, the US-APWR DCD is acceptable with respect to the use of codes and standards described therein, with the exception of the confirmatory items indicated in this SER section.

5.2.1.2 Compliance with Applicable Code Cases

5.2.1.2.1 Introduction

This section discusses the staff's review of the use of Code cases in accordance with the ASME BPV Code and ASME *Code for Operation and Maintenance of Nuclear Power Plants* (OM Code) in the design, fabrication, construction, and testing of RCS SSCs as described in DCD Tier 2, Section 5.2.1.2, "Compliance with Applicable Code Cases." Also discussed is the applicant's statement in this DCD section that the COL applicant should specify those ASME Code Cases to be used at the plant referencing the certified design, that are in effect at the time of the COL application that are applicable to RG 1.147 (ISI Code cases) and RG 1.192 (OM Code cases).

In general, a Code case applicable to nuclear power plants is developed by ASME based on inquiries from the nuclear industry associated with code clarification, modification or alternatives to the code. A Code case is the official method of handling a reply to an inquiry when a study indicates that the Code wording needs clarification, or when the reply modifies the existing requirements of the Code, or grants permission to use alternative methods. Code cases normally apply to a specific edition and addenda of the Code and used to expire after a period of time (e.g., 3 years), unless annulled or reaffirmed. The ASME Boiler and Pressure Vessel Standards Committee eliminated Code case expiration dates since March 11, 2005. This means that all Code cases listed in March 2005 and beyond will remain available for use until annulled by the

ASME Codes and Standards Committee. ASME Code cases acceptable to NRC staff are published in RGs 1.84, 1.147 and 1.92 in accordance with requirements of 10 CFR 50.55a(b)(4), (b)(5) and (b)(6).

5.2.1.2.2 Summary of Application

DCD Tier 1: The Tier 1 information associated with this section is found in DCD Tier 1, Section 2.4.2.

DCD Tier 2: The application identifies applicable Code Addenda for RCS Class 1 components in Table 5.2.1-1. The application also identifies ASME Code cases for RCPB Class 1 components in Table 5.2.1-2. The DCD also describes that other ASME Code cases in effect at the time of the DC may be used for pressure boundary components.

DCD Tier 2 (Revision 2) Section 5.2.1.2, "Compliance with Applicable Code Cases," references Table 5.2.1-2, "ASME Code Cases," for the use of ASME Code Cases for the US-APWR. DCD Tier 2, Section 5.2.1.2, states that acceptable Code Cases for ASME BPV Code, Section III can be found in RG 1.84, "Design and Fabrication Code Case Acceptability, ASME Section III, Division 1." This RG lists those Section III cases oriented to design, fabrication, materials, and testing, which are acceptable to the staff for implementation in the licensing of nuclear power plants.

DCD Tier 2, Section 5.2.1.2, states that the ASME Section III Code cases (design) acceptable for use in the US-APWR design, subject to the limitations in 10 CFR 50.55a, and as listed in RG 1.84, include Code Case N-71-18, "Additional Materials for Subsection NF, Class 1, 2, 3, and MC Supports Fabricated by Welding."

The application states that the ASME Section XI Code cases (repair/replacement and inservice inspection) acceptable for use in the US-APWR design are subject to the limitations in 10 CFR 50.55a and are listed in RG 1.147. However, there are no Section XI Code cases invoked for Class 1 pre-service or inservice requirements for the US-APWR design. The application also states that the ASME OM Code cases (inservice testing) acceptable for use in the US-APWR design are subject to the limitations in 10 CFR 50.55a and are listed in RG 1.192.

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

Technical Specifications: There are no Technical Specifications associated with this area of review.

5.2.1.2.3 Regulatory Basis

The relevant requirements of the Commission's regulations for this area of review, and the associated acceptance criteria, are given in Section 5.2.1.2 of NUREG-0800, the SRP, and are summarized below. Review interfaces with other SRP sections can be found in Section 5.2.1.2 of NUREG-0800.

1. 10 CFR Part 50, Appendix A, GDC 1, which requires 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. 10 CFR 50.55a, which establishes minimum quality standards for the design, fabrication, erection, construction, testing, and inspection of certain components of boiling- and pressurized-water reactor nuclear power plants by requiring conformance with appropriate editions of specified published industry codes and standards. This requirement is applicable to both pressure- retaining and non-pressure- retaining SSCs that are related to the RCPB, as well as other systems important to safety. Where generally recognized codes and standards are used, they must be identified and evaluated to determine their applicability and adequacy.

Acceptance criteria adequate to meet the above requirements include:

1. RG 1.84, “Design and Fabrication Code Case Acceptability, ASME Section III, Division 1,”as it relates to ASME Section III Code cases.

2. RG 1.147, “Inservice Inspection Code Cases Acceptability, ASME Section XI, Division 1,” as it relates to ASME Section XI Code cases.

3. RG 1.192, “Operation and Maintenance Code Case Acceptability – ASME OM Code,” as it relates to ASME OM Code cases.

5.2.1.2.4 Technical Evaluation

The NRC staff has reviewed the US-APWR DCD for compliance in accordance with 10 CFR Part 52. The only acceptable ASME Code cases that may be used for the design of ASME Code Class 1, 2, 3 piping systems in the US-APWR standard plant are those that are either conditionally or unconditionally approved in RG 1.84, and that are in effect at the time of the design certification, or determined to be conditionally acceptable.

The COL applicant may submit with its application future ASME Code cases that are endorsed in RG 1.84 at the time of the COL application, provided that the Code Cases do not alter the safety evaluation on the US-APWR certified design.

DCD Tier 2, Section 5.2.1.2, states that any Code case conditionally approved in RG 1.84 for Class 1 components meets the condition established in the RG. DCD Tier 2, Table 5.2.1-2, lists ASME Code Case N-71-18, “Additional Materials for Subsection NF, Class 1, 2, 3 and MC Supports Fabricated by Welding, Section III, Division 1,” for use in the design of supports for specific nuclear power plant components. In **RAI 253-2063, Question 05.02.01.02-01 (also identified as RAI 2063, Question 8279)**, the NRC staff requested that the applicant specify in the DCD the components that will be fabricated using Code Case N-71-18 and the material specifications and grades that will be used. In its response to this RAI in a letter dated April 17, 2009, the applicant stated that DCD Table 5.2.1-2 identifies the RCPB component supports to which Code Case N-71-18 and Code Case N-249-14, “Additional Materials for Subsection NF, Class 1, 2, 3, and MC Supports Fabricated Without Welding, Section III, Division I,” may be applied. The applicant stated that these components will be fabricated from carbon steel material such as SA-36, consistent with the application and criteria of the ASME Code cases. The NRC staff finds that the information provided by the applicant clarifies the application of Code Cases N-71-18 and N-249-14, and the material to be used in these applications. Therefore, **RAI 253-2063, Question 05.02.01.02-01,** is resolved and closed.

US-APWR DCD Tier 2, Section 5.2.1.2 states that ASME Code cases for Class 2 and 3 piping are covered in Section 3.12, "Piping Design Review." 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. These ASME Code cases are listed as acceptable in RG 1.84. DCD Tier 2, Section 5.2.1.2, states that other ASME Code cases may be used in the US-APWR DC if they are either conditionally or unconditionally approved in RG 1.84. However, the staff notes that the only acceptable ASME Code cases that may be used for the design of ASME Code Class 1, 2, and 3 piping systems in the certified design plant are those either conditionally or unconditionally approved in RG 1.84, and that are in effect at the time of the DC, or determined to be conditionally acceptable in the NRC SER on the DC application. A COL applicant may submit, with its COL application, future ASME Code cases that are endorsed in RG 1.84 at the time of the application, provided that they do not alter the staff's safety findings on the certified design. In addition, the COL applicant should specify those ASME Code cases to be used at the plant referencing the certified design, that are in effect at the time of the COL application that are applicable to RG 1.147 and RG 1.192. In **RAI 253-2063, Question 05.02.01.02-02 (also identified as RAI 2064, Question 8280)**, the NRC staff requested that the applicant clarify the need for the COL applicant to identify ASME Code cases to be used at the plant referencing the US-APWR design certification in the COL application. In its response to this RAI in its letter dated April 17, 2009, the applicant stated that the DCD would be revised to specify the need for the COL applicant to identify ASME Code cases to be used at the plant referencing the US-APWR design certification in the COL application. Subsequently, the staff found that Revision 2 to US-APWR DCD Tier 2, Section 5.2.1.2, includes the following provisions:

> A COL applicant may submit, with its COL application, future ASME Code Cases that are endorsed in RG 1.84 at the time of the application, provided that they do not alter the NRC staff's safety findings on the certified design. In addition, the COL applicant should specify those ASME Code Cases to be used at the plant referencing the certified design, which are in effect at the time of the COL application that are applicable to RG 1.147 and 1.192.

The staff finds that Revision 2 to the US-APWR DCD clarifies the responsibility of the COL applicant to identify ASME Code cases that are planned to be applied for a US-APWR plant at the time of the COL application. Therefore, **RAI 253-2063, Question 05.02.01.02-02,** is resolved and closed.

US-APWR DCD Tier 2, Section 5.2.1.2 states that the COL applicant addresses ASME Code cases that are approved in RG 1.147 and RG 1.192. DCD Tier 2, Section 5.2.6, "Combined License Information," lists COL information items COL 5.2(1), (2), and (3) that specify that the COL, applicant should address ASME Code cases approved in RGs 1.84, 1.147, and 1.192, respectively. In **RAI 253-2063, Question 05.02.01.02-03 (also**

identified as RAI 2065, Question 8281), the NRC staff requested that the applicant specify in the DCD the ASME BPV Code, Section III and Section XI, Code cases, and ASME OM Code, Code cases currently planned to be applied as part of the US-APWR design. In its response to **RAI 253-2063, Question 05.02.01.02-03**, in its letter dated April 17, 2009, the applicant stated that ASME BPV Code Section XI, Code Cases N-307-3 and N-613-1 for Reactor Vessel (RV), Steam Generator, and Pressurizer, and N-729-1 for RV upper head are currently planned to be used. The applicant stated that Code Case OMN-13 for the ASME OM Code is currently planned to be applied to the inservice testing of snubbers as stated in DCD Tier 2, Section 3.9.3.4.2.6. The applicant indicated that DCD Tier 2, Table 5.2.1-2, would be revised to list these Code cases. The applicant noted that the ASME BPV Code, Section III Code cases that may be used as part of the US-APWR RCBP Class 1 component design are listed in DCD Tier 2, Table 5.2.1-2. The applicant stated that other ASME BPV Code, Section III Code cases that may be used are N-318-5 and N-392-3 for Class 2 and 3 piping as stated in DCD Tier 2, Section 3.12.2.2, and N-4-11 for Core Support Structures as stated in DCD Tier 2, Section 4.5.2.1. The applicant noted that Table 5.2.1-2 applies to RCPB components and does not list these other Code cases. As planned in the RAI response, Revision 2 to DCD Tier 2 lists ASME Code Cases N-307-3, N-613-1, N-729-1, and OMN-13 in Table 5.2.1-2. The staff finds that Revision 2 to the US-APWR DCD clarifies the use of ASME Code Cases currently planned for the US-APWR standard design. Therefore, **RAI 253-2063, Question 05.02.01.02-03**, is resolved and closed.

In a similar RAI question, **eRAI 2063 / Question 9595**, the staff requested the applicant to provide all the ASME Section III, Section XI and OM Code cases that were used or will be used (for instance, Code Cases N-284-2, N-759-2, N-729-1, if used) for US-APWR component design certification. As discussed above regarding RAI 253-2063, Question 05.02.01.02-3, in its response to **RAI 253-2063, Question 05.02.01.02-03**, dated April 17, 2009, the applicant indicated that it planned to use the ASME Section XI Code Cases N-307-3, N-613-1 and N-729-1 for preservice and inservice inspection and the ASME OM Code Case OMN-13 for inservice testing of snubbers as stated in the DCD Subsection 3.9.3.4.2.6 and that DCD Tier 2 would be revised to add these Code cases to Table 5.2.1-2 of Section 5.2.1.2. Therefore, on the basis of the response to **RAI 253-2063, Question 05.02.01.02-03** and the associated changes to the DCD in Revision 2, **eRAI 2063, Question 9595**, is resolved and closed**.**

In **RAI 291-2301, Questions 05.02.01.02-04** and **Question 05.02.01.02-05**, the staff requested the applicant to address additional ASME Code cases that might need to be listed in DCD Tier 2, Table 5.2.1-2. In its response to these RAIs in its letter dated April 17, 2009, the applicant reported that some of the referenced Code cases had been incorporated into the ASME BPV Code and did not need to be listed in Table 5.2.1-2. For some other referenced Code cases, the applicant stated that those Code cases will not be applied to RCPB components in the US-APWR design. As noted above, the applicant revised Table 5.2.1-2 to list several additional Code cases. The staff finds that the applicant's response and the DCD revision clarify the use of Code cases in the US-APWR design. Therefore, **RAI 291-2301, Questions 05.02.01.02-04 and -05**, are resolved and closed.

However, that staff also noted that DCD Tier 2, Section 5.3.2.1, "Limit Curve," states that the methods outlined in ASME Code Section XI, Appendix G, including defect sizes and safety factors, are applied in the analyses for protection against non-ductile failure.

ASME Code Section XI, Appendix G, is applied rather than ASME Code Section III, Appendix G, as it is referenced by 10 CFR Part 50, Appendix G, and also incorporates several ASME Code cases including N-588, N-640 and N-641. In **eRAI 2301, Question 9594**, the staff requested the applicant to confirm whether these Code cases mentioned above were used in the US-APWR design. If yes, they should be listed in Table 5.2.1-2 for the applicable ASME Code cases. The applicant was also requested to discuss how the use of these Code cases is acceptable in the US-APWR design. In its response to **RAI 291-2301, Question 05.02.01.02-04**, dated April 17, 2009, which also addresses **eRAI 2301, Question 9594,** the applicant indicated that Code cases N-588, N-640 and N-641 have been incorporated into Appendix G of ASME Section XI. The US-APWR design therefore directly uses Appendix G of Section XI of the ASME Code in lieu of the Code cases. The staff agrees with the applicant that these Code cases no longer need to be used for the US-APWR design and are not required to be contained in Table 5.2.1-2, "Applicable Code Cases." Therefore, **eRAI 2301, Question 9594**, is resolved and closed by the response to **RAI 291-2301, Question 05.02.01.02-04**.

DCD Tier 2, Table 5.2.1-2, provides the applicable ASME Code cases for Classes 1, 2 and 3 pressure boundary piping and components. ASME Code cases acceptable to the NRC staff are published in RG 1.84, 1.147 and 1.92 in accordance with requirements of 10 CFR 50.55a(b)(4), (b)(5) and (b)(6). In its review, the staff found that Table 5.2.1-2 listed Code Case 2142-2. However, the staff noted that this Code case is a non-nuclear code case and is therefore not accepted by the NRC in RG 1.84. Therefore, in **RAI 315-2289, Question 05.02.01.02-06 (eRAI 2289 / Question 9554)**, the staff requested the applicant to discuss how Code Case 2142-2, "F-Number Grouping for Ni-Cr-Fe Filler Metals Section XI," (applicable to all Sections, including Section III, Division 1, and Section XI) is acceptable for use in the design of US-APWR Class 1 components in accordance with 10 CFR 50.55a(a)(3)(i) and (ii) when it is not accepted by NRC in the RGs mentioned above. In its response to **RAI 315-2289, Question 05.02.01.02-06 (eRAI 2289/Question 9554)**, dated April 28, 2009, the applicant stated that Code Case 2142-2 has been incorporated in ASME Section XI. However, the applicant explained that the material specification of this welding material is incorporated into the 2006 Edition of ASME Section II Part C. The applicant further explained that in DCD Tier 2, the applicable Code addenda of ASME Section II for the US-APWR RCS Class 1 components are specified to be the 2001 Edition with 2003 Addenda, as stated in DCD Table 5.2.1-1. Thus, the applicant contended it is necessary to use Code Case 2142-2. The staff considers this ASME Code case to be acceptable because it includes the weld metal to be used in the welding of Ni-Cr-Fe Alloy 690, which the staff endorsed and accepted for use in its safety evaluation report (SER) for the Electric Power Research Institute (EPRI), "Advanced Light Water Reactor Utility Requirements Document," Volume III. On that basis, the staff finds that the use of Code Case 2142-2 in this application is justified. Therefore, **RAI 315-2289, Question 05.02.01.02-06**, is resolved and closed.

Section 50.55a of 10 CFR Part 50 requires that the Code edition and addenda to be applied to ASME Class 1, 2 and 3 piping and components must be determined by the rules of the ASME Section III paragraph NCA-1 140(2), which disallows use of Code Edition and Addenda in the Design Specifications that are (a) earlier than 3 years prior to the date the construction permit application is docketed or (b) earlier than the latest Edition and Addenda endorsed by the regulatory authority at the time the construction permit application is docketed. This requirement is not satisfied by the code of record for the US-APWR. For instance, the code used for the US-APWR DC is the 2001 Edition through the 2003 Addenda while the COL application date for Comanche Peak, Units 3 and 4, is September 19, 2008. This implies a violation of NCA-1 140(2)(a). To resolve the issue, ASME approved a Code Case N-782 in January 2009, which allows the Code Edition and Addenda endorsed in a design certification or licensed by the regulatory authority to be used for systems and components as an alternative rule to NCA-1 140(2)(a) and (2)(b). Also, 10 CFR 50.55a(b)(4) conditionally allows the application of ASME Code cases listed in the NRC RG 1.84, Revision 34, without prior NRC approval. However, Code Case N-782 is not listed for acceptance in Revision 34 of RG 1.84.

In order to apply the alternative rule to the requirements of NCA-1140(2)(a) and (b), in **RAI 575-4422, Question 05.02.01.02-07**, the staff requested the applicant to provide justification for inclusion of Code Case N-782 in the DCD in accordance with 10 CFR 50.55a(3)(i) and (ii). In its response, dated May 7, 2010, the applicant indicated that the DCD would be revised to add Code Case N-782 in Section 5.2.1.2, Table 5.2.1-2, as shown in the attached markup. However, the applicant still needs to provide justification showing how the code case is applicable and acceptable for the US-APWR in accordance with 10 CFR 50.55a(a)(3)(i) and (ii). **This is Open Item 05.02.01.02-07.**

5.2.1.2.5 Combined License Information Items

DCD, Tier 2, Section 5.2.6, "Combined License Information," and Table 1.8-2, "Compilation of All Combined License Applicant Items," of the DCD include the following COL information items related to Section 5.2.1.2:

Table 5.2.1.2-1
US-APWR Combined License Information Items

Item No.	Description	Section
5.2(1)	ASME Code cases that are approved in RG 1.84; the COL applicant addresses the addition of ASME Code cases that are approved in RG 1.84.	5.2.1.2
5.2(2)	ASME Code cases that are approved in RG 1.147; the COL applicant addresses Code cases invoked in connection with the inservice inspection program that are in compliance with RG 1.147.	5.2.1.2
5.2(3)	ASME Code cases that are approved in RG 1.192; the COL applicant addresses Code cases invoked in connection with the operation and maintenance that are in compliance with RG 1.192.	5.2.1.2

COL information items not identified in Table 1.8-2 of the DCD: **None**

5.2.1.2.6 Conclusions

The NRC staff concludes that with the exception of OI 05.02.01.02-07, the information provided in the US-APWR DCD with respect to Section 5.2.1.2, "Compliance with Applicable Code Cases," is sufficient to support compliance with the requirements of 10 CFR Part 50, Appendix A, GDC 1, that nuclear power plant SSCs important to safety shall be designed, fabricated, erected and tested to quality standards commensurate with the importance of the safety function to be performed. Therefore, with the exception of the OI cited above, the US-APWR design certification application is acceptable with respect to the use of ASME Code Cases for the RCPB in the US-APWR design. A COL applicant may identify within its COL application the planned use of additional Code Cases provided they do not alter the staff's safety findings on the US-APWR design.

5.2.2 Overpressure Protection

5.2.2.1 Introduction

The overpressure protection systems provide for overpressure protection in the RCS and the primary side of the main steam system through the use of pressurizer safety valves and the main steam safety valves, along with the resulting action of the reactor protection system.

Overpressure protection system pressure relieving devices include the pressure-relieving devices installed in: RCS, primary side of auxiliary or emergency systems connected to the RCS, blowdown or heat dissipation systems connected to the discharge of these pressure-relieving devices, and SGs of the main steam supply system (MSS). The auxiliary system connected to the RCS utilized for overpressure protection consists only of the containment spray/residual heat removal (CS/RHR) pump suction relief valves, which protect the RCS during low-temperature operations when the RHRS is in service.

5.2.2.2 Summary of Application

DCD Tier 1: The DCD Tier 1 information associated with this section is found in Tier 1, Section 2.4.2. Section 2.4.2 identifies that the pressurizer safety valves provide overpressure protection for the RCS. In addition, DCD Tier 1 information related to overpressure protection is also provided in DCD Sections 2.4.5 (RHRS) and 2.7.1.2 (MSS).

DCD Tier 2: The Tier 2 information associated with this section is found in Tier 2, Section 5.2.2.1. In addition, Tier 2 information associated with overpressure protection is discussed in DCD Section 10.3.2 (MSS). Also, RHRS overpressure protection is described in DCD Section 5.2.2.1.2 under low-temperature overpressure protection (LTOP).

DCD Section 5.2.2.1.1 states that the functional design of the overpressure protection which is described in DCD Section 3.1 is in conformance with the requirements of GDC 15 and 31.

The overpressure protection system is able to perform its function assuming any single failure and loss of offsite power. The basis for the sizing of the pressurizer safety valves includes that the pressurizer safety valve capacity prevents exceeding 110 percent of RCS design pressure for the bounding events.

For LTOP, the DCD describes the administrative and plant design features that provide for overpressure protection during low-temperature reactor operation. The design features include annunciation to alert the operator to arm the cold overpressure mitigation system and maximizing the use of a pressurizer steam bubble which moderates maximum pressure and rate of pressure increase for specified transients. The DCD also describes methods of LTOP protection when the RHRS is aligned to the RCS to provide RHR from the core. The CS/RHR design features include suction line relief valves that open to mitigate overpressure transients when the RCS temperature is less than 350 °F.

The LTOP is designed to prevent exceeding the applicable TS and Appendix G limits for the RCS while operating at low temperatures. The LTOP system is operable below the enabling temperature as described in ASME Code, Section III, Appendix G, limit calculations. The LTOP system is able to perform its function assuming any single active component failure as demonstrated through calculations that demonstrate that the required pressure relief capacity can be achieved. The analyses will assume the most limiting allowable operating conditions and systems configuration for the postulated cause of the overpressure event. Controls, alarms, and interlocks exist, in accordance with the IEEE Standard 603 guidance, to alert the operator to enable the system at the correct condition during cooldown. Positive indication confirms that the LTOP system is enabled and alarms will annunciate when the protective action is initiated.

The LTOP system is testable in accordance with the TS surveillance requirements for valve operability. The LTOP system will meet the requirements of RG 1.26 and Section III of the ASME Code and is designed to function during an operating basis earthquake in accordance with RG 1.29.

Chapter 15 describes the overpressure protection evaluation for RCS, except LTOP. The pressurizer relief capacities are determined from the overpressure transient conditions in conjunction with the resulting action of the reactor protection system.

For low-temperature conditions in which the RCS is in water-solid conditions, over-pressurization transients can be caused by either mass input, or heat input. The events upon which relieving capability is based include inadvertent safety injection (SI), charging/letdown flow mismatch, inadvertent actuation of the pressurizer heaters, loss of RHR cooling, and inadvertent start of one RCP. In accordance with ASME Code, Section III, Appendix G, anticipated mass input transients are evaluated and determine that inadvertent SI is the most limiting event. Likewise, the most limiting heat input transient is an inadvertent RCP start in a loop where the SG temperature is 50 °F higher than the other temperatures in the loop. To prevent or minimize the likelihood of occurrences of these conditions, administrative controls exist to provide plant operation precautions.

ITAAC: The ITAAC associated with Tier 2, Section 5.2.2 are given in Tier 1, Section 2.4.2.2. They include inspections to confirm that the vendor code plate rating is greater than or equal to system relief requirements ($1.728X10^6$lb/hr), and testing each pressurizer safety valves to ensure that they lift below the set pressure maximum (2,485 pounds per square inch gauge [psig]) and above the set pressure minimum (2,435 psig).

Technical Specifications: US-APWR TS 3.4.10 requires that a four pressurizer safety valves OPERABLE with a lift setting of ≥ 2435 psig and ≤ 2485 psig when in Modes 1, 2, and 3 and in Mode 4 when all RCS cold leg temperatures are greater than the LTOP arming temperature specified in the pressure-temperature limits report (PTLR). The US-APWR TS 3.4.11 requires that two safety depressurization valves (SDVs) and associated block valves are OPERABLE when in Modes 1, 2, and 3. TS 3.4.12 requires that the LTOP system shall be OPERABLE with a maximum of two SI pumps and one charging pump capable of injecting into the RCS and accumulators isolated when the RCS temperature is less than the LTOP arming temperature specified in the PTLR in Mode 4, or in Mode 5, or Mode 6 when the reactor vessel head is on.

Topical Reports: Non-LOCA [loss-of-coolant accident] Methodology Topical Report, MUAP-07010-NP (Non-Proprietary), July 2007, (Agencywide Document Access and Management System (ADAMS) Accession No. ML083650173)

5.2.2.3 Regulatory Basis

The relevant requirements of the Commission's regulations for this area of review, and the associated acceptance criteria, are given in Section 5.2.2 of NUREG-0800 and are summarized below. Review interfaces with other SRP sections can be found in Section 5.2.2 of NUREG-0800.

1. GDC 1, as it relates to SSCs important to safety being designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed.

2. GDC 15, as it relates to designing the RCS and associated auxiliary, control, and protection systems with sufficient margin to assure that the design conditions of the reactor coolant pressure boundary (RCPB) are not exceeded during any condition of normal operation, including AOOs.

3. GDC 30, as it relates to components, that are part of the RCPB being designed, fabricated, erected, and tested to the highest quality standards practical.

4. GDC 31, as it relates to designing the RCPB with sufficient margin to assure 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 fractures is minimized.

5. 10 CFR 50.34(f)(2)(x) and 10 CFR 50.34(f)(2)(xi) require that RCS safety/relief valves (SRVs) meet Three Mile Island (TMI) Action Plan Items II.D.1 and II.D.3 of NUREG-0737.

6. 10 CFR 52.47(a)(8) provides the requirement for DC reviews to comply with the technically relevant portions of the TMI requirements in 10 CFR 50.34(f).

7. 10 CFR 50.55a, as it relates to meeting codes and standards.

8. 10 CFR 52.79(a)(17) provides the requirement for COL applications to comply with the technically relevant information in 10 CFR 50.34. This includes the

TMI-related requirements specified by 10 CFR 50.34(f)(2)(x) and 10 CFR 50.34(f)(2)(xi).

9. Branch Technical Position (BTP) 5-2, "Overpressurization Protection of Pressurized-Water Reactors When Operating at Low Temperatures."

Acceptance criteria adequate to meet the above requirements include:

The applicable provisions of the ASME Code (e.g., ASME Code, Section II, NB-2000; ASME Code Section III, NB-2000, NB-3500, NB-7000, and NB-7511.1) and by acceptable application of material Code cases, as described in RG 1.84.

5.2.2.4 Technical Evaluation

The staff reviewed the overpressure protection systems for the US-APWR. The primary functions of the US-APWR overpressure protection systems are to prevent the RCS of the primary side of SG and the MSS of the secondary side of the SG from exceeding 110 percent of the of their design pressure during power operations. In addition, during startup and shutdown operations, LTOP of the RCS is provided by the CS/RHR pump suction relief valves to mitigate pressure transients originating in the RCS to maximum pressure values determined by the relief valve set pressure. This design feature was also reviewed by the staff.

In reference to overpressure protection systems, the applicant stated that the "combinations of these systems provide compliance with the overpressure protection requirements of the ASME Boiler and Pressure Vessel Code, Section III, Paragraph NB-7200, 'Overpressure Protection Report for Pressurized Water Reactor Systems.'" In its review of the overpressure protection systems of this DCD section with regard to compliance with ASME III NB-7200 the staff determined that additional documentation is required to support applicant's positions on the following ASME III NB-7200 requirements: "…(c) the range of operating conditions, including the effect of discharge piping back pressure; (k) consideration of set pressure and blowdown limitations, taking into account opening pressure tolerances and overpressure of the pressure relief device; and (l) consideration of burst pressure tolerance and manufacturing design range of the rupture disk device." Therefore, in **RAI 103-1448, Questions 05.02.02-01** and **-02**, the staff requested that the applicant provide the references or documentation that support compliance with NB-7200 requirements.

In its response to **RAI 103-1448, Questions, 05.02.02-01** and **-02**, dated December 25, 2008, the applicant adequately addressed the US-APWR design compliance with ASME Section III, NB-7200, requirements (c), (k), and (l). The response included a discussion of the pressurizer safety valve backpressure compensation and rupture disk device burst pressure tolerance and manufacturing design range with respect to ASME criteria compliance. In addition, the applicant provided a discussion of the sizing of the pressurizer safety valves with a description of the operating conditions, set pressure, opening pressure tolerances and overpressure of the pressure relief device assumed in the analysis that is used in the MAREL-M plant transient analysis code in Non-LOCA Methodology Topical Report, MUAP-07010-NP (Non-Proprietary), July 2007. Therefore, the staff finds the applicant's explanation acceptable and **RAI 103-1448**, **Question 05.02.02-01**, is resolved and closed.

In DCD Section 5.2.2.1.1, "Design Bases for Overpressure Protection of RCS," the applicant states that "The functional design of the overpressure protection is in conformance with the requirements of GDC 15…" where "compliance with GDC is discussed in [DCD] Section 3.1." The staff review of DCD Section 3.1, "Conformance with NRC General Design Criteria," found the applicant discussion was in generalities with insufficient specific information to support its position. However, in conjunction with DCD Section 5.2.2.1.1 and DCD Chapter 15, "Transient and Accident Analyses," the staff review of the bounding events as discussed and summarized below concludes that compliance with GDC 15 is satisfied during power operation pending approval of DCD Chapter 15.

At power operations, overpressure protection is provided for the RCS by the pressurizer safety valves for the following bounding events: (1) Loss of external electrical load, (2) Loss of normal feedwater flow, (3) Reactor coolant pump shaft break, (4) Uncontrolled rod cluster control assembly bank withdrawal from a subcritical or low-power startup condition, and (5) Spectrum of rod cluster control assembly ejection accidents. Therefore, to assure total steam relief, the optimum sizing analysis of the pressurizer safety valves was selected based on a complete loss of steam flow to the turbine with the reactor operating at 102 percent of the design nuclear steam supply system (NSSS) thermal power with the assumptions of loss of feedwater flow, no credit for operation of the pressurizer level control system, pressurizer spray system, rod control system, turbine bypass system, or main steam relief valves, and no reactor trip. Steam relief through the main steam safety valves is considered. The sizing analysis results in a pressurizer safety valve capacity that is in excess of the capacity required to prevent exceeding 110 percent of system design pressure for the above listed events. Before concluding that the pressurizer safety relief valves and their setpoints meet the ASME criteria for over-pressure protection, the staff requested additional information from the applicant to provide the references for the methodology, analysis, and results that support the applicant's position that the ASME criteria are satisfied.

The applicant stated in its response to **RAI 103-1448, Question 05.02.02-02**, dated December 25, 2008, that the "sizing methodology results in a pressurizer safety valve capacity in excess of the capacity required to prevent exceeding 110 percent of the RCS design pressure" while excluding the first reactor trip in its analysis. This is in compliance with ASME Code, Paragraph NB-7311. To comply with the ASME Code requirements, the analysis assures that the pressurizer steam space volume does not exceed 103 percent of design pressure corresponding to the pressure when the safety valve is fully open for the most limiting AOO (loss of load event). The analysis considers piping downstream of the safety valve discharge, pressure difference between the safety valve inlet and the highest pressure in the RCS (pump discharge during operation), pressure drops around the RCS piping, and elevation heads in the RCS. The staff finds the applicant response acceptable and **RAI 103-1448**, **Question 05.02.02-02**, is resolved and closed.

During low-power operations, at temperatures below approximately 350 ^{0}F, the RCS is aligned to the RHRS which provides a means of protecting the RCPB from over-pressurization in addition to removing residual heat from the core, providing a path for letdown to the purification subsystem, and controlling RCS pressure when the pressurizer is operating in a water-solid mode. The RHRS is designed with CS/RHR pump discharge and suction self-actuated water relief valves to mitigate pressure transients originating within the system or from transients transmitted from the RCS.

In DCD Subsection 5.2.2.1.2, the LTOP design is discussed in addressing BTP 5-2. The applicant states that "the system is designed and installed to prevent exceeding the applicable technical specifications and Appendix G limits for the RCS while operating at low temperatures. The system is capable of relieving pressure during all anticipated over-pressurization events at a rate sufficient to satisfy the technical specification limits, particularly while the RCS is in a water-solid condition." In **RAI 103-1448, Question 05.02.02-03**, the staff requested the applicant to provide the references related to the analysis or supporting documentation which demonstrates the above requirements are satisfied.

In its response to **RAI 103-1448, Question 05.02.02-03**, dated December 25, 2008, the applicant provided Attachment A to the response letter, which summarizes the LTOP analysis for the US-APWR. The analysis focused on two types of events, mass input and heat input, that would cause an overpressure transient in the RCS at low temperature. The applicant used the MARVEL-M plant transient analysis code to calculate the transient responses of primary coolant pressure for cold overpressure event.

For the mass input case, the applicant assumed the inadvertent actuation of two high-head injection pumps as the initiating event with the RCS solid and both RHR relief valves operable while operating at initial primary coolant temperature and pressure of 100^{o} F and 400 psig, respectively. The results indicate a maximum primary coolant pressure increase of 120 psi.

For the heat input case, the applicant assumed the inadvertent start of one reactor coolant pump as the initiating event with the RCS solid and both RHR relief valves operable while operating at initial conditions with the primary coolant temperature and pressure and secondary coolant temperature of 100 ^{o}F, 400 psig, and 150 ^{o}F, respectively. The result was a maximum primary coolant pressure increase of 95.1 psi.

Based on the results, the maximum pressure (mass input) is 520 psi. From Figures 5.3-2 and 5.3-3, representative P-T Limit Curves for Heatup and Cooldown, it is demonstrated, at approximately 125 ^{o}F, that the allowable pressure limit is flat and minimized with the upper limit of approximately 620 psi. Since the maximum analyzed pressure is well below the upper limit, the LTOP system satisfies the requirements. Whenever the reactor coolant temperature is below 95 ^{o}F, the upper limit slopes downward and approaches the RHR operating pressure of 400 psi. Therefore, special conditions must be taken to prevent over pressurization from another system. The applicant addressed this situation by proposing that sufficient open area, such as reactor head removal, should be provided to avoid exceeding the pressure limit. With the RCS temperatures below 90^{o} F, the staff believes this is indicative of an outage condition and most likely the reactor head would be off. Therefore, the staff finds the response acceptable and **RAI 103-1448, Question 05.02.02-03**, is resolved and closed.

In addition, BTP 5-2 states that the system should be able to perform its function assuming any single active component failure. The applicant stated that "Analyses using appropriate calculation techniques will demonstrate that the system provides the required pressure relief capacity assuming the most limiting single active failure. The cause for initiation of the event (e.g., operator error, component malfunction) will not be considered as the single active failure." Therefore, the analysis, at the time of the postulated cause of the overpressure event will assume the most limiting allowable operating conditions and systems configuration. In **RAI 103-1448, Question 05.05.02-04**, the staff requested the applicant to provide the references in regard to the analysis or supporting documentation which demonstrates the above requirements are satisfied.

The applicant stated in its response to **RAI 103-1448, Question 05.02.02-04**, dated December 25, 2008, that the LTOP system for [the] US-APWR consist[s] of CS/RHR pump suction relief valves, which are spring-loaded relief valves. As spring-loaded relief valves, they are passive components, thus, the LTOP system is a passive system. Therefore, any single active failure does not affect the LTOP system. Also, the applicant referred to Attachment A to its letter in response to RAI 130-1448, which was discussed and reviewed in regard to Question 05.02.02-3. Therefore, the staff concluded that the applicant's response was adequate and **RAI 103-1448, Question 05.02.02-04**, resolved and closed.

Also, in DCD Section 5.2.2.1.2, the applicant states that the system satisfies BTP 5-2 provision that the design of the system should use the guidance of IEEE Standard 603, which states that the system should be capable of being manually enabled. However, an alarm should be provided to alert the operator to enable the system at the correct plant condition during cooldown. Positive indication should be provided to indicate when the system is enabled. An alarm should activate when the protective action is initiated. In **RAI 103-1448, Question 05.02.02-05**, the staff requested the applicant to identify the plant (condition) parameter that is monitored during cooldown to alert the operator to enable the system and, if applicable, the TS related to the parameter.

In its response to **RAI 103-1448**, **Question 05.02.02-05**, dated December 25, 2008, the applicant provided an explanation that partially addressed the RAI question. In reference to "An alarm should activate when the protective action is initiated," the applicant adequately addressed this statement by noting that the CS/RHR pump suction relief valves are equipped with direct position indication and alarm in the control room. However, the applicant failed to address the specific request to "Identify the plant (condition) parameter that is monitored during cooldown to alert the operator to enable the system…" with respect to the DCD Subsection 5.2.2.1.2 statement that "The system may be manually enabled; however, an alarm will be provided to alert the operator to enable the system at the correct plant condition during cool down."

The primary concern during startup and shutdown conditions at low temperature, especially in a water-solid condition, is to prevent the reactor vessel pressure-temperature conditions from exceeding the limitations established in the technical specifications for the protection against brittle fracture. Plant malfunctions or operator errors could create an inadvertent over-pressurization such as the many events described in NUREG-0138 that have occurred at operating plants.

Therefore, the initial response to **RAI 103-1448**, **Question 05.02.02-05,** required additional clarification to address the guidance of IEEE Standard 603 as described in the BTP 5-2 provisions to identify plant conditions/indications monitored in the control room (CR) in regard to initiation of LTOP operation: (1) The plant parameter that is monitored during cooldown to alert the operator to enable the system and (2) The alarm (positive indication) when the LTOP system is enabled. This was identified as **Open Item 05.02.02-01**. However, in a conference call on January 25, 2011, the applicant agreed to revise the initial response to include a discussion of the LTOP initiation. In its revised response, dated February 25, 2011, the applicant stated that the initiation/operation of the LTOP system is dependent upon the RHR system, which is locked-out whenever the RCS water temperature is above the enable temperature but automatically activates below the enable temperature. Below the enable temperature, the CS/RHR Pump Hot Leg Isolation Valves (RHS-MOV[motor-operated valve]-001A, B, C, D and RHS-MOV-002A, B, C, D) automatically open, placing the RHR and LTOP systems in service. The isolation valves' position indication in the MCR provides the operator with positive indication of the operational status of the RHR and LTOP systems. The RHR and LTOP systems cannot be manually activated by the operator. The staff finds the response acceptable; thus, **RAI 103-1448**, **Question 05.02.02-05**, **and OI 05.02.02-01**, are resolved and closed.

Furthermore, in DCD Section 5.2.2.1.2, the applicant states that the design satisfies the BTP 5-2 conditions that (1) the overpressure protection system does not depend on the availability of offsite power to perform its function and (2), that, if pressure relief is from a low-pressure system not normally connected to the primary system, the design ensures that the interlocks that would isolate the low-pressure system from the primary coolant system do not defeat the overpressure protection function. However, from the information presented, the staff was unable to confirm that these BTP 5-2 provisions were satisfied. Therefore, in **RAI 103-1448, Question 05.02.02-6**, the staff requested the applicant to provide references or documents with a detailed system description and piping and instrumentation diagrams (P&IDs) to support its position.

In its December 25, 2008, response to **RAI 103-1448, Question 05.02.02-6**, the applicant explained that the LTOP system does not require offsite power to perform its function since the primary component, the pump suction relief valve, is a spring-loaded, passive device that depends on mechanical forces and not electrical power to cause actuation. Thus, offsite power does not affect the operation of the relief valve. In respect to the interlocks that control the motor-operated isolation valves between the primary system and the relief valves, the interlocks are set to open whenever the reactor coolant pressure is below the RHR operating pressure and remain locked open during LTOP conditions. In addition, these interlocks are periodically tested to confirm proper operation in accordance with surveillance requirement (SR) 3.4.12.7 of the Technical Specifications.

Therefore, the staff concluded that the applicant's response was adequate and **RAI 103-1448, Question 05.02.02-6**, is resolved and closed.

In DCD Section 5.2.2.2.2.1, the applicant discusses low-temperature transient evaluation during water-solid conditions. While at relatively low temperatures, potential overpressurization transients to the RCS can result from mass or heat input with both types resulting in more rapid pressure changes when the RCS is in a water-solid condition. The CS/RHR pump suction relief valves provide low-temperature overpressure protection for the reactor coolant system where the valve is sized to prevent overpressure during the following credible events with a water-solid pressurizer [i.e., full of liquid water with no steam bubble]: (1) Inadvertent safety injection; (2)

Charging/letdown flow mismatch; (3) Inadvertent actuation of the pressurizer heaters; (4) Loss of residual heat removal cooling; and (5) Inadvertent start of one reactor coolant pump. The applicant states that the most limiting mass input transient is an inadvertent safety injection, and the most limiting heat input transient is an inadvertent reactor coolant pump startup in a loop where the steam generator temperature is 50°F higher than the other temperatures in the loop. The range of RCS temperatures is 70 °F to 280 °F. And the anticipated mass and heat input transients are evaluated to demonstrate conformance with ASME III, Appendix G. To complete the review of this section, in **RAI 103-1448**, **Question 05.02.02-07**, the staff requested applicant to provide the references that support compliance with ASME III, Appendix G. Also, the analysis is based on a range of RCS temperatures from 70 °F to 280 °F; whereas the LTOP operation starts below approximately 350 ^{0}F. The staff asked the applicant to include an explanation of why the analysis was not based on a range of RCS temperatures from 70 °F to 350 ^{0}F.

In its December 25, 2008, response to **RAI 103-1448**, **Question 05.02.02-07**, the applicant confirmed that the temperature values of 280 ^{o}F and 330 ^{o}F provided in Subsection 5.2.2.2.2.1 was an editorial error. The staff confirmed the temperature values were revised in Revision 2 of the US-APWR DCD, Subsection 5.2.2.2.2.1, with a range of 70 ^{o}F to 350 ^{o}F on the reactor coolant side and 120 ^{o}F to 400 ^{o}F on the SG side. Therefore, **RAI 103-1448**, **Question 05.02.02-07**, is resolved and closed.

The applicant states in DCD Subsection 5.2.2.8 that each pressurizer safety valve discharge line includes a temperature indication and alarm in the main control room of steam discharge due to either leakage or actual valve operation. In addition, each pressurizer safety valve and CS/RHR suction relief valve is equipped with open and closed indications in the main control room in compliance with recommendations of TMI action plan item II.D.3 in10 CFR 50.34(f)(2)(xi). The staff concurs that the conditions of TMI action plan item II.D.3 in10 CFR 50.34(f)(2)(xi), which requires direct indication of relief and safety valve position (open or closed) is provided in the control room, are satisfied.

In DCD Section 5.2.2.9, “System Reliability,” the design of the pressurizer safety relief valves (SRV) is described as pop-type valves with backpressure compensation features, that are spring-loaded and self-actuated by fluid pressure, when the set-point pressure is exceeded. Therefore, the applicant states that full credit is taken for the pressurizer SRVs. The staff concurs that full credit is allowed because the pressurizer SRVs satisfy ASME Code Article NB-7511.1, which requires that valves open automatically by direct action of the fluid pressure as a result of forces acting against a spring.

The applicant discusses administrative controls during LTOP plant operation in DCD Section 5.2.2.2.2.2. The staff review of this section finds the administrative controls are adequate to protect the reactor coolant pressure boundary. Therefore, the staff finds the administrative controls acceptable.

Also, in **RAI 103-1448**, **Question 05.02.02-08**, the staff requested clarification in regard to DCD Tier 1, Table 2.4.2-2, listing of valves and the designation of RCS MOVs. In its December 25, 2008, response to this RAI question, the applicant provided an acceptable explanation and the staff considers **RAI 103-1448, Question 05.02.02-08**, resolved and closed.

5.2.2.5 Combined License Information Items

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

Table 5.2.2-1
US-APWR Combined License Information Items

Item No.	Description	Section
5.2(10)	Safety and relief valve information; the COL applicant addresses the actual throat area of the pressurizer safety valves and the CS/RHR pump suction relief valves.	5.2.2.4

COL information items not identified in Table 1.8-2 of the DCD: None.

5.2.2.6 Conclusions

Based on the staff's review and evaluation of the material provided in DCD Tier 2 Section 5.2.2 and the staff's approval and closure of the open item, the staff concludes that the US-APWR overpressure protection design meets the requirements of GDC 1, GDC 15, GDC 30, GDC 31, 10 CFR 50.34(f)(2)(x) and 10 CFR 50.34(f)(2)(xi), 10 CFR 52.47(a)(8), 10 CFR 50.55a, 10 CFR 52.79(a)(17), and the acceptance criteria in BTP 5-2.

5.2.3 Reactor Coolant Pressure Boundary Materials

5.2.3.1 Introduction

This section describes material specifications issues common to the RCPB components. Material specification discussions for the RV are provided in the SER for DCD Tier 2, Section 5.3. RCPB materials are described as being fabricated in accordance with the requirements of GDC-1, GDC-30 of 10 CFR Part 50, Appendix A, and 10 CFR 50.55a. Meeting these regulatory requirements encompasses meeting the provisions for Class 1 components in ASME Code, Section III (Ref. 5.2-4, Ref. 5.2-3) and conforming to the applicable ASME Code, Section II material specifications (Ref. 5.2-21).

5.2.3.2 Summary of Application

DCD Tier 1: The Tier 1 information associated with this section is found in DCD Tier 1, Section 2.4.2.

DCD Tier 2: In DCD Tier 2, Section 5.2.3, the applicant provides a subject area description summarized here in part as follows:

> DCD Tier 2, Section 5.2.3, "Reactor Coolant Pressure Boundary Materials," describes the material specifications issues common to the RCPB components. The specifications include the grade or type and final metallurgical conditions, where

applicable, for the ferritic steels, austenitic steels, and nickel-base-alloys used. Table 5.2.3-1 of the DCD Tier 2 lists the typical material specifications used for the US-APWR RCPB components. The materials used in the RCPB satisfy the applicable material requirements of ASME Code, Section III.

All ferritic low-alloy and carbon steel surfaces that may come into contact with the reactor coolant and are used in RCPB applications are covered with stainless steel or nickel-chromium-iron cladding for corrosion resistance.

Austenitic stainless steel base materials for RCPB applications are solution-heat-treated to avoid sensitization and stress corrosion cracking (SCC). Nickel-chromium-iron [Alloy 690] materials for RCPB applications are thermally treated to enhance their resistance to primary water stress corrosion cracking (PWSCC).

Reactor coolant system (RCS) water chemistry is specified to minimize corrosion and is shown in DCD Tier 2, Table 5.2.3-2.

In order to avoid cold cracking and embrittlement of the welded materials, all welding is conducted using procedures in compliance with the rules of ASME Code, Sections III and IX. Welding materials used for the fabrication and installation welds cannot contain less than 5 percent delta ferrite, which reduces the susceptibility of stainless steel welds to hot cracking as called for by ASME Code, Section III.

ITAAC: The ITAAC associated with DCD Tier 2, Section 5.2.3 are discussed in DCD Tier 2, Sections 14.3.4.3 and 14.3.4.4 and delineated in DCD Tier 1, Section 2.4. ITAAC related to the review of DCD Tier 2, Section 5.2.3 for materials, fabrication, inspection and testing of reactor coolant pressure piping and components are located in DCD Tier 1, Table 2.4.2-5.

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

5.2.3.3 Regulatory Basis

The relevant requirements of the Commission regulations for this area of review, and the associated acceptance criteria, are given in Section 5.2.3 of NUREG-0800 and are summarized below. Review interfaces with other SRP sections can be found in Section 5.2.3 of NUREG-0800.

1. GDC 1 and 30, found in Appendix A to 10 CFR Part 50, as they relate to quality standards for design, fabrication, erection and testing.

2. GDC 4, as it relates to the compatibility of components with environmental conditions.

3. GDC 14 and 31, as they relate to minimizing the probability of rapidly propagating fracture and gross rupture of the RCPB.

4. Appendix B to 10 CFR Part 50, Criterion XIII, as it relates to onsite material cleaning control.

5. Appendix G to 10 CFR Part 50, as it relates to materials testing and acceptance criteria for fracture toughness of the RCPB.

6. 10 CFR 50.55a, as it relates to quality standards applicable to the RCPB.

Acceptance criteria adequate to meet the above requirements include:

1. ASME Boiler and Pressure Vessel Code, Section II, "Materials," Parts A, B and C; Section III, "Rules for Construction of Nuclear Facility Components," and Section IX, "Welding and Brazing Qualifications."

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

3. RG 1.34, "Control of Electroslag Weld Properties."

4. RG 1.36, "Nonmetallic Thermal Insulation for Austenitic Stainless Steel."

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

6. RG 1.43, "Control of Stainless Steel Weld Cladding of Low-Alloy Steel Components."

7. RG 1.44, "Control of the Use of Sensitized Stainless Steel."

8. RG 1.50, "Control of Preheat Temperature for Welding of Low-Alloy Steel."

9. RG 1.71, "Welder Qualification for Areas of Limited Accessibility."

10. RG 1.84, "Design, Fabrication, and Materials Code Case Acceptability, ASME Section III."

11. NUREG-0313, Revision 2, "Technical Report on Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping."

12. NUREG/CR-4513 (ANL-90/42), "Estimation of Fracture Toughness of Cast Stainless Steels During Thermal Aging in LWR Systems"

13. American Society for Testing Materials (ASTM), A-262-1970, "Detecting Susceptibility to Intergranular Attack in Stainless Steels."

14. ASTM, A-708-1974, "Detection of Susceptibility to Intergranular Corrosion in Severely Sensitized Austenitic Stainless Steel."

15. NUREG-1823, "U.S. Plant Experience with Alloy 600 Cracking and Boric Acid Corrosion of Light-Water Reactor Pressure Vessel Materials."

5.2.3.4 Technical Evaluation

5.2.3.4.1 Materials Specifications

The specifications for pressure-retaining ferritic materials, nonferrous metals, and austenitic stainless steels, including weld materials, that are used for each component in the RCPB must meet GDC 1, “Quality Standards and Records”; GDC 30, “Quality of Reactor Coolant Pressure Boundary”; and 10 CFR 50.55a, “Codes and Standards”; as they relate to quality standards for design, fabrication, erection, and testing. These requirements are met by complying with the appropriate provisions of the ASME Code, Section III and by specifying the Code Cases that are included in RG 1.84.

The staff reviewed DCD Section 5.2.3.1, “Materials Specifications,” and Table 5.2.3-1 “Materials Specifications for RCPB Components,” to determine the suitability of the RCPB materials and conformance with ASME Code, Section III; GDC 1, GDC 30 and 10CFR 50.55a. ASME Code Section III specifies the use of materials listed in ASME Code Section II that may be used in accordance with ASME Code, Section III.

DCD Table 5.2.1-1 identifies the 2001 Edition through 2003 Addenda of ASME Code, Sections II and III as being applicable to RCPB components. The staff verified that all materials specifications and grades listed in Table 5.2.3-1 meet ASME Code Section III requirements or are listed in RG 1.84, except as noted below.

For the use of ERNiCrFe-7 (UNS N06054) weld filler material, the applicant will employ non-nuclear ASME Code Case 2142-2. Code Case 2142-2 is not listed in RG 1.84. The aforementioned weld filler material, identified as ERNiCrFe-7A (UNS 06054) in subsequent editions of ASME Code (2004 and beyond) has been approved for use by the NRC staff for repair and replacement activities at several operating plants. In addition, the use of this filler material was approved for use in the AP600 design, as described in NUREG-1512 Final Safety Evaluation Report Related to Certification of the AP600 Standard Design, Section 5.2.1.2. The staff, therefore, finds the applicant’s use of ERNiCrFe-7 (UNS N06054), as described in Code Case 2142-2, acceptable.

The applicant specified the use of “G” classification weld filler materials in Table 5.2.3-1. G classification weld filler materials contain chemical compositions specified by the buyer in accordance with ASME Code, Section II, Part C. In order for the staff to determine that these filler materials are appropriate for their intended use in the US-APWR design, the staff requested, in **RAI 224-2217, Question 05.02.03-04(b)**, that the applicant provide chemical compositions for “G” classification weld filler materials listed in Table 5.2.3-1. In addition, the staff requested that the applicant address the weld filler material classifications listed in Table 5.2.3-1 that are not in accordance with ASME Code. The applicant responded in a letter dated October, 4, 2010, and provided its proprietary weld filler metal chemistry requirements for all “G” classification weld filler materials. The staff reviewed the information provided by the applicant and finds it acceptable because the specified weld filler material chemistries are compatible with the materials being welded and meet all ASME Code requirements. The applicant’s response also included proposed revisions to weld filler materials classifications. The staff reviewed the proposed revisions to weld filler material classifications in Table 5.2.3-1, and finds them acceptable because the revised classifications meet ASME Code requirements. However, the applicant’s proposed revision to Table 5.2.3-1 does not include supplementary chemistry requirements for weld filler materials used in the reactor pressure vessel (RPV) core beltline region as identified in DCD Table 5.3-1. Therefore, in **RAI 644-5077, Question 05.02.03-26**, the staff requested that the applicant modify Table 5.2.3-1 to include supplementary weld filler material chemistry requirements that are consistent with those found in Table 5.3-1 for weld filler materials used in the RPV core beltline region. The applicant responded in a letter dated February 22, 2011, and provided a proposed revision to Table 5.2.3-1 that includes supplementary chemistry requirements for weld filler materials used in the RPV core beltline region as identified in DCD Table 5.3-1**.** The staff finds this acceptable because supplementary core beltline weld filler metal chemistry requirements listed in Table 5.2.3-1 are consistent with those listed in Table 5.3-1. The staff will verify that the appropriate modifications are made in Table 5.2.3-1 in the next revision to the DCD**. RAI 224-2217, Question 05.02.03-4(b)** is resolved and closed and will be tracked as **Confirmatory Item 05.02.03-26.**

With the exception of the above confirmatory item, the staff finds that all materials specifications listed in DCD Section 5.2.3.1 and Table 5.2.3-1 meet ASME Code Section III materials requirements and are, therefore, acceptable.

5.2.3.4.2 Compatibility of Materials with Reactor Coolant

As described in SRP Section 5.2.3, the staff has reviewed the compatibility of RCPB materials with coolant. This review includes an evaluation of the composition and preparation of materials, and of the concentrations of various solutes in the coolant. The latter is also described in SRP Section 9.3.4, which is referenced in Section 5.2.3. The Electric Power Research Institute (EPRI) Guidelines represent the principal authority on coolant chemistry, and are cited as such in SRP Section 9.3.4, although not mentioned specifically in GDC 4, GDC 14, or RG 1.44. Although the staff does not formally review or issue a safety evaluation of the various EPRI water chemistry guidelines (including the *PWR Primary Water Chemistry Guidelines*), the guidelines are recognized as representing industry best practices in water chemistry control. Extensive experience in operating reactors has demonstrated that following the EPRI guidelines minimizes the occurrence of corrosion related failures. Further, the EPRI guidelines are periodically revised to reflect evolving knowledge with respect to best practices in chemistry control. Therefore, the staff accepts the use of the EPRI *PWR Primary Water Chemistry Guidelines* as a basis for a recommended primary water chemistry program for a standard reactor design.

Composition and preparation of materials, and coolant chemistry are discussed in more detail in the two sub-sections that follow.

5.2.3.4.3 Material Composition and Preparation.

The applicant states that all materials contacting the coolant are either constructed of, or clad with, stainless steel or nickel-chromium-iron alloy. Clad surfaces of low-alloy or carbon steel include the pressure vessel, pressurizer, and nozzles of various components. Material specifications require heat treatment of austenitic stainless steel and Ni-Cr-Fe alloy materials to maximize resistance to SCC and intergranular corrosion, consistent with the requirements of RG 1.44.

5.2.3.4.4 Reactor Coolant Chemistry

It is essential that coolant purity be maintained carefully, since many dissolved species can enhance material corrosion or otherwise damage internal components. In **RAI 224-2067, Question 05.02.03-01**, the staff asked the applicant to clarify how the standard and limiting values for RCS chemical parameters in the US-APWR are consistent with EPRI primary water chemistry guidelines. In its October 2, 2009, response, the applicant committed to revise the DCD to be compatible with the latest industry best practices, as represented by the most recent edition of the EPRI Guidelines. In addition, it committed to include a COL information item to ensure the COL applicant references the latest edition of the EPRI *PWR Primary Water Chemistry Guidelines.* The staff confirmed that in Revision 2 of the DCD, the applicant added a reference to the most recent EPRI guidelines in DCD Section 5.2.3.2.1 and created a new COL information item for the COL to specify the applicable edition of the EPRI PWR Primary Water Chemistry Guidelines (COL 5.2(12)). Since the applicant has included a commitment in the DCD ensuring the US-APWR primary water chemistry will conform to the latest EPRI guidelines and thus will follow industry best practices, the staff finds the response to **RAI 224-2067, Question 05.02.03-01**, acceptable. Accordingly, **RAI 224-2067, Question 05.02.03-01**, is resolved and closed.

DCD Table 5.2.3-2 states that the standard value for pH at 25 °C is between 4.2 and 10.5. However, DCD Section 9.3.4.2.3.2 says pH will be maintained in the range 7.2-7.4, which seemed inconsistent with Table 5.2.3-2. In **RAI 350-2675, Question 05.02.03-17**, the staff requested the applicant to clarify whether the pH limits given in Table 5.2.3-2 are intended to cover all operating modes and to explain when it would be necessary to allow a pH as high as 10.5. In its response to **RAI 350-2675, Question 05.02.03-17**, dated June 18, 2009, the applicant indicated that Table 5.2.3-2 is meant to be taken at 25 °C, and includes all phases of the operating cycle, whereas Section 9.3.4.2.3.2 refers to normal operation at high temperature (285-300 °C). In the former case, the low pH of 4.2 would occur during refueling operations, when highly borated water would enter the RCS. A pH as high as 10.0-10.5 might be achieved during hot functional testing (HFT), which requires lithium hydroxide (LiOH) addition to pure water. The applicant plans to use a LiOH concentration of 0.5 parts per million (ppm), which it claims will bring pH to "about 10." The plot supplied to describe HFT indicates a pH somewhat below 10, which is consistent with the staff's independent calculation of pH=9.3. In any event, this explanation of the pH range in Table 5.2.3-2 is reasonable.

Additionally, in its response to **RAI 350-2675, Question 05.02.03-17**, the applicant indicated that it intends to follow a modified pH control approach as described by the EPRI Guidelines and provided a proposed revision to DCD Fig. 5.2.3-1 to provide greater illustration of pH variations. A reference to Figure 5.2.3-1, which shows the modified pH regime with clarifications in response to **RAI 350-2675, Question 05.2.03-17**, was added to Section 9.3.4.2.3.2, in Revision 2 of the DCD. The EPRI Guidelines mention that pH at nominal operating temperature (usually around 285 °C) should be above 6.9 and not exceed 7.4, with a target of operating in the range 7.1-7.3. In its response to **RAI 350-2675, Question 05.02.03-17**, the applicant provided additional explanation of the pH range that it intends to follow, and maintained that it is consistent with EPRI Guidelines. The applicant also noted that a change in operating temperature from 285 °C to 300 °C produces a rise of 0.1 pH unit (An independent calculation by the staff indicated a change of 0.17 pH unit.). By either calculation, a small temperature change could easily move the pH out of its standard operating range. However, the applicant's plan to stay above pH=7.1 gives sufficient margin to ensure pH remains above the critical minimum of pH=6.9. As discussed below, exceeding the upper limit does not threaten RCPB integrity.

The applicant's response to **RAI 350-2675, Question 05.02.03-17**, suggests the possibility that reactor coolant pH could reach 7.4 (at 285 °C) or 7.5 (at 300 °C) during operation. A pH of 7.4 is the upper limit mentioned in the EPRI Guidelines, and a pH of 7.5 actually exceeds this limit. However, the EPRI Guidelines do not mandate this as an absolute upper limit, and even suggest that operation at this elevated pH may be beneficial in reducing stress corrosion cracking. The principal disadvantage of such operation is economic—it costs more because it requires more LiOH. The EPRI Guidelines also note that long-term effects of operation at these higher pH values are not well known, and that careful monitoring should be undertaken. Such monitoring is already part of the US-APWR chemistry program. Thus, exceeding the recommended upper pH does not exceed any prescribed limits in the EPRI Guidelines, nor does it compromise the integrity of the RCPB. The staff finds the applicant's response to **RAI 350-2675, Question 05.02.03-17**, acceptable because the applicant provided clarifying information on the control of RCS pH, sufficient to allow the staff to conclude the pH will be adequately controlled to support RCS materials integrity. Accordingly, **RAI 350-2675, Question 05.02.03-17**, is resolved and closed.

5.2.3.4.5 Control Parameters

The EPRI Guidelines specify certain inventories as control parameters, which require strict adherence to limits in order to achieve material protection. In addition, they mention three Action Levels, which denote progressive severity if violated. Action Level 1 calls for correction within 1 week, Action Level 2 calls for reduction in power and Action Level 3 calls for shutdown. Control parameters consist of the concentrations of chlorine (Cl), fluorine (F), sulfate (SO_4), lithium (Li), hydrogen (H_2), and oxygen (O_2). For each of these except Li and SO_4, DCD Table 5.2.3-2 gives limiting values equal to, or more stringent than, the Action Level 2 values from the EPRI Guidelines. (For Cl and F, these values are also mentioned in RG 1.44 as strict limits.) For Li concentration, no exact limits are provided in the EPRI Guidelines, as this component is determined by the pH control. For SO_4, the applicant has stated in its October 2, 2009, response to **RAI 224-2067 Question 05.02.03-03**, that limiting values exactly corresponding to EPRI Action Level 2 will be added to the DCD Table 5.2.3-2. The staff finds the response to RAI 224-2067 Question No. 05.03.02-3 acceptable because the applicant committed to modify the DCD to include sulfate limits consistent with the EPRI guidelines. Subsequently, the staff confirmed the sulfate limits were added to Table 5.2.3-2 in Revision 2 of the DCD. The EPRI Guidelines only state Action Level 1 limits for H_2 and O_2, and Table 5.2.3-2 standard values are consistent with, or more stringent than, these limits.

The EPRI Guidelines stipulate sampling three times/week for all but sulfate (once/week) and dissolved O (as stipulated in plant Technical Specifications). However, they note that sampling frequencies may vary and that they will be determined in the plant Technical Specifications.

The staff concludes that the applicant has provided appropriate limits for the RCS chemistry control parameters since the limits are the same as, or more stringent than, the limits recommended by the EPRI Guidelines.

5.2.3.4.6 Diagnostic Parameters

EPRI Guidelines specify certain parameters as "diagnostic," which do not have mandated limits, but which should nevertheless be monitored. These are listed as conductivity, pH, B, Si, Zn, and suspended solids. A brief discussion of each follows:

1. The use of Zn seems to help prevent crack propagation, although no limits have been established. The value in the DCD is within the range of amounts that have been used successfully at other plants.

2. EPRI Guidelines mention that no deposits have been observed if Si is below 1 ppm; they suggest a plant specific target of 3 ppm (p. 3-16). The DCD recommended value is 1 ppm, which is consistent with the EPRI Guidelines. It should be observed that this value is not mandated, but prudent.

3. For suspended solids, EPRI Guidelines state that normal operational values are typically < 10 parts per billion (ppb), but recognize that this value varies widely and cannot be mandated. The DCD specifies a standard value of 350 ppb, which seems very generous. The problem is complicated by the difficulty of obtaining truly accurate samples (see Ref. 1, Appendix F). However, in Section 4.2.3 of the EPRI PWR Primary Water Chemistry Guidelines, suspended solids are classified as a parameter having negligible effect on reactor coolant pressure boundary or fuel cladding integrity. Further, Table 3.8 of the EPRI PWR Primary Water Chemistry Guidelines recommends suspended solids be less than 350 ppb prior to reactor criticality. Based on the above, the staff finds the proposed limit acceptable.

4. Conductivity is used as an auxiliary measurement to assess general ionic activity. No limits are mandated, as this information must be interpreted locally.

Thus, the staff finds the DCD limits for the diagnostic parameters acceptable because they are generally consistent with EPRI Guidelines.

5.2.3.4.7 Zinc Addition

DCD Section 5.2.3.2.1 states that a soluble zinc (Zn) compound depleted of Zn-64 may be added to the reactor coolant as a means to reduce radiation fields within the primary system. DCD Section 5.2.3.2.1 further states that when used, the target system zinc concentration is normally maintained to a concentration no greater than 10 ppb. The staff reviewed several reports documenting industry experience with zinc addition in PWRs, which indicate that there is no concern with crud deposition for plants with low-duty or medium-duty cores, and, in fact, zinc addition typically leads to thinner, more evenly distributed crud on fuel. However, there is currently insufficient operating experience with zinc addition in plants with high-duty cores to be able to conclude that zinc injection would not cause a problem with crud deposition in such plants. Core duty is a measure of the amount of sub-cooled nucleate boiling (SNB) occurring in the core. Plants with high-duty cores are those with high fluid temperatures and high surface heat flux at the fuel clad causing a portion of the total heat transfer to the coolant to occur by SNB. Although favorable for thermal efficiency, the combination of high temperature and SNB leads to more severe duty on the fuel, and surface boiling is known to enhance the formation of corrosion product deposits (crud) at the cladding surface. The tendency for SNB can be quantified by means of the High Duty Core Index (HDCI), calculated in accordance with Appendix F of EPRI PWR Axial Offset Anomaly Guidelines. Cores with an HDCI of ≥ 150 are considered to be high-duty plants, medium-duty plants have HDCI of 120-149, and a plant with HDCI ≤ 119 is considered a low-duty plant. Staff calculations based on thermal-hydraulic data from DCD Chapter 4 indicate the US-APWR core may be considered high-duty. There may be alternate methods to determine the amount of SNB other than the HDCI, such as detailed thermal hydraulic computer models.

Potential problems with crud deposition could include excessively thick fuel crud, or uneven crud thickness that could lead to crud-induced power shift (CIPS), also known as axial offset anomaly (AOA).

DCD Section 4.2.1.7 describes the fuel surveillance program for the US-APWR, which will specify the inspection items, inspection criteria, methodology, schedule, for a number of different aspects, including crud deposition. However, the inspection method and acceptance criteria for crud on the fuel are not described.

Therefore, in **RAI 509-4114, Question 05.02.03-18**, the staff requested the applicant to provide an evaluation of the CIPS risk, and how the zinc addition is considered

In its January 29, 2010, response to **RAI 509-4114, Question 05.02.03-18**, the applicant indicated that the US-APWR core is a marginally high-duty core based on the criteria of Appendix F of the EPRI "PWR Axial Offset Anomaly Guidelines." However, the applicant noted that the HDCI for the US-APWR core is significantly lower than the HDCI of Japanese plants that have experienced CIPS. Based on the experience in Japanese plants injecting zinc, the applicant concluded that the risk of CIPS will be lower and the crud loading will be lower than if zinc were not injected. The applicant also indicated that it expects less crud on fuel surfaces than could be expected without zinc, based on the EPRI Zinc Application Guidelines, which reported that long-term zinc injection reduces the risk of CIPS by reducing out-of-core corrosion and release rate.

The staff reviewed the industry experience with zinc addition in PWRs and agrees that the industry experience supports the conclusion that zinc addition generally reduces the risk of CIPS, even in high-duty plants.

The staff finds the applicant's evaluation of the risk of CIPS resulting from zinc addition to be acceptable because the applicant considered operating experience with plants having a higher risk of CIPS, and because the applicant has appropriately applied the EPRI guidance related to zinc addition. Therefore **RAI 509-4114, Question 05.02.03-18**, is resolved and closed.

The staff finds the proposed reactor coolant chemistry acceptable because the pH control and diagnostic parameters are generally consistent with the latest EPRI *PWR Primary Water Chemistry Guidelines*. The staff considers consistency with the EPRI Guidelines an acceptable method of ensuring GDC 14 will be met for the reactor coolant pressure boundary, since the guidelines are recognized as representing industry best practice in water chemistry control RAI509-4114 is resolved and closed.

5.2.3.4.8 Fabrication and Processing of Ferritic Materials

The staff reviewed DCD, Section 5.2.3.3, to ensure that the RCPB components satisfy applicable requirements regarding prevention of RCPB fracture, control of welding, and non-destructive examination (NDE) of tubular products.

The acceptance criteria for fracture toughness properties of ferritic materials are the requirements of 10 CFR Part 50, Appendix G, "Fracture Toughness Requirements." These criteria satisfy the requirements of GDC 14, "Reactor Coolant Pressure Boundary," and GDC 31, "Fracture Prevention of Reactor Coolant Pressure Boundary."

Appendix G to 10 CFR Part 50 requires the pressure-retaining components of the RCPB to be made of ferritic materials that meet the requirements for fracture toughness during system hydrostatic tests, and any condition of normal operation, including AOOs. These requirements are met by complying with ASME Code, Section III, Subarticle NB-2300. The fracture toughness requirements of GDC 14 and 31 are also met through compliance with the acceptance standards in Subarticle NB-2300 of ASME Code, Section III. The US-APWR design complies with these Code requirements, as stated in DCD Section 5.2.3.3, and, therefore, satisfies the requirements of 10 CFR Part 50, Appendix G, and Appendix A, GDC 14 and GDC 31.

The acceptance criteria for control of ferritic steel welding are met through compliance with the applicable provisions of the ASME Code and with the regulatory positions of RG 1.34, RG 1.43, RG 1.50, RG 1.71, and ASME Code Section III, Appendix D, "Nonmandatory Preheat Procedures."

The amount of specified preheat for the welding of ferritic steels should be in accordance with ASME Code, Section III, Appendix D, Article D-1000. Appendix D is supplemented by positions described in RG 1.50 for the control of preheat temperatures for low-alloy steels. RG 1.50 recommends, in part, that the welding procedure be qualified at the minimum preheat temperature to be specified on the welding procedure specification and that preheating be maintained until post-weld heat treatment is performed.

DCD Section 5.2.3.3.2 states that minimum preheat for carbon steel (P No.-1) and low-alloy steel (P No.-3) materials is generally kept higher than those listed in Appendix D. However, the applicant states that the minimum preheat temperature for low-alloy steels may be below that recommended in Appendix D. Taking into consideration the weldability and quality of the product, preheating above 122 °F is applied to the first and second passes of circumferential joint welding. A minimum preheating temperature of 250 °F is applied to subsequent passes in accordance with ASME Code, Section III, Appendix D. In **RAI 644-5077 Question, 05.02.03-27**, the staff requested, in part, that the applicant provide a technical basis for the use of reduced preheats, less than those recommended in ASME Code, Section III, Appendix D, Article D-1000, for the first and second passes of circumferential joint welding. The applicant responded in a letter dated October, 4, 2010, and provided proprietary information related to testing that the applicant has performed and to the applicant's fabrication experience using a minimum reduced preheat of 122 °F for the first and second passes of circumferential joint welding. The staff reviewed the information provided by the applicant and determined that the applicant's use of reduced preheat is acceptable because the proprietary technical information provided by the applicant and its fabrication experience using reduced preheat provide reasonable assurance that delayed hydrogen cracking will not occur. Further, the staff notes that all full-penetration vessel welds require radiographic examination in accordance with ASME Section III as well as ultrasonic preservice examination, which would detect delayed hydrogen cracking in the unlikely event it were to occur. The applicant's response also provided a proposed revision to DCD Section 5.2.3.3.2, which states that all welding procedures are qualified at the minimum preheat temperature. The staff finds this acceptable because qualifying welding procedures at the minimum preheat temperature to be used during fabrication is consistent with the recommendations of RG 1.50. In addition, the staff requested, as part of **RAI 644-5077, Question 05.02.03-27**, that the applicant provide a note in DCD Table 1.9-1 describing any alternatives to the recommendations listed in RG 1.50. In its response, the applicant indicated that the only exception to RG 1.50 is the use of post-weld backing and the applicant provided a proposed revision to Table 1.9-1 noting its alternative, which is acceptable. The applicant's alternative is described and evaluated below. The staff will verify that that the appropriate modifications are made in DCD Revision 3. **Confirmatory Item 05.02.03-27.**

RG 1.50 Section C, Regulatory Position 2, recommends that for low-alloy steel production welds, the preheat temperature should be maintained until post-weld heat treatment (PWHT) has been performed. DCD, Section 5.2.3.3.2 states that hydrogen is removed by either post heating [post-weld baking] at a temperature and time sufficient to preclude the effects of hydrogen-assisted cracking, or by maintaining preheat until PWHT is performed. Post-weld baking is maintained at a temperature of 450-550 °F for a period of 4 hours minimum. The staff notes that post-weld baking has been successfully used in several other applications, such as fossil fuel electric generation facilities, as well as petrochemical facilities, with materials that are much more sensitive to hydrogen cracking than those materials used within the RCPB of a nuclear power plant. Post-weld baking is an effective measure to prevent delayed hydrogen cracking in welds that do not go directly from preheat to PWHT. In addition, the staff has accepted this type of alternative for other reactor designs such as the ABWR and ESBWR. The staff, therefore, considers the applicant's alternative to RG 1.50, acceptable, given that it provides reasonable assurance that delayed hydrogen cracking will not occur in the time that a weld is completed through completion of PWHT.

ASME Code, Section III, requires adherence to the requirements of ASME Section IX for welder qualification for production welds. However, there is a need for supplementing this section of the Code for welds made in locations of restricted direct physical and visual accessibility. RG 1.71 provides the necessary supplement to ASME Code, Section IX, in this respect. RG 1.71 Regulatory Position C.(1) applies when physical conditions restrict the welder's access to a production weld to less than 30 centimeters (12 inches) in any direction from the joint and which would affect electrode manipulation, or bead progression, or require an indirect means of weld pool observation (such as a mirror). DCD Sections 5.2.3.3.2 and 5.2.3.4.4 state that RG 1.71 is applied to field welds. The applicant further stated that these additional requirements do not apply to shop fabrication where the welders' physical position relative to the welds is controlled. The staff finds this acceptable because the applicant will follow the guidance provided in RG 1.71 for welds made in locations of restricted direct physical and visual accessibility.

RG 1.43 provides guidance acceptable to the staff for the control of underclad cracking in stainless steel corrosion resistant weld overlay cladding of low-alloy steel components. The applicant has indicated, in Section 5.2.3.3.2, that the US-APWR Class 1 components, that will be clad, are made from SA-508 Grade 3 Class 1 or 2 and SA-533 type B Class 1 or 2 materials. These materials are heat treated by quenching and tempering, and fine grain size of five (5) or finer is required. These materials are resistant to underclad cracking as described in RG 1.43. The staff finds this acceptable because the applicant will follow the guidance provided in RG 1.43 by using fine grain materials that are resistant to underclad cracking.

RG 1.34 provides staff guidance on the control of electroslag weld properties. However, RG 1.34 does not apply to electroslag welding used for cladding operations. The only use of electroslag welding in the US-APWR is for cladding operations. Therefore, RG 1.34 does not apply to the US-APWR.

For NDE of ferritic steel tubular products, compliance with applicable provisions of the ASME Code meets the requirements of GDC 1, GDC 30, and 10 CFR 50.55a regarding quality standards. The applicable provisions of ASME Code, Section III, are paragraphs NB-2550 through NB-2570. The applicant states, in DCD Sections 5.2.3.3.3, that NDE performed on ferritic steel tubular products of the RCPB compliance with the requirements of ASME Code Section III, paragraphs NB-2550 through NB-2570.

5.2.3.4.9 Fabrication and Processing of Austenitic Stainless Steels

Process control techniques must be included during all stages of component manufacturing and reactor construction to meet GDC 1, GDC 4; and 10 CFR Part 50, Appendix B, Criterion XIII, "Handling, Storing, and Shipping." These requirements prevent sensitization of austenitic stainless steels and minimize the exposure of the reactor coolant system and connected systems to contaminants that could lead to SCC, and reduce the likelihood of component degradation or failure through exposure to contaminants.

The requirements of GDC 4 and 10 CFR Part 50, Appendix B, Criterion XIII, are met through compliance with the applicable provisions of the ASME Code and with the regulatory positions of RG 1.31, RG 1.34; RG 1.36, RG 1.37, RG 1.44, and RG 1.71.

For NDE of austenitic stainless steel tubular products, compliance with applicable provisions of the ASME Code meets the requirements of GDC 1, GDC 30, and 10 CFR 50.55a regarding quality standards. The applicable provisions of ASME Code, Section III, are paragraphs NB-2550 through NB-2570. The applicant states, in DCD Sections 5.2.3.4.5, that NDE performed on stainless steel tubular products of the RCPB will comply with the requirements of ASME Code Section III, paragraphs NB-2550 through NB-2570.

DCD Table 5.2.3-1 indicates that some reactor coolant piping components will be fabricated from SA-182 or SA-336 Grade F316 or F316LN forged austenitic stainless steel material. These specifications do not contain limitations on grain size. Given that grain size can affect the material properties of a component and the ability to perform ultrasonic examination, the staff requested in **RAI 289-2217, Question 05.02.03-09**, that the applicant modify DCD Section 5.2.3 to include the maximum grain size for forged stainless steel components within the entire RCPB and a basis for the grain size specified. In addition, the staff requested that the applicant provide a basis for the grain size selected based on the ability to perform ultrasonic test (UT) examinations. In its May 13, 2009, response to **RAI 289-2217, Question 05.02.03-09**, the applicant stated that specifications for limitations on grain size and their bases for forged austenitic stainless steel materials in US-APWR are required to implement ASME Section II requirements. Maximum grain size limitations are specified by the applicant to the manufacturer. The applicant further stated that grain size specifications are considered proprietary information, and its specification has proven to be sufficient for inspection purposes because forged austenitic stainless steel materials in MHI operating plants are manufactured according to the same specification, and UT inspection results for these parts have been satisfactory. The applicant provided, as proprietary information, its maximum grain size requirement. The staff finds this acceptable because the applicant will specify an appropriate grain size for austenitic stainless steel forging material which that will facilitate UT examinations. Therefore, **RAI 289-2217, Question 05.02.03-09**, is resolved and closed.

DCD Table 5.2.3-1 indicates that cast austenitic stainless steels (CASS) will be used to fabricate RCPB components and Table 4.5.2 indicates that CASS will be used for the guide funnel of the CRDM thermal sleeve. CASS components used in light-water reactors (LWRs) can be susceptible to thermal aging embrittlement. In **RAI 289-2217, Question 05.02.03-10**, the staff requested that the applicant: (1) discuss the consideration of the thermal embrittlement mechanism in the design and material selection for RCPB components and reactor vessel internal components, (2) discuss the need for inspections to detect this aging effect, and (3) verify that δ-ferrite content is calculated using Hull's equivalent factors or a method producing an equivalent level of accuracy. In addition, the staff requested that the applicant modify the DCD to address the above.

In its response to **RAI 289-2217, Question 05.02.03-10**, dated March 25, 2009, the applicant stated that it may use ASTM A-800 to calculate the ferrite content of CASS materials used in the US-APWR design. The applicant contends that the methods used to calculate ferrite content in A-800 are essentially the same as the Hull's equivalent factor method. The applicant did not specify the maximum ferrite content limit for various CASS materials. The use of ASTM A-800 is inconsistent with the staff's position that ferrite content is calculated using Hull's equivalent factors as indicated in NUREG/CR-4513, Revision 1, "Estimation of Fracture Toughness of Cast Stainless Steels During Thermal Aging in LWR Systems," issued May 1994. For ferrite content above 12 percent, ASTM A-800 may produce non-conservative ferrite levels lower than those calculated using Hull's equivalent factors. Regarding ferrite limits for CASS components. DCD Table 5.2.3-1 identifies CF3A, CF3M, CF8 and CF8M materials for use in the RCPB. Table 4.5-2 identifies the use of CF8 for the guide funnel of the CRDM thermal sleeve. To be consistent with staff guidance, CF3A and CF8 materials, with service conditions above 482 ºF, should have a ferrite content of ≤20% to be considered not susceptible to thermal aging embrittlement. Materials with higher levels of molybdenum, such as CF3M and CF8M, should have a ferrite content of ≤14% to be considered not susceptible to thermal aging embrittlement. The staff requested in **RAI 540-4176, Question 05.02.03-22**, that the applicant modify the DCD to require the use of Hull's equivalent factors to calculate the ferrite content of CASS RCPB components and CASS reactor vessel internals. In addition, the staff requested that the applicant modify the DCD to limit the ferrite content of RCPB and reactor internal CASS components as discussed above. The staff also requested in **RAI 540-4176**

Question No. 05.02.03-22, that the applicant address CASS ESF components that interface with the RCPB that have operating conditions above 482 °F. The applicant's response included a proposed revision to DCD Section 5.2.3.4, which states, "Ferrite content of the cast austenitic stainless steel will be controlled to be less than or equivalent to 14 percent for 316 materials and 20 percent for 304 materials calculating the δ -ferrite level using Hull's equivalent factor method."

In order to provide clarity regarding the applicant's requirements for CASS materials to prevent thermal aging embrittlement, the staff requested, in **RAI 644-5077, Question No. 05.02.03-28**, that the applicant modify DCD Section 5.2.3.4 to state the following:

> For cast austenitic stainless steel components used in the RCPB, RPV internals and ESF systems with service temperatures greater that 482 °F, the delta ferrite content is limited to less than or equal to 20% for low-molybdenum (0.5wt% maximum)-content, statically cast materials; less than or equal to 14 percent for high-molybdenum-(2.0-3.0 wt%)-content, statically cast materials; and less than or equal to 20 percent for high-molybdenum-content, centrifugally cast materials. Ferrite content will be calculated using Hull's equivalent factors method as described in NUREG/CR-4513 Revision 1 (May 1994).

In its February 22, 2011 response to this RAI question, the applicant provided a proposed revision to DCD Tier 2, Section 5.2.3.4 that includes the above staff-recommended language to provide clarity regarding the applicant's use of CASS materials. The staff finds this acceptable because the next revision of the DCD is to include appropriate limitations on CASS materials to mitigate potential degradation due to thermal aging embrittlement of CASS components. The staff will verify that the appropriate modifications are made in DCD Revision 3. This is **Confirmatory Item 05.02.03-28.**

RG 1.31 contains guidance pertaining to the delta ferrite content in austenitic stainless steel welds to minimize the presence of microfissures, which could have an adverse effect on the integrity of components. DCD Section 5.2.3.4.4 addresses the applicant's use of the guidance provided in RG 1.31 and states that welding materials contain a minimum of 5 percent ferrite, which is consistent with the guidance in RG 1.31. In addition, DCD Table 1.9.1-1 indicates that for the austenitic stainless steel welding in the RCPB, the applicant conforms to the guidance provided in RG 1.31, with no exceptions. Accordingly, the staff finds that the applicant's control of ferrite in welds conforms to the recommendations of RG 1.31 and is, therefore, acceptable.

The applicant's compliance with the guidelines in RG 1.34 related to electroslag welding is discussed above under the staff's evaluation of fabrication and processing of ferritic materials.

The acceptance criteria for compatibility of austenitic stainless steel with thermal insulation are based on RG 1.36, "Nonmetallic Thermal Insulation for Austenitic Stainless Steel," to satisfy GDC 14 and 31 relative to prevention of failure of the RCPB. Reactor coolant system components and piping primarily use reflective metal insulation (RMI). DCD Subsection 5.2.3.2.3 states that nonmetallic insulating materials, which may be applied to small segments of pipe, comply with RG 1.36. In addition, DCD Table

9.2.1-1 indicates that for RCPB components the US-APWR design conforms to the guidance in RG 1.36 with no exceptions.

RG 1.37 provides staff guidance for quality assurance requirements for cleaning of fluid systems and associated components of water-cooled nuclear power plants. The controls for abrasive work on austenitic stainless steels surfaces should, as a minimum, be equivalent to the controls described in ASME NQA-1-1994, which is referenced in RG 1.37. The applicant stated in DCD Section 5.2.3 that surface cleanliness achieved by its procedures satisfies the requirements of RG 1.37. The staff reviewed the applicant's description of its cleaning and contamination protection measures described in DCD Section 5.2.3.4.1, and finds them acceptable as they are consistent with the recommendations described in RG 1.37 and the controls listed in ASME NQA-1-1994.

The staff's evaluation of the applicant's use of RG 1.71 is discussed above, under fabrication and processing of ferritic materials.

DCD Table 5.2.3-1 lists standard and low-carbon grades of austenitic stainless steels for use in the RCPB. DCD Sections 5.2.3.4.1 and 5.2.3.4.2 address RG 1.44 and present the methods and controls used by the applicant to avoid sensitization and to prevent intergranular attack of austenitic stainless steels. In addition, Table 1.9.1-1 states that the US-APWR conforms to the guidance provided in RG 1.44 with no exceptions identified for the RCPB materials. The staff finds this acceptable because the applicant will follow staff guidance provided in RG 1.44 to control the use of sensitized austenitic stainless steel. However, the staff notes that stagnant or dead-end PWR coolant environments that contain oxygen can contribute to stress corrosion cracking.

In **RAI 289-2217, Question 05.02.03-12**, the staff requested that the applicant discuss its use of standard grades of stainless steel (non-low-carbon). In its May 13, 2009 response, the applicant stated that the basic policy [standard MHI practice] for austenitic stainless steel in the RCPB of the US-APWR is to select low-carbon grade materials permitting market conditions and availability at the time of procurement. The staff notes that the use of standard grades is acceptable per RG 1.44 as long as the reactor coolant has a controlled oxygen content of less than 0.10 ppm dissolved oxygen. However, elevated dissolved oxygen content in stagnant or dead-end primary coolant environments has contributed to stress corrosion cracking of austenitic stainless steel components in operating PWRs. The staff requested in **RAI 540-4176, Question 05.02.03-23**, that the applicant verify, and state in the DCD, that a stagnant or dead-end reactor coolant environment will not exist in any portion of the RCBP that could result in elevated dissolved oxygen above 0.1 ppm at all operating temperatures above 200ºF during normal operation. The applicant responded in a letter dated March 1, 2010, and stated that it had submitted a revised response to **RAI 289-2217, Question 05.02.03-12**, which addresses **RAI 540-4176, Question 05.02.03-23**. The applicant's revised response to **RAI 289-2217, Question 05.02.03-12**, dated March 1, 2010, states that the maximum carbon content of stainless steel used in the RCPB is 0.05 percent (heat analysis) and 0.06 percent (product analysis). The applicant also stated that if, during the detailed design process, significant stagnant areas are generated where the dissolved oxygen is evaluated to be elevated to over 0.10 ppm, the applicant will evaluate the possibility of stress corrosion cracking. The staff expects that any RCPB piping or components that may be subjected to a dissolved oxygen content environment over 0.10 ppm will be fabricated from low-carbon stainless steel material (≤ 0.03%

Carbon). Therefore, the staff requested, in **RAI 644-5077, Question 05.02.03-29**, that the applicant modify DCD Section 5.2.3 in order to provide clarity, to state the following:

> During the detailed design of RCPB piping and components, MHI will determine if there are local areas where flow stagnation may be present resulting in dissolved oxygen content greater than 0.10 ppm. For piping and components where the above condition exists, stainless steel with a carbon content less than or equal to 0.03 percent will be used.

The applicant's response to **RAI 644-5077, Question 05.02.03-29**, dated February 22, 2011, provided a proposed revision to CD Tier 2, Section 5.2.3.4.1, which included the staff's recommended modification to the language in Section 5.2.3.4.1. The staff finds this acceptable because the applicant will ensure that portions of the RCPB that may have stagnant flow conditions will use low carbon stainless steel to reduce the potential for stress corrosion cracking of RCPB components. The staff will verify that the appropriate modifications are made in DCD Revision 3. Accordingly, **RAI 289-2217, Question 05.02.03-12,** and **RAI 540-4176, Question 05.02.03-23,** are closed and **RAI 644-5077, Question 05.02.03-29**, will be tracked as **Confirmatory Item 05.02.03-29**

5.2.3.4.10 Dissimilar Metal Welds

In **RAI 289-2217, Question 05.02.03-14,** and **RAI 540-4176, Question 05.02.03-24**, the staff requested that the applicant describe fabrication process controls employed to limit the effects of cold work and residual stress, caused by grinding/repair or other fabrication processes on surfaces that come into contact with RCS fluids. In the applicant's responses to the above referenced RAIs, dated May 13, 2009 and June 4, 2010, respectively, the applicant stated that dissimilar-metal welds (DMWs) use Alloy 690 filler materials [Alloy 52] to avoid stress corrosion cracking. In addition, the applicant stated that components fabricated from stainless steel and nickel-based alloys are designed so that residual stress and cold-work effects are minimized as much as possible. The staff expects that fabrication process controls will be employed on the surface of dissimilar metal welds that are in contact with reactor coolant to limit the effects of cold work caused by grinding. In addition, the staff expects that special fabrication controls will be utilized during the repair of dissimilar metal welds to limit residual stress. Therefore, the staff requested in **RAI 644-5077, Question 05.02.03-30**, that the applicant discuss its process controls for the surface of all DMWs in contact with reactor coolant (i.e. safe end welds and CRDM nozzle to RPV head welds). In addition, the staff requested that the applicant discuss process controls employed during the repair of DMWs to limit residual stress, thus making the repaired welds less susceptible to stress corrosion cracking.

The applicant's response to **RAI 644-5077, Question 05.02.03-30**, dated February 22, 2011, provided, as proprietary information, MHI's fabrication processes to limit cold work and residual stress. The staff reviewed the applicant's fabrication controls and finds them acceptable because the applicant will take appropriate steps, such as controls on grinding and on the surface conditioning of ground surfaces in contact with reactor coolant. The process control procedures require the applicant to apply these controls to austenitic stainless steel and nickel-based alloy components and welds. **RAI 289-2217, Question 05.02.03-14, RAI 540-4176, Question 05.02.03-24,** and **RAI 644-5077 Question No. 05.02.03-30** are therefore resolved and closed.

In **RAI 289-2217, Questions 05.02.03-15** and **-16**, the staff requested, in part, that the applicant discuss welding process controls employed to reduce weld metal dilution to retain the maximum percentage of chromium possible in order to decrease the susceptibility of components to stress corrosion cracking. In its May 13, 2009, response, the applicant stated that heat input is controlled to prevent hot cracking and minimize weld metal dilution. The staff finds this acceptable with regard to heat input; but the applicant did not discuss how it verifies that its welding procedures and process controls will result in dissimilar metal welds with an acceptable level of chromium to resist stress corrosion cracking. Therefore, in **RAI 644-5077, Question No. 05.02.03-31**, the staff requested that the applicant discuss testing that has been or will be performed to verify the minimum chromium content in dissimilar metal welds and that the minimum chromium content is sufficient to resist stress corrosion cracking.

In its initial response to **RAI 644-5077, Question 05.02.03-31**, dated November 8, 2010, and a revised response, dated February 22, 2011, the applicant provided proprietary electron probe microanalysis (EPMA) results from a steam generator safe-end weld mockup, which shows the chromium content of a typical US-APWR full-penetration DMW that utilizes buttering in the weld joint design. The applicant's EPMA results show a minimal loss of chromium at the buttering/low-alloy steel interface due to dilution of the low-alloy steel material into the buttering layer. In addition, the staff notes that at the point where reactor coolant comes into contact with the low-alloy-steel side of the weld joint, the Alloy 52 weld metal is diluted with the stainless steel cladding and not the low-alloy steel. Thus the loss of chromium, due to dilution, will be even less than that shown in the applicant's EPMA results for the Alloy 52/low-alloy steel interface. The staff finds this acceptable because the applicant has shown that its welding process controls appropriately limit weld metal dilution thus reducing DMWs' susceptibility to stress corrosion cracking. The applicant also stated that the weld configuration and welding process used on the mockup that was tested are consistent with the weld configuration and welding process used on safe-end DMW's in the US-APWR design. The staff finds this acceptable because the fabrication of the mock-up tested is consistent with actual fabrication techniques that the process controls prescribe. Therefore **RAI 289-2217, Questions 05.02.03-15** and **-16,** and **RAI 540-4176, Question 05.02.03-31**, are resolved and closed.

5.2.3.4.11 ITAAC

The staff reviewed the Tier 1 ITAAC associated with Tier 2 Section 5.2.3, "Reactor Coolant Pressure Boundary Materials," listed in DCD Tier 1, Table 2.4.2-5, "Reactor Coolant System Inspections, Tests, Analyses, and Acceptance Criteria." The staff verified the applicable ITAAC appropriately address the verification of compliance with applicable ASME Code, Section III, requirements related to materials, fabrication, welding and non-destructive examination of piping and components, thus meeting the requirements of 10 CFR52.47(b)(1). Further, staff evaluation of the applicants compliance with 10 CFR52(b)(1), as it relates to the RCPB, is located in Section 14.3. of this FSER.

5.2.3.5 Combined License Information Items

The following is a list of COL information item numbers and descriptions from Table 1.8-2 of the DCD related to the review of DCD Section 5.2.3.

Table 5.2.3-1
US-APWR Combined License Information Items

Item	Description	Section
5.2(12)	The COL applicant should specify the applicable version of the EPRI "Primary Water Chemistry Guideline" that will be implemented.	5.2.3.2.1

The staff finds the above listing to be complete. Also, the list adequately describes actions necessary for the COL applicant or holder. No additional COL information items need to be included in DCD Tier 2, Table 1.8-2 for reactor coolant pressure boundary materials because the proposed ITAAC and initial plant test program assure that the RCPB will be constructed in accordance with the certified design.

5.2.3.6 Conclusions

Except for the confirmatory items discussed above, based on the information provided by the applicant and for the reasons set forth above, the staff concludes that the RCPB materials are acceptable and meet the relevant requirements of GDC 1, GDC 4, GDC 14, GDC 30, and GDC 31; Appendices B and G to 10 CFR Part 50, and 10 CFR 50.55a.

5.2.4 Inservice Inspection and Testing of the RCPB

5.2.4.1 Introduction

Components that are part of the RCPB must be designed to permit periodic inspection and testing of important areas and features to assess their structural and leak tight integrity. Periodic inspections of the RCPB are required so that aging effects or other incipient degradation phenomena may be identified and preventive measures promptly taken to preclude potential loss of reactor coolant or impairment of reactor core cooling. 10 CFR 50.55a, "Codes and Standards," requires that SSCs be designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with the importance of the safety function they are intended to perform. Therefore, 10 CFR 50.55a incorporates by reference Sections III and XI of the ASME Boiler and Pressure Vessel Code (B&PV), as well as the OM Code. Section XI defines, for each component Code Class, the specific inservice inspection (ISI) requirements (e.g., methodology, periodicity, acceptance criteria). ISI includes a preservice inspection (PSI) prior to initial plant startup.

5.2.4.2 Summary of Application

DCD Tier 1: The Tier 1 information associated with this section is found in Tier 1 Section 2.3, where it states that the piping and piping supports are analyzed and designed to the requirements of the ASME Code Section III, based on Code classification and ASME Service Level. Table 2.4.2-5 states that pressure boundary welds in ASME Code Section III piping, identified in Table 2.4.2-3, meet ASME Code Section III requirements. The as-built pressure boundary welds will be inspected and nondestructively examined in accordance with ASME Section III requirements.

DCD Tier 2: The applicant has provided a Tier 2 description of its ISI program for Class 1 RCPB components in Section 5.2.4, summarized here in part as follows:

> The physical arrangement of ASME Code Class 1 components is designed to allow personnel and equipment access to the extent practical to perform the required inservice examinations specified by the ASME Code Section XI and mandatory appendices. Design provisions, in accordance with Section XI, Article IWA-1500, are incorporated in the design processes for Class 1 components. Piping and pipe support locations, insulation, hangers, and stops are designed so as not to interfere with inspection equipment personnel. Where this cannot be done, the components are easily and quickly removable with minimal special handling equipment.
>
> Removable insulation and shielding are provided on those piping systems requiring volumetric and surface examination. Removable hangers are provided, as necessary and practical, to facilitate ISI examinations. Working platforms are provided in areas requiring inspection and servicing of pumps and valves. Temporary or permanent working platforms, walkways, scaffolding, and ladders are provided to facilitate access to piping and component welds. The components and welds requiring inspections allow for the application of the required ISI examination methods. Such design features include sufficient clearances for personnel and equipment, maximized examination surface distances, favorable materials, weld-joint simplicity, elimination of geometrical interferences, and proper weld surface preparation.
>
> The application addresses: 1) examination categories and methods (e.g., visual, liquid penetrant, magnetic particle, eddy current, ultrasonic, radiography), 2) inspection intervals, 3) evaluation of examination results, 4) system pressure tests, 5) code exemptions, 6) relief requests, 7) code cases, 8) preservice inspection program, and 9) combined license information.

ITAAC: The ITAAC associated with DCD Tier 2, Section 5.2.4 are given in DCD Tier 1, Section 2.4.2, Item 4.b in Table 2.4.2-5, "RCS Inspections, Tests, Analyses, and Acceptance Criteria," which indicates that inspections will be performed of the as-built piping and that a design report exists, which concludes that the piping as indicated on Table 2.4.2-3 as ASME Code Section III has been reconciled with the design documents.

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

5.2.4.3 Regulatory Basis

The relevant requirements of the Commission's regulations for this area of review, and the associated acceptance criteria, are given in Section 5.2.4 of NUREG-0800 (the SRP) and are summarized below. Review interfaces with other SRP sections can be found in Section 5.2.4 of NUREG-0800.

1. GDC 32 of Appendix A to 10 CFR Part 50, as it relates to periodic inspection and testing of the RCPB.

2. 10 CFR 50.55a, as it relates to the requirements for testing and inspecting Code Class 1 components of the RCPB as specified in Section XI of the ASME Code.

3. ASME Code Case N-729-1, as modified by 10 CFR 50.55a(g)(6)(ii)(D) for reactor vessel head inspection requirements.

Acceptance criteria adequate to meet the above requirements include:

1. RG 1.26, as it relates to the quality group classification of components.

2. RG 1.147, as it relates to ASME Section XI Code Cases acceptable for use.

3. NRC Bulletin 88-05, as it relates to the establishment of a program to detect and correct potential RCPB corrosion caused by boric acid leaks.

5.2.4.4 Technical Evaluation

This portion of the staff safety evaluation report (SER) addresses the ISI program requirements while Section 3.9 of the staff SER addresses the IST program requirements. The DCD states that this subsection describes the ISI and Inservice Testing (IST) for Quality Group A components of the US-APWR RCPB. These components are defined as ASME Code Class 1 pressure-retaining components (other than the steam generator tubes) including vessels, piping, pumps, valves, bolting, and supports within the RCPB. The US-APWR components meet the definition for Quality Group A components presented in RG 1.26, “Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants.” Subsection NB of Section III of the ASME Code presents the construction requirements for Class 1 components, and Subsection IWB of Section XI presents their preservice and inservice inspection requirements.

DCD Section 5.2.4 states that the 2001 edition with the 2003 Addenda of the ASME Code is used to determine the requirements for the initial and subsequent inspection intervals in the ISI and IST programs. The preservice and inservice inspections meet the requirements set forth in Section XI of the ASME Code as specified in 10 CFR 50.55a(g) with exceptions as permitted under 10 CFR 50.55a(g)(6)(i).

5.2.4.4.1 Arrangement and Accessibility of Systems and Components

SRP Section 5.2.4 states that the design and arrangement of system components are acceptable if adequate clearance is provided in accordance with the ASME Code, Section XI, Subarticle IWA-1500, “Accessibility.” In addition, 10 CFR 50.55a(g)(3)(i) requires Class 1 components including supports to be designed and to be provided with access to enable the performance of inservice examination of these components. In addition, the regulation requires that these components meet the preservice examination requirements set forth in the editions and addenda of Section XI for the ASME Code of record.

DCD Tier 2, Section 5.2.4.1.1, describes accessibility for inspection and states that the physical arrangement of ASME Code Class 1 components is designed to allow personnel and equipment access "to the extent practical" [sic] to perform the required inservice examinations specified by the Code and mandatory appendices. The DCD also states that space is also provided per IWA-1500(e) for necessary operations "to the extent practical" [sic] associated with repair/replacement activities. Piping arrangement allows for adequate separation of piping welds so that space is available to perform ISI. Welds in piping that passes through walls are located such that there is sufficient clearance and access into the wall penetration to perform weld examination. The DCD also states that design features include sufficient clearances for personnel and equipment, maximized examination surface distances, and favorable materials, weld joint simplicity, elimination of geometric interferences, and proper weld surface preparation.

10 CFR 50.55a(g)(3)(i) and (3)(ii) require that for a boiling- or pressurized-water-cooled nuclear power facility whose construction permit under 10 CFR Part 50, or design certification, design approval, combined license, or manufacturing license under 10 CFR Part 52 was issued on or after July 1, 1974, components (including supports) classified as Class 1, 2, and 3 must be designed and be provided with access to enable the performance of inservice examination and must meet the preservice examination requirements set forth in the editions and addenda of Section XI of the ASME Code incorporated by reference. The staff had concerns that designing the RCPB with access "to the extent practical" to perform ISI and Repair/Replacement activities is not consistent with the regulation, nor with the statement in the DCD, "…design features include sufficient clearances for personnel and equipment, maximized examination surface distances, and favorable materials, weld joint simplicity, elimination of geometric interferences, and proper weld surface preparation." Based on the above concern, in **RAI 254-2075, Question 05.02.04-01**, the staff requested the applicant to describe and provide justification for any conditions in which it believes Code ISI examinations would be impractical for reasons of design, geometry, or materials of construction**.**

In its response to **RAI 254-2075, Question 05.02.04-01**, dated October 2, 2009, the applicant stated that ASME Class 1 components are designed to provide access for the examinations required by ASME Section XI and mandatory appendices. In addition, the applicant provided more detail on accessibility in its responses to **RAI 254-2075, Questions 05.02.04-03** through **05.02.04-08**. Finally, the applicant stated that the DCD will be changed to remove the phrase "to the extent practical" from Subsection 5.2.4.1. Based on the applicant's commitment to remove the phrase "to the extent practical" from the DCD in areas involving design and accessibility, the staff finds that the applicant's proposed DCD revision would render the design as described in the DCD in compliance with 10 CFR 50.55a with regard to Class 1 component accessibility and is therefore acceptable. The staff will verify that Section 5.2.4.1 is revised accordingly in the next revision of the DCD. This is **Confirmatory Item 05.02.04-01.**

The staff notes that a significant number of dissimilar metal welds and austenitic welds in the current fleet of operating PWRs have suffered primary-water, stress-corrosion cracking (PWSCC). Specific regulations for performing ultrasonic examination of dissimilar metal and austenitic weld qualification under 10 CFR 50.55a(b)(2)(xv) and 10 CFR 50.55a(b)(2)(xvi) require that one-sided access for ultrasonic examinations be qualified. If this cannot be accomplished, supplemental nondestructive examinations on the side that has not been qualified must be performed in order to satisfy 10 CFR 50.55a(g)(3) and obtain 100 percent coverage. However, the DCD does not provide specific design considerations to address this issue. Based on the above, in **RAI 254-2075, Question 05.02.04-08**, the staff requested that the applicant provide a proposed DCD revision to address this concern in order for the staff to be able to make a reasonable assurance finding of the acceptability of the design in this regard.

In its response to **RAI 254-2075, Question 05.02.04-08**, dated October 2, 2009, the applicant stated that a sentence will be added to the last paragraph of Subsection 5.2.4.1.1 of the DCD to state that any changes to the design of US-APWR components by the COL applicant should include a discussion of the provisions for preservice accessibility to perform ISI for Class 1 components consistent with the requirements of IWA-1500 and 10 CFR 50.55a(g)(3). Since the COL applicant is going to preserve accessibility, the modifications under 10 CFR 50.55a(b)(2) are unnecessary. The staff concludes that the changes will be in compliance with the requirements of the ASME Code and 10 CFR 50.55a, and is therefore, acceptable. The staff will verify that Subsection 5.2.4.1.1 is revised accordingly in the next revision of the DCD. This is **Confirmatory Item 05.02.04-08.**

5.2.4.4.2 Examination Categories and Methods

The examination categories and methods specified in the DCD are acceptable if they meet the requirements in ASME Code, Section XI, Article IWB-2000, “Examination and Inspection.” Every area subject to examination falling within one or more of the examination categories in Article IWB-2000 must be examined, at least to the extent specified. The requirements of Article IWB-2000 also list the methods of examination for the components and parts of the pressure-retaining boundary.

The applicant’s examination techniques and procedures used for preservice inspection or ISI of the system are acceptable if they meet the following criteria:

- The methods, techniques, and procedures for visual, surface, or volumetric examination are in accordance with Article IWA-2000, “Examination and Inspection,” and Article IWB-2000, “Examination and Inspection,” of ASME Code, Section XI.

- The methods, procedures, and requirements regarding qualification of NDE personnel are in accordance with Article IWA-2300, “Qualification of Nondestructive Examination Personnel.”

- The methods, procedures, and requirements regarding qualification of personnel performing ultrasonic examination reflect the requirements provided in Appendix VII, “Qualification of Nondestructive Examination Personnel for Ultrasonic Examination,” to Division 1 of ASME Code, Section XI. In addition, the performance demonstration for ultrasonic examination systems reflects the requirements provided in Appendix VIII, “Performance Demonstration for Ultrasonic Examination Systems,” to Division 1 of ASME Code, Section XI.

DCD Section 5.2.4.1.2, "Examination Categories and Methods," discusses examination techniques, categories, and methods. The visual, surface, and volumetric examination techniques and procedures are consistent with the requirements of Subarticle IWA-2200 and Table IWB-2500-1 of ASME Code, Section XI. Examination categories and requirements are established according to Subarticle IWB-2500 and Table IWB-2500-1 of ASME Code, Section XI. The PT method or the magnetic particle method is used for surface examinations. Radiography, ultrasonic, or eddy current techniques (manual or remote) are used for volumetric examinations. DCD Section 5.2.1.1, indicates that the baseline Code used for the US-APWR DC is the 2001 edition with the 2003 Addenda of the ASME Code, Section XI. This edition and addenda of ASME Code, Section XI, requires the implementation of Appendix VII for qualification of NDE personnel for ultrasonic examination, and the implementation of Appendix VIII for performance demonstration for ultrasonic examination of reactor pressure boundary piping, RV welds, and RV head bolts. However, the DCD did not discuss qualification of NDE personnel to the acceptance criteria of the SRP. Therefore, in **RAI 254-2075, Question 05.02.04-02**, the staff requested that the applicant provide a proposed revision to the DCD to address this issue.

In its response to **RAI 254-2075, Question 05.02.04-02**, dated October 2, 2009, the applicant stated that DCD Subsection 5.2.4.1 will be revised to state that personnel performing nondestructive examinations will be qualified and certified in accordance with ASME Section XI, IWA-2300, "Qualification of Nondestructive Examination Personnel." The staff concludes that the examination methods and categories applied to Class 1 components comply with the requirements of ASME Code, Section XI and are acceptable. Furthermore, the staff finds that the proposed changes to DCD Subsection 5.2.4.1 as discussed above, regarding qualification of nondestructive examination personnel are in compliance with the SRP acceptance criteria and the ASME Code, and are therefore acceptable. The staff will verify that Subsection 5.2.4.1 is revised accordingly in the next DCD revision. This is **Confirmatory Item 05.02.04-02.**

5.2.4.4.3 Inspection Intervals

The required examinations and pressure tests must be completed during each 10-year interval of service, hereinafter designated as the inspection interval. In addition, the scheduling of the program must comply with the provisions of Article IWA-2000, "Examination and Inspection," concerning inspection intervals of the ASME Code, Section XI. DCD Tier 2, Section 5.2.4.1.3, "Inspection Intervals," discusses inspection intervals. Subarticles IWA-2400 and IWB-2400 of ASME Code, Section XI, define inspection intervals. The inspection intervals specified for US-APWR components are consistent with the definitions in Section XI of the ASME Code and, therefore, are acceptable.

5.2.4.4.4 Evaluation of Examination Results

SRP Section 5.2.4 states that standards for examination evaluation in the program for flaw evaluation are acceptable if in agreement with the requirements of ASME Code, Section XI, Article IWB-3000, "Acceptance Standards." SRP Section 5.2.4 also states that the proposed program regarding repairs of unacceptable indications or replacement of components containing unacceptable indications is acceptable if in agreement with the requirements of ASME Code, Section XI, Article IWA-4000, “Repair/Replacement Activities.” The criteria that establish the need for repair or replacement are described in ASME Code, Section XI, Article IWB-3000, “Acceptance Standards.”

DCD Tier 2, Section 5.2.4.1.4, “Evaluation of Examination Results,” discusses the evaluation of examination results. Examination results are evaluated according to ASME Code, Section XI, IWA-3000 and IWB-3000, with flaw indications being evaluated according to IWB-3400 and Table IWB-3410-1. Repair procedures, if required, are evaluated according to ASME Code, Section XI. Based on this method of evaluating examination results, and the use of the appropriate ASME Code rules for repair, the applicant’s DCD description of its prescribed practice for evaluation of examination results for US-APWR Class 1 components meets the SRP acceptance criteria and is therefore acceptable.

5.2.4.4.5 System Pressure Tests

The pressure-retaining Code Class 1 component leakage and hydrostatic pressure test program is acceptable if the program is in accordance with the requirements of Section XI, Article IWB-5000, and the technical specification requirements for operating limitations during heatup, cooldown, and system hydrostatic pressure testing. The pressure tests verify pressure boundary integrity in conjunction with ISI.

DCD Tier 2, Section 5.2.4.1.5, “System Pressure Tests,” states that Class 1 systems and components are pressure tested in accordance with IWB-5000 of the ASME code and the technical specification requirements for operating limitations during heat-up, cooldown, and system hydrostatic testing. Since the applicant’s methodology for performing pressure testing of the Class 1 boundary and components meets the acceptance criteria of the ASME Code, as stated in the SRP, the methodology for performing system pressure testing is, therefore, acceptable.

5.2.4.4.6 Code Exemptions

The SRP states that exemptions from Code examinations should be permitted if the criteria in Subarticle IWB-1220, “Components Exempt from Examination,” are met. DCD Tier 2 Section 5.2.4.1.6 states that ASME Section XI Code exemptions are permitted by Subarticle IWB-1220. The staff could not determine if the applicant was invoking any exemptions. Based on the above, the staff requested additional information.

In its October 2, 2009, response to RAI 254-2075, 5.2.4-3, the applicant stated that the DCD Subsection 5.2.4.1.6 will be changed to add a sentence that no additional exemptions to the ASME Section XI, IWB-1220 criteria are necessary based on the current design. The staff concludes that the applicant is not requesting additional exemptions to the design and its proposed changes to the DCD are in compliance with ASME Section XI. **This is Confirmatory Item 05.02.04-03**.

In addition to the above, the staff notes that the applicant listed all appropriate component exemptions with the exception of the Code exemption for buried components under IWB-1220(d). DCD Section 5.2.4 states that the “design standards include provisions for placement of Class 1 piping and components, and establishing minimum structural clearances around them, such that adequate access for inservice inspection is maintained. These provisions preclude locating welds or portions of welds such that they would otherwise be exempt from examination due to their inaccessibility because they are encased in concrete, buried underground, located inside a penetration, or encapsulated by a guard pipe." This practice is consistent with designing for accessibility to perform inservice examinations, which meets the regulations under 10 CFR 50.55a(g)(3)(i), and is more stringent than the ASME Code because the DCD eliminates an exemption listed under IWB-1220(d) pertaining to buried components. The staff's review of the exemptions listed in DCD Tier 2 are in accordance with Subarticle IWB-1220 and 10 CFR 50.55a(g)(3)(i), meeting the acceptance criteria in the SRP, and are, therefore, acceptable.

5.2.4.4.7 Relief Requests

The NRC staff reviewed whether the applicant has demonstrated that any ASME Code requirement is impractical due to design, geometry, or materials of construction. DCD, Section 5.2.4.1.7 states that the COL applicant discusses any requests for relief from ASME Code requirements that are impractical as a result of limitations of component design, geometry, or materials of construction. In such cases, specific information is provided which identifies the applicable Code requirements, justification for the relief request, and the inspection method to be used as an alternative. The staff could not determine if the US-APWR design incorporated relief requests from impractical examinations as a result of component design, geometry, or materials of construction. Based on the above, the staff requested additional information.

In its October 2, 2009, response to RAI 5.2.4-4, the applicant stated that no relief request is expected for PSI and first interval ISI examinations for US-APWR Class 1 components. The applicant stated that DCD Subsection 5.2.4.1.7 will be changed to state that as such, and that approved ASME Code Cases as listed in RG 1.147 may be used. The staff concludes that changes are in compliance with the requirements of 10 CFR 50.55a, and are, therefore, acceptable. **This is Confirmatory Item 05.02.04-4.**

5.2.4.4.8 Code Cases

The NRC staff reviewed ASME code cases for acceptability and compliance with RG 1.147. DCD Section 5.2.4.1.8 states that code cases referenced by the COL application that may have been invoked in connection with the ISI program are in compliance with RG 1.147. The staff could not determine if the DCD uses any code cases for the design. In addition, the reactor pressure vessel head inspections, at one time, were required to be performed in accordance with NRC Order EA-03-009 and First Revised Order EA-03-009 by conditional implementation of Code Case N-729-1, "Alternative Examination Requirements for PWR Reactor Vessel Upper Heads with Nozzles Having Pressure-Retaining Partial-Penetration Welds." Furthermore, the DCD states, "COL applicants that reference the US-APWR design may invoke ASME Code Case N-729-1, with conditions cited in the two orders, until subsequent Nuclear Regulatory Commission (NRC) requirements supersede the order."

The NRC requirement for inspecting the reactor vessel head was discussed in a proposed rule and a final rule amending 10 CFR 50.55a in 10 CFR 50.55a(g)(6)(ii)(D) related to reactor vessel head inspections. Based on the evolving nature of the rulemaking, the staff requested additional information.

In its October 2, 2009, response to **RAI 254-2075, Question 05.02.04-05**, the applicant stated that DCD Section 5.2.4.1.8 would be revised to include the implementation of ASME Code Case N-729-1 with the conditions specified in 10 CFR 50.55a(g)(6)(ii)(D) of the final amended rule for 10 CFR 50.55a (73 FR 52748). The proposed revision will also include the implementation of ASME Code Cases N-613-1 and N-307-3. The staff finds that the changes comply with 10 CFR 50.55a, and are therefore, acceptable. In addition, incorporation of ASME Code Cases N-613-1 and N-307-3 into the ISI program is acceptable because these two code cases were approved by the NRC staff in RG 1.47 and are acceptable for use. **This is Confirmatory Item 05.02.04-05.**

5.2.4.4.9 Augmented ISI to Protect Against Postulated Piping Failures

The SRP (NUREG-0800) states that the ISI program is reviewed to verify that the high-energy system piping between containment isolation valves receives an augmented ISI that meets four criteria. As the DCD does not address this aspect of augmented ISI; the staff could not determine the extent to which the augmented ISI program monitored components for postulated piping failures. Based on the above, in **RAI 254-2075, Question 05.02.04-06**, the staff requested the applicant to provide a description of the augmented ISI that is used for high-energy system piping between containment isolation valves. The staff also requested the applicant, alternatively, to revise the DCD to reflect that no high-energy piping (including Class 1 piping) penetrates the containment, and no augmented ISI is required to protect against postulated failures of such piping between containment isolation valves, if that is, in fact, the case. If so, the applicant was requested to discuss how postulated pipe breaks at the containment boundary are considered, including a discussion of single failure of one containment isolation valve to close.

In its response to **RAI 254-2075, Question 05.02.04-06**, dated April 17, 2009, the applicant stated that the ISI program contains information addressing areas subject to inspection, method of inspection, and extent and frequency of inspection of high-energy fluid systems described in DCD Chapter 3, Sections 3.6.1 and 3.6.2. It also stated that the design criteria, including the requirements for inspection of all piping in the pre-stressed concrete containment vessel (PCCV) penetration are addressed in Subsection 3.6.2.1.1.1 of the DCD. The staff reviewed Subsection 3.6.2.1.1.1 and found that a sufficient level of detail was present to make a reasonable assurance finding regarding the ability of the augmented ISI program to protect against postulated piping failures. Refer to the staff's FSER under Subsection 3.6.2.1.1.1. The staff considers the applicant's response acceptable since the applicant clarified its augmented ISI program. Accordingly, **RAI 254-2075, Question 05.02.04-06**, is resolved and closed.

5.2.4.4.10 Other Inspection Programs

The SRP states that for PWR plants the applicant must establish an inspection program to detect and correct potential RCPB corrosion caused by boric acid leaks as described in NRC Generic Letter 88-05. DCD Section 5.2.4 does not discuss any aspect of a boric acid leak detection program in accordance with NRC Generic Letter 88-05. Based on the above, in **RAI 254-2075, Question 05.02.04-07**, the staff requested the applicant to provide information on boric acid leak detection and to revise the DCD accordingly.

In its response to **RAI 254-2075, Question 05.02.04-07**, dated April 17, 2009, the applicant stated that the boric acid leak detection program will provide guidance for inspecting the integrity of bolting and threaded fasteners. For the reactor vessel closure head, this program includes surface examination requirements of Code Case N-729-1, with the conditions stated in 10 CFR 50.55a. The applicant stated that DCD Subsection 5.2.4.1 will be changed to state that the inservice inspection/ inservice testing (ISI/IST) programs detail the areas subject to examination and the method, extent and frequency of examinations, including a program to detect and correct potential RCPB corrosion caused by boric acid leaks, as described in NRC Generic Letter 88-05. For the reactor vessel closure head, this program includes surface examination requirements of Code Case N-729-1, with the conditions of 10 CFR 50.55a. The staff finds that the ISI program methodologies for inspection, evaluation and frequencies, as supplemented by GL 88-05 and Code Case N-729-1, and with conditions specified in 10 CFR 50.55a as described in the applicant's proposed DCD revision are in compliance with the requirements, and are therefore acceptable. The staff will verify that Section 5.2.4.1 is revised accordingly in the next revision to the DCD. **This is Confirmatory Item 05.02.04-07**.

5.2.4.4.11 Preservice Inspection and Testing Program

The NRC staff verified that the preservice inspection (PSI), ISI and IST programs are fully described and that implementation milestones have been identified. The applicant stated in DCD Section 5.2.4.2 that the PSI program for Class 1 components conforms to the guidelines of NB-5280 of Section III, Division 1 of the ASME Code and the preparation of the inspection and testing program is the responsibility of the COL applicant. In DCD Chapter 13, "Operational Programs," under Section 13.4.1, "Combined License Information," COL information item COL 13.4(1), the applicant stated that a COL applicant referencing the US-APWR design is to develop a description and schedule for the implementation of operational programs. The COL applicant is to "fully describe: the operational programs as defined in SECY-05-0197 and provide commitments for the implementation of operational programs required by regulation. In some instances, programs may be implemented in phases, which the COL applicant is to include in their submittal." The PSI program conforms to NB-5280 which is the acceptance criterion specified by the SRP and the COL information item that specifies the implementation milestones to be specified by the COL applicant. Based on the above, the staff finds that the PSI, ISI and IST programs for ASME Code Class 1 components as described in the DCD meet the SRP acceptance criteria and are, therefore, acceptable.

5.2.4.5 Combined License Information Items

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

Table 5.2.4-1
US-APWR Combined License Information Items

Item No.	Description	Section
13.4(1)	A COL applicant that references the U.S. APWR design certification will fully describe: the operational programs as defined in SECY-05-0197 and provide commitments for the implementation of operational programs required by regulation. In some instances, programs may be implemented in phases, which the COL applicant is to include in its submittal.	5.2.4

COL information items not identified in Table 1.8-2 of the DCD: None

5.2.4.6 Conclusions

With the exception of the confirmatory items identified above, the staff concludes the following: The design of the reactor coolant system incorporates provisions for access for ISI examinations in accordance with 10 CFR 50.55a(g)(3) and the 2001 Edition with the 2003 Addenda of the ASME Code, Section XI. Suitable equipment will be developed and installed to facilitate the remote inspection of these areas of the RCPB that are not readily accessible to inspection personnel. The final ISI program will consist of a preservice examination plan and an inservice inspection plan which addresses ASME Section XI criteria, augmented inspections, and boric acid corrosion control. The conduct of periodic inspections and pressure testing of pressure-retaining components of the reactor coolant pressure boundary are specified in accordance with the requirements in applicable subsections of Section XI of the ASME Code and provides reasonable assurance that evidence of structural degradation or loss of leak-tight integrity occurring during service will be detected in time to permit corrective action before the safety function of a component is compromised. Compliance with the inservice inspections required by this Code constitutes an acceptable basis for satisfying in part the requirements of GDC 32.

The staff concludes the ASME Code Class 1 ISI program description is acceptable and meets the inspection and testing requirements of GDC 32 and 10 CFR 50.55a. This conclusion is based on the applicant’s meeting the requirements of the ASME Boiler and Pressure Vessel Code, Section XI, “Rules for Inservice Inspection of Nuclear Power Plant Components,” as reviewed by the staff and determined to be appropriate for this application.

5.2.5 RCPB Leakage Detection

5.2.5.1 Introduction

The RCPB leakage detection systems are designed to detect and, to the extent practical, identify the source of reactor coolant leakage. Diverse measurement methods include monitoring of sump level and flow, containment airborne radioactivity, and containment air cooler condensate flow. Additional methods used to indicate leakage inside containment RCS inventory balance and localized humidity and temperature monitoring.

5.2.5.2 Summary of Application

DCD Tier 1: Section 2.4.7 in DCD Tier 1 contains a design description for the RCPB leakage detection system. Associated ITAAC are delineated in Table 2.4.7-1 in Section 2.4.7 of DCD Tier 1.

DCD Tier 2: The applicant has provided a DCD Tier 2 description in Section 5.2.5, summarized here in part, as follows:

> The application states that the US-APWR RCPB leakage monitoring system provides a means of detecting and identifying the source of reactor coolant leakage and monitoring leaks from the reactor coolant and associated systems. This system provides information which permits the plant operators to take corrective action if a leak is evaluated as detrimental to the safety of the facility.
>
> The application describes the design bases for the system and states that they meet the requirements of GDC 30, RG 1.45, and RG 1.29. The application further explains that RCPB leakage is classified as either identified or unidentified leakage in accordance with the guidance of Regulatory Position 1 of RG 1.45. Identified leakage includes leakage into closed systems such as pump seals or valve packing leaks that are captured, flow metered, and the leakage conducted to a sump or collecting tank. Leakage into the containment atmosphere for which the location is identified and which does not interfere with the unidentified leakage detection system or is identified as leakage from other than the RCPB is also identified leakage. Identified leakage is also defined as leakage into auxiliary systems and secondary systems. All other leakage is unidentified leakage.
>
> The application describes the methods for detecting identified intersystem leakage. The RCPB leakage detection system consists of temperature, pressure, humidity, radiation, flow and level sensors with associated instrumentation, power supplies and logic used to detect, indicate, record and alarm leakage from the reactor primary pressure boundary. Abnormal leakage is detected, indicated and alarmed to alert the operators. Leakage into auxiliary systems connected to or interfacing with the RCPB is detected by increasing auxiliary system level, temperature, and pressure indications or lifting of relief valves accompanied by increasing values of monitored parameters in the relief valve discharge path. The application provides specific detail for the intersystem leakage detection systems for the RHRS, SIS/accumulators, SI pump discharge, SIS direct vessel injection line, RHR emergency letdown lines, reactor head seal, and the CCW system.
>
> Unidentified reactor coolant leakage to the containment is monitored and quantified by monitoring the containment sump level, the containment airborne particulate and airborne gaseous radioactivity monitors, and condensate flow rate from the air coolers. Indirect and qualitative indications of coolant leakage are monitored by humidity, temperature, and pressure measurements of the containment atmosphere. Indications of reactor coolant leakage from these instruments are provided in the MCR with alarms. Additional leakage detection methods include the charging pump

operation, containment humidity monitoring, and reactor coolant inventory. The application provides the bases for the limits of reactor coolant leakage rates within the RCPB. They are in accordance with Regulatory Position 9 of RG 1.45.

ITAAC: RCPB leakage detection system ITAAC are delineated in Table 2.4.7-1 in Section 2.4.7 of DCD Tier 1, Revision 2.

Technical Specifications: The Technical Specifications associated with DCD Tier 2, Section 5.2.5, are given in DCD Tier 2, Chapter 16, Sections 3.4.15 and B.3.4.15. Technical Specifications provided in DCD Tier 2, Chapter 16, Sections 3.4.13 and B3.4.13 also pertain to the Leakage Detection System (DCD Tier 2, Section 5.2.5) in that they specify operational leakage limits for the RCS.

5.2.5.3 Regulatory Basis

The relevant requirements of the Commission regulations for this area of review, and the associated acceptance criteria, are given in Section 5.2.5 of NUREG-0800 and are summarized below. Review interfaces with other SRP sections can be found in Section 5.2.5 of NUREG-0800.

1. GDC 2, "Design Bases for Protection Against Natural Phenomena," as it relates to SSCs being designed to withstand the effects of natural phenomena, such as earthquakes, tornadoes, hurricanes, floods, seiches, and tsunami without loss of capability to perform their safety functions.

2. GDC 30, "Quality of Reactor Coolant Pressure Boundary," as it relates to the components, which are part of the RCPB being designed, fabricated, erected, and tested to the highest quality standards practical. Means shall be provided for detecting and, to the extent practical, identifying the location of the source of reactor coolant leakage.

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

Acceptance criteria adequate to meet the above requirements include:

1. RG 1.29, "Seismic Design Classification," as it relates to identifying and classifying system portions that should be designed to withstand the effects of a safe shutdown earthquake.

2. RG 1.45, "Guidance on Monitoring and Responding to Reactor Coolant System Leakage," as it relates to the selection of RCPB leakage detection systems.

5.2.5.4 Technical Evaluation

The staff reviewed the RCPB leakage detection system described in Section 5.2.5 of the US-APWR DCD, Revision 2, in accordance with Section 5.2.5 “Reactor Coolant Pressure Boundary Leakage Detection,” Revision 2, March 2007, of NUREG-0800, the SRP. DCD Section 5.2.5 was reviewed to determine if its description of the US-APWR design was adequate to provide reasonable assurance that the regulatory requirements and acceptance criteria cited in Subsection 5.2.5.3 above are met by the design.

In Section 5.2.5.4.1.2 of Revision 1 to the DCD Tier 2, fourth paragraph lists the following radionuclides: Na- 24, Cr-51, Zr-65 [*sic*], Mn-54, 56, Co-58, 60, and Fe-55, 59. The staff notes that there is no such nuclide as “Zr-65.” By comparing it with the nuclides discussed in DCD Section 11.1, it appears that “Zr-65” might be a typographic error for “Zn-65.” In **RAI 165-1967, Question 05.02.05-06**, the staff requested that the applicant acknowledge the above error and revise the DCD accordingly**.**

The applicant responded to **RAI 165-1967, Question 05.02.05-06**, in a letter dated February 20, 2009, in which the applicant concurred that “Zr-65” is a typographical error and provided a markup of the proposed change to “Zn-65” to be made to the Tier 2 DCD in Section 5.2.5.4.1.2. Subsequently, the staff confirmed that in Section 5.2.5.4.1.2 of Revision 2 to the DCD Tier 2, the reference to “Zr-65” had been replaced with “Zn-65.” Therefore, **RAI 165-1967, Question 05.02.05-06, is closed**.

In Revision 2 of the DCD Tier 2, Section 5.2.5.5, the applicant states that leak detection monitoring has no safety-related function. In Section 5.2.5.1, the applicant states that the leakage monitoring system is designed in accordance with the requirements of GDC 30 and the guidance provided in RG 1.29, “Seismic Design Classification,” Revision 4, March 2007, and RG 1.45, “Reactor Coolant Pressure Boundary Leakage Detection Systems, Revision 1,” May 2008.

During the staff review of Revision 2 to the DCD Tier 2, Section 5.2.5, a number of discrepancies regarding reference to RG 1.45 were noted. In supplemental **RAI 549-4390, Question 05.02.05-12,** the staff requested that the applicant correct the inconsistencies, including the title of the referencedRG, and correct the regulatory position identifiers to align with those provided in RG 1.45, Revision 1. In the letters, dated April 9, 2010 and February 5, 2010, the applicant responded **RAI 549-4390, Question 05.02.05-12**, and Question 14.02-120, revising DCD Tier 2, Section 5.2.5, to correct the inconsistencies identified above. Based on the marked-up DCD revision, the staff has determined that **RAI 549-4390, Question 05.02.05-12**, is resolved and closed. This is **Confirmatory Item 05.02.05-12**.

<u>GDC 2</u>

In order for the RCPB leakage detection system to meet the requirements of GDC 2 as it relates to SSCs being capable of withstanding natural phenomena, RG 1.29, Regulatory Positions C.1 (for safety-related SSCs) and C.2 (for non-safety-related SSCs) provide an acceptable method to meet this criterion. Regulatory Position C.2 indicates that the failure of non-safety-related portions of SSCs that could affect the SSCs’ safety-related function(s) should be designed to maintain the safety function following a seismic event. In addition, RG 1.45 states that at least one of the leakage detection systems should be capable of performing its function following a seismic event that does not require plant shutdown.

In Revision 2 of the DCD Tier 2 Section 5.2.5.5, the applicant states that leak detection monitoring has no safety-related function. Further, the DCD indicates that the containment airborne particulate radioactivity monitor is seismic Category I, and the containment airborne gaseous radioactivity monitor, the containment air cooler condensate flow rate monitoring system, the containment sump level, and flow monitoring system are qualified for seismic events not requiring a plant shutdown.

The staff reviewed the DCD information and verified the applicant's statement that RCS leakage detection systems have no safety-related function. Therefore, RG 1.29, Regulatory Position C.1, is not applicable. Further, RG 1.29, Regulatory Position C.2, and the RG 1.45 regulatory position cited above are satisfied because the containment airborne particulate radioactivity monitor is seismic Category I and the other leak detection methods identified above are qualified for seismic events not requiring a plant shutdown. Based on the above, the staff has determined that the design of the RCPB leakage detection system satisfies GDC 2 regarding protection from the effects of natural phenomena such as earthquakes without loss of its capability to perform its function.

GDC 30

Section 5.2.5 of the SRP requires the RCPB leakage detection system to meet the requirements of GDC 30 as it relates to providing the means for detecting and, to the extent practical, identifying the location of the source of reactor coolant leakage. Subsection II of SRP Section 5.2.5 indicates that meeting the guidelines of RG 1.45 is an acceptable means of satisfying GDC 30. There is operating experience regarding RCPB leakage detection that is not included in Revision 0 of RG 1.45, 1973, but is addressed in Revision 1 of RG 1.45, May 2008. This operating experience, including the Davis-Besse RCS leakage and reactor vessel corrosion event and inadequate response time for gaseous radiation monitor, will be discussed below in this SER.

DCD Tier 2 Section 5.2.5 indicates that identified and unidentified leakage will be collected and monitored separately without masking the flow rates between the two types. The staff has determined that this design feature will allow the RCPB leakage detection system to meet the guidance specified in the regulatory position of RG 1.45 that flow rates from identified sources are monitored separately from the flow rates from unidentified sources.

The applicant has stated that the sensitivity and response time of leakage detection equipment for unidentified leakage is such that a leakage rate, or its equivalent, of 0.5 gallon per minute (gpm) can be detected in less than an hour. The leak detection capability of 0.5 gpm within one hour, which was derived from the need to support leak-before-break application (DCD Tier 2 Section 3.6.3), is more stringent than the RG 1.45 criterion of one gpm within one hour. Thus, the staff has determined that the system meets the regulatory position specified in RG 1.45 that the leakage detection system should be capable of detecting one gpm leakage in less than one hour.

The applicant has also indicated that the methods employed for detecting leakage to the containment from unidentified sources include containment sump level monitoring, containment airborne particulate radioactivity monitoring, and condensate flow rate monitoring from air coolers, which are in the technical specifications and have the detection capability discussed above. In addition, the humidity, temperature, and pressure monitoring of the containment atmosphere can provide alarms and indirect indication of leakage to the containment. Based on the above, the staff has determined that these diverse leakage detection methods conform to those listed in the regulatory position of RG 1.45 that plant technical specifications should identify at least two independent and diverse instruments, that have the capability discussed above. In addition to the monitoring systems in the technical specifications, the plant should have other diverse methods, which may not necessarily have the quantitative detection capability specified above.

The applicant indicated in the DCD that Identified leakage into the containment is detected and monitored by directing coolant leakage from pump seals or valve packing to the containment coolant drain tank. Leakage through a steam generator from the primary side is detected by radiation monitors and liquid sampling. Further, the identified leakage and the steam generator leakage criteria are specified in the technical specifications. The staff has determined that these design features meet the guidance in RG 1.45 that the source and location of reactor coolant leakage should be identifiable to the extent practical and the plant should measure the leakage rate and that the plant should monitor critical components of the RCPB for leaks.

In DCD Tier 2, Section 5.2.5, the applicant states that reactor coolant system leakage into auxiliary systems connected to the RCPB is detected by increasing levels, temperature, and pressure indications or the lifting of relief valves in the interfacing system. Intersystem leakage is monitored in the residual heat removal system (suction side and emergency letdown), the safety injection system (accumulators, pump discharge, and direct vessel injection line), the reactor head seal, and the CCW system. Monitoring RCS leakage into the CCW system is accomplished by using radiation monitors and an increase in the CCW surge tank level. Leakage through a steam generator from the primary side is detected by radiation monitors and liquid sampling. Based on the above design, the staff has determined that the intersystem leakages are monitored using methods described above by the applicant that is consistent with the regulatory position in RG 1.45 that the plant should monitor intersystem leakage for systems connected to the RCPB.

Procedures for Conversion and Alarms in the MCR

RG 1.45 states that the plant should provide output and alarms from leakage MCR. Procedures for converting the instrument output to a leakage rate should be readily available to the operators. The alarm should provide operators an early warning signal so that they can take corrective actions. The applicant states in the DCD that indications and alarms for the leakage detection system are provided in the MCR, and those procedures for converting the various indications to a common leakage equivalent will be available to operating personnel. The staff has determined that the application meets the above guidance in RG 1.45 with the exceptions discussed below. The staff could not find the procedures anywhere in Revision 1 of the DCD that were promised for the operator to convert the instrument indications of various leakage detection (e.g., containment radioactivity monitors, containment sump level monitor, containment air cooler condensate flow rate monitor) into common leakage rate. Therefore, in **RAI 165-**

1967, Question 05.02.05-03, the staff requested the applicant to provide the following information.

- Identify a combined license (COL) information item to require the COL applicant to provide operators the procedures, chart, or graph that permits rapid conversion of instrument indications of various leakage detection instruments into common leak rate (gpm).

- Define the alarm set points and demonstrate the set points are sufficiently low to provide an early warning for operator actions prior to Technical Specification (TS) limits.

The applicant responded to **RAI 165-1967, Question 05.02.05-03**, in a letter dated February 20, 2009. In that response, the applicant promised to provide leakage conversion procedures as part of the Operating and Emergency Operating Procedures described in DCD Section 13.5.2.1. The applicant provided a markup of Tier 2 DCD, Section 5.2.5.6 stating that the leakage conversion procedure will be developed. Since the Operating and Emergency Operating Procedures described in DCD Section 13.5.2.1 are to be developed by the COL applicant, the staff can not confirm the existence and evaluate the adequacy of the procedures. The applicant did not have a COL information item to require COL applicants to provide operators with the procedures discussed above. Consequently, the staff requested the applicant to provide the above information in **Supplemental RAI 438-3079, Question 05.02.05-07**.

The applicant responded to **RAI 438-3079, Question 05.02.05-07** in a letter dated September 14, 2009. In that response, the applicant agreed to add a COL information item specifically requiring a COL applicant to provide operators with procedures for converting leakage detection instrument readings into common leakage rates. In addition, the applicant stated that alarm set point values will be established for each leakage detection system by the COL applicant based on the actual instruments utilized. The applicant also provided a markup of the changes to the DCD.

In the staff review of Section 5.2.6 of Revision 2 to the DCD Tier 2, it was noted that COL 5.2(14) had been added to the list of Combined License Information items. This new COL information item requires the COL applicant to address and develop a milestone schedule for the preparation and implementation of procedures for conversion into common leakage rates. This COL item will lead the COL applicant to address the procedures in the application, and the NRC staff to review the COL information item in meeting the guidance in Regulatory Positions C.3.3 and C.3.4 of RG 1.45, Revision 1, regarding procedures for conversion of instrument output into leakage rates and alarms for early warning signal. Therefore, **RAI 165-1967, Question 05.02.05-03 and RAI 438-3079, Question 05.02.05-07 are closed**. Based on the resolution of the above RAIs, the staff has determined that the regulatory positions of RG 1.45 discussed above for the alarms and indications in the MCR are satisfied.

In DCD Tier 2 Section 5.2.5, the applicant states that periodic testing of leakage detection systems is conducted to verify the operability and sensitivity of the detection equipment. The staff has determined that this provision is consistent with the guidance in RG 1.45 that the leakage monitoring systems should have provisions to permit calibration and testing during plant operation.

In Chapter 16 of Tier 2 in the DCD, the applicant has provided TS for the leakage detection systems that specify allowable leakage limits for identified, unidentified, RCPB leakage, intersystem (primary to secondary) leakage and requirements for instruments of diverse monitoring principles to be operable during plant operating modes 1, 2, 3, and 4. This meets the guideline presented in RG 1.45 that plant TS should include the limiting conditions for identified, unidentified, RCPB leakage, and intersystem leakage and they should address the availability of various types of instruments to ensure adequate coverage during all phases of plant operation (see the additional discussion of these TS items below).

Based on the above evaluation relative to RG 1.45, the staff concludes the design of the RCPB leakage detection system satisfies GDC 30 as it relates to providing the means for detecting and, to the extent practical, identifying the location of the source of reactor coolant leakage.

Inspections, Tests, Analyses and Acceptance Criteria (ITAAC)

The staff reviewed other sections of Revision 1 to the DCD as part of the evaluation of the RCPB leakage detection system. From this review, the staff determined that there were no DCD Tier 1 entries for this system, and there were no ITAAC items for this system provided in the DCD. In DCD Tier 2 Section 5.2.5.5, Safety Evaluation, the applicant states that leak detection monitoring has no safety-related function. However, since the leakage detection systems are monitoring the integrity of the RCPB, which is safety-related, the staff requested the applicant to provide Tier 1 ITAAC in **RAI 165-1967, Question 05.02.05-01**. The ITAAC should verify the sensitivity, response time, and alarm limit for the RCPB leakage detection instrument.

The applicant responded to RAI 165-1967, Question 05.02.05-01, in a letter dated February 20, 2009. In that response, the applicant promised to provide a new ITAAC item for the RCPB leakage detection system in Tier 1 of the DCD. It also provided a proposed markup of the Reactor Coolant Pressure Boundary Leakage Detection ITAAC. However, the markup provided for the proposed ITAAC did not address the means for determining the acceptability of sensitivity, response time, and alarm limits for the RCPB leakage detection instruments. The staff determined that the applicant’s response was only partially responsive and that the ITAAC for the RCPB leakage detection system needs to specify the acceptance criteria for instrument sensitivity, response time, and alarm setpoints. Consequently, the staff requested the applicant to provide the above information in **Supplemental RAI 438-3079, Question 05.02.05-08**.

The applicant responded to RAI 438-3079, Question 05.02.05-08 in a letter dated September 14, 2009. In that response, the applicant committed to revise the new ITAAC for the RCPB leakage detection system for the containment sump level channels and standpipe level channels to add acceptance criteria that verify the instruments are capable of meeting the required sensitivity, response times and setpoints. In addition, the applicant provided a markup of the revised ITAAC. The staff has confirmed that the inclusion of the new ITAAC for the RCPB leakage detection system and the promised revisions are contained in Revision 2 of the DCD Tier 1 as Section 2.4.7. However, the acceptance criteria for the containment radiation particulate monitor, RMS-RE-040, do not address the means for verifying the sensitivity, response time and alarm setpoint. Instead, ITAAC in Tier 1 Section 2.4.7 refers to another ITAAC item in Revision 2 of the DCD Tier 1, Section 2.7.6.6, for checking the radiation monitoring equipment.

After reviewing the referenced ITAAC requirements in Section 2.7.6.6, the staff discovered that the acceptance criteria in the referenced section do not verify instrument sensitivity, response time or alarm setpoints. Staff concerns in **RAI 438-3078, Question 05.02.05-08,** remained partially unresolved. As a result of staff's review of the response to Question 05.02.05-08, **the staff issued supplemental RAI 478-3837, Question 05.02.05-11,** requesting the applicant to include testing and acceptance criteria that specifically address the means for verifying the sensitivity, response time and setpoints of the containment radiation particulate monitor, RMS-RE-040.

The applicant responded to **RAI 478-3837, Question 05.02.05-11**, in a letter dated December 2, 2009. In that response the applicant promised to revise DCD Tier 1, Section 2.4.7, Table 2.4.7-1 that specifies the ITAAC for the RCPB leakage detection system to provide a means for verifying that the containment radiation particulate monitor, RMS-RE-040, can detect leakage with the required sensitivity, response time and setpoint. The applicant also provided a markup of the revised ITAAC in their response. The staff reviewed the applicant's response and determined that the proposed changes adequately address the requirement for providing a means to verify the sensitivity, response time and setpoints for all of the primary RCPB leakage detection system instruments. Therefore, **RAI 478-3837, Question 05.02.05-11 is resolved**, and thus also are **RAI 165-1967, Question 05.02.05-01, and RAI 438-3078, Question 05.02.05-08**. Incorporation of the proposed revision in the next revision of the DCD is **Confirmatory Item 05.02.05-11**.

Initial Testing Program

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, Subsection II, "Acceptance Criteria," states that the DC applicant can meet the above requirements by conforming to the criteria stated in RG 1.68, "Initial Test Programs for Water-Cooled Nuclear Power Plants."

The staff reviewed Chapter 14 of Tier 2 in Revision 1 of the US-APWR DCD to ensure the applicant conformed to initial plant test requirements. Chapter 14, "Verification Programs," of the US-APWR DCD lists Section 14.2.12.1.71, "RCS Leak Rate Preoperational Test," and Section 14.2.12.2.1.10, "RCS Final Leak Test." RG 1.68, Appendix A, Items 1.j.(5) and 2.d identify RCS leak detection systems being required for the initial testing. The initial test program for US-APWR is evaluated in Section 14.2 of this SER, and evaluation of the RCPB leakage detection initial test program in this section is an extension of the evaluation provided in Section 14.2. The objective of the RCPB leakage detection initial test program was found to be appropriate since it is to demonstrate the capability of the leak detection systems. However, the staff was not able to find the sensitivity and response time of the leak detection systems being included in the above tests. In **RAI 165-1967, Question 05.02.05-05**, the staff requested the applicant to identify the tests to demonstrate the sensitivity, response time, and alarm limit of the leak detection systems.

The applicant responded to **RAI 165-1967, Question 05.02.05-05**, in a letter dated February 20, 2009. In that response, the applicant referred to responses to previous RAI questions (RAI Question Nos. 14.02-69 and 14-02-101).

The staff reviewed the applicant's response to the previous RAI questions, and determined that the responses to RAI Question 33, No. 14.02-69, and RAI Question 102-1391, No. 14.02-101, adequately address the concern that leak detection system sensitivity, response time and alarm set points were not included in the RCS leak test programs. Specifically, the applicant's response to RAI Question 102-1391, No. 14.02-101 states that, as part of the pre-requisites for the test, RCS leakage detection instrumentation will be calibrated to ensure that the sensitivity and response time of the leakage detection equipment is consistent with the design capability. In addition, the applicant's response to RAI Question 14.02-69, shows that verification of the alarm set points will be accomplished by "using simulated signals to each channel of the RCPB leak detection systems" to verify alarm and automatic functions as part of the test method for RCPB leak detection systems preoperational test protocols. Therefore, **RAI Question 05.02.05-5**, is resolved and closed.

In the staff review of the Verification Programs of Revision 2 to the DCD Tier 2, Chapter 14, it was noted that the promised changes to Section 14.2.12.1.71, "RCS Leak Rate Preoperational Test," and Section 14.2.12.2.1.11, "RCS Final Leak Test," had been implemented. In addition, a new section, Section 14.2.12.1.115, "RCPB Leak Detection System Preoperational Test," has also been included. This new section specifies the methods to determine the sensitivity and alarm setpoints for the liquid leakage detection instruments. The radiation monitors are addressed in Section 14.2.12.1.78, "Process and Effluent Radiological Monitoring System, Area Radiation Monitoring System and Airborne Radioactivity Monitoring System Preoperational Test," of Revision 2 to the DCD Tier 2. Based on the above, the staff has confirmed the DCD changes are consistent with the response to **RAI 165-1967, Question.05.02.05-05**; thus, **RAI 165-1967, Question 05.02.05-05**, is resolved and closed.

Technical Specifications (TS)

DCD Section 5.2.5.8 states that the limits imposed on identified and unidentified reactor coolant leakage are provided in DCD Chapter 16, which describes the TS. TS 3.4.13 and 3.4.15 address the allowed limits of RCS leakage and the required leakage detection instrumentation for this system. Limiting Condition for Operation (LCO) 3.4.13 specifies the leakage limits during operating modes 1, 2, 3, and 4:

- No pressure boundary leakage is allowed
- One gallon per minute of unidentified leakage is allowed
- Up to 10 gpm of identified leakage is allowable
- Primary to secondary leakage through any one steam generator is limited to 150 gallons per day

The staff has determined that the limits specified above were considered sufficient to meet the guidance of RG 1.45 regulatory position that plant TS should include the LCO for identified, unidentified, RCPB, and intersystem leakage.

Also in Chapter 16 of the DCD Tier 2 (Revision 1), LCO 3.4.15 requires instruments of diverse monitoring principals be operable and sets monitoring conditions and restoration time limits if any of the primary leakage monitoring systems become unavailable. The staff has determined that these TS requirements are adequate to keep the RCPB leakage detection system in compliance with the regulatory position of RG 1.45 that plant TS should address the availability of various types of instrumentations to ensure adequate coverage during all phases of plant operation.

In reviewing the TS, the staff identified a potential problem in the TS with respect to gaseous radiation monitors. The applicant indicated that its gaseous radiation monitor could detect 1-gpm leakage within one hour and included the gaseous radiation monitor in the TS. NRC Information Notice (IN) 2005-24 indicated that this 1-hour response time at 1-gpm may not be a conservative estimate for the gaseous radiation monitor. Revision 1 of RG 1.45 provides guidance to deal with this issue. In light of IN 2005-24 and Revision 1 of RG 1.45, the staff issued **RAI 165-1967, Question 05.02.05-02,** requesting the applicant to reevaluate the sensitivity and response time of the radiation monitor using a realistic source term in the RCS and to determine whether it is proper to keep the gaseous radiation monitor in the TS.

The applicant responded to **RAI 165-1967, Question 05.02.05-02** in a letter dated February 20, 2009. In that response, the applicant proposed to delete references to the containment airborne gaseous radioactivity monitor from the TS and provided a markup of the changes to be made in the Tier 2 DCD, Chapter 16 TS. The staff reviewed the applicant's response to the RAI question, and determined that the proposed change to the TS regarding the use of gaseous radiation monitors in the RCPB leakage detection system is consistent with RG 1.45 Revision 1 Regulatory Position 2.3 regarding the use of gaseous radiation monitors. Therefore, **RAI 165-1967, Question 05.02.05-02,** was resolved and closed. The proposed change to the TS was confirmed by the staff in Revision 2 of the DCD. Furthermore, the staff reviewed the markup provided for the proposed change, and it seemed that perhaps <u>two</u> particulate containment atmosphere radioactivity monitoring instrumentation channels would be required by LCO 3.4.15. It was uncertain whether this was a simple oversight by the applicant or if it intended that two airborne particulate monitors would be employed to provide redundancy. Consequently, the **staff requested clarification from the applicant about the number of airborne particulate monitors required by the TS in supplemental RAI 438-3079, Question 05.02.05-09.**

The applicant responded to **RAI 438-3079, Question 05.02.05-09** in a letter dated September 14, 2009. In that response, the applicant agreed that the word "both" had been inadvertently left as part of the markup response to RAI Question 05.02.05-2 and should have been removed. The applicant provided a revised markup for that section that indicates only a single particulate containment atmosphere radioactivity instrument is required. The staff reviewed this change and finds that it correctly identifies the number and type of containment radiation monitoring instruments to be included as part of the RCPB leakage detection system. The staff also verified that all of these changes have been incorporated into Revision 2 of the DCD. Therefore, **RAI 438-3079, Question 05.02.05-09**, is resolved and closed.

In addition, the staff has determined that the limits imposed by the applicant in LCO 3.4.13 and LCO 3.4.15 and the associated Surveillance Requirements are consistent with the recommended set of Standard TS for a PWR as provided in NUREG-1431, "Standard Technical Specifications Westinghouse Plants," Volume1, Revision 3.0, June 2004.

COL Information Items

There were no COL Information Items pertaining to the RCPB leakage detection system provided in Revision 1 of the DCD Tier 2, Section 5.2.6, "Combined License Information." As a result of the staff review, the applicant identified two COL information items, which are discussed below.

First, as discussed above regarding the procedures to convert various leak detection instrument indications to a common leakage equivalent for operating personnel, the DC applicant should specify a COL information item in the DCD. In RAI 165-1967 Question 05.02.05-3, the staff requested the applicant to provide such a COL information item. The applicant's subsequent response and the addition of COL 5.2(14) in Revision 2 of the DCD Tier 2, Section 5.2.6, allowed RAI 165-1967 Question 05.02.05-3 to be closed. The second COL information item has to do with the procedures for prolonged low-level leakage.

<u>Procedures for Prolonged Low-Level Leakage</u>

An important safety issue lesson learned from the Davis Besse reactor vessel head leakage event of 2002 indicated that small RCS leakage and boron corrosion, if it lasted for a long time, could be a significant safety concern. This concern was not recognized in Revision 0 of RG 1.45 because it was published in 1973. It is discussed in Revision 1 of RG 1.45, May, 2008, and the regulatory guidance regarding how to use procedures to manage the small RCS leakage is provided. Revision 1 of the DCD did not address such procedures that would be developed to adequately provide operators an early warning mechanism and provide guidance in response to a low-level leakage (well below TS limits) event such as the one that occurred at Davis-Besse. In **RAI 165-1967, Question 05.02.05-04**, the staff requested the applicant to provide a COL information item regarding the requirement to develop procedures for determining the existence of and operator response to prolonged low-level leakage conditions.

The applicant responded to **RAI 165-1967, Question 05.02.05-04,** in a letter dated February 20, 2009. In that response, the applicant committed to establish procedures that specify operator actions in response to prolonged low-level leakage conditions as part of the Operating and Emergency Operating Procedures described in DCD Section 13.5.2.1. Although the applicant provided a markup of changes to Tier 2 DCD, Section 5.2.5.8, stating that the leakage management procedure will be developed, it did not include a markup of the procedure; Thus, no evaluation of the existence and adequacy of the procedure was possible. In addition, the applicant did not add a COL information item to develop procedures for determining the existence of and operator response to prolonged low-level leakage conditions. Consequently, the staff requested the applicant to provide the above procedures and COL information item in **Supplemental RAI 438-3079, Question 05.02.05-10.**

The applicant responded to **RAI 438-3079, Question 05.02.05-10** in a letter dated September 14, 2009. In that response, the applicant committed to add a COL information item specifically requiring a COL applicant to provide operators with procedures for response to prolonged low-level leakage rates.

In the review of Section 5.2.6 of Revision 2 to the DCD Tier 2, the staff confirmed that COL 5.2(15) had been added to the list of Combined License Information items. This new COL information item requires the COL applicant to address and develop a milestone schedule for the preparation and implementation of procedures for operator response to prolonged low-level leakage rates. This COL item will lead the COL applicant to address, and the NRC staff to review the COL application in meeting the guidance in Regulatory Positions C.3.1 and C.3.2 of RG 1.45, Revision 1 that the plant should periodically analyze the trend in the unidentified and identified leakage rates and that the plant should establish procedures for responding to leakage. Therefore, RAI **165-1967, Question 05.02.05-04, and RAI 438-3079, Question 05.02.05-10, are closed**.

5.2.5.5 Combined License Information Items

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

Table 5.2.5-1
US-APWR Combined License Information Items

Item No.	Description	Section
	Note that Section 5.2.5 does not identify any COL information items. However, Section 5.2.6 lists all the COL information items pertinent to Section 5.2, including the two that follow, which are associated with RCPB leakage detection.	5.2.6
COL 5.2(14)	*Procedures for conversion into common leakage rate:* *The COL applicant addresses and develops a milestone schedule for preparation and implementation of the procedure.*	5.2.6
COL 5.2(15)	*Procedures for operator response to prolonged low-level leakage:* *The COL applicant addresses and develops a milestone schedule for preparation and implementation of the procedure.*	5.2.6

5.2.5.6 Conclusions

With the exception of **Confirmatory Items 05.02.05-11** and **05.02.05-12**, the staff concludes that the RCPB Leakage Detection System meets the requirements of GDC 2 and 30 and the guidelines of SRP Section 5.2.5.

5.3 Reactor Vessel

The RPV contains the reactor fuel and the vessel internals which direct the flow of reactor coolant. The RPV has four inlet nozzles, four outlet nozzles, and four direct vessel injection nozzles located in a horizontal plane just below the reactor vessel flange, but above the top of the fuel. The reactor coolant enters the RPV through the inlet nozzles and is guided downward into the annulus between the core basket and the vessel shell and then directed upward through the core, acquiring thermal energy. The reactor coolant leaves the RPV through the outlet nozzles. The RPV closure head contains nozzles for the control rod drive mechanism, in-core instrumentation, and thermocouple and RPV level instrumentation system penetrations. There are no penetrations in the RPV lower head.

5.3.1 Reactor Vessel Materials

5.3.1.1 Introduction

Section 5.3.1 of the US-APWR DCD describes the RPV materials of construction, their specifications, and applicable construction codes. This section also refers to the applicable GDC satisfied by the US-APWR RPV. Additionally, this section describes the provisions for performing NDE to ensure that the manufacturing processes are performed consistently with the construction codes. This section also describes the strategy for minimizing the effects of irradiation embrittlement on the RPV beltline region and weld material. The applicant presents its methods for conducting material surveillances over the life of the RPV to ensure material condition is within predicted values. Finally, RPV head closure bolting material is described within this section.

The purpose of this review is to assess the adequacy of material specifications used for the RPV and applicable attachments and appurtenances, such as the shroud support, studs, control rod drive housings, vessel support skirt, stub tubes, and instrumentation housings. Their adequacy for use in the construction of such components is assessed on the basis of the mechanical and physical properties of the materials, the effects of irradiation on these materials, their corrosion resistance, and their fabric ability. Similarly, the specifications for austenitic steel and nonferrous metals specified for the above applications are reviewed with respect to mechanical properties, stress-corrosion resistance, and fabric ability. The review also focuses on the processes used for manufacturing and fabricating the RPV, methods for non-destructive examination of the RPV, controls for ferritic and austenitic stainless steels, fracture toughness, RPV material surveillance program, monitoring programs, and RPV fasteners.

5.3.1.2 Summary of Application

DCD Tier 1: The Tier 1 information associated with this section is found in Tier 1, Section 2.4.1.

DCD Tier 2: The applicant has provided a Tier 2 description of the materials used in the RPV in Section 5.3.1, summarized here in part as follows:

Materials, Special Processes, Special Methods and Special Controls

Material specifications for the RPV pressure boundary meet the requirements of ASME Code, Sections II and III and are described in DCD Subsection 5.2.3. The RPV is manufactured mainly from low-alloy steel (ferritic) forgings in accordance with ASME Code, Section III requirements for Class 1 components. The pressure-boundary forging specification is SA-508 Grade 3 Class 1. The forgings are heat-treated by quenching and tempering, and fine-grain size of 5 or finer is required. The forgings are also vacuum-degassed to lower the hydrogen level and improve the quality of the metal. In order to reduce the effects of irradiation embrittlement on beltline-region ferritic materials, the copper, nickel and phosphorus contents are limited as specified in DCD Table 5.3-1. The RPV lower shell, transition ring and the weld line between the lower shell and transition ring are fully or partially within the beltline region.

Welding material for the RPV conforms to ASME Code, Section III requirements and the applicable welding specification, or satisfies requirements for other welding materials as permitted in ASME Code, Section IX. The welding material specifications are described in DCD Subsection 5.2.3. Special controls for welding ferritic and austenitic-stainless steels are also described in DCD Subsection 5.2.3

NDE are carried out to satisfy the requirements of ASME Code, Section III (during manufacturing) and Section XI (pre-service examinations). NDE acceptance criteria are in accordance with ASME Code, Section III (during manufacturing) and Section XI (pre-service examinations). The application discusses the proposed special methods for NDE used to satisfy ASME Code Section III NB-2500 requirements. The acceptance standards for the NDE during manufacturing are in accordance with ASME Code Section III NB-5300 and that the NDE is conducted in accordance with NB-5200. The application describes the methods of NDE, which include techniques of ultrasonic examination, liquid penetrant examination, and magnetic particle examination.

The applicant claims conformance to the RGs listed in paragraph 5.3.1.3 of this SER. These RGs impose special controls on ferritic and austenitic-stainless steels.

Fracture Toughness

Ferritic RPV pressure-boundary materials satisfy the fracture toughness requirements of 10 CFR Part 50, Appendix G. Material test coupons and test specimens are prepared and tested in accordance with ASME Code, Section III. For forging material, except for bolting, the Charpy V-notch (CVN) impact specimens are oriented normal to the principal working direction. For bolting and bars, axial test specimens are used.

The nil ductility reference temperature (RT_{NDT}) for each of these materials is determined by first determining a nil-ductility transition temperature (T_{NDT}) through a drop weight test. Next, CVN tests are carried out at temperatures not greater than T_{NDT} + 60 °F. Where the CVN test results exhibit at least 35 mils of lateral expansion and not less than 50 ft-lbs of absorbed energy, the T_{NDT} is defined as the nil ductility reference temperature, RT_{NDT}. The maximum value for this RT_{NDT} for RPV pressure boundary ferritic materials is 10 °F. The maximum RT_{NDT} for the

RPV beltline region ferritic materials is 0°F for forgings and -20°F for weld materials.

Full transverse CVN curves are generated for beltline region base and weld material. As a minimum, three specimens are tested in the upper-shelf region (100 percent shear). In accordance with the requirements of 10 CFR Part 50, Appendix G, the minimum initial upper-shelf energy (USE) for beltline–region, base material in the transverse direction and weld material along the weld is 75 ft-lb. The predicted Charpy USE at the end of the RPV life is estimated using RG 1.99, and expected to be greater than 50 ft-lb, as required by 10 CFR Part 50, Appendix G, at 1/4-thickness (1/4-T) of the RPV beltline-region shell.

Material Surveillance

The program evaluates the effects of radiation on the fracture toughness properties of RPV ferritic pressure boundary materials by the transition temperature and fracture mechanics approach, in accordance with ASTM E-185-82 and 10 CFR 50 Appendix H. In addition to tensile and CVN test specimens, 1/2-T CT fracture toughness test specimens are used as they will provide additional information concerning the pressure-temperature (P-T) limits of the RPV beltline materials. Implementation of the RPV material surveillance program for a specific plant is to be addressed by the COL applicant.

The surveillance program for the RPV consists of six capsules. The capsules include test specimens for the RPV weld metal, base metal and heat-affected zone (HAZ) metal. Base-metal test specimens are taken from locations near the fracture-toughness test specimens, and are oriented in both the longitudinal and transverse direction compared to the principal forging direction of the forging material. Weld test plates for surveillance program specimens have their principal working direction parallel to the weld line so that specimens for the HAZ area are normal to the principal working direction. A minimum of 9 tensile test specimens, 48 CVN test specimens and 6 CT fracture test specimens are contained in each capsule. The capsules are located in guide baskets that are attached to the outside of the core barrel. The guide baskets are located at the orientation shown in DCD Figure 5.3-1 so that the lead factors (ratio of the neutron flux at the location of the capsule to that at the RPV inner surface at the peak fluence location) of the surveillance capsules are between 2 and 3. This range for the lead factors is specified to monitor the embrittlement properties of the RPV materials in the future and takes into consideration the recommendations of ASTM E-185-82. Plant-specific orientation and the resulting lead factors for the capsules will be addressed by the COL applicant for each plant. Additionally, the application describes the functions for each of the types of the test specimens and the significance of their test results with respect to meeting the requirements of ASME Code Section III NB-2220, NB-2210, and NB-2322. The specimens are tested in order to determine RPV material properties such as RT_{NDT}, and fracture toughness to predict changes in properties over the life of the RPV. The applicant provides a recommended general capsule withdrawal schedule in DCD Subsection 5.3.1.6 based on the requirements of ASTM E185-82. The use of such capsules and their withdrawal schedule will be addressed by the COL applicant for each plant.

RPV Fasteners

The stud bolts, nuts and washers that attach the reactor vessel closure head flange to the vessel flange are ASME Code, Section II, SA-540 Grade B24 material and satisfy the fracture toughness requirements of ASME Code, Section III and 10 CFR Part 50, Appendix G. The allowable tensile strength is between 145 ksi and 170 ksi. The surface Brinell hardness shall be 302 HB minimum to 311 HB maximum. Also discussed with threaded fasteners are the threaded holes into which the bolts and studs are installed.

ITAAC: Item 3 in Table 2.4.1-2, "Reactor System Inspections, Tests, Analyses, and Acceptance Criteria," indicates that inspections of the as-built system will be conducted to verify that the functional arrangement of the RPV is as shown in Figure 2.4.1-3. Items 1, 4, 5, 6 and 7 in Table 2.4.1-2 state that applicable components listed as ASME Section III in Table 2.4.1-1, including the RPV, will be designed, fabricated and inspected in accordance with ASME Code Section III.

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

Technical Reports: MHI Technical Report MUAP-09018, "Calculation Methodology for Reactor Vessel Neutron Flux and Fluence"

5.3.1.3 Regulatory Basis

The relevant requirements of the Commission regulations for this area of review, and the associated acceptance criteria, are given in Section 5.3.1 of NUREG-0800 and are summarized below. Review interfaces with other SRP sections can be found in Section 5.3.1 of NUREG-0800.

1. GDC 1 and GDC 30 found in Appendix A to 10 CFR Part 50, as they relate to quality standards for design, fabrication, erection, and testing of SSCs.

2. GDC 4, as it relates to the environmental compatibility of components

3. GDC 14, as it relates to prevention of rapidly propagating fractures of the reactor coolant pressure boundary (RCPB).

4. GDC 31, as it relates to material fracture toughness.

5. GDC 32, as it relates to the requirements for a materials surveillance program.

6. 10 CFR 50.55a, as it relates to quality standards for design, and determination and monitoring of fracture toughness.

7. 10 CFR 50.60, "Acceptance criteria for fracture prevention measures for light-water nuclear power reactors for normal operation," as it relates to RCPB fracture toughness and material surveillance requirements of 10 CFR Part 50, Appendix G, and Appendix H.

8. 10 CFR Part 50, Appendix B, Criterion XIII, as it relates to onsite material cleaning control.

9. 10 CFR Part 50, Appendix G, as it relates to materials testing and acceptance criteria for fracture toughness.

10. 10 CFR Part 50, Appendix H, as it relates to the determination and monitoring of fracture toughness.

Acceptance criteria adequate to meet the above requirements include:

1. Regulatory Guide (RG) 1.31, "Control of Ferrite Content in Stainless Steel Weld Metal," as it relates to the control of welding in fabricating and joining safety-related austenitic stainless steel components and systems.

2. RG 1.34, "Control of Electroslag Weld Properties," as it relates to acceptable solidification patterns and impact test limits and the criteria for verifying conformance during production welding.

3. RG 1.37, "Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water Cooled Nuclear Power Plants," as it relates to the quality of water used for final cleaning or flushing of finished surfaces during installation.

4. RG 1.43, "Control of Stainless Steel Weld Cladding of Low-Alloy Steel Components," as it relates to criteria to limit the occurrence of under-clad cracking in low-alloy steel safety-related components clad with stainless steel.

5. RG 1.44, "Control of the Use of Sensitized Stainless Steel," as it relates to the compatibility of RCPB materials with the reactor coolant and the avoidance of stress corrosion cracking.

6. RG 1.50, "Control of Preheat Temperature for Welding Low Alloy Steel"

7. RG 1.71, "Welder Qualification for Areas of Limited Accessibility," as it relates to welder requalification.

8. RG 1.99, "Radiation Embrittlement of Reactor Vessel Materials," as it relates to RPV fracture toughness.

9. RG 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," as it relates to the RPV material surveillance program.

5.3.1.4 Technical Evaluation

The staff reviewed DCD, Tier 2, Section 5.3.1, "Reactor Vessel Materials," in accordance with SRP Section 5.3.1.

The NRC staff reviewed the US-APWR RV materials to ensure that they meet the relevant requirements of GDC 1, GDC 30, and 10 CFR 50.55a as related to the material specifications, fabrication, and NDE. Compliance with these requirements will determine whether the RV materials are adequate to ensure a quality product commensurate with the importance of the safety function to be performed. The material specifications for the US-APWR design are in accordance with ASME Code, Section III, requirements and Appendix G to 10 CFR Part 50. In addition, the design and fabrication of the RV conforms to ASME Code, Section III, Class 1, requirements. Furthermore, the RV and its appurtenances are fabricated and installed in accordance with ASME Code, Section III, NB-4000. The NDE of the RV and its appurtenances is conducted in accordance with ASME Code, Section III, NB-5200 requirements. The staff finds this acceptable because compliance with ASME Code, Section III, and Appendix G to 10 CFR Part 50 constitutes an adequate basis for satisfying the requirements of GDC 1, GDC 30, and 10 CFR 50.55a as they relate to the material specifications, fabrication, and NDE of RV materials.

Section 5.2.3 of this report provides the staff's evaluation of the welding of ferritic steels and austenitic stainless steels and addresses compliance with GDC 4.

Steels from nondomestic sources could have different characteristic responses to radiation embrittlement, particularly those steels with high phosphorous and sulfur contents. The methodology of RG 1.99, Revision 2, may not apply to steels with high phosphorous and sulfur contents. DCD Tier 2, Table 5.3-1, indicates that restrictive maximum content limits would be imposed on the critical residual elements (copper, nickel, phosphorous, etc.) in the materials of the RV beltline. Specified limits for RV materials used in the core beltline region are the following:

- Base Materials: 0.05 percent maximum copper, 0.005 percent maximum phosphorus, 0.05 percent maximum vanadium, and 1.00 percent maximum nickel (forging).

- Weld Materials: 0.08 percent maximum copper, 0.012 percent maximum phosphorus, 0.05 percent maximum vanadium and 0.95 percent maximum nickel.

The staff finds the applicant's limits to be acceptable because the chemical content controls imposed on the RV materials meet the guidelines of RG 1.99, Revision 2.

The RV studs, nuts, and washers for the main closure flange are manufactured using ASME SA-540 Grade B24 (4340 Mod) Class 3 and their testing conforms to guidance in RG 1.65.

The tests for fracture toughness of RV materials specified in the DCD are in accordance with ASME Code, Section III, paragraph NB-2300, and Appendix G to 10 CFR Part 50. The staff confirmed that the applicant's initial Charpy V-notch minimum upper-shelf fracture energy levels for the RV beltline base metal transverse direction and welds are greater than 101.7 N-m (75 ft-lbs). DCD, Tier 2, Table 5.3-4, indicates that the EOL values for the USE are greater than 67.8 N-m (50 ft-lbs) for the beltline forgings and welds. The staff performed an independent calculation to estimate the decrease in shelf energy using the calculational methods described in RG 1.99 for the beltline forgings and welds and found the applicant's EOL values of USE meet ASTM E185 limits and are, thus, acceptable.

The predicted EOL Charpy USE and adjusted reference temperature for the RV materials are in accordance with the requirements of Appendix G to 10 CFR Part 50. The fracture toughness tests required by the ASME Code and Appendix G provide reasonable assurance that adequate safety margins against the possibility of non-ductile behavior or rapidly propagating fracture can be established for all pressure-retaining components of the RV. This methodology ensures adequate safety margins are maintained during operating, testing, maintenance, and postulated accident conditions. Compliance with the provisions of Appendix G to 10 CFR Part 50 satisfies the requirements of GDC 14 and 31 and 10 CFR 50.55a regarding the prevention of fracture of the RV. Therefore, the staff finds that the applicant's fracture toughness design of its reactor vessel materials is acceptable.

The design of an RV must consider the potential embrittlement of RV materials as a consequence of neutron irradiation and the thermal environment. GDC 32 requires that the RCPB components shall be designed to permit an appropriate material surveillance program for the RV. Appendix H to 10 CFR Part 50 provides requirements for such a program.

To meet the requirements of GDC 32, the US-APWR design includes provisions for a material surveillance program to monitor changes in the fracture toughness caused by exposure of the RV beltline materials to neutron radiation. Appendix H to 10 CFR Part 50 requires that the surveillance program for the US-APWR RV meet ASTM E-185. The US-APWR surveillance capsule program includes six specimen capsules. The staff verified that the surveillance test materials will be prepared from samples taken from the materials used in fabricating the beltline of the RV. In addition, the staff verified that the base metal, weld metal, and HAZ materials included in the program will be those predicted to be most limiting in terms of setting pressure-temperature (P/T) limits for operation of the reactor to compensate for radiation effects during its lifetime. The staff finds that the materials selection, withdrawal, and testing requirements for the US-APWR design are in accordance with ASTM E 185-82. Compliance with the materials surveillance requirements of Appendix H to 10 CFR Part 50 and ASTM E-185 satisfies the requirements of GDC 32 for an appropriate surveillance program for the RV. Thus, the US-APWR design meets the requirements of GDC 32 regarding a RV beltline material neutron embrittlement surveillance program

Staff evaluation of closure studs is discussed in Section 3.13 of this SER.

US-APWR DCD Section 5.3.1.6.2, "Neutron Flux and Fluence Calculations," states that calculation methods for the neutron flux and fluence are described in Subsection 4.3.2.8. However, the methodology should also be submitted to the NRC for its review and approval**.** The staff requested in **RAI 284-2214, Question 05.03.01-01**, that the applicant discuss plans for submitting to the NRC the methodology used to calculate fluence. The applicant responded in a letter dated April 23, 2009, that the methodology to calculate the fluence would be provided as a technical report, scheduled to be submitted by July 31, 2009. Subsequently, the applicant submitted MHI Technical Report MUAP-09018, "Calculation Methodology for Reactor Vessel Neutron Flux and

Fluence," to the NRC for review. The staff is reviewing this technical report, but the review is not yet complete. The review results will be presented in Section 4.3 of this SER. The staff is tracking this item as **RAI 284-2214, Question 05.03.01-01**.

US-APWR DCD Table 5.3-1 "Chemical Composition Requirements for Reactor Vessel Materials" limits copper, nickel, phosphorous, and vanadium content for weld and base material being used in the beltline region of the reactor vessel. But RG 1.99 recommends that sulfur content also be controlled to low levels, and base material specifications such as ASME (ASTM) SA-508 (A508) and SA-533 (A533) have supplementary requirements that recommend that lower sulfur content limits be imposed on material used in the beltline region or when otherwise agreed upon between the manufacturer and the purchaser. The staff requested in **RAI 284-2214, Question 05.03.01-02**, that the applicant describe how controls on sulfur will be imposed on weld and base materials purchased for use in the beltline. The applicant responded that DCD Subsection 5.3.1.1 and Table 5.3-1 will be revised to specify limits on the sulfur content (0.005 percent for the beltline region forging and 0.01 percent for the beltline region as-welded weld material). The staff finds this response acceptable since these requirements meet (and exceed) the recommendations of RG 1.99 and ASME SA-508, Supplement S9. Subsequently, the staff confirmed that in Revision 2 of the DCD, the applicant revised Section 5.3.1.1 and Table 5.3-1 to specify the sulfur limits for the reactor vessel beltline materials as stated in its response to **RAI 284-2214, Question 05.03.01-02**. Accordingly, **RAI 284-2214, Question 05.03.01-02**, is resolved and closed.

5.3.1.5 Combined License Information Items

The following is a list of item numbers and descriptions from Table 1.8-2 of the DCD pertaining to Section 5.3.1:

Table 5.3.1-1
US-APWR Combined License Information Items

Item No.	Description	Section
5.3(2)	A COL applicant that references the US-APWR design certification will provide a reactor vessel material surveillance program.	5.3.1.6
5.3(3)	A COL applicant that references the US-APWR design certification will address the orientation and resulting lead factors for the plant-specific surveillance capsules.	5.3.1.6
5.3(5)	Pre-service and Inservice Inspection; The COL applicant provides the information for pre-service and inservice inspection described in Subsection 5.2.4.	5.2.4

5.3.1.6 Conclusion

With the exception of the resolution of **RAI 284-2214, Question 05.03.01-01**, which depends on the approval of MUAP-09018, the staff concludes that the US-APWR RV material specifications, RV manufacturing and fabrication processes, NDE methods of the RV and its appurtenances, fracture toughness testing, material surveillance, and RV fasteners are acceptable and meet the material testing and monitoring requirements of Section III of the ASME Code, Appendices G and H to 10 CFR Part 50, and 10 CFR 50.55a, which provide an acceptable basis for satisfying the requirements of GDC 1, 14, 30, 31, and 32.

5.3.2 Pressure-Temperature Limits, Pressurized Thermal Shock, and Charpy Upper-Shelf Energy Data and Analyses

5.3.2.1 Introduction

Neutron radiation embrittlement causes a reduction in the ductility of the reactor vessel (RV) materials, which is most severe in the beltline region due to its exposure to the highest average neutron flux at power. This reduction is measured in terms of the nil ductility reference temperature RT_{NDT}. The presence of elements such as copper, nickel and phosphorus is controlled to limit reductions in ductility and fracture toughness in the steel that forms the RV. Pressure-temperature (P-T) limits, derived using linear-elastic fracture mechanics principles provide margins of safety to prevent non-ductile fracture during normal operation, heat-up, cooldown, AOOs, and system hydrostatic, pre-service and inservice leakage tests.

Pressurized Thermal Shock (PTS) events are potential transients in a pressurized-water RV that can cause severe overcooling of the vessel wall, followed by immediate re-pressurization. The thermal stresses caused when the inside surface of the RV cools rapidly, combined with high-pressure stresses, will increase the potential for fracture if a flaw is present in a low-toughness material. The materials most susceptible to PTS are the materials in the RV beltline where neutron radiation gradually embrittles the material over time.

5.3.2.2 Summary of Application

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

DCD Tier 2: The applicant provided a Tier 2 description of how it addresses P-T limits, PTS, and Charpy upper-shelf energy in Section 5.3.2, summarized here in part as follows:

> RCS pressure and temperature limits for heatup, cooldown, low-temperature operation, criticality, and hydrostatic testing, as well as heatup and cooldown rates shall be established and documented per the TS provided in Chapter 16 Section 3.4.3, “RCS Pressure and Temperature (P-T) Limits” and Section 3.4.12, “Low Temperature Overpressure Protection (LTOP).” The analytical methods used to determine the RCS pressure and temperature are described in Chapter 5. Revised P-T limits will be provided to the NRC for each reactor vessel fluence period. Generic P-T Limits curves are provided. However, for a specific US-APWR plant, these limit curves are plotted based on actual material composition requirements and the COL applicant addresses the use of these plant-specific curves. Plant operating procedures are established by the COL applicant so that actual transients do not exceed the established P-T limits.

For beltline region materials, a screening criterion is established in accordance with 10 CFR 50.61 whereby the pressurized thermal shock reference temperature (RT_{PTS}) is not to exceed the screening criterion of 270 °F for forgings, plates, and axial welds, and 300 °F for circumferential welds. The RT_{PTS} values for the US-APWR beltline region materials are calculated using the RT_{NDT} requirements of the materials, the fluence values at the inner diameter of the shell, and the chemical composition requirements of the material. The application contains a table specifies the minimum Charpy upper-shelf energy values for the RV beltline materials.

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

Technical Specifications: The TS associated with Section 5.3.2 are given in DCD Chapter 16, Sections 3.4.3, B.3.4.3, 3.4.12, and B 3.4.12. In addition, TS 5.6.4 states that the RCS pressure and temperature limits for heatup, cooldown, low-temperature operation, criticality, and hydrostatic testing as well as heatup and cooldown rates shall be established and documented in the pressure-temperature limits report (PTLR).

5.3.2.3 Regulatory Basis

The relevant requirements of the Commission regulations for this area of review and the associated acceptance criteria are given in Section 5.3.2 of NUREG-0800 and are summarized below. Review interfaces with other Standard Review Plan (SRP) sections can be found in Section 5.3.2 of NUREG-0800.

1. 10 CFR 50.55a, as it relates to quality standards for the design, fabrication, erection, and testing of SSCs important to safety.

2. 10 CFR 50.60, as it relates to compliance with the requirements of Appendix G to 10 CFR Part 50.

3. 10 CFR 50.61, as it relates to fracture toughness criteria for PWRs relevant to PTS events.

4. General Design Criterion (GDC) 1, found in Appendix A to 10 CFR Part 50, as it relates to quality standards for design, fabrication, erection, and testing.

5. GDC 14, as it relates to ensuring an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture of the reactor coolant pressure boundary (RCPB).

6. GDC 31, as it relates to ensuring that the RCPB will behave in a non-brittle manner and that the probability of rapidly propagating fracture is minimized.

7. GDC 32, as it relates to the reactor vessel materials surveillance program.

8. Appendix G to 10 CFR Part 50, as it relates to material testing and fracture toughness.

Acceptance criteria adequate to meet the above requirements include:

1. Regulatory Guide (RG) 1.99 as it relates to RV beltline material properties.

2. RG 1.190 as it relates to the calculation of fluence estimates.

5.3.2.4 Technical Evaluation

5.3.2.4.1 Pressure-Temperature (P-T) Limits

Pressure-temperature (P-T) limits, derived using linear-elastic fracture mechanics principles provide margins of safety to prevent non-ductile fracture during normal operation, heat-up, cooldown, AOOs, system hydrostatic, preservice and inservice leakage tests. In **RAI 258-2334, Question 05.03.02-01**, the staff requested that the applicant confirm its decision to submit either P-T limits or a generic PTLR using bounding material properties and projected fluence. In its April 23, 2009, response, the applicant stated that it would submit a generic PTLR, following the guidelines of GL96-03, as part of the US-APWR design certification, and that all COLAs referencing the US-APWR DCD would address the use of the generic US-APWR PTLR that is approved by the NRC. Subsequently, in a letter dated February 1, 2010, the applicant submitted Technical Report MUAP-09016, Revision 1 (ADAMS Accession No. ML100470913), which contains bounding P-T limit curves based on the bounding material properties in the design certification, and a generic PTLR. The staff's evaluation of MUAP-09016, Revision 1, is not yet complete, and being tracked as part of Open Item 05.03.02-1. The staff will update this SER to reflect the final disposition of the report.

MUAP-09016 describes how the P-T limits for the US-APWR reactor vessel have been developed based on the evaluation of the beltline and closure flange regions. However, MUAP-09016 does not mention any consideration given to the vessel nozzles. In RAI 694-5355, Question **05.03.02-11**, the staff asked the applicant to clarify how the analysis performed to develop the P-T limits for the US-APWR design has considered the entire reactor vessel, including the beltline, closure flange, and nozzle regions. In its March 11, 2011, response to **RAI 694-5355, Question 05.03.02-11**, the applicant stated that the fracture mechanics evaluation on the US-APWR reactor vessel is subsequently performed using the stress analysis results (Technical Report MUAP-09005). The applicant stated that the result, which is to be available in June, 2011, will show that the evaluation based on the beltline and closure flange regions is conservative. This is **Open Item 05.03.02-02**.

5.3.2.4.2 Pressurized Thermal Shock (PTS)

PTS events are potential transients in a pressurized-water RV that can cause severe overcooling of the vessel wall, followed by immediate re-pressurization. The thermal stresses caused when the inside surface of the RV cools rapidly, combined with the high-pressure stresses, will increase the potential for fracture if a flaw is present in a low-toughness material. The materials most susceptible to PTS are the materials in the RV beltline where neutron radiation gradually embrittles the material over time.

The PTS rule established screening criteria to serve as a limiting level of RV material embrittlement beyond which operation cannot continue without further plant-specific evaluation. The screening criteria are given in terms of reference temperature, RT_{PTS}. The screening criteria are 132.2 °C (270 °F) for plates, forgings, and axial welds, and 148.9 °C (300 °F) for circumferential welds. The RT_{PTS} is defined by the following equation:

$RT_{PTS} = RT_{NDT\,(U)} + \Delta RT_{PTS} + M$, where:

$RT_{NDT\,(U)}$	= initial reference temperature
ΔRT_{PTS}	= mean value in the adjustment in reference temperature caused by irradiation
M	= margin to be added to cover uncertainties in the initial reference temperature, copper and nickel contents, fluence and calculational procedures

The US-APWR beltline forging material will contain a maximum of 0.05 weight-percent copper and 1.00 weight-percent nickel and the weld metal will contain a maximum of 0.08 weight-percent copper and 0.95 weight-percent nickel. The initial RT_{NDT} is < 0 °F for the reactor vessel beltline forging material and <-20 ^{o}F for the welds. DCD Table 5.3-4 provides projected RT_{PTS} values at 60 effective full-power years (EFPY) of 24.8 ^{o}C (76.7 ^{o}F) for beltline region forgings and 64.7 ^{o}C (148.6 ^{o}F) for beltline region welds. Thus, the staff finds that the US-APWR design has met the PTS screening criteria of 10 CFR 50.61. In addition, DCD Section 5.3.2.3 states that RT_{PTS} values will be calculated based on plant-specific material property. This is COL information Item 5.3(4).

5.3.2.4.3 Upper-Shelf Energy (USE)

DCD Section 5.3.2.4 states that the change in the USE due to radiation embrittlement for the US-APWR reactor vessel is predicted in accordance with the requirements of RG 1.99 (Ref. 5.3-16). The tests for fracture toughness of RV materials specified in the DCD are in accordance with ASME Code, Section III, Paragraph NB-2300 and 10 CFR Part 50, Appendix G. The initial and EOL USE for the US-APWR RV base material and weld are shown in DCD Tier 2 Table 5.3-4. DCD Table 5.3-4 indicates that the initial USE levels for the US-APWR RV materials are greater than 75 ft-lbs and the projected EOL USE values are greater than 50 ft-lbs. Therefore, the staff finds that the applicant's USE values meet the requirements of Appendix G to 10 CFR Part 50 and are, thus, acceptable.

5.3.2.5 Combined License Information Items

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

Table 5.3.2-1
US-APWR Combined License Information Items

Item No.	Description	Section
5.3(1)	*Pressure-Temperature Limit Curves; The COL applicant addresses the use of plant-specific reactor vessel P-T limit curves. Generic P-T limit curves for the US-APWR reactor vessel are shown in Figures 5.3-2 and 5.3-3, which are based on the conditions described in Subsection 5.3.2. However, for a specific US-APWR plant, these limit curves are plotted based on actual material composition requirements and the COL applicant addresses the use of these plant-specific curves.*	5.3.2
5.3(4)	*Reactor Vessel Material Properties verification; The COL applicant verifies the USE and RT_{NDT} at EOL, including a PTS evaluation based on actual material property requirements of the reactor vessel material and the projected neutron fluence for the design-life objective of 60 years.*	*5.3.2*

5.3.2.6 Conclusions

Pressure-Temperature Limits

With the exception of **OI-05.03.02-01 and OI-05.03.02-02**, the staff finds that MUAP-09016, the US-APWR PTLR, conforms to the staff's technical criteria for pressure-temperature limit reports (PTLRs) as defined in GL 96-03. In addition, the staff concludes that the applicant has met the fracture toughness criteria of Appendix G to 10 CFR Part 50. The change in fracture toughness properties of the RV beltline materials during operation will be determined through a material surveillance program developed in conformance with Appendix H to 10 CFR Part 50. The use of operating limits as determined by the criteria defined in Section 5.3.2 of the SRP provides reasonable assurance that non-ductile or rapidly propagating failure will not occur. This constitutes an acceptable basis for satisfying the requirements of 10 CFR 50.55a, Appendix A to 10 CFR Part 50, and GDC 1, 14, 31, and 32.

Pressurized Thermal Shock

The staff concludes that the US-APWR RV meets the relevant requirements of 10 CFR 50.61 because calculations show that the RV beltline materials will be substantially below the PTS screening criteria after 60 years of operation. The COL applicant will address RT_{PTS} values based on plant-specific material properties and projected neutron fluence for the plant design objective of 60 years.

Upper Shelf Energy (USE)

The staff evaluated the initial Charpy USE values for the proposed reactor vessel materials in accordance with the acceptance criterion specified in paragraph IV.A.1.a of Appendix G to 10 CFR Part 50. The staff also evaluated the EOL projected Charpy USE values for the reactor vessel materials in accordance with the acceptance criterion specified in paragraph IV.A.1.a of Appendix G to 10 CFR Part 50. The staff concludes that the US-APWR RV materials meet the relevant requirements of 10 CFR Part 50, Appendix G.

5.3.3 Reactor Vessel Integrity

5.3.3.1 Introduction

While most of the features and topics addressed in this section are being reviewed separately in other sections of this FSER, the integrity of the reactor vessel is of such importance that a special summary review of all factors relating to the integrity of the reactor is warranted. The information in each area of the application is reviewed for completeness and consistency with requirements to ensure reactor vessel integrity.

5.3.3.2 Summary of Application

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

DCD Tier 2: The applicant has provided a Tier 2 description of the reactor vessel integrity in Section 5.3.3, summarized here in part as follows:

> The RV is a pressure boundary component whose function is to support and enclose the reactor core internals. With the reactor core internals, the RV guides the flow of reactor coolant, and also maintains a volume of coolant around the core. The RV is a vertical cylindrical pressure vessel consisting of a vessel flange and upper shell, lower shell, transition ring, hemispherical bottom head, and removable upper closure head. It is fabricated by welding the vessel flange and upper shell, lower shell, transition ring, and bottom head. The inlet, outlet, and direct vessel injection (DVI) nozzles are welded to the upper shell. The closure head consists of a bolting flange and hemispherical top head dome. It is secured to the RV flange by fifty eight (58) stud bolts and nuts, using stud tensioners. The RV closure head flange and RV flange is sealed by two metallic O-rings, which are designed to prevent leakage through the inner or outer O-ring at any operating condition. The top head dome has penetrations for the control rod drive mechanisms, in-core instrumentation systems, and head vent. Sixty nine (69) control rod drive mechanism nozzles are inserted through the penetration holes in the RV closure head and are welded by J-groove welds. The instrumentation systems in the reactor are inserted though the in-core instrumentation system nozzles located on the closure head and also welded by J-groove welds.
>
> Visual inspections of the closure head are carried out during each refueling outage. Selective inspections of the internal cladding, closure head nozzles, and the gasket seating surface can be done with the use of optical devices. The closure head stud

bolts, nuts, and washers can be inspected periodically using visual, magnetic particle and/or ultrasonic examinations. The upper shell cladding is accessible for inspection during refueling in certain areas above the inlet and outlet nozzles. If necessary, the lower core internals can be removed so that the entire vessel inner surfaces are accessible. In addition, full-penetration welds in many areas of the installed RV are accessible for NDE.

The application describes that the reactor vessel forms a part of the reactor coolant pressure boundary (RCPB) system and is designed, fabricated, tested, and installed in accordance with the requirements of 10 CFR 50.55a, 10 CFR 50.61, GDC 1, GDC 4, GDC 14, GDC 30, GDC 31, and GDC 32. Further, the requirements of ASME Code, Section III are satisfied through the reactor vessel's design and fabrication.

The application states the safety design bases of the reactor vessel as housing the heat generating reactor core, fission product barrier, providing proper reactor internals alignment and support, directing flow through the core, allowing access for control rod drive mechanisms and in-core instrumentation, supporting refueling activities, and an interface with attached systems.

The application further describes that for maintenance and refueling activities, the closure head with integral lifting lugs, consists of a bolting flange and hemispherical top head dome, which also has the penetrations for the control rod drive mechanisms and in-core instrumentation. The closure head also provides for venting when required. The application provides critical dimensions for the reactor, including vessel length, diameter, weight, thicknesses at various locations, and volume. There are no penetrations located below the top of the reactor core, minimizing the potential for a loss of coolant accident from uncovering the core. The reactor vessel design pressure and temperature is 2,485 pounds per square inch gauge (psig) and 650 °F, respectively. The reactor vessel is clad through various described processes in order to protect the base metal against the effects of a boric acid attack.

The DCD provides further detail on provisions for performing NDE, methods of shipment, precautions against foreign material intrusion, and installation procedures. Tables are provided that identify chemical composition requirements for the reactor vessel materials, inspection plan for reactor vessel materials, inspection plan for reactor vessel welds, end-of life, the nil ductility reference temperature (RT_{NDT}) and upper shelf energy (USE) for beltline materials, and reactor vessel design data. Figures are provided depicting orientation of surveillance capsules, representative pressure-temperature (P-T) limit curves, and various reactor vessel outline views.

ITAAC: There are no ITAAC entries specific to this area of review.

Technical Specifications: There are no Technical Specifications specific to this area of review.

5.3.3.3 Regulatory Basis

The relevant requirements of the Commission's regulations for this area of review, and the associated acceptance criteria, are given in Section 5.3.3 of NUREG-0800 and are summarized below. Review interfaces with other SRP sections can be found in Section 5.3.3 of NUREG-0800.

1. General Design Criteria (GDC) 1 and GDC 30 found in Appendix A to Part 50, as they relate to quality standards for design, fabrication, erection, and testing of structures, systems and components.

2. GDC 4, as it relates to the compatibility of components with environmental conditions.

3. GDC 14, as it relates to prevention of rapidly propagating fractures of the reactor coolant pressure boundary (RCPB).

4. GDC 31, as it relates to material fracture toughness.

5. GDC 32, as it relates to the requirements for a materials surveillance program.

6. 10 CFR 50.55a, as it relates to quality standards for design, and determination and monitoring of fracture toughness.

7. 10 CFR 50.60, "Acceptance criteria for fracture prevention measures for light water nuclear power reactors for normal operation," as it relates to RCPB fracture toughness and material surveillance requirements of 10 CFR Part 50, Appendix G, and Appendix H.

8. 10 CFR Part 50, Appendix B, Criterion XIII, as it relates to onsite material cleaning control.

9. 10 CFR Part 50, Appendix G, as it relates to materials testing and acceptance criteria for fracture toughness.

10. 10 CFR Part 50, Appendix H, as it relates to the determination and monitoring of fracture toughness.

Acceptance criteria adequate to meet the above requirements include:

1. RG 1.31, "Control of Ferrite Content in Stainless Steel Weld Metal," as it relates to the control of welding in fabricating and joining safety-related austenitic stainless steel components and systems.

2. RG 1.34, "Control of Electroslag Weld Properties," as it relates to acceptable solidification patterns and impact test limits and the criteria for verifying conformance during production welding.

3. RG 1.43, "Control of Stainless Steel Weld Cladding of Low-Alloy Steel Components," as it relates to criteria to limit the occurrence of under-clad cracking in low-alloy steel safety-related components clad with stainless steel.

4. RG 1.44, "Control of the Use of Sensitized Stainless Steel," as it relates to the compatibility of RCPB materials with the reactor coolant and the avoidance of stress corrosion cracking.

5. RG 1.71, "Welder Qualification for Areas of Limited Accessibility," as it relates to welder requalification.

6. RG 1.99, "Radiation Embrittlement of Reactor Vessel Materials," as it relates to RPV fracture toughness.

7. RG 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," as it relates to the RPV material surveillance program.

5.3.3.4 Technical Evaluation

Although the staff reviewed most areas separately in accordance with the other SRP sections, the integrity of the RV is of such importance that a special summary review of all factors relating to RV integrity was warranted. The staff reviewed the fracture toughness of the ferritic materials for the RV, the P-T limits for the operation of the RV, and the materials surveillance program for the RV beltline. SRP Section 5.3.3 provides the acceptance criteria and references that form the bases for this evaluation.

The staff reviewed the information in each area to ensure that inconsistencies that would reduce the certainty of RV integrity did not exist. The following is a list of the areas reviewed and the sections of this report in which they are discussed:

- RCPB materials (Section 5.2.3)
- RCPB inservice inspection (ISI) and testing (Section 5.2.4)
- RV materials (Section 5.3.1)
- P-T limits, PTS and USE (Section 5.3.2)

The staff has reviewed the integrity of the US-APWR RV in the following areas: design, materials of construction, fabrication methods, inspection requirements, shipment and installation, operating conditions, inservice surveillance and operational programs. The RV will be designed and fabricated to the high standards of quality required by Section III of the ASME Code and the pertinent Code Cases, and thus, satisfies the quality standards requirements of GDC 1, GDC 30, and 10 CFR 50.55a.

Annealing of the RV provides assurance that fracture toughness properties can be restored to satisfy the fracture toughness requirements of GDC 31. In US-APWR DCD, Revision 1, Section 5.3.2.1.7, the applicant has stated that annealing of the RV at site for radiation embrittlement is not considered necessary as the RV is predicted to maintain an equivalent safety margin in accordance with the procedures of 10 CFR Part 50, Appendix G. This is acceptable because it meets the requirements of Section IV.2.B of Appendix G to 10 CFR Part 50.

The special considerations relating to fracture toughness and radiation effects limit the basic Code-approved materials that are currently acceptable for most parts of RVs to SA 533 Grade B Class 1, SA 508 Class 2, and SA 508 Class 3. The US-APWR design utilizes SA 508 Class 3 for the RV, and, thus, the materials of construction are acceptable to the staff. The acceptability of fabrication methods and inspection requirements are discussed in Section 5.3.1 of this SER.

The US-APWR RV is shipped horizontally with saddle-type shipping skids attached. All RV openings are sealed to prevent moisture from entering and desiccant is placed inside the RV to keep the inside dry. The RV closure head assembly is shipped with saddle-type shipping skids and a shipping cover attached. All closure head and shipping cover openings are sealed to prevent moisture from entering and desiccant is placed inside the closure head. Desiccant is usually placed in a wire mesh basket attached to the shipping cover. Prior to shipment, the interior of the closure head assembly is normally purged with inert gas. After arrival at the site and prior to installation, the RV and closure head assembly is examined for cleanliness and damage. During installation, RV integrity is maintained by different measures; for example, by applying access control to personnel entering the RV, weather protection using temporary tents, and periodic cleaning.

The staff found that the methods described to protect the integrity of the RV during shipping, arrival onsite, and installation are acceptable because they ensure that the as-built characteristics of the reactor vessel are not degraded by improper handling. However, the applicant did not describe the measures taken to ensure the proper cleanliness and freedom from contamination of the RV prior to shipment. In **RAI 225-2029**, **Question 05.03.03-01**, the staff requested that the applicant describe the methods used to ensure the proper cleanliness and freedom from contamination of the RV prior to shipment. In its April 17, 2009, response, the applicant stated that measures taken to ensure proper cleanliness and freedom from contamination of the RV prior to shipment are discussed in DCD Subsection 5.3.1.4.3. The applicant proposed to revise US-APWR DCD, Revision 1, Section 5.3.3.5, to so state, i.e., that prior to shipment, the RV is cleaned and protected from contamination as described in Subsection 5.3.1.4.3, and that the outer surfaces of the RV are also normally protected with temporary coverings prior to shipment and that further details concerning the cleanliness and protection against contamination for austenitic stainless steel parts of the RV are discussed in Subsection 5.2.3.4. The staff finds that the applicant has appropriately addressed the RAI and has verified that the proposed revisions are included in Section 5.3.3.5 of the US-APWR DCD. Therefore **RAI-225-2029, Question 05.03.03-01**, is resolved and closed.

The RV will operate under conditions, procedures, and protective devices that ensure that the design conditions will not be exceeded during normal reactor operation, maintenance, testing, and anticipated transients. Section 5.3.2 of this SER discusses compliance with the acceptance criteria in Appendix G to 10 CFR Part 50 and 10 CFR 50.61 for PTS.

The applicant further stated that the RV will be subjected to periodic inspection to demonstrate that its high initial quality has not deteriorated significantly under service conditions. The internal and external surfaces of the US-APWR RV are designed to be accessible for periodic inspection using visual and NDE techniques. Various design considerations permit these inspections. For example, the reactor internals are completely removable and the tools and storage space required to enable these inspections have been provided. The bare top head dome and closure head nozzles beyond the peripheral nozzles can be accessed by removing the top head removable insulation panels. Access is also provided to RV nozzle safe ends. The insulation covering field welds between the RV nozzle safe ends and the main coolant piping can be removed. Full penetration welds of the closure head, vessel shell, vessel nozzles, nozzle safe ends, transition ring, and bottom head are all accessible for NDE from either the inside clad, outside surface, or both. Reactor cavity and thermal insulation allows access to the outside surface of the RV. Thus the staff finds that the US-APWR design appropriately enables access for inspections of the RV. The acceptability of RCPB ISI and Testing is discussed in Section 5.2.4 of this SER.

The RV will be subjected to surveillance to monitor for neutron irradiation damage so that the operating limitations may be adjusted. The Reactor Vessel Material Surveillance Program requirements have been addressed in Section 5.3.1 of this SER.

5.3.3.5 Combined License Information Items

No additional information is required to be provided by a COL applicant in connection with Section 5.3.3 of the US-APWR DCD, Revision 1.

5.3.3.6 Conclusions

The staff concludes that the structural integrity of the US-APWR RV meets the requirements of GDC 1, 4, 14, 30, 31, and 32 of Appendix A to 10 CFR Part 50; Appendices G and H to 10 CFR Part 50; and 10 CFR 50.55a. Therefore, the staff finds the structural integrity of the US-APWR RV to be acceptable. The basis for this conclusion is that the design, materials, fabrication, inspection, and quality assurance requirements of the US-APWR plants conforms to the applicable NRC regulations and RGs discussed above, as well as to the rules of Section III of the ASME Code. The US-APWR design meets the fracture toughness requirements of the regulations and Section III of the ASME Code, including requirements for surveillance of RV material properties throughout its service life, in accordance with Appendix H to 10 CFR Part 50. In addition, operating limitations on temperature and pressure will be established for the plant in accordance with Appendix G, “Protection Against Non-ductile Failure,” of the ASME Code, Section III, and Appendix G to 10 CFR Part 50.

5.4 RCS Component and Subsystem Design

The review of reactor thermal-hydraulic systems includes the review of the various components and subsystems associated with the reactor coolant system (RCS). RCS design bases, descriptions, evaluations, and necessary tests and inspections for these components and subsystems (including radiological considerations from the viewpoint of how radiation affects operation, and the viewpoint of how radiation levels affect the operators and their capabilities of operation and maintenance) are to be evaluated for the following subsystems and components: reactor coolant pumps, steam generators, RCS piping and valves, main steamline flow restriction, RHR system, pressurizer, pressurizer relief tank, and RCS high-point vents. The applicant has provided a DCD Tier 2 description in Section 5.4, which provides information regarding the performance requirements and design features of these subsystems and components. Information about the RCS support is provided in Section 3.8.3.1. Information about the pressurizer safety valve is provided in Section 5.2.2, and information about the safety depressurization valve (SDV) and depressurization valve (DV) is given in Section 5.4.12.

The RCS component and subsystem design discusses the design bases, fabrication and inspection, and various operational conditions for the reactor coolant pumps in Section 5.4.1, steam generators in Section 5.4.2, reactor coolant piping in Section 5.4.3, main steam line flow restrictor in Section 5.4.4, RHR system in Section 5.4.7, pressurizer in Section 5.4.10, pressurizer relief tank in Section 5.4.11, and RCS high-point vents in Section 5.4.12. The application also identifies that DCD Sections 5.4.5 and 5.4.9 are reserved by the NRC according to RG 1.206, and DCD Sections 5.4.6 and 5.4.8 are not applicable to the US-APWR. Note that RG 1.206 also reserves Section 5.4.10, but does not have a section for the pressurizer. NUREG-0800 does not have a section 5.4.10 and has no other section for the pressurizer. Therefore, the applicant used Section 5.4.10 of the US-APWR DCD to provide the information on the pressurizer.

5.4.1 Reactor Coolant Pumps

5.4.1.1 Reactor Coolant Pump Flywheel Integrity

5.4.1.1.1 Introduction

The RCPs provide forced flow circulation of the reactor coolant to transfer heat from the reactor core to the SGs. The RCPs circulate the water through the reactor vessel and SGs at a sufficient rate to ensure proper heat transfer and prevent fuel damage. The RCPs form part of the RCPB during all modes of operation, thereby retaining the circulated reactor coolant and entrained radioactive substances.

RCP flywheels have large masses and rotate at 1200 revolutions per minute (rpm) during normal reactor operation. A loss of flywheel integrity could result in high-energy missiles and excessive vibration of the RCP assembly. The safety consequences could be significant because of possible damage to the reactor coolant system, the containment, or the engineered safety features. RCP flywheel failure can also result in reduction or loss of forced coolant flow.

The staff review of the US-APWR DCD related to this section ensures that the potential for failures of the flywheels of reactor coolant pumps are minimized and the materials are adequate to assure a quality product commensurate with the importance to safety.

5.4.1.1.2 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 5.4.4.1, summarized here in part, as follows:

> RCP flywheels are designed, manufactured, and inspected to minimize the possibility of generating high-energy fragments (missiles) under any anticipated operating or accident condition, consistent with the intent of the guidelines set forth in Section 5.4.1.1 of NUREG-0800 and RG 1.14. Centrifugal forces at operating speed are calculated to determine developed stresses. By convention, flywheels are tested at 125 percent of the maximum synchronous speed of the motor. The flywheel has been analyzed for a variety of speed-related failure modes and is designed to operate at less than one-half of the lowest of the critical speeds.
>
> The application describes the flywheel as consisting of two thick plates bolted together, fabricated from SA-533, Grade B. Class 1 steel. The flywheel material is produced by a process that minimizes flaws in the material and improves its fracture toughness properties. Nondestructive examination (NDE) is performed to meet the requirements of Section III of the ASME Code. RG 1.14 recommendations are used in an inspection program for the flywheels and references Section XI of the ASME Code.

ITAAC: US-APWR DCD, Revision 1, Tier 1, Section 2.4, Table 2.4.2-5 provides an ITAAC commitment No. 10.b, which requires that the as-built reactor coolant pump flywheel assembly be tested at 125 percent of operating speed.

Technical Specifications: The Technical Specifications associated with DCD Tier 2, Section 5.4.1.1 are given in DCD Tier 2, Chapter 16, Section 5.5.7, “RCP Flywheel Inspection Program.” This program provides for the inspection of each RCP flywheel in accordance with the recommendations of Regulatory Position C.4.b of RG 1.14, Revision 1, August 1975. In lieu of Position C.4.b(1) and Position C.4.b(2), a qualified in-place ultrasonic test (UT) examination over the volume from the inner bore of the flywheel to the circle one-half of the outer radius or a surface examination (magnetic particle test [MT] and/or liquid penetrant test [PT]) of exposed surfaces of the removed flywheels may be conducted at 20 year intervals.

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

Technical Reports: MHI Report MUAP-07035, Revision 0 and MUAP-09017, Revision 0

5.4.1.1.3 Regulatory Basis

The relevant requirements of the Commission's regulations for this area of review, and the associated acceptance criteria, are given in Section 5.4 and Section 5.4.1.1 of NUREG-0800 and are summarized below. Review interfaces with other SRP sections can be found in Section 5.4 and Section 5.4.1.1 of NUREG-0800.

1. GDC 1 and 10 CFR 50.55a(a)(1), as they relate to pump flywheel design, materials selection, fracture toughness, preservice and inservice inspection programs, and overspeed test procedures to determine their adequacy to assure a quality product commensurate with the importance of the safety function to be performed.

2. GDC 4, as it relates to protecting safety-related SSCs of nuclear power plants from the effects of missiles that might result from RCP flywheel failure.

Acceptance criteria adequate to meet the above requirements include:

1. RG 1.14, "Reactor Coolant Pump Flywheel Integrity," as it relates to the RCP flywheel design, materials selection and fabrication, preservice inspection program, inservice inspection program, and overspeed test of each pump flywheel assembly.

5.4.1.1.4 Technical Evaluation

The staff reviewed US-APWR DCD, Revision 1, Tier 2, Section 5.4.1.1, describing the materials used in the fabrication of the reactor coolant pump flywheel, so that the structural integrity of the reactor coolant pump flywheel is maintained in the event of design overspeed transients or postulated accidents. The staff reviewed this information using the guidelines in Standard Review Plan (SRP) Section 5.4.1.1, "Pump Flywheel Integrity (PWR)." The following evaluation addresses the acceptance criteria outlined in SRP Section 5.4.1.1.

Material Selection, Fabrication and Fracture Toughness

US-APWR DCD, Revision 1, Tier 2, Section 5.4.1.1.2, states that the reactor coolant pump flywheel uses plate material in accordance with ASME Code, SA-533, Grade B, Class 1 steel. The material is manufactured using an electric arc furnace with vacuum degassing to minimize flaws in the material and to improve its fracture toughness. In addition, the US-APWR DCD incorporates the cut surfaces guidance of RG 1.14 in that it specifies that all cut surfaces are to be removed by machining to a depth of 1/2 inch minimum below the cut surface to minimize any loss of fracture toughness during fabrication. The staff finds the design material and manufacturing methods acceptable because the material is produced by vacuum melting and degassing, which is an acceptable method of producing material of improved purity and toughness. Also, the staff finds that the processing and fabrication of the flywheel material will provide a suitable material that will maintain its toughness to resist brittle fracture, and thereby meets the guidelines in RG 1.14.

However, DCD Section 5.4.1.1.2 does not discuss the guidance in RG 1.14 on the prohibition of welding, including tack welding and repair welding, unless the welds are inspectable and considered as potential sources of flaws in the fracture analysis.

Therefore, in **RAI 274-2126, Question 05.04.01.01-01**, the staff told the applicant that DCD Section 5.4.1.1.2 should discuss how the guidance in RG 1.14 on prohibiting welding will be addressed for the US-APWR flywheel design. In its response to **RAI 274-2126, Question 05.04.01.01-01**, dated April 28, 2009, the applicant proposed to include a statement in DCD Section 5.4.1.1.2, Revision 1, to prohibit welding on flywheels, including tack welding and repair welding. The staff finds that this welding prohibition will preserve the material properties, especially toughness, and is in accordance with the guidance in RG 1.14. Subsequently, the staff confirmed that the prohibition of welding on the flywheel was incorporated in Revision 2 of the DCD. Therefore, **RAI 274-2126, Question 05.04.01.01-01**, is resolved and closed.

RG 1.14, "Reactor Coolant Pump Flywheel Integrity," states that past evaluations have shown that ASME Code, SA-533, Grade B, Class 1, and SA-508, Classes 2 and 3, materials generally have suitable toughness for typical flywheel applications, provided stress concentrations are kept within reasonable limits and the nil-ductility reference temperature (RT_{NDT}), determined in accordance with Article NB-2331(a) of Section III to the ASME Code, is at least 50 °C (90 °F) below the lowest temperature at which operating speed is achieved. In **RAI 274-2126, Question 05.04.01.01-02**, the staff requested the applicant to discuss how the RT_{NDT} value of the material to be procured for the RCP flywheels will meet the RG 1.14 guidance. In its response to **RAI 274-2126, Question 05.04.01.01-02**, dated April 28, 2009, the applicant stated that the RT_{NDT} value of the material procured for the flywheel will be determined using the results of the Charpy impact test at different temperatures as stated in DCD Subsection 5.4.1.1.3, "Material Acceptance Criteria." The staff notes that these temperatures are based on the requirements in ASME Code, Section III, Paragraph NB-2331, as outlined in RG1.14. This method of determining fracture toughness is consistent with the indirect method specified in SRP Section 5.4.1.1, Paragraph II.2 using ASME Code, Section III, Paragraph NB-2330. In addition, MHI Report MUAP-07035, Revision 0, used the RT_{NDT} value of 10ºF in the critical speed analyses as discussed below, which demonstrates that the flywheel has sufficient toughness. The staff also notes that the minimum operating temperature is at least 50 °C (90 °F) above the RT_{NDT} value, based on MHI Report MUAP-09017-P, Revision 0, which is referenced in Section 5.4.1.1.2 of Revision 2 of the DCD. Therefore, the staff finds that the material will have sufficient toughness for this flywheel application. Accordingly, **RAI 274-2126, Question 05.04.01.01-02**, is resolved and closed.

Pre-service Inspection

US-APWR DCD, Revision 1, Tier 2, Section 5.4.1.1.2 states that finished flywheels are inspected, which includes a 100 percent ultrasonic inspection in accordance with ASME Code, Section III, and a surface inspection using liquid penetrant or magnetic particle examination of finished machined bore, keyways and drilled holes. These areas are considered to have high stress concentrations. In addition, US-APWR DCD, Revision 1, Tier 2, Section 5.4.1.1.1 states that the flywheel is tested at 125 percent of operating speed.

RG 1.14, Section C.4.a, states that following the spin test, each finished flywheel receives a check of critical dimensions, and a non-destructive examination. The non-destructive examination includes surface examination of areas of high stress concentrations using procedures in accordance with NB-2540, and acceptance criteria in

NB-2545 or NB-2546 of Section III to the ASME Code, and a 100-percent volumetric examination using procedures and acceptance criteria specified in accordance with NB-2530 or NB-2540 of Section III to the ASME Code.

However, DCD Sections 5.4.1.1.1 and 5.4.1.1.2 do not address this RG 1.14 guidance. Therefore, in **RAI 274-2126, Question 05.04.01.01-03**, the staff requested the applicant to revise DCD Sections 5.4.1.1.1 and 5.4.1.1.2 to address this RG 1.14NDE guidance

In its response to **RAI 274-2126, Question 05.04.01.01-03**, dated April 28, 2009, the applicant stated that DCD Subsection 5.4.1.1.2 covers PSI; so it would be revised to reflect the NRC comments, addressing RG 1.14 guidance. The response included a markup of the proposed revision, specifically stating that the surface and volumetric examinations will be performed after the overspeed test so that any flaws that have initiated or grown during the overspeed test can be detected. In addition, the flywheel will also be inspected for critical dimensions after the overspeed test to detect any dimensional changes. However, the applicant's response did not provide the acceptance criteria to be used (i.e., Section III of the ASME Code, NB-2545, NB-2546, NB-2530 and NB-2540). In a letter dated February 25, 2011, the applicant provided the ASME acceptance criteria to be used (NB-2545 or NB-2546 for liquid penetrant or magnetic particle testing and NB-2530 for ultrasonic testing) and that the inspections will be performed after the spin test, which the staff finds acceptable since it meets the guidelines in RG 1.14, and was incorporated in Revision 2 of the US-APWR DCD. However, the staff notes that the DCD includes the statement, "Qualified test procedure and the acceptance criteria should be decided with respect to this test procedure". Since this statement provides no meaningful information, the staff requests that this statement be removed or clarify the intent of this statement. In a letter dated April 14, 2011, the applicant amended their response by proposing to delete the above statement in the DCD. The staff finds this deletion acceptable since it provided no meaningful information. Therefore, **RAI 274-2126, Question 05.04.01.01-03**, is resolved and closed. The staff will confirm that the proposed revision to Section 5.4.1.1.2 is incorporated in the next revision of the DCD. This will be tracked as **Confirmatory Item 05.04.01.01-01.**

Since the flywheel will be inspected in accordance with the ASME Code, and meets the guidelines of RG 1.14 as detailed in SRP Section 5.4.1.1, paragraph II.3, the staff finds the preservice inspection provides an acceptable initial flywheel condition. The initial flywheel condition, along with the flywheel analysis, provides a baseline for future inservice inspections to ensure that no flaws will propagate resulting in the fracture of the flywheel and generation of potential missiles.

Flywheel Design

The flywheel is required to be designed to withstand normal conditions, anticipated transients, the design basis loss of coolant accident, and the safe shutdown earthquake without loss of structural integrity, and the potential of generating a missile. DCD, Revision 1, Section 5.4.1.1.1, states that the flywheel is designed to minimize the possibility of generating high-energy missiles under any anticipated operating or accident condition consistent with the guidelines of RG 1.14 and SRP Section 5.4.1.1. In addition, MHI Report MUAP-07035, Revision 0, verifies that the normal speed of the flywheel is less than one-half of the lowest critical speeds for failure modes of ductile facture, non-ductile fracture and excessive deformation. This report also confirms that the lowest critical speed is greater than the predicted loss-of-coolant-accident (LOCA) overspeed. The staff reviewed the evaluation in MUAP-07035, Revision 0, to the regulatory position of RG 1.14 for the flywheel design based on these critical speeds. In addition, Section 2.0 of MUAP-0735, Revision 0, provides the physical design and dimensions of the flywheel. In its response to **RAI No. 274-2126, Question 05.04.01.01-04**, dated April 28, 2009, the applicant clarified that the flywheel assembly is composed of two steel discs that are bolted together. Therefore, **RAI 274-2126, Question 05.04.01.01.04**, is resolved and closed.

For the ductile fracture analysis, MUAP-07035, Revision 0, uses the elastic stress analysis method of the ASME Code, Section III, to predict the critical speed based on the ductile fracture of the flywheel. The ASME Code states that the stress limits for the general primary membrane stress intensity, $P_{m,}$ should be equal to 0.7 of the minimum specified ultimate tensile strength of the flywheel material (S_u) and the primary membrane plus primary bending stress intensity (P_m + P_b) should be equal to 1.05 S_u. The staff verified that the minimum calculated limiting speed (3486 rpm) assuming a 0.50-inch crack is at least twice the normal operating speed (1200 rpm). The staff finds that the critical speed for ductile fracture meets the criterion in RG 1.14. In addition, the staff confirmed that the critical speed for ductile fracture (3486 rpm) is greater than the LOCA overspeed, which is less than 1500 rpm, assuming a leak-before-break scenario. Therefore, the staff finds the ductile fracture analysis in MUAP-07035, Revision 0, acceptable.

For the non-ductile fracture analysis, the report uses the linear elastic stress analysis method of the ASME Code, Section III, to predict the critical speed for non-ductile fracture of the flywheel. The analysis in the report uses the closed form solution for a radial full-depth crack emanating from the bore of a rotating disc to calculate the applied stress intensity factor (K). The fracture toughness of the flywheel material was determined to be 102 ksi √in by using the lower bound stress intensity factor for critical static crack initiation (K_{IC}) curve of Appendix A to Section XI of the ASME Code, while using a material RT_{NDT} value of 10 °F. Assuming a crack depth of 0.50 inch, the analysis showed that the lowest calculated critical speed is 2489 rpm. Half of this speed, approximately 1244 rpm is higher than the operating speed of 1200 rpm. Thus, the lowest calculated critical speed meets the pertinent criterion of RG 1.14 and is therefore acceptable.

MUAP-07035, Revision 0, also provided a fatigue crack growth that was determined from the crack growth rate in Appendix A of Section XI to the ASME Code. An initial crack length of 0.50 inch was assumed, with an assumed loading cycle of 3000 starts and stops for a 60-year plant life. A crack growth of 0.039 inch was calculated. Since the fatigue crack growth of 0.039 inch for a 60-year period is minimal, fatigue is not a major contributor for crack growth.

However, the critical crack size is needed to determine if the fatigue crack growth of a missed flaw for the inspection interval will still meet the critical crack size criterion of the ASME Code. Therefore, in **RAI 274-2126, Question 05.04.01.01-5**, the staff requested that the applicant provide the allowable crack length for the flywheel using the criteria of Section XI of the ASME Code, and discuss what crack length and depth the current ultrasonic examination techniques can detect to determine if the fatigue crack growth of a missed flaw for the inspection interval will still meet the critical crack size criterion of the ASME Code. In response to this RAI question, the applicant stated in a letter dated April 28, 2009, that in lieu of the deterministic fracture mechanics analysis in MUAP-07035, Revision 0, a probabilistic fracture mechanics (PFM) analysis, was being performed to justify an inservice inspection interval of 20 years for the flywheel. The PFM analysis was to provide the initial flaw size distribution and flaw acceptance criteria, and was to have been provided in a technical report by June 30, 2009.

MHI Technical Report MUAP-09017-P, Revision 0 was received by the staff on July 15, 2009. In addition, Section 5.4.1.1.2 of Revision 2 to the DCD referenced MUAP-09017-P, Revision 0. The staff's review of MUAP-09017-P, Revision 0, revealed that it provided a fracture mechanics evaluation using Monte Carlo simulations on the growth of pre-existing defects, using various parameters to provide a comparison of the probability of failure of the RCP flywheel based on different inspection intervals. These conditions included stress, stress intensity factor, crack size, fatigue crack growth rate, fatigue crack growth threshold, fracture toughness, tensile strength, RT_{NDT}, operating speed and temperature, in addition to the probability of detecting a flaw. Also, Figure 1-3 of MUAP-09017-P, Revision 0, showed the keyway corners have sufficient radius to minimize notching effects, thereby reducing crack initiation sites. However, this does not resolve the issue related to **RAI 274-2126, Question 05.04.01.01-5**, since it cannot be determined from this information what the critical crack size is, and whether it can be detected during an inspection performed during the proposed inspection interval. Therefore, in **RAI 5663, Question 05.04.01.01-8**, the staff requested that the applicant provide the critical crack size and detection capability of the ultrasonic inspection technique for the flywheel. The staff identifies this as **Open Item 05.04.01.01-2**.

MUAP-07035, Revision 0, provided an analysis to calculate the change of the flywheel inner and outer radii at the flywheel overspeed condition of 1500 rpm. The results show that the largest change for the inner radius is 0.004 inch and the largest change for the outer radius is 0.007 inch. Although the average shrink-fit of the flywheel to the shaft is about 0.00118 inch, the flywheel uses keys, so the minor deformation would not result in any adverse conditions, such as excessive vibrational stresses and imbalance of the flywheel. There are three keyways positioned in the flywheel bore that maintain the centering of the flywheel, to prevent imbalance or separation of the flywheel from the shaft. Therefore, the staff finds this acceptable.

Overspeed Testing

US-APWR DCD, Revision 1, Tier 2, Section 5.4.1.1.1 states that the flywheel is tested at 125 percent of operating speed. The design overspeed of the flywheel is determined to be 1500 rpm based on 125 percent of normal operating speed of 1200 rpm. The flywheel will be tested at 1500 rpm, and is therefore acceptable since it will be tested at the design overspeed prior to service to ensure the flywheel will maintain its structural integrity during an overspeed event, in accordance with the guidelines in Section C.3 of RG 1.14. In addition, US-APWR DCD, Revision 1, Tier 1, Section 2.4, Table 2.4.2-5 provides ITAAC Commitment No. 10.b, which prescribes that the as-built RCP flywheel assembly be tested at 125 percent of operating speed. This ITAAC is considered acceptable since it provides the necessary information for the flywheel spin test to be performed, including the acceptance criteria. However, the staff notes that SRP Section 5.4.1.1, Paragraph II.3, states that the preservice inspection results, which are performed after the spin test, should be appropriately documented to establish the initial flywheel conditions, accessibility and practicality of the inspection program to be used as baseline information for future inservice inspections. In response to **RAI 274-2126, Question 05.04.01.01-06**, the applicant stated in a letter dated April 28, 2009, that it would provide a program in the ITAAC to address how the preservice inspection results of the as-built flywheel assembly are documented as a baseline for future inspections. Subsequently, the staff confirmed that in Revision 2 of the DCD, Tier 1, Section 2.4, Table 2.4.2-5, the applicant revised ITAAC Commitment No. 10.b to address the documentation of the inspection in a report. Therefore, the staff finds this acceptable. Accordingly**, RAI 274-2126, Question 05.04.01.01-06,** is resolved and closed.

Inservice Inspection

DCD, Revision 1, Tier 2, Section 5.4.1.1.2, states that the inservice inspection program of the flywheel is based on the guidelines of RG 1.14 and ASME Code Section XI. The inspection program is further discussed in DCD, Revision 2, Tier 2, Chapter 16, TS 5.5.7, "Reactor Coolant Pump Flywheel Inspection Program." TS 5.5.7 in DCD, Revision 2, Tier 2, Chapter 16, states that the RCP flywheel inspection program is per the recommendations of Regulatory Position C.4.b of RG 1.14, except that in lieu of Regulatory Position C.4.b(1) and C.4.b(2), a qualified in-place ultrasonic examination over the volume from the inner bore of the flywheel to the circle one-half of the outer radius or a surface examination (magnetic particle and/or liquid penetrant) of exposed surfaces of removed flywheels may be conducted at 20-year intervals. Since this does not meet the guidelines of RG 1.14, the staff requested in **RAI 274-2126, Question 05.04.01.01-07**, that the applicant provide justification for the proposed inspection option. In addition, the applicant was asked to provide justification for making the inspection interval 20 years, which differs from the guidance in RG 1.14 and SRP Section 5.4.1.1, Subsection II.6.B. In its response to **RAI 274-2126, Question 05.04.01.01-07**, dated April 28, 2009, the applicant stated that a probabilistic fracture mechanics analysis with a detailed justification to confirm that the fracture probability of the flywheel is low enough to allow an inservice inspection interval of 20 years would be provided in a technical report by June 30, 2009. MHI Technical Report MUAP-09017-P, Revision 0, was received by the staff on July 15, 2009. In addition, as stated above, Section 5.4.1.1.2 of Revision 2 to the US-APWR DCD references MUAP-09017-P, Revision 0.

The staff's review of MUAP-09017 revealed that it documents the performance of a fracture mechanics analysis using Monte Carlo simulations to calculate the probability of a flywheel failure, based on the crack growth from cyclic loading (fatigue) due the starts and stops of the RCP flywheel. A failure is postulated to occur when the maximum stress intensity factor exceeds the fracture toughness of the flywheel material. This analysis is used to show the effects of inspection intervals on the probability of a flywheel failure. Based on the staff's review, this report supplements the MUAP-07035, Revision 0, analysis that the flywheel material has sufficient toughness to prevent a ductile, non-ductile and fatigue failure of the flywheel. With the exception of **Open Item 05.04.01.01.-2**, the staff finds that MUAP-07035, Revision 0, demonstrates that a fatigue crack would not grow to the critical crack size during a 60 year period. Therefore, with the exception of **Open Item 05.04.01.01-2**, the staff finds that an inspection interval of 20 years is sufficient to detect a flaw before it grows to the critical crack size. In addition, the staff notes that by way of precedent, current operating

Westinghouse plants have RCP flywheel inspection intervals of 20 years based on Westinghouse technical report WCAP-15666-A. Therefore, based on the ability of the 20 year inspection interval to detect a flaw prior to growing to a critical crack size and the precedent for this interval with current operating reactor coolant pumps, the staff finds the 20 year inspection interval acceptable. However the acceptability of the applicant's inspection plan is still dependent upon resolution of Open **Item 05.04.01.01-02**. Therefore, **RAI 274-2126, Question 05.04.01.01-07**, is closed/unresolved, but is being tracked as part of **OI 05.04.01.01-02**.

5.4.1.1.5 Combined License Information Items

There are no COL information items applicable to this issue.

5.4.1.1.6 Conclusions

Based on the above evaluation, with the exception of the **Confirmatory Item 05.04.01.01-01,** the staff concludes that the material selection, fabrication practices, and preservice inspection provide reasonable assurance that the materials used for the reactor coolant pump flywheel structures will preclude inservice deterioration and maintain its structural integrity. In addition, the flywheel analysis results, upon resolution of **Open Item 05.04.01.01-2**, along with the corresponding inservice inspection will ensure that flaws do not propagate beyond those predicted for the service life of the flywheel. Therefore, with the exception of the open items discussed above, the staff concludes that Section 5.4.1.1 of the US-APWR DCD is otherwise acceptable because it meets the requirements of 10 CFR 50.55a, GDCs 1 and 4 of Appendix A to 10 CFR Part 50, and Section III of the ASME Code, in that the probability of a flywheel failure is sufficiently small, thereby minimizing the potential of generating missiles from the reactor coolant pump flywheel.

5.4.1.2 Reactor Coolant Pumps

5.4.1.2.1 Introduction

The RCPs provide forced flow circulation of the reactor coolant to transfer heat from the reactor core to the steam generators. The primary purpose of the RCPs is to ensure adequate reactor cooling flow rate to maintain a departure from nucleate boiling ratio (DNBR) greater than the limit that is evaluated in the safety analysis. The RCPs form part of the reactor coolant pressure boundary during all modes of operation, thereby retaining the circulated reactor coolant and entrained radioactive substances.

The four RCPs, one in each coolant loop, are vertical-shaft, single-stage, mixed-flow, centrifugal pumps with diffusers, driven by totally enclosed, squirrel-cage induction motors. The RCPs raise the pressure of the reactor coolant flowing out of the SGs through the crossover pipes and return it at sufficient pressure/flow head to the reactor vessel via the cold leg pipes to maintain the desired/design reactor coolant flow.

5.4.1.2.2 Summary of Application

DCD Tier 1: The Tier 1 information regarding the RCPs is in DCD Tier 1, Section 2.4.2.

DCD Tier 2: The applicant has provided Tier 2 information regarding the RCPs in DCD Tier 2, Sections 5.4.1.2, 5.4.1.3, 5.4.1.4 and 5.4.1.5. The details are in the following DCD Tier 2 Sections:

5.4.1.2 Reactor Coolant Pump Design Bases: The RCP is in the reactor containment and ensures adequate reactor cooling flow rate to maintain a departure from nucleate boiling ratio (DNBR) greater than the limit that is evaluated in the safety analysis. The RCP is designed, fabricated, and tested according to the requirements of 10 CFR Part 50, Section 50.55a, GDC 1 and ASME Code, Section III (Ref. 5.4-7, 14). The pump is designed with margin in integrity and exhibits safe operation under all postulated events. In the event of loss of offsite power (LOOP), the pump is able to provide adequate flow during coastdown conditions because of the pump assembly rotational inertia, which is provided by the flywheel (top of the motor), the motor rotor, and other rotating parts. This forced flow and the subsequent natural circulation effect in the reactor coolant system (RCS) adequately cool the core. Figure 5.4.1-1 shows the RCP and Table 5.4.1-1 provides the design parameters of the RCP.

5.4.1.3 Pump Assembly description

5.4.1.3.1 Design Description

5.4.1.3.2 Description of Operation

5.4.1.3.3 Loss of Seal Injection

5.4.1.3.4 Loss of Component Cooling Water

5.4.1.4 Design Evaluation

5.4.1.4.1 Pump Performance

5.4.1.4.2 Coastdown Capability

5.4.1.4.3 Bearing Integrity

5.4.1.4.4 Locked Rotor

5.4.1.4.5 Critical Speed

5.4.1.4.6 Missile Generation

5.4.1.4.7 Pump Overspeed Considerations

5.4.1.4.8 Anti-reverse Rotation Device

5.4.1.4.9 Shaft Seal Leakage

DCD Section 5.4.1.5, “Test and Inspection,” states: “The design and construction of the RCP is made to comply with the ASME Code, Section III (Ref. 5.4-14). The

RCP is designed to allow inspections as stipulated by ASME Code, Section XI (Ref. 5.4-15). Tests and inspections of the RCP are given in Table 5.4.1-2. The pump casing with support feet is cast in one piece. Internal parts can be removed from the casing for visual access to the pump casing.

ITAAC: DCD Tier 1, Section 2.4.2, Table 2.4.2-5 lists the ITAAC pertinent to the RCPs.

Technical Specifications: The TS related to the RCPs are in DCD Tier 2, Chapter 16, US-APWR TS, in the following TSD sections and in the corresponding bases and surveillance sections:

- 3.4.1, "RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits," which requires RCS total flow rate to be ≥ 460,000 gpm and "greater than or equal to the limit specified in the COLR [Core Operating Limits Report]."

- 3.4.4, "RCS Loops – MODES 1 and 2"

- 3.4.5, "RCS Loops- MODE 3"

- 3.4.6, "RCS Loops – MODE 4"

- 3.4.7,"RCS Loops – MODE 5, Loops Filled"

- 3.4.18, "RCS Loops – Test Exceptions".

5.4.1.2.3 Regulatory Basis

The relevant requirements of the Commission's regulations for this area of review, and the associated acceptance criteria, are given in Section 5.1.3 and Section 5.4.1.1.3 of this FSER.

5.4.1.2.4 Technical Evaluation

RG 1.206 designates Section 5.4.1 for RCPs, but NUREG-0800 does not have a section 5.4.1. However, the RCPs are identified in SRP Section 5.4, "Reactor Coolant System Component and Subsystem Design," Subsection I, "Area of Review," Part 1, "Reactor Coolant Pumps or Circulation Pumps [BWR]," and are classified within the various components and subsystems associated with the RCS. The specific areas of review of other SRP sections related to RCPs are provided below. General information about the RCPs and the staff's evaluation and conclusion regarding US-APWR's design features and performance requirements are discussed in the sections of this report related to the following SRP sections:

- 3.9.2, "Dynamic Testing and Analysis of Systems, Components, and Equipment"
- 3.9.6, "Functional Design, Qualification, and Inservice Testing Programs for Pumps, Valves, and Dynamic Restraints"
- 9.2.2, "Component Cooling Water System"
- 3.5.1.2, "Internally Generated Missiles (Inside Containment)"

- 3.9.1, “Special Topics for Mechanical Components”
- 3.9.3, “ASME Code Class 1, 2, and 3 Components, Component Supports, and Core Support Structures”
- 3.10, “Seismic and Dynamic Qualification of Mechanical and Electrical Equipment”
- 5.2.3, “Reactor Coolant Pressure Boundary Materials”
- 5.2.4, “Reactor Coolant Pressure Boundary Inservice Inspection and Testing”
- 5.4.1.1, “Pump Flywheel Integrity (PWR)”
- 7.2, “Reactor Trip System”
- 7.3, “Engineered Safety Features”
- 7.4, “Safe Shutdown”
- 7.5, “Information Systems Important to Safety”
- 15.3.1-15.3.2, “Loss of Forced Reactor Coolant Flow Including Trip of Pump Motor and Flow Controller Malfunctions”
- 15.3.3-15.3.4, “Reactor Coolant Pump Rotor Seizure and Reactor Coolant Pump Shaft Break”

The results of the staff evaluation of DCD Sections 5.4.1.2 though 5.4.1.4 are as follow:

Injection Water/Pump Seals

The RCP seal section consists of three seals. The first is the hydrostatic seal; the second and third are mechanical seals. The No. 2 and No. 3 seals are assembled in a cartridge. The seals are designed to prevent release of reactor coolant to the atmosphere.

Injection water is continuously supplied by the chemical and volume control system (CVCS) to the RCP through a connection on the main flange to help maintain proper operating temperatures. The injection water enters a plenum (narrow annulus between the thermal barrier and the pump shaft) in the thermal barrier, and then flows in two directions. Downward flow goes to the thermal barrier/heat exchanger and into the RCS. The upward flow goes through the seal and is then bled off and returned to the CVCS. The thermal barrier heat exchanger cools the reactor coolant that enters the RCP plenum in the event of loss of injection.

The seal water flow requirement, during normal plant operation, is specified in DCD Tier 2 Table 9.3.4-2. The seal water flow rate is 32 gpm with a seal water return flow rate of 12 gpm. The spent fuel pit (SFP) serves as a safety-related alternate water source for reactor coolant pump seal injection.

Component Cooling Water System (CCWS)

Component cooling water (CCW) is supplied to the external upper bearing oil cooler and to the integral lower bearing oil cooler. RCP thermal barrier heat exchanger leakage into the CCWS is monitored to automatically close the motor-operated valve (MOV) on the outlet line upon receiving a high-flow-rate signal, thus, isolating and preventing in-leakage of reactor coolant from further contaminating the CCWS.

Internal Motor Parts Cooling

The RCP motor internal parts are cooled by internal air. Fans located on each end of the rotor draws air in through cooling slots in the motor frame. The air is then routed to the external water/air coolers that are supplied by the CCWS. Each motor has two such coolers, mounted diametrically opposed to each other. The coolers are sized to maintain optimum motor-operating temperature.

Shaft and Frame Vibration

Vibration sensors mounted on the RCP shaft and frame continuously monitor the vibration levels. Two probes are mounted on top of the seal housing to measure shaft displacement. One probe is placed parallel to the pump discharge while the other probe is located 90° apart. The frame vibration monitoring system consists of two velocity seismoprobes. The seismoprobes installed on the motor flange are placed 90° apart. The measured signals of shaft vibration and frame vibration are transmitted to and recorded in the MCR. If the vibrations exceed their set point, alarms are generated to inform plant operators of the abnormal conditions.

Spool Piece

A spool piece mounted between the motor coupling and the pump coupling can be removed easily to disassemble the seal system without displacing the motor. The pump internals can be dismantled from the pump casing after the removal of the motor and the motor stand.

Oil Spillage Protection System

The oil spillage protection system is attached to the RCP motor and is provided to contain and channel lubricating oil to a common collection point.

Loss of Seal Injection

The CVCS continuously supplies seal water to the reactor coolant pump seals, as required by the reactor coolant pump design. With regard to loss of RCP seal injection flow, the applicant states that loss of injection water flow is detected with a flow meter at the seal injection line. This condition will normally lead to an increase in seal and bearing inlet temperature and an increase in the No. 1 seal leak rate because of reactor coolant leakage flow into the RCP seals. Under these conditions, however, the CCWS continues to provide flow to the thermal barrier heat exchanger; which cools the reactor coolant leaking in. Therefore, the pump is able to maintain a safe operating temperature and operate safely long enough for safe shutdown of the pump. To complete its review, in **RAI 5718, Question 05.04.01.02-01,** the staff asked the applicant (1) if upon loss of CVCS seal injection, is there a low-flow alarm from the flow meter in the MCR, and (2) how long can the RCP operate in this condition. This is identified as **Open Item 05.04.01.02-01**.

Also, during a station blackout (SBO), RCP seal integrity is maintained until the charging pumps can be powered from an alternate source and seal water injection can be restarted. **RAI 5576, Question 09.02.02-82**-was communicated to the applicant during the public meeting held on March 3, 2011. This is an Open Item for Chapter 9.

Loss of Component Cooling Water

With regard to loss of CCW to the thermal barrier, applicant states that should the loss of CCW occur, seal injection flow from the CVCS would continue to be provided to the RCP. The applicant also states that the pump is designed so that the seal injection flow is sufficient to prevent damage to the seals with a loss of thermal barrier cooling. The loss of CCW to the motor bearing oil coolers would result in an increase in oil temperature and a corresponding rise in motor bearing metal temperature. However, the applicant states that the motor is designed to withstand a loss of CCW for up to 10 minutes. In the design, several transmitters are provided to monitor CCW flow in RCP motors and RCP thermal barriers. These transmitters are designed to provide flow indication and actuate low-flow alarms in the control room. The applicant states: "Instructions are prepared for loss of CCW and/or seal injection to the RCPs and/or loss of CCW to their motors." The applicant states that should CCW flow be lost and not restored within 10 minutes, the RCPs would be tripped following the reactor trip. However, the staff considers the applicant's description of this event to be insufficient because CCW is common to the CVCS, the thermal barrier heat exchanger, and the motor bearing oil coolers of the RCP. No mention is made of seal injection flow temperature, which would also increase upon loss of CCW. Therefore, to complete the review, in **RAI5718, Question 05.04.01.02-02**, the staff requested the applicant to provide more detail on the following:

(1) Since the CCWS provides cooling water to both the CVCS (pump motor, seal water cooler, and oil cooler) and the thermal barrier, explain what is meant by loss of CCW, and
(2) If "loss of CCW" includes cooling flow to CVCS, explain how seal injection flow would provide adequate cooling for "up to 10 minutes" to prevent seal damage. Include the estimated seal water injection temperature.
(3) The DCD states: "Instructions are prepared for loss of CCW and seal injection." Are these instructions operating procedures? If so, identify the DCD section that governs the preparation of these instructions.

This is identified as **Open Item 05.04.01.02-02.**

RCP Seal Protection

In the event that the CCWS to RCP is isolated by a containment spray actuation signal and the seal water injection from the CVCS is also lost, the DCD states that the containment isolation valves on the CCWS supply and return lines can be manually reopened from the MCR to restore RCP seal cooling. See RAI 5576 related to SRP Sections 5.4, 8.4, 9.2.2, and 9.3.4.

Oil Spillage Protection System

The oil spillage protection system is attached to the RCP motor and is provided to contain and channel lubricating oil to a common collection point.

Motor/Pump Sensors

The staff considers the RCP sensors are sufficient to monitor the motor/pump for temperature, vibration and speed. In addition, the RCP oil lift pump pressure and lower/upper oil reservoir levels are monitored.

- Oil lift pump pressure
- Lower and upper oil reservoir level
- Thrust bearing upper and lower shoe temperature
- Stator winding temperature
- Upper and lower radial bearing temperature
- Upper frame vibration transmitter (X) and (Y)
- Upper frame vibration indicator and alarm
- Lower frame vibration transmitter (X) and (Y)
- Lower frame vibration indicator and alarm
- Shaft vibration transmitter (X) and (Y)
- Shaft vibration indicator and alarm
- Frame vibration recorder
- Shaft vibration recorder
- Speed

Pump Performance

The RCP is designed to maintain the required flow rate. Initial plant testing confirms total delivery capability of the RCP. ITAAC and initial startup testing are part of the RCP initial plant testing. The RCP hydraulic performance is checked and RCP motor mechanical performance is tested at over-speeds up to and including 125% of normal speed.

The controlled-leakage shaft seal system is designed such that the No. 1 seal includes a runner, which rotates with the shaft, and a non-rotating seal ring attached to the seal housing, both equipped with a silicon nitride faceplate-clamped holder. The flow path is formed between the interface of the seal ring and seal runner. The seal ring can move axially on the seal insert to follow the seal runner. This controlled gap is thin and has a high spring ratio in which the gap is constantly maintained.

The No. 2 seal is designed and tested to maintain full system pressure for a sufficient time to secure the pump. In the case of No. 1 seal failure, the No. 1 seal leak-off line is automatically closed. If the No. 1 seal fails during normal operation, the No. 2 seal minimizes the leakage rate. The No. 2 seal is able to withstand full system pressure and the No. 3 seal ensures the backup function. Thus, the three-seal design ensures that leakage into the atmosphere is not excessive. The staff finds that the three-seal design is adequate because the features of the three-tier seal provide a sufficient barrier to limit the leakage to acceptable levels.

If loss of offsite power (LOOP) occurs, injection flow to the pump seals and CCWS flow to the thermal barrier and motor stop. Standby power sources are automatically triggered by LOOP so that CCW flow and seal injection flow are automatically restored. The RCP seal integrity during SBO is discussed in DCD Tier 2 Section 8.4.

Coastdown Capability

In the event of LOOP, the RCPs are necessary to provide adequate flow to protect the reactor. The RCP is designed with a flywheel (See Section 5.4.1.1 of this report), which, along with the other rotating parts, provides high rotational inertia, thereby providing reactor cooling during the coastdown period. Coastdown capability is maintained during the worst-case scenario, which is when safe shutdown due to earthquake and loss of offsite electrical power occur simultaneously. Coastdown flow and core flow transients are described in Section 15.3 of this report. The reactor trip system ensures that RCP operation does not exceed assumptions used for analyzing loss of coolant flow and also ensures that adequate core cooling is provided to permit an orderly reduction in power if flow from a RCP is lost during operation.

Bearing Integrity

The RCP bearings are designed to be self-aligning, hydrodynamic journal bearings that provide long life with negligible wear. The bearings are designed to keep the bearing load within the limit established by analysis under severe conditions experienced during seismic events.

In addition, the bearings are designed to preclude excessive wear between rotating parts and stationary parts and therefore providing sufficient stiffness to limit shaft motion. Low oil level in the motor lube oil sumps triggers an alarm in the control room. Each motor bearing has built-in temperature detectors with a high-bearing-temperature setpoint that triggers an alarm in the control room. This requires pump shutdown. If the pump is not shut down and the bearing proceeds to failure, the low melting point of Babbitt metal on the pad surfaces prevents sudden seizure of the shaft. In this event, the motor continues to operate, since it is designed to have sufficient reserve capacity to drive the pump under such conditions. However, the high torque required to drive the pump would cause the motor to draw excessive current. Such a sustained overload would lead to the motor being shut down by an overcurrent trip of its circuit breaker.

Locked Rotor

If the pump impeller comes in contact with a stationary part, it is assumed that instantaneous seizure of the impeller would occur. In this postulated case, the pump shaft just below the coupling to the motor would fail in torsion and loss of reactor coolant flow would occur. In this event, the motor would continue to run without overspeed and the flywheel would maintain its integrity because it is supported on the shaft by two bearings. The RCP is designed to endure the piping load caused by the locked rotor. Flow transients related to a locked-rotor event are described in Section 15.3 of this report.

Shaft seizure can only be caused by impeller contact with a stationary part. Seizure of the shaft would not occur due to excessive contact in other locations; for instance, excessive rubbing at the bearing would result in breakaway of the graphite in the bearing and any seizure at the seal would result in breakage of the anti-rotating pin of the seal ring. The motor has enough power to rotate the pump shaft in both cases. Also, in these cases, signs of excessive rubbing would be detected in the increase of vibration and temperature sensor output. Increased vibration and high No. 1 seal inlet temperature and high No. 1 seal leakoff temperature would be observed. High vibration level indicates mechanical trouble.

Critical Speed

The critical speed is analyzed under severe conditions as described in Chapter 15 of this report.

Missile Generation

RCP missile generation analysis is performed to confirm that the pump will not produce missiles under all anticipated accident conditions. Missile generation is further discussed in Section 3.5.1.2 of this report.

Pump Overspeed Considerations

An RCP overspeed condition may occur due to an electrical fault requiring immediate trip of the generator. The turbine control system and the turbine intercept valves, however, limit the overspeed to less than 120%. As additional backup, the turbine protection system has a mechanical overspeed protection trip, usually set at about 110% of rotating speed. In case a generator trip de-energizes the pump buses, the RCP motors are supplied from offsite power.

In the case of LOCA, the leak-before-break (LBB) model is applied for reactor coolant loop (RCL) piping and RCL branch piping with nominal diameter 6 inches or larger. Thus, the pump overspeed associated with the postulated pipe rupture accident need not be considered when LBB is demonstrated. The LBB model is discussed in Subsection 3.6.3 of this report.

If a turbine trip occurs, the generator and the RCP remain connected to the external network for 15 seconds to prevent a pump overspeed condition.

Anti-reverse-Rotation Device

The RCP motor is designed with an anti-reverse-rotation device. The anti-reverse mechanism consists of pawls mounted on the outside diameter of the flywheel, a serrated ratchet plate mounted on the motor frame, a spring return for the ratchet plate, and three shock absorbers.

At a lower forward speed, the pawls drop and bounce across the ratchet plate. As the motor continues to slow, the pawls drag across the ratchet plate. After the motor has slowed and comes to a stop, the dropped pawls engage the ratchet plate, and as the motor tries to rotate in the opposite direction, the ratchet plate also rotates until it is stopped by the shock absorbers. The rotor remains in this position until the motor is reenergized. When the motor is started, the ratchet plate is returned to its original position by the spring return.

As the motor begins to rotate, the pawls drag over the ratchet plate. When the motor reaches sufficient speed, the pawls bounce into an elevated position and are held in that position by friction resulting from centrifugal forces acting upon the pawls. While the motor is running at full speed, there is no contact between the pawls and ratchet plate. The staff considers the anti-reverse rotation device to be acceptable to prevent the motor from reverse rotation because the ratcheting design features is a proven industry method to prevent reverse rotation.

Test and Inspection

The design and construction of the RCP is made to comply with the ASME Code, Section III. The RCP is designed to allow inspections as stipulated by ASME Code, Section XI. Tests and inspections of the RCP are given in DCD Tier 2 Table 5.4.1-2.

5.4.1.2.5 Combined License Information

There are no COL information items for this area of review.

5.4.1.2 Conclusions

The staff reviewed DCD Tier 1, Section 2.4.2 and Tier 2 Sections 5.4.1.2 through 5.4.1.4 in accordance with SRP Section 5.4.0, “Reactor Coolant System Component and Subsystem Design,” Revision 2. Approval of the design of the US-APWR RCPs is dependent upon approval of the DCD sections listed in the evaluation section above that are related to RCPs, and satisfactory resolution of the open items identified above. Therefore the staff cannot finalize its conclusion regarding the acceptability of the design of the US-APWR RCPs until those issues are resolved. The staff will update its evaluation of the RCP design and the resulting conclusions in this FSER upon satisfactory resolution of the design issues.

5.4.2 Steam Generators

5.4.2.1 Steam Generator Materials

5.4.2.1.1 Introduction

The steam generators transfer heat from the reactor core to the secondary system to produce the steam required for turbine operation.

5.4.2.1.2 Summary of Application

DCD Tier 1: The Tier 1 information associated with this section is found in DCD Tier 1, Section 2.4.2.

DCD Tier 2: The applicant has provided a subject area description in DCD Tier 2, Section 5.4.2.1, summarized here in part as follows:

> Section 5.4.2 describes the SG materials and design. The US-APWR SGs are vertical inverted U-tube recirculation-type heat exchangers, model number 91TT-1. The general assembly of this model is shown in Figure 5.4.2-1. Each SG contains 6,747 tubes, which are arranged in a triangular pitch and hydraulically expanded for the full depth of the tubesheet at each end. Tube support plates support the straight

sections of the tubes, and anti-vibration bars support the U-bend region. The U-tubes, tube-to-tubesheet welds and the tubesheet form the reactor coolant pressure boundary (RCPB) between the primary and secondary sides of the SG.

On the primary side, reactor coolant enters through the channel head inlet nozzle, flows through the tubes, and exits through the channel head outlet nozzle. Inside the tubes, the heat is transferred to the secondary water on the outside of the tubes to produce steam. Feedwater enters through the feedwater nozzle and mixes with recirculating water. The feedwater nozzle includes a thermal sleeve and elevated feedwater ring. The steam generated on the outside of the tubes and the steam-water mixture from the tube bundle flow into centrifugal moisture separators and secondary moisture separators. The dry steam exits the SG through a steam outlet nozzle equipped with a flow restrictor.

The SG tubes are fabricated from thermally treated Alloy 690 (TT690) based on superior corrosion resistance compared to mill-annealed Alloy 600 and thermally treated Alloy 600 used in earlier SG models. The tube support plates and anti-vibration bars are fabricated from stainless steel. The primary and secondary water chemistry specifications are based on industry guidelines and operating experience for compatibility with the materials of construction. The primary coolant water chemistry specification is discussed in Section 5.2.3.2 and the secondary water chemistry specification is discussed in Section 10.3.5. Hand-holes at the top of the tubesheet, hand-holes at the uppermost tube support plate elevation, and man-ways in the upper shell provide access to the secondary side for cleaning, inspection, repair, and retrieval of foreign or loose parts.

ITAAC: The ITAAC associated with DCD Tier 2, Section 5.4.2.1 are discussed in DCD Tier 2, Section 14.3.4.4 and delineated in DCD Tier 1, Section 2.4.2. Item 2 in Table 2.4.2-5, “Reactor Coolant System Inspections, Test, Analyses, and Acceptance Criteria,” states that the functional arrangement of the RCS is as described in DCD, Tier 1, Section 2.4.2.1, “Design Description,” and as shown in Figure 2.4.2-1. Item 3a states that the components identified in Table 2.4.2-2 are constructed of material meeting ASME Code requirements. Item 4a states that these components are designed and constructed in accordance with ASME Code requirements. Item 5a states that the pressure boundary welds in ASME Code, Section III components meet ASME Code, Section III requirements for NDE of the as-built pressure boundary welds. Item 6a states that these components retain their pressure boundary integrity at their design pressure. Item 7a states that Seismic Category I components can withstand seismic design-basis loads without loss of safety function.

Technical Specifications: The Technical Specifications associated with DCD Tier 2, Section 5.4.2.1 are given in DCD Tier 2, Chapter 16, Section 3.4.17.

US-APWR Interface Issues identified in the DCD: Several sections of the application include information related to US-APWR plant interfaces that will be addressed by COL applicants. According to DCD Tier 2, sections with this information are 5.2.3, 10.3.6, 3.13, 10.4.6, 10.4.8, 9.3.4, 5.2.1.1, 5.2.1.2, 3.9.3, 5.2.4 and 5.4.2.2.

5.4.2.1.3 Regulatory Basis

The relevant requirements of the Commission's regulations for this area of review, and the associated acceptance criteria, are given in Section 5.4.2.1 and BTP 5-1, "Monitoring Secondary Side Water Chemistry in PWR Steam Generators," of NUREG-0800 and are summarized below. Review interfaces with other SRP sections can be found in Section 5.4.2.1 of NUREG-0800.

1. GDC 1 of Appendix A to 10 CFR Part 50 requires, in part, that SSCs important to safety be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed. If generally recognized codes and standards are used, they shall be identified and evaluated to determine their applicability, adequacy, and sufficiency and shall be supplemented to provide adequate assurance that these SSCs will perform their safety functions and that records will be maintained.

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

3. GDC 14 requires that the RCPB be designed, fabricated, erected, and tested to have an extremely low probability of abnormal leakage, rapidly propagating failure, and gross rupture.

4. GDC 15 requires that the RCS and associated auxiliary control and protection systems be designed with sufficient margin to ensure that the design conditions of the RCPB are not exceeded during any condition of normal operation, including anticipated operational occurrences.

5. GDC 30 requires, in part, that components that are part of the RCPB be designed, fabricated, erected, and tested to the highest quality standards practical.

6. GDC 31 requires, in part, 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, thereby minimizing the probability of rapidly propagating fracture.

7. 10 CFR 50.55a(c), 10 CFR 50.55a(d), and 10 CFR 50.55a(e) generally require that certain groups of components, including those comprising the RCPB meet the requirements of Section III of the ASME Code.

8. Appendix B to 10 CFR Part 50 applies to the SG materials. Of particular note is Criterion XIII, which requires, in part, that measures be established to control the cleaning of material and equipment in accordance with work and inspection procedures to prevent damage or deterioration.

9. Appendix G to 10 CFR Part 50 requires that RCPB pressure-retaining components that are made of ferritic materials meet ASME Code requirements for fracture toughness during system hydrostatic tests and any condition of normal operation, including anticipated operational occurrences.

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

Acceptance criteria adequate to meet the above requirements include:

1. ASME Boiler and Pressure Vessel Code, Section II, "Materials," Parts A, B and C; Section III, "Rules for Construction of Nuclear Facility Components," and Section IX, "Welding and Brazing Qualifications."

2. Certain welding-related information in NUREG-0313, Rev. 2, "Technical Report on Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping," pertinent to PWRs.

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

4. RG 1.34, "Control of Electroslag Weld Properties."

5. RG 1.36, "Nonmetallic Thermal Insulation for Austenitic Stainless Steel."

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

7. RG 1.43, "Control of Stainless Steel Weld Cladding of Low-Alloy Steel Components."

8. RG 1.44, "Control of the Use of Sensitized Stainless Steel."

9. RG 1.50, "Control of Preheat Temperature for Welding of Low Alloy Steel."

10. RG 1.65, "Materials and Inspections for Reactor Vessel Closure Studs."

11. RG 1.71, "Welder Qualification for Areas of Limited Accessibility."

12. RG 1.84, "Design, Fabrication, and Materials Code Case Acceptability, ASME Section III."

13. ASTM A-708, "Standard Recommended Practices for Detection of Susceptibility to Intergranular Corrosion in Severely Sensitized Austenitic Stainless Steel."

14. Nuclear Energy Institute (NEI) Publication NEI 97-06, "Steam Generator Program Guidelines."

5.4.2.1.4 Technical Evaluation

<u>Steam Generator Design and Materials</u>

The staff reviewed DCD Tier 2, Section 5.4.2.1, “Steam Generator Materials,” in accordance with SRP Section 5.4.2.1, “Steam Generator Materials,” to ensure that the integrity of the SG materials is maintained, and that the SG materials meet the relevant requirements of GDC 1, GDC 4, GDC 14, GDC 15, GDC 30, GDC 31, and Appendix B to 10 CFR Part 50. These requirements are met by complying with appropriate requirements of the ASME ASME Code and guidance in RGs by specifying design features shown to preserve SG tube integrity, and by specifying water chemistry practices that limit degradation of SG materials.

<u>Selection, Processing, Testing, and Inspection of Materials</u>

The steam generator materials proposed for the US-APWR are listed in DCD, Tier 2, Section 5.4.2.4.1, Section 5.2.3.1, and Table 5.2.3-1. The materials proposed are ferritic carbon and low-alloy steels, austenitic and martensitic stainless steels, and nickel-chromium-iron alloys. The staff reviewed these material selections in terms of their adequacy, suitability, and compliance with ASME Code Sections II and III. As discussed in SRP Section 5.4.2.1, for purposes of compliance with GDC 1 and GDC 30, the materials used for the steam generators are acceptable if they are selected fabricated, tested, and inspected (during fabrication/manufacturing) in accordance with the ASME Code.

The primary side of each SG is designed and fabricated to comply with ASME Code Class 1 requirements. The secondary side of each SG is designed and fabricated to comply with ASME Code Class 2 requirements. These design criteria are identified in DCD Chapter 3, Table 3.2-2. The applicant proposes to use thermally treated Alloy 690 (Alloy 690 TT, ASME SB-163) for the tubing material. This material is listed in Section II of the ASME Code and is, thus, permitted by 10 CFR 50.55a. In addition, this material is appropriate based on operating experience. Alloy 690 TT tubes were first used in U.S. operating plant steam generators in 1989 and have thus far resisted degradation by corrosion mechanisms. The tubes have a nominal outside diameter of 0.75 inch and a nominal wall thickness of 0.043 inch, which is typical for Alloy 690 tubes at operating plants. The tubes are arranged in a triangular pattern with a tube spacing (pitch) of 1.00 inch. The applicant indicated, in its response to **RAI 265-2172, Question 05.04.02.01-01**, dated March 25, 2009, that thermal treatment will be applied after the final mill anneal in accordance with EPRI guidelines (TR-016743). This thermal treatment produces the metallurgical structure characteristic of the tubing used in operating U.S. PWRs with Alloy 690 SG tubes. The applicant also noted that this tube material has been used in all MHI steam generators manufactured since 1990. **RAI 265-2172, Question 05.04.02.01-01** is resolved and closed.

Alloy 690 is also used for the channel head divider plate. Welding consumables matched to Alloy 690 (SFA-5.11/Alloy 152 or SFA-5.14/Alloy 52) are used for the cladding on the primary face of the tubesheet and for welds to the Alloy 690 TT components. In its response to **RAI 265-2172, Question 05.04.02.01-02**, March 25, 2009, the applicant stated the thickness of the tubesheet alone is 26.5 inches, and the weld cladding thickness is at least 0.15 inch. **RAI 265-2172, Question 05.04.02.01-02,** is resolved and closed.

Austenitic stainless steels are used for the primary nozzle safe ends. This is the only stainless steel steam generator component that forms part of the reactor coolant pressure boundary. The staff finds this application of austenitic stainless steel acceptable because this material is listed in Section II of the ASME Code, is appropriate for its intended application, and, thus, meets the requirements associated with stainless steel pressure boundary materials, which are discussed further in Section 5.2 of the DCD and Section 5.2.3 of this report.

The proposed bolting materials are high-strength, low-alloy, ferritic steels, ASME specifications SA-193 and SA-194. These materials are listed in Section II of the ASME Code and are, thus, permitted by 10 CFR 50.55a. The staff's review of threaded fasteners is discussed in Section 3.1.3 of this report.

For the reasons discussed above, the staff finds that the steam generator materials are acceptable and meet the requirements of GDC 1, GDC 30 and 10 CFR 50.55a. The materials used in the fabrication of the steam generators have been identified and meet the requirements of 10 CFR 50.55a.

Steam Generator Design

The staff reviewed the adequacy of the design and fabrication process proposed for the US-APWR steam generators to determine whether crevice areas are limited, residual stresses are limited in the tubesheet crevice region, corrosion-resistant materials are used, corrosion allowances are specified, and suitable bolting materials are used. As discussed above, the steam generators are designed to comply with ASME Code Class 1 and Class 2 design for the primary and secondary sides. Compliance with Code Class 1 and Class 2 design includes consideration of an additional thickness to allow for corrosion. Since the potential for degradation depends partly on the materials and water chemistry, provisions for limiting degradation are further discussed below.

Crevices around steam generator tubes have caused corrosion in earlier steam generator designs. In order to minimize or eliminate crevice areas, the APWR design includes expansion of the tubes into the tubesheet for the entire tube sheet thickness, and the tube support plates and flow distribution baffle are designed with broached trifoil holes. Experience with operating steam generators has shown that hydraulically expanding the tubes over the full tube sheet thickness serves to limit the crevice size, and that broached, trifoil holes increase flow to limit crevices from forming along the tube during operation. In addition, the tube support plates and flow distribution baffle are made of Type 405 ferritic stainless steel to limit corrosion and buildup of corrosion products that may create local environments and stresses on tube surfaces and result in corrosion processes, including stress corrosion cracking. Operating experience with replacement steam generators of similar design indicates Type 405 stainless steel does not corrode in the secondary water environment (reference: NUREG-1841, "U.S. Operating Experience with Thermally Treated Alloy 690 Steam Generator Tubes").

In order to reduce the stress formed in small-diameter U-bend sections of SG U-tubes, a thermal stress relief heat treatment is normally applied to the U-bend area. In its response to **RAI 265-2172, Question 05.04.02.01-03**, dated March 25, 2009, the applicant stated that the U-bend stress relief will be in accordance with the EPRI guidelines for steam generator tube specifications (TR-016743). The applicant further stated that for the 0.75-inch diameter tubing for the US-APWR, U-bends with a bend radius less than 7.5 inches will receive the stress relief heat treatment**. RAI 265-2172, Question 05.04.02.01-03**, is resolved and closed.

The staff reviewed the design of the feedwater inlet ring with respect to integrity and loose parts. In this design the ring is above the feedwater inlet nozzle and the feedwater is introduced through perforated nozzles at the top of the feedwater ring. The applicant discussed the feedwater inlet materials in a March 25, 2009, response to **RAI 265-2172, Question 05.04.02.01-08**. The feedwater ring and perforated nozzle material is 2.25Cr-1Mo steel, and the feedwater nozzle thermal sleeve is Alloy 690. The perforated nozzles have 0.20–inch diameter hole to capture foreign material that may enter with the feedwater. Experience with operating steam generators indicates these materials and design features provide reasonable assurance that the internal feedwater components will not degrade and contribute to loose parts in the tube bundle. The staff notes that the steam generator design includes provisions for detecting and removing loose parts, as discussed below. **RAI 265-2172, Question 05.04.02.01-08**, is resolved and closed.
For the reasons discussed above, the staff determined that the steam generators, in addition to being designed to ASME Code Class 1 and 2 criteria, are designed to promote flow and limit crevice areas between tubes and tube supports, use appropriate corrosion-resistant materials for tube supports, limit residual stresses in the tubesheet crevice region, and use corrosion-resistant materials. Designing the steam generators to these criteria meets, in part, the requirements of GDC 14, GDC 15, and GDC 31.

The staff's review of mechanical and flow-induced vibration under SRP Section 3.9.2 is discussed in Section 3.9.2 of this FSER.

Fabrication and Processing of Ferritic Materials

To comply with GDC 14, GDC 15, and GDC 31, the fracture toughness of the RCPB (Class1) ferritic materials for the SGs must resist rapidly propagating failure and ensure that the design conditions will not be exceeded during operation. The pressure-retaining ferritic materials selected for use in SGs are acceptable with respect to fracture toughness if they (1) comply with Appendix G to 10 CFR Part 50, 10 CFR 50.55a(c), (d), and (e), and (2) follow the provisions of Appendix G to Section III of the ASME Code. For Class 1 and Class 2 steam generator components, the regulations cited above require the use of Section III of the ASME Code. Article NB-2300, Article NC-2300, and Appendix G of Section III of the ASME Code address fracture toughness requirements for Class 1 and Class 2 components. The US-APWR design complies with these Code requirements, as stated in DCD Tier 2, Section 5.2.3.3.1, and, therefore, complies with the requirements related to fracture toughness. The staff's review of the fracture toughness of RCPB Materials is discussed in more detail in Section 5.2.3 of this report.

To comply with GDC 1 and GDC 30, the welding of RCPB ferritic steel for the steam generators must meet the requirements of 10 CFR 50.55a(c), (d), and (e). Ferritic steel RCPB welding must also meet the requirements of paragraph D-1210 of Appendix D to ASME Code Section III, as well as adhere to the guidance in RG 1.50, RG 1.34, RG 1.71, and RG 1.43. The US-APWR design follows these requirements and RGs, as stated in DCD Tier 2, Section 5.2.3.3.2, and, therefore, satisfies the requirements of 10 CFR Part 50 related to SG welding. The staff's review of welding of RCPB materials is discussed in more detail in Section 5.2.3 of this report.

Fabrication and Processing of Austenitic Stainless Steel

To comply with GDC 1, 14, 15, 30, and 31, the use of austenitic stainless steel must include limiting the susceptibility to stress corrosion cracking and performing welding according to quality standards. The requirements of GDC 4 and 10 CFR Part 50, Appendix B, Criterion XIII, are met through compliance with the applicable provisions of the ASME Code and with the regulatory positions of RG 1.31, RG 1.34; RG, 1.36, RG 1.37, RG 1.44, and RG 1.71. The US-APWR steam generator pressure boundary design includes Type F316LN or F316LN (SA-182) forgings for the primary nozzle safe ends.

The design of the US-APWR SGs meets the requirements and RGs listed above, as discussed in Tier 2 Sections 1.9, 5.2.3.2, 5.2.3.3, 5.2.3.4, and 5.4.2.1 of the DCD, and therefore satisfies the requirements of 10 CFR Part 50 related to the fabrication and processing of austenitic stainless steel for steam generator pressure boundary applications. The staff's review of stainless steel fabrication and processing for reactor coolant pressure boundary components is discussed in more detail in Section 5.2.3 of this report.

Compatibility of Materials with the Primary and Secondary Coolant and Cleanliness Control

The steam generator components that form the RCPB and the supporting structural components must be compatible with the reactor coolant and secondary coolant in order to meet the requirements of GDC 4. In its March 25, 2009, response to **RAI 265-2172, Question 05.04.02.01-09**, the applicant proposed modifying DCD Section 5.4.2.1.5 to identify EPRI as the source of industry guidelines that will be used for primary and secondary water chemistry. The US-APWR bases primary water chemistry control on the EPRI "PWR Primary Water Chemistry Guidelines," which is acceptable as discussed in Section 9.3.4 of this report. Secondary water chemistry for the US-APWR for both operating and shutdown conditions is based on the EPRI "PWR Secondary Water Chemistry Guidelines," NUREG-0800, BTP 5-1, NUREG-1431 (the standard technical specifications), and NEI 97-06. This approach to secondary water chemistry control is consistent with the acceptance criteria in SRP Section 5.4.2.2. The staff's review of secondary water chemistry control is discussed further in Section 10.4.6 of this report. The staff confirmed that the applicant incorporated the proposed modification in Section 5.4.2.1.5 of Revision 2 of the DCD. Therefore **RAI 265-2172, Question 05.04.02.01-09**, is resolved and closed.

As discussed above, the steam generator materials include nickel-base Alloy 690, martensitic stainless steel, austenitic stainless steel, and ferritic carbon and low-alloy steels. The laboratory and operating experience for these materials and environments, summarized, for example, in NUREG/CR-6923, indicates no general corrosion is expected for Alloy 690 or stainless steels in the primary or secondary coolant. Inservice inspections performed on Thermally Treated Alloy 690 steam generator tubing placed in service between 1989 and 2004 have revealed no instances of corrosion-related degradation of the tubing or stainless steel support structures. This experience is documented in NUREG-1841.

For carbon and low-alloy steels that are exposed only to secondary coolant that conforms to the EPRI water chemistry guidelines, very low general corrosion rates are expected, as discussed in, for example, the EPRI guidelines and NUREG/CR-6923. In a March 25, 2009, response to **RAI 265-2172, Question 05.04.02.01-05**, the applicant stated that steel containing 2.25 weight percent chromium (2.25Cr-1.0Mo) would be used for the portions of the steam generator susceptible - based on velocity - to flow-accelerated corrosion (FAC). The applicant identified these locations as the feedwater distribution ring and portions of the primary moisture separators. Studies such as EPRI TR-1008047 indicate this level of chromium reduces the FAC rate in steam and feedwater systems by a factor of approximately 65 compared to chromium-free steel. **RAI 265-2172, Question 05.04.02.01-05,** is resolved and closed.

Onsite cleaning and cleanliness controls for the steam generators are acceptable because they meet the regulatory provisions of RG 1.37. This is consistent with the requirements of Criterion XIII of Appendix B to 10 CFR Part 50.

The controls placed on the primary and secondary coolant chemistry limit the susceptibility of the steam generators to corrosion in the operating environment so that the inservice inspection program can manage any degradation that may occur. In addition, the proposed secondary water chemistry program conforms to the latest revision of the Standard Technical Specifications. These water chemistry controls meet, in part, the requirements of GDC 4 to ensure that the materials are compatible with the environment.

<u>Provisions for Accessing the Secondary Side of the Steam Generator</u>

The design for accessibility is considered acceptable if it provides adequate secondary-side access for tools to remove corrosion products (e.g., on the tubesheet and support plate crevices) and foreign objects that may affect tube integrity (such as loose parts). The staff reviewed the information in the DCD related for secondary-side accessibility. The design provides openings for access to the tube bundle, feedwater ring, feedwater nozzle, moisture separators, steam nozzle, and part of the shell interior surface. The openings are designed and located to facilitate cleaning, inspection, repairs, and foreign object search and retrieval. There are four hand holes distributed around the shell at the top of the tubesheet, two hand holes at the uppermost support plate elevation, and two manways located in the upper shell. In its response to **RAI 265-2172, Question 05.04.02.01-10**, dated March 25, 2009, the applicant provided additional details about the sizes and locations of the access openings. The manway and handhole sizes are 18 inches and 8 inches, respectively. There are two primary and two secondary manways. There are six handholes – four at the top of the tubesheet region and two at the U-bend region. The staff finds this level of access acceptable for inspection, cleaning, and foreign object removal because it closely resembles configurations used successfully in replacement steam generators at operating plants. In addition, the staff finds this level of access acceptable because tools may be inserted to inspect and remove corrosion products, contaminants that may lead to corrosion, and foreign objects (including loose parts) that may affect tube integrity. Therefore, incorporating this level of accessibility in the steam generator design meets, in part, the requirements of GDC 14 and GDC 15. Therefore, **RAI 265-2172, Question 05.04.02.01-10**, is resolved and closed.

5.4.2.1.5 Combined License Information

There are no COL information items for Section 5.4.2.1.

5.4.2.1.6 Conclusions

On the basis of its review of DCD Tier 2, Section 5.4.2.1, the staff concludes that the US-APWR steam generator materials specified in the design as described in DCD Tier 2, Section 5.4.2.1, satisfy the acceptance criteria for materials selection, design, fabrication, compatibility with the service environments, and secondary-side accessibility. The staff further concludes that these materials as specified are acceptable and meet the requirements of GDCs 1, 4, 14, 15, 30, and 31, as well as the requirements of 10 CFR Part 50, Appendices B and G.

5.4.2.2 Steam Generator Program

5.4.2.2.1 Introduction

The SG program is intended to ensure that structural and leakage integrity of the tubes are maintained during operation and postulated accident conditions.

5.4.2.2.2 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 5.4.2.2, summarized here in part, as follows:

> The application describes the SG design that provides for access to internal components for inspection and maintenance. The application identifies the location and number of man-ways and hand-holes making primary side and secondary side internals accessible for full length eddy current testing, and secondary shell, wrapper, and support inspections. The SG program as described in the application also describes provisions for top-of-tubesheet sludge lancing, foreign object search and retrieval.
>
> The steam generator program is based on NEI 07-06 for maintaining structural and leakage integrity of the SG U-tubes. The major program elements are degradation assessment, tube inspection, tube integrity assessment, tube plugging, primary-to-secondary leak monitoring, foreign material exclusion (including loose parts management), maintenance of SG secondary-side integrity, contractor oversight, self-assessment, reporting, and maintaining compatibility between the SG tubes and coolant.

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

Technical Specifications: The Technical Specifications associated with DCD Tier 2, Section 5.4.2.2 are given in DCD Tier 2, Chapter 16, Sections 3.4.13, 3.4.17 and 5.6.7, and bases sections B 3.4.13 and B.3.4.17. In addition, TS 5.5.9 specifies the content of the SG program.

5.4.2.2.3 Regulatory Basis

The relevant requirements of the Commission's regulations for this area of review, and the associated acceptance criteria, are given in Section 5.4.2.2 of NUREG-0800 and are summarized below. Review interfaces with other SRP sections can be found in Section 5.4.2.2 of NUREG-0800.

1. GDC 32 of Appendix A to 10 CFR Part 50. GDC 32 requires, in part, that the designs of all components that are part of the reactor coolant pressure boundary (RCPB) permit periodic inspection and testing of critical areas and features to assess their structural and leaktight integrity.

2. 10 CFR 50.55a(b)(2)(iii) specifically addresses the inspection of SG tubes and states that if the plant TS include inspection requirements that differ from those in Article IWB-2000 of Section XI of the ASME Code, the TS govern.

3. 10 CFR 50.55a(g) requires that inservice inspection (ISI) programs meet the applicable inspection requirements in Section XI of the ASME Code. The SG program is a portion of the ISI program. In addition, 10 CFR 50.55a(b)(2)(iii) specifically addresses SG tubes and states that if the plant TS include inspection requirements that differ from those in Article IWB-2000 of Section XI of the ASME Code, the TS govern.

4. 10 CFR 50.36 applies to the SG program in the TS.

5. Appendix B to 10 CFR Part 50 applies to the implementation of the SG program. Of particular note are Criteria IX, XI, and XVI.

6. 10 CFR 50.65, as it relates to the SG program.

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 is built 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. NEI 97-06, "Steam Generator Program Guidelines."

2. NUREG-1430, NUREG-1431, NUREG-1432, PWR Standard Technical Specifications.

3. RG 1.121, "Bases for Plugging Degraded PWR Steam Generator Tubes."

5.4.2.2.4 Technical Evaluation

The staff reviewed DCD Tier 2, Section 5.4.2.2, "Steam Generator Program," in accordance with SRP Section 5.4.2.2, "Steam Generator Program," to ensure that the SG tube bundle, as part of the RCPB, is designed to permit periodic inspection and testing of the tubes and critical areas, and includes features to assess the structural and leakage integrity of the tubes, as required by GDC 32.

The SG Program is based on NEI 97-06 and incorporates prevention of degradation, inspection, evaluation, corrective action, leakage monitoring, and maintaining performance criteria that define SG tube integrity. The applicant stated that the SG program complies with the following NRC regulations and requirements: GDC 32, 10 CFR 50.55a(g), 10 CFR 50.36, 10 CFR 50.65 and Appendix B to 10 CFR Part 50. Compliance with 10 CFR 50.55a requires the ISI program (including the SG Program) to meet the requirements of Section XI of the ASME Code. The applicant stated that if the plant technical specifications differ from the requirements in Article IWB-2000 of Section XI of the Code, the technical specifications will govern. This is acceptable to the staff because it is consistent with the ASME Code, with 10 CFR 50.55a(b)(2)(iii), and with the guidance in SRP Section 5.4.2.2.

The applicant proposed a Steam Generator Program that closely follows the latest revision of the Standard Technical Specifications (NUREG-1430, NUREG-1431, and NUREG-1432), which is Revision 3.1, dated December 2005. Several RAI questions resulted in clarification and wording changes in the DCD and TS for consistency with the STS. The applicant proposed these changes in its April 17, 2009, response to **RAI 293-2173, Questions 05.04.02.02-01, -02, -03, and -04**. The staff also identified an instance in which a US-APWR-specific TS requirement for SG tubes differed from the STS. STS 5.5.9.d.1 requires inspection of 100 percent of the tubes during the first refueling outage following SG replacement, but has no similar requirement for installation of original SGs at a new plant. In its April 17, 2009, response to **RAI 293-2173, Question 05.04.02.02-05**, on this subject, the applicant proposed to revise TS 5.5.9.d.1 to read, “Inspect 100 percent of the tubes in each SG during the first refueling outage following SG installation.” This wording is acceptable to the staff because it applies to both initial installation and replacement. The staff confirmed that all of the proposed changes were incorporated in Revision 2 of the DCD. Therefore **RAI 293-2173, Questions 05.04.02.02-01** through **-05**, are resolved and closed.

The proposed TS also differ from the STS with respect to the surveillance frequency for RCS operational leakage (TS 3.4.13). The TS provide the information in brackets, as options to be selected by a COL applicant, which state the frequency will be either 72 hours or in accordance with the Surveillance Frequency Control Program (SFCP). This is acceptable because 72 hours is consistent with the STS and the use of the SFCP will be described by a COL applicant and reviewed by the NRC staff.

Preservice inspection (PSI) of the tubes will be conducted by examining the full length of each tube after fabrication and prior to placing the SGs in service (i.e.., after the field hydrostatic test as suggested in Section 3.2.1 of the EPRI Steam Generator Examination Guidelines referenced in NEI 97-06.) In its April 17, 2009, response to **RAI 293-2173, Question 05.04.02.02-06**, the applicant affirmed this by revising DCD Section 5.4.2.2.2 to read, “Preservice inspection in accordance with Section XI of the ASME Code and EPRI PWR SG examination guidelines will be performed over the entire length of each tube in each SG.” This is acceptable because the PSI will be performed after SG fabrication, which supports the objective of discriminating between service-related degradation and manufacturing imperfections and is consistent with industry practice. In addition, in its April 17, 2009, response to **RAI 293-2173, Question**

05.04.02.02-07, the applicant removed the reference to eddy current techniques for PSI and added a statement that, "…preservice inspection will be performed using the techniques expected to be used during inservice inspection." This is acceptable because it allows for the possibility of using inspection methods other than eddy current and it supports the goal of identifying inservice degradation. The staff confirmed that these changes related to PSI specification were incorporated into Revision 2 of the DCD. Therefore, **RAI 293-2173, Questions 05.04.02.02-06 and -07**, are resolved and closed.

The tube repair criteria establish a minimum acceptable SG tube wall thickness that accounts for flaw growth and uncertainty in measuring the size of the flaw. The repair criteria are based on maintaining tube structural and leakage integrity. Tubes with flaws exceeding the repair criteria will be removed from service. For the US APWR, the generic steam generator tube repair criteria in the TS were determined using the methodology specified in RG 1.121. This is acceptable because it satisfies the staff's recommendations in SRP Section 5.4.2.2. As indicated above, however, a COL applicant may submit an analysis that proposes different tube repair criteria than those in the DCD. The staff notes that the applicant refers to the "plugging criteria" in Section 5.4.2.2.2 of the DCD, and "repair criteria" in Chapter 16 (TS). The staff considers "repair criteria" the correct term because it is consistent with NEI 97-06 and the STS. This was communicated to the applicant in **RAI 293-2173. Question 05.04.02.02-2**. In its April 17, 2009, response to **RAI 05.04.02.02-2**, the applicant proposed changing the DCD to use the word "plugging," and the staff confirmed that this change was incorporated into Revision 2 of the DCD. Therefore as stated above, **RAI 293-2173, Question 05.04.02.02-2** is resolved and closed.

As discussed in Section 5.4.2.1 above, the US APWR steam generators are designed to be accessible for inspection. All tubes can be inspected from the primary side using currently available nondestructive examination techniques. The steam generators are also designed with access to the secondary side for inspection, cleaning, and evaluation of conditions such as loose parts. On this basis, the staff finds the design of the US APWR steam generators acceptable as it relates to providing access to allow inservice inspection.

5.4.2.2.5 Combined License Information

There are no COL information items for this area of review.

5.4.2.2.6 Conclusions

On the basis of its review of DCD Tier 2, Section 5.4.2.2, the staff concludes that the US-APWR Steam Generator Program as described in the DCD satisfies the acceptance criteria for accessibility for periodic inspection and testing of critical areas for structural and leakage integrity. The staff also concludes that the proposed technical specifications are consistent with the standard technical specifications for domestic PWRs with certain exceptions that the staff reviewed and found acceptable. Therefore, the staff concludes that the SG Program is acceptable and meets the requirements of GDC 32, 10 CFR 50.55a, 10 CFR 50.36, and 10 CFR 65.

5.4.3 Reactor Coolant Piping

5.4.3.1 Introduction

The major RCS piping comprises the hot-leg pipe in each of the four reactor coolant loops from each RV outlet nozzle to each SG reactor coolant inlet nozzle, the “crossover” pipes from each SG reactor coolant outlet nozzle to each RCP suction nozzle, and the cold-leg pipes from each RCP discharge nozzle to each RV inlet nozzle. In addition, RCS piping includes the post-accident high-point vent line; and the pressurizer surge, spray, and relief lines. Other systems that interface with the RCS such as the RHR system (DCD Tier 2, Section 5.4.7), SIS piping (DCD Tier 2, Section 6.3), and CVCS piping (DCD Tier 2, Section 9.3.4) constitute a part of the reactor coolant pressure boundary, but are not part of the RCS piping.

The primary function of the major RCS piping is to provide a closed-loop flow path and means of thermal energy transfer via hot coolant from the reactor to the SGs primary-side inlets, then for cooler coolant at reduced pressure from the SG primary-side outlets to the suctions of the RCPs, and finally, at increased pressure, from the RCP discharge back to the reactor. As part of the RCPB, the piping also has the safety functions to contain coolant inventory and to form part of the second fission product barrier. It also provides for connections for the pressurizer surge and spray lines, and other connected systems.

The RCS piping contains the coolant under operating temperature and pressure conditions and limits leakage to the containment. The RCS piping contains demineralized, borated water that is circulated at the flow rate and temperature consistent with achieving the design reactor core thermal and hydraulic performance. The RCS piping is designed to withstand operating modes of the plant and other anticipated system interactions in terms of pressure and temperature. As part of the RCPB, the RCS piping is designed as Safety Class 1 and is fabricated to meet the requirements of the ASME Code, Section III, Class 1.

5.4.3.2 Summary of Application

DCD Tier 1: The Tier 1 information associated with Section 5.4.3 is found in DCD Tier 1, Section 2.3, “Piping Systems and Components,” and Section 2.4, “Reactor Systems.” Section 2.3 describes piping stress analysis, protection against the dynamic effects of pipe rupture, LBB analysis and component stress analysis. Sections 2.3 and 2.4 describe the applicant’s proposed ITAAC associated with the RCS piping in accordance with SRP Section 14.3, "Inspections, Tests, Analyses, and Acceptance Criteria." In section 2.4, Table 2.4.2-3, provides the RCS piping characteristics.

DCD Tier 2: The applicant has provided a DCD Tier 2 description in Section 5.4.3, summarized here in part, as follows:

> The RCS piping is designed to withstand operating modes of the plant and other anticipated system interactions in terms of pressure and temperatures. The piping is designed as Safety Class 1 and is fabricated to meet the requirements of the ASME Code, Section III, Class 1. The materials used to fabricate the piping are specified to

minimize corrosion and erosion while ensuring compatibility with the operating conditions. The stress limitations stipulated in ASME Code, Section III, are followed in the design of the piping. Subsection NB of the ASME Code, Section III, is the basis for the calculation of the RCS piping thickness. Because the reactor coolant piping materials are forged from austenitic stainless steel, thermal aging embrittlement is not a concern compared to stainless steel castings. The reactor coolant piping is designed using the LBB concept.

RCS piping covers the piping sections that connect the reactor vessel, the SGs, and the RCPs. The RCS piping configuration is shown in Figure 5.4.3-1. Piping sections connected to the reactor coolant piping and primary components are as follow:

- CVCS charging line from the designated check valve up to one reactor coolant cold leg.
- CVCS letdown line and excess letdown line from the reactor coolant crossover leg to the designated isolation valve.
- Pressurizer spray lines from the reactor coolant cold legs up to the spray nozzle on the pressurizer vessel.
- RHRS pump suction lines from the reactor coolant hot legs up to the designated isolation valve.
- RHRS discharge lines from the designated check valve to the reactor coolant cold legs.
- Accumulator lines from the designated check valve to the reactor coolant cold leg.
- Safety injection lines from the designed check valve to the reactor vessel nozzle.
- Drain, sample and instrument lines to the designated isolation valve.
- Pressurizer surge line from one reactor coolant hot leg to the pressurizer vessel surge nozzle.
- Pressurizer spray scoop, reactor coolant temperature element installation boss, and the temperature element well itself.
- All branch connection nozzles attached to reactor coolant loops.
- Pressure safety lines from nozzles on top of the pressurizer vessel up to and including the pressurizer safety valves.
- Pressure safety depressurization lines from nozzles on top of the pressurizer vessel up to and including the pressurizer safety depressurization valves.

- Auxiliary spray line from the designed isolation valve up to the main pressurizer spray line.

- Vent line from the reactor vessel closure head to the designed isolation valves.

The reactor coolant piping major design data are shown in Table 5.4.3-1. With the exception of the flanged safety valves and the reactor vessel closure head vent line, all joints and connections are welded. Piping connections for auxiliary systems are above the horizontal centerline of the reactor coolant loop piping, except for the following:

- The residual heat removal pump suction lines, which are 45 degrees down from the horizontal centerline.

- Loop drain lines, CVCS letdown line, CVCS excess letdown line, and the connection for temporary level measurement of water in the RCS during refueling and maintenance operation.

- The differential pressure taps for flow measurement, which are downstream of the SGs of the first 90 degrees elbow.

- Two of the three wells for narrow-range resistance temperature detector, which are instrumented at locations 120 degrees apart around the reactor coolant hot leg with one located at the top of the reactor coolant hot leg.

- The sample nozzle located at the horizontal centerline of the reactor coolant hot leg and cold leg.

Parts encroaching into the primary coolant loop flow path are limited to the following:

- The spray line inlet nozzles extend into the reactor coolant cold leg in the form of a scoop so that the velocity head of the reactor coolant loop flow adds to the spray driving force.

- The narrow-range and wide-range temperature detectors are in resistance temperature detector wells that extend into the reactor coolant hot leg and cold leg.

- The sample nozzles extend into the reactor coolant hot leg and cold leg to obtain a representative sample of the reactor coolant.

The RCS piping material is selected to minimize corrosion in the reactor coolant water Chemistry. A periodic analysis of the coolant chemistry is performed to verify that the reactor coolant water quality meets the specifications. Water quality is maintained to minimize corrosion by using the CVCS and process and post-accident sampling system (PSS), described in Chapter 9. The austenitic stainless steel material for the RCS piping has high erosion resistance. The RCS piping and its nozzles are sized based on the flow rates of previous plants long experiences. The flow rate is far below the tolerable erosion values due to water for this material. The fabrication processes, material selection and RCS water chemistry as described in Subsection 5.2.3 are made to minimize the possibility for stress corrosion cracking.

Tables 5.4.3-2 and 5.4.3-3 summarize the tests and inspections of the Reactor Coolant Piping materials and weldments respectively. Piping and fittings are examined over the entire volume of the material in compliance with the applicable requirements of ASME Code, Section III. The welds and their adjacent surfaces are smoothed to allow preservice and inservice inspection. Accessible surfaces of the piping, fittings and its welds are liquid penetrant examined in accordance with the ASME Code, Section III criteria.

ITAAC: The detailed ITAAC for piping systems, including the RCS piping, is given in Tables 2.3-2 in DCD Tier 1 Section 2.3. Detailed ITAAC for reactor systems, including RCS piping are given in Table 2.4.2-5 of Section 2.4.2 of DCD Tier 1. ITAAC associated with the RCS equipment, components, and piping and that comprise a portion of the containment isolation system (CIS) are described in Table 2.11.2-2.

Technical Specifications: The Technical Specifications associated with DCD Tier 2, Section 5.4.3, are given in DCD Tier 2, Chapter 16, Sections 3.4 and B 3.4.

Initial Plant Testing: As appropriate, verification, by observations and measurements that the piping and component movements, vibrations, and expansions are in compliance with Class 1, 2, and 3 system requirements, as defined by the ASME Code. Also covered are other high-energy piping systems inside Seismic Category I structures, and high-energy portions of systems, the failure of which could reduce the functioning of any Seismic Category I plant feature to an unacceptable level. Preoperational test 14.2.12.1.8, "RCS Cold Hydrostatic Preoperational Test," is performed to ensure the integrity of the RCS piping and welds. The acceptance criterion requires that no leaks at welds or piping exist within the RCS test boundaries during the final inspection.

5.4.3.3 Regulatory Basis

The relevant requirements of the Commission's regulations for this area of review, and the associated acceptance criteria, are given in Section 5.4.0 of NUREG-0800, the SRP. They are summarized in Subsections 5.1.3 and 5.4.3.4 of this FSER.

5.4.3.4 Technical Evaluation

RG 1.206 designates Section 5.4.3 for reactor coolant piping but NUREG-0800 does not have a section 5.4.3. However, the reactor coolant piping is identified in SRP Section 5.4, "Reactor Coolant System Component and Subsystem Design," Subsection I, "Area of Review," Part 3, "Reactor Coolant System Piping and Valves," and is classified within the various components and subsystems associated with the RCS. The specific areas of review of other SRP sections related/interfaced to reactor cooling piping are provided below. Detailed information about the reactor coolant piping and the staff's evaluation and conclusion regarding US-APWR's reactor coolant piping design features and performance requirements, as well as analytical methods used and structural integrity, are discussed in the sections of this report related to the following SRP sections:

- SRP 3.9.1, “Special Topics for Mechanical Components”
- SRP 3.9.6, “Functional Design, Qualification, and Inservice Testing Programs for Pumps, Valves, and Dynamic Restraints”
- SRP 3.12, “Piping Design Review”
- SRP 5.2.3, “Reactor Coolant Pressure Boundary Materials”
- SRP 5.2.4, “Inservice Inspection and Testing of the RCPB”
- SRP 5.2.5, “Reactor Coolant Pressure Boundary (RCPB) Leakage Detection”
- SRP 5.4.7, “Residual Heat Removal System”
- SRP 6.3, “Emergency Core Cooling System”
- SRP 6.1.1, “Engineered Safety Feature Materials”

5.4.3.5 Combined License Information

There are no combined license information items for Section 5.4.3 of the DCD.

5.4.3.6 Conclusions

The staff reviewed DCD Tier 2, Section 5.4.3, in accordance with SRP 5.4.0, “Reactor Coolant System Component and Subsystem Design,” Revision 2. The staff’s conclusion regarding the acceptability of the RCS piping design depends upon approval of the DCD sections listed in the evaluation section above that are related to RCS piping. Therefore, the staff cannot finalized its conclusions regarding the design of the US-APWR RCS piping until the approval of the corresponding DCD sections and any associated outstanding issues. The staff will update its conclusions regarding the design of the RCS piping when all the related design issues are resolved.

5.4.4 Main Steam Line Flow Restrictor

5.4.4.1 Introduction

The steam generator (SG) steam outlet nozzle is designed with an integral flow limiter. This design produces a minimal penalty on steam pressure during normal operation but would choke the flow in the event of a design-basis main steam line break event. Each SG contains a flow restrictor in its stem outlet nozzle.

5.4.4.2 Summary of Application

DCD Tier 1: The Tier 1 information associated with this section is found in Tier 1 Section 2.7.1.2, “Main Steam Supply System (MSS).” This section includes the applicant's proposed ITAAC associated with the MSS system which includes the SG steam outlet nozzle flow restrictors related to DCD Tier 2, Section 5.4.4, "Main Steam Line Flow Restrictor" in accordance with SRP Section 14.3, "Inspections, Tests, Analyses, and Acceptance Criteria."

DCD Tier 2: DCD Tier 2, Section 5.4.4.1 Contains the following Tier 2 information, cited here in its entirety:

> The US-APWR SG steam outlet nozzle is designed with an integral flow limiter. This design produces a minimal penalty on steam pressure during normal operation but will choke the flow in the event of a design-basis main steam line break. The benefits of the flow restrictor are as follows:
>
> - Reduction of maximum containment pressure
> - Reduction in the rate of energy release to containment
> - Extension of the cool-down rate of RCS
> - Reduction of the main steam line pipe whip loading
> - Reduction of forces on SG tubes, tube supports, and internals
>
> The flow restrictor consists of seven Alloy TT690 venturi inserts which are inserted into seven holes in the steam outlet nozzle forging of the SG. They are welded to the Alloy 690 clad surface of the nozzle.
>
> The pressure loss during normal operation is 5.9 psi under the condition of 5.0×10^6 lb/h/SG steam flow. Requirements for materials of construction and manufacturing of the flow restrictor are in accordance with Safety Class 3 of the ANSI/ANS-51.1-1981.

5.4.4.3 Regulatory Basis

The relevant requirements of the Commission's regulations are summarized in Sections 5.1.3 and 5.4.2.3 of this FSER and also are given in the SRP sections listed in Subsection 5.4.4.4 below.

5.4.4.4 Technical Evaluation

The staff reviewed DCD Tier 2, Section 5.4.3, in accordance with SRP Section 5.4.0, "Reactor Coolant System Component and Subsystem Design," Revision 2.

Each main steam line flow restrictor consists of seven Alloy TT690 venturi inserts which are inserted into seven holes in the steam outlet nozzle forging of the SG. The flow restrictor is welded to the Alloy 690 clad surface of the nozzle. With seven venturis in each flow restrictor, the equivalent throat area of each SG outlet nozzle is 1.4 ft^2 (throat diameter equivalent to 16 inches) and the resultant pressure drop during normal operation is 5.9 psi under the condition of 5.0×10^6 lb/h/SG steam flow. Requirements for materials of construction and manufacturing of the flow restrictor are in accordance with Safety Class 3 of the ANSI/ANS-51.1-1981.

The staff reviewed the safety analysis of the design-basis event of steam system piping failure described in DCD Tier 2, Section 15.1.5. The analysis uses the 1.4 ft^2 nozzle flow area of the main steamline flow restrictors for each SG. The analysis results show that the acceptance criteria specified in SRP Section 15.1.5 are met. Therefore, the SG flow restrictor has an equivalent throat area of 1.4 ft^2. Also, ITAAC Item 13(b)(ii) in DCD Tier 1, Table 2.7.1.2-5, requires a verification that the installed flow restrictor within each SG main steam discharge nozzle does not exceed 1.4 ft^2. This is consistent with the safety analysis value and, therefore, the staff finds it to be acceptable.

RG 1.206 reserves Section 5.4.4, but does not have a section for the main steam line flow restrictors. Nonetheless, Section 5.4.4 of the US-APWR DCD contains the information on the steam line flow restrictors. Although NUREG-0800 does not have a section 5.4.4, the main steam line flow restrictors are identified in SRP Section 5.4, "Reactor Coolant System Component and Subsystem Design," Subsection I, "Area of Review," Part 4, "Main Steam Line Flow Restrictions," and is classified within the various components and subsystems associated with the RCS. The specific areas of review of other SRP sections related to the main steam line flow restrictors are provided below. Detailed information about the main steam line flow restrictors and the staff's evaluation regarding the US-APWR's main steam line flow restrictor design features and performance requirements are discussed under the sections of this report related to the following SRP sections:

- SRP 15.1.5, "Steam System Piping Failures Inside and Outside of Containment (PWR)"

5.4.4.5 Combined License Information Items

There are no COL information items for this area of review.

5.4.4.6 Conclusions

On the basis of its reviews related to this section, including the safety analysis of the design-basis event of steam system piping failure described in DCD Tier 2, Section 15.1.5 and also, ITAAC Item 13(b)(ii) in DCD Tier 1, Table 2.7.1.2-5, the staff concludes that the design of the main steam line flow restrictors as described in DCD Tier 2, Section 5.4.3 is acceptable with the exception of approval of DCD Section 15.1.5.

5.4.5 Reserved per RG 1.206

The DCD indicates also that Section 5.4.5 is reserved per RG 1.206 and the SRP has no Section 5.4.5. Therefore, there is no information in the section to be reviewed.

5.4.6 Reactor Core Isolation Cooling System

This section is applicable to boiling-water reactors (BWRs) only; therefore, there is no information to be reviewed in this section.

5.4.7 Residual Heat Removal System

5.4.7.1 Introduction

The RHR system (RHRS) transfers reactor core decay heat and residual heat from the RCS to the essential service cooling water system (ESWS) through the component cooling water system (CCWS) to reduce the temperature of the reactor coolant during normal plant shutdown and cooldown conditions at a rate that assures that the acceptable fuel design limit and the design condition of RCPB are not exceeded.

The RHRS is also used to transfer refueling water between the refueling cavity and the refueling water storage pit (RWSP) at the beginning and end of the refueling operations.

The RHRS is designed to operate during plant startup, normal power operation, plant shutdown and cooldown, and at the beginning of a refueling operation to provide LTOP for RCS and RHRS components, RCS flow to the CVCS during normal plant startup operations to control RCS pressure, borated water from the RWSP to the refueling cavity at the beginning of a refueling operation, and cooling of the in-containment RWSP during normal plant operations when required.

The RHRS is a safety-related system consisting of four independent loops while sharing portions of the RHRS with the containment spray system (CSS).

5.4.7.2 Summary of Application

DCD Tier 1: The Tier 1 information associated with this section is found in DCD Tier 1, Section 2.4.5, "Residual Heat Removal System." This section includes the applicant's proposed inspections, tests, analyses, and acceptance criteria (ITAAC) associated with the SSCs related to Standard Review Plan (SRP) Section 5.4.7, "Residual Heat Removal (RHR) System" in accordance with SRP Section 14.3, "Inspections, Tests, Analyses, and Acceptance Criteria."

DCD Tier 2: The applicant has provided a DCD Tier 2 system description in Section 5.4.7, summarized here in part, as follows:

The RHRS is designed to perform the following functions:

- The RHRS pressure boundary and pressure boundary components are designed to meet GDC 2, GDC 4, GDC 5, GDC 19 Control Room, GDC 34, and RG 1.139.

- The RHRS is designed to cool the reactor by removing fission products, decay heat, and other residual heat from the reactor core and the RCS after the initial phase of the normal plant shutdown and cool down. Heat is transferred from the RCS through the steam generators (SGs) during the initial cooldown phase.

- The RHRS is designed to ensure that the reactor core decay heat and other residual heat are safely removed from the reactor using four independent subsystems. Any two of the four subsystems have a 100 percent capability for safe shutdown.

- Each containment spray (CS)/RHR pump and the RHRS motor-operated valves (MOVs) receive electrical power from safety buses so that the RHRS safety functions are maintained during a loss of offsite power (LOOP).

- Each CS/RHR pump and isolation valve of one train is connected to a separate electrical train so that the RHRS safety functions are maintained during a single failure of the electrical train. This prevents the loss of two or more trains during an electrical failure.

- Assuming the four units of the CS/RHR heat exchangers and four units of the CS/RHR pumps are in service and the supplied essential service water temperature to the component cooling water heat exchanger is at 95 °F, the RHRS is designed to provide the capability of reducing reactor coolant temperature during a normal shutdown as follows:

  - Reduce reactor coolant temperature to 140 °F within 24 hours after reactor shutdown

  - Reduce reactor coolant temperature to 130 °F within 45 hours after reactor shutdown

  - Reduce the reactor coolant temperature to 120 °F within 90 hours after reactor shutdown.

Assuming that two units of the CS/RHR heat exchangers and two units of the CS/RHR pumps are in service (assuming a single failure in one train and a second train being out of-service for preventive maintenance, testing or postulated accident) and the component cooling water (CCW) heat exchangers are supplied with the essential service water temperature at 95°F, the RHRS is capable of reducing the reactor coolant temperature from 350 °F to 200 °F within 36 hours of the reactor shutdown with two subsystems.

The RHRS as shown in Figure 5.4.7-1 consists of four independent subsystems, each of which receives electrical power from one of four safety buses. The piping and instrumentation diagram (P&ID) for the four loop RHRS is as shown in Figure 5.4.7-2. Each subsystem includes one CS/RHR pump, one CS/RHR heat exchanger, associated piping, valves, and instrumentation necessary for operational control. Table 5.4.7-2 provides the important system design parameters and CS/RHR pump characteristic curve is as shown in Figure 5.4.7-3. The CS/RHR heat exchangers and the CS/RHR pumps have functions in both the CS system and the RHRS. The mode diagram is as shown in Figure 5.4.7-4.

The RHRS is placed in operation when the pressure and temperature of the RCS are approximately 400 pounds per square inch gauge (psig) and 350 °F, respectively. The inlet lines to the RHRS are connected to the hot legs of four reactor coolant loops (RCLs), while the return lines are connected to the cold legs of each of the RCLs. The RHRS suction lines are isolated from the RCS by two normally-closed motor-operated valves with power lockout capability, which are connected in series and located inside the containment. Each discharge line is isolated from the RCS by two check valves located inside the containment and by a normally closed MOV.

- Transfer of reactor core decay heat and other residual heat from the RCS for the purpose of reducing the temperature of the reactor coolant during normal plant shutdown and cool down conditions,

- Transfer of refueling water between the refueling cavity and the RWSP at the beginning and end of the refueling operations,

- 100 percent capability provided for safe shutdown with two of any four RHR independent subsystem combinations,

- Safety functions preserved during the LOOP by providing electrical power from safety buses to each containment spray (CS)/RHR subsystem,

- Safety functions maintained during single failure of electrical train by providing independent separate electrical trains to each CS/RHR subsystem,

- RCS temperature reduction capability with all four CS/RHR subsystems in service and fully operable during a normal shutdown, within the following time frame:

    - To 140 °F within 24 hours
    - To 130 °F within 45 hours
    - To 120 °F within 90 hours

- RCS temperature reduction capability, from 350 °F to 200 °F within 36 hours, with two subsystems in service and fully operable during a normal shutdown,

- Isolation capability of the RHRS from the RCS during normal operation above a preset pressure by RHRS suction line isolation valves,

- Control RCS pressure during normal plant startup operations by pumping a portion of the reactor coolant to the chemical and volume control system (CVCS),

- Cooling capability of the in-containment RWSP by the RHRS to limit the RWSP temperature to not greater than 120 °F during normal plant operations,

- Over-pressurization and LTOP capacity for RCS/RHRS components caused by transients, loss of equipment and potential operator error during plant startup and shutdown provided by RHRS suction line equipped pressure relief valves,

- Designed/designated system for a single nuclear power unit,

- Fully controlled/operated by the control room during single failure for normal operation with the exception of restoring power to the suction isolation valves,

- Mid-loop or drain down operation to allow for maintenance or inspection activities of the reactor head, steam generator and/or reactor coolant pump seals,

- Interface system LOCA protection provided by two RHRS suction line motor operated valves in series with power lockout capability,

- Capability to withstand high pressure and discharge the RCS inventory to the in-containment RWSP,

- CS/RHR pump overheating and loss of flow damage prevention provided by minimum flow lines,

- Leakage detection system protection to minimize the leakage from those portions of the RHR system outside of the containment that contain or may contain radioactive material following an accident.

ITAAC: The ITAAC associated with Tier 2 Section 5.4.7 are given in Table 2.4.5-5 of DCD Tier 1 Section 2.4.5. The ITAAC associated with those components shared with the CSS performing their containment spray functions are provided in Tier 1, Section 2.11.3.

Technical Specifications: The Technical Specifications associated with DCD Tier 2 Section 5.4.7 are given in Tier 2 Chapter 16 Sections:
- 3.9.5 "Residual Heat Removal (RHR) and Coolant Circulation - High Water Level"
- 3.9.6 "Residual Heat Removal (RHR) and Coolant Circulation – Low Water Level"
- 3.4.12 "Low Temperature Overpressure Protection (LTOP) System"
- In addition, Technical Specifications of DCD Tier 2 Sections which interface with Section 5.4.7 having specific RHRS requirements include 3.4.6, 3.4.7, 3.4.8, and 3.4.14.
- 3.4.15 and B.3.4.15

Initial Plant Testing: DCD Tier 2, Chapter 14, identifies the RHRS tests to be performed as part of the initial plant testing program including: Test 14.2.12.1.22, "Residual Heat Removal System (RHRS) Preoperational Test" and Test 14.2.12.1.76, "Remote Shutdown Preoperational Test" (Test demonstrates the capability to cool down the plant from the hot standby condition to the cold shutdown condition by remotely controlling RHRS). Also, system accessibility will be provided for periodic testing and in-service inspection of the RHRS.

5.4.7.3 Regulatory Basis

The relevant requirements of the Commission's regulations for this area of review, and the associated acceptance criteria, are given in Section 5.4.7 of NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," and are summarized below. Review interfaces with other SRP sections can be found in Section 5.4.7 of NUREG-0800.

1. GDC 2, as it relates to the seismic design of SSCs whose failure could cause an unacceptable reduction in the capability of the RHRS, specifically based on meeting Regulatory Position C-2 of RG 1.29 or its equivalent.

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

3. GDC 5, as it relates to the requirement that any sharing among nuclear power units of SSCs important to safety will not significantly impair their safety function.

4. GDC 19, as it relates to control room requirements for normal operations and shutdown

5. GDC 34, as it relates to requirements for an RHRS.

6. NUREG-0737, Task Action Plan, Item III.D.1.1, equivalent to 10 CFR 50.34(f)(2)(xxvi) for applicants subject to 10 CFR 50.34(f), as it relates to the provisions for a leakage detection and control program to minimize the leakage from those portions of the RHRS outside of the containment that contain or may contain radioactive material following an accident.

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 is built 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. The system or systems must satisfy the functional, isolation, pressure relief, pump protection, and test provisions specified in BTP 5-4, “Design Requirements of the Residual Heat Removal System.”

2. To meet the requirements of GDC 4, design features and operating procedures should be provided to prevent damaging water hammer caused by such mechanisms as voided lines.

3. Interfaces between the RHRS and the component or service water systems should be designed so that operation of one does not interfere with, and provides proper support (where required) for, the other. In relation to these and other shared systems (e.g., emergency core cooling and containment heat removal systems), the RHRS must conform to GDC 5.

4. When the RHRS is used to control or mitigate the consequences of an accident, it must meet the design requirements of an engineered safety feature system. This includes meeting the guidelines of RG 1.82, “Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident,” regarding water sources for long term recirculation cooling following a loss-of-coolant accident.

5. NRC Generic Letter GL 88-17, “Loss of Decay Heat Removal”

5.4.7.4 Technical Evaluation

The staff reviewed the US-APWR DCD Tier 2 Section 5.4.7 (Revision 2), "Residual Heat Removal System," including applicable technical specifications, related ITAAC and initial test requirements to determine the acceptability of the design. The staff's evaluation of the DCD information was performed against the regulatory requirements and acceptance criteria listed in Section 5.4.7.3 of this FSER.

DCD Tier 1 Section 2.4.5 identified above in Section 5.4.7.2 was also reviewed by the staff. The Tier 1 section of the DCD identifies 29 ITAAC items applicable to the RHRS, including verifications of as-built configuration and subsystem physical separation, inspections of ASME Code test reports, inspections of Seismic Category I as-built equipment, verifications of Class 1E equipment/component test performance, verifications of equipment operability, verifications of parameter displays in the main control room and the remote shutdown station, and component functional performance checks. The information contained in DCD Tier 1 Section 2.4.5 is consistent with the design descriptions provided in DCD Tier 2 Section 5.4.7. The staff has reviewed the ITAAC and found them to be consistent with the certification description provided in DCD Tier 1, Section 2.4.5, with the exception of CS/RHR pump minimum-flow line. The CS/RHR pump minimum-flow line, equipped with flow measurement, is connected between the downstream side of the CS/RHR heat exchanger and the upstream side of the CS/RHR pump suction. DCD Tier 1, Sections 2.4.5 and 2.11.3 ITAAC do not address testing of the minimum-flow line. NRC Bulletin 88-04 and NRC Generic Letter 89-04 identify concerns involving the design of minimum-flow lines, particularly with regard to the potential for pump damage while running pumps in the minimum-flow configuration. In order to complete its review of the RHRS design, the staff requested additional information regarding the minimum-flow line and testing method in **RAI 163-1923, Question 05.04.07-1**.

In a letter dated February 19, 2009, in response to the RAI question, the applicant explained that each CS/RHR pump has a dedicated full-flow line that is used for in-service testing and therefore in-service testing does not rely on the minimum flow line. Provided with this insight, the staff finds that the RHRS design satisfies the requirements of 10 CFR 50.55a(g) pertaining to RHRS pump testing and the guidelines of GL 88-04 and GL 89-04. In response to the staff's concerns that a single failure could result in a "dead-head," no-flow condition of the CS/RHR pumps during mini-flow operation, the applicant described how each CS/RHR pump has a dedicated minimum-flow line and each train is independent. It was also noted that the mini-flow path has no active components and therefore the likelihood of a failure is very low. The staff finds that this response adequately addresses the concern about a single failure resulting in a no-flow condition. Therefore, **RAI 163-1923, Question 05.04.07-1,** is resolved and closed.

In addition, the US-APWR RHRS is provided with a leakage detection system designed to detect and minimize the leakage from those portions of the RHRS outside of the containment structure that contain or may contain radioactive material following an accident. The DCD also states that plant programs and procedures will detect, monitor, and control RHR leakage. DCD Tier 2 Section 5.2.5.3.1 describes the RHRS leakage detection instrumentation; and the requirements for a leakage control program, including schedule for re-testing, are addressed in DCD Tier 2 Chapter 16, Technical Specification

5.5.2, “Programs and Manuals – Primary Coolant Sources Outside Containment.” However, in order for the staff to complete its review of the RHRS design and determine that the design basis meets the design requirements of 10 CFR 50.34(f)(2)(xxvi), additional information regarding the RHRS leakage detection system and initial testing of the leakage detection system was requested in **RAI 163-1923, Question 05.04.07-2**.

The applicant responded to **RAI 163-1923, Question 05.04.07-2,** in a letter dated February 19, 2009. The response explained that the RHRS leakage detection system is part of the Equipment and Floor Drainage Systems described in DCD Tier 2, Section 9.3.3, and the initial testing of the leakage detection is described in DCD Tier 2, Section 14.2.12.1.77, “Miscellaneous Leakage Detection System Preoperational Test.” After reviewing these materials, the staff finds that the RHRS design basis includes a leakage control program, an initial test program, a schedule for re-testing the systems and a description of the actions to be taken to minimize leakage. Therefore, the RHRS design basis meets the design requirements of 10 CFR 50.34(f)(2)(xxvi) and **RAI 163-1923, Question 05.04.07-2,** is resolved and closed.

The Technical Specifications identified in FSER Section 5.4.7.2 above were reviewed by the staff and found to be consistent with the descriptions and requirements provided in DCD Tier 2, Section 5.4.7. In addition, Technical Specification 3.9.5, “Residual Heat Removal (RHR) and Coolant Circulation – High Water Level,” and Technical Specification 3.9.6, “Residual Heat Removal (RHR) and Coolant Circulation – Low Water Level,” are consistent with the limiting conditions for operations suggested by the staff in NUREG-1449, “Shutdown and Low-Power Operation at Commercial Nuclear Power Plants in the U.S.”

DCD Section 5.4.7.2.3.6 describes the mid-loop and drain-down operations of the RCS with RHRS providing core cooling. Section 5.4.7.2.3.6.E explains that when the water level of RCS abnormally drops and no RHR pumps can be operated because of air intake, the operator can supply water from the SFP to the reactor vessel by gravity via the RHRS connection to the SFP. In **RAI 163-1923, Question 05.04.07-3**, the staff requested additional information regarding SFP water supply to the reactor vessel through the RHRS and whether the approximately 34,042 gallons injected to the RHRS would be sufficient inventory to recover RHR level to allow the restart of the RHR pumps. The RHRS initial plant test, Test 14.2.12.1.22, “Residual Heat Removal System (RHRS) Preoperational Test,” includes mid-loop operational testing. The staff also asked the applicant whether this test includes verifying operation of SFP gravity drain injection to the reactor vessel when the RCS drops to a level that affects RHRS pump operation (10 CFR 52.47(b)(1)).

The applicant responded to **RAI 163-1923, Question 05.04.07-3,** in a letter dated February 19, 2009. The applicant stated that the purpose of the SFP gravity drain is not to allow restart of the RHR pumps but rather to compensate for transpiration due to decay heat by supplying water to the core. Furthermore, the applicant noted that the initial SFP volume does not matter since the SFP water is supplied from the RWSP using the refueling recirculation pump during gravity injection. In the response, the applicant also committed to revising the Subsection 14.2.12.1.22 of the DCD to add testing the SFP gravity drain injection to startup test 14.2.12.1.22. The staff finds this acceptable. Therefore, **RAI 163-1923, Question 05.04.07-03**, is resolved and closed.

In DCD Section 5.4.7, it is not apparent that the applicant describes the reasons for the RHRS satisfying the requirements of GDC 4. In **RAI 163-1923, Question 5.4.7-4**, the staff requested that the applicant provide the reference that describes the RHRS in regard to satisfying GDC 4 requirements that SSCs important to safety be designed to accommodate the effects of and be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accident conditions, including dynamic effects of such events as flow instabilities and water hammer.

In a letter dated February 19, 2009, the applicant responded to **RAI 163-1923, Question.05.04.07-4**, by proposing a revision to Section 5.4.7.1 of the DCD to include an explicit description of how the design meets GDC 4 with a reference to Section 3.11 of the DCD which details the US-APWR environmental qualification (EQ) program. After reviewing the proposed revision and the EQ program, the staff finds the RHRS satisfies GDC 4**.** Therefore, **RAI 163-1923, Question 05.04.07-4**, is resolved and closed. The staff will confirm that the proposed revision to Section 5.4.7.1 is incorporated in the next revision of the DCD. This will be tracked as **Confirmatory Item 05.04.07.04**.

To complete the review of the RHRS in accordance with SRP Section 5.4.7 and RG 1.68, the applicant was asked to provide the P&IDs of the RHRS and systems that interface with the RHRS in **RAI 163-1923, Question 05.04.07-5**. The applicant responded with a list of the interfacing P&IDs. The staff found no discrepancies when cross-checking the interfacing system P&IDs with the RHR P&IDs to ensure major flow paths and components are properly aligned between system designs. **RAI 163-1923, Question 05.04.07-5**, is resolved and closed.

DCD Section 5.4.7.1, "Design Bases," states, "The RHRS pressure boundary and pressure boundary components are designed to meet ... RG 1.139." However in June 2008, the NRC withdrew RG 1.139, "Guidance for Residual Heat Removal" (see Federal Register Notice 73 FR 32750). Since March 2007, SRP 5.4.7 has referenced BTP 5-4, "Design Requirements of the Residual Heat Removal System." Therefore the staff issued **RAI 163-1923, Question 05.04.07-6**, to request that the applicant verify that DCD Section 5.4.7 meets the provisions of BTP 5-4 as stated in SRP 5.4.7. The RAI also asked the applicant to provide the results of this verification and if necessary, revise DCD Section 5.4.7 and provide markups of the revisions.

The applicant responded to **RAI 163-1923, Question 05.04.07-6**, in a letter dated February 19, 2009, with a proposed revision to the DCD and a list of what subsections in the DCD address compliance with BTP 5-4. After reviewing the listed subsections, the staff indentified multiple areas where more information is needed to show that the provisions of BTP 5-4 are met. In **RAI 464-3520, Question 05.04.07-9**, the staff requested this additional information.

In a letter dated November 4, 2009, the applicant responded to **RAI 464-3520, Question 05.04.07-9,** but the information was still too general and did not address certain specific technical issues of importance. Hence, in a teleconference with the applicant on January 28, 2010, the staff asked for further justification showing that specific BTP 5-4 provisions are being met. Specifically of interest was Functional Requirement C of BTP 5-4, which states that the RHRS should be capable of being operated from the control room with either only onsite or offsite power available. The applicant referred to the reference in its response to **RAI 464-3520, Question 05.04.07-9**, to DCD Subsection 5.4.7.2.3.4, addressing safe shutdown of the plant . DCD Subsection 5.4.7.2.3.4 states that safe shutdown capability is available at anytime to the operator using normal shutdown systems if available. If normal shutdown systems are not available, then the operator will maintain a hot-standby condition until normal shutdown systems can be made operational. Backup systems are to be used if normal systems cannot be made available. The staff finds that this response satisfies BTP 5-4 Functional Requirement C.

However, more clarification was needed to show that certain other provisions of BTP 5-4 are met, in particular, Part (d) of BTP 5-4, which addresses boron mixing under natural circulation conditions. Initially, the DCD discussed boron mixing for the prototype plant in Chapter 14, Subsection 14.2.8.2.1, which referenced Subsection 14.2.12.2.3.9. However, in Subsection 14.2.12.2.3.9, there was no discussion regarding adequate boron mixing during natural circulation testing. Therefore in **RAI 548-4331, Question 05.04.07-12**, the staff asked the applicant to explain why DCD Subsection 14.2.12.2.3.9 does not address BTP 5-4(d) to confirm that adequate boron mixing can be achieved under natural circulation conditions.

In its response to **RAI 548-4331, Question 05.04.07-12**, dated April 6, 2010, the applicant stated that the prototype (first plant) testing for natural circulation described in DCD Revision 2, Subsection 14.2.8.2.1, includes verification of boron mixing. The applicant stated that it was revising DCD Subsection 14.2.8.2.1 to refer to Subsection 14.2.12.2.3.9 for details of the natural circulation test abstract, which are applicable to the prototype test. The applicant further stated that it was also revising Subsection 14.2.12.2.3.9 to address boron mixing. The applicant provided the proposed revisions to Subsections 14.2.8.2.1 and 14.2.12.2.3.9. However, in the natural circulation test description in the proposed revision to Subsection 14.2.12.2.3.9, there was no discussion of how adequate mixing would actually be determined. Therefore, the staff issued **RAI 617-4843, Question 05.04.07-13**, asking for the number of primary liquid sampling points to be used in the test and what numerical value is to be used for the acceptance criterion. In its response to **RAI 617-4843, Question 05.04.07-13**, dated September 14, 2010, the applicant referred to the two hot-leg sampling lines discussed in DCD Section 9.3.2 to address the number of sampling points and indicated the maximum difference in boron concentration that will be allowed between the sampling points. Based on all changes and information provided from **RAI 163-1923, Question 05.04.07-6**, **RAI 464-3520, Question 05.04.07-9, RAI 548-4331, Question 05.04.07-12**, and **RAI 617-4843, Question 05.04.07-13**, the applicant has provided sufficient information regarding how adequate boron mixing is to be determined and how the other provisions of BTP 5-4 are met. These RAIs are therefore resolved and closed. The staff will confirm incorporation of the proposed revisions to DCD Subsections 14.2.8.2.1 and 14.2.12.2.3.9 in the next revision of the DCD. This issue will be tracked as a Chapter 14 confirmatory item.

To address concerns regarding Subsection 15.6.5 large-break LOCA analyses, the staff issued **RAI 464-3520, Question 05.04.07-7**. The applicant was asked to discuss how the temperature of the RWSP is monitored and controlled if no high-temperature alarm exists in order to avoid exceeding 120 °F which is the temperature used in the accident analyses. In its response to **RAI 464-3520, Question 05.04.07-07**, dated November 4, 2009, the applicant explained that temperature indication and a low-temperature alarm is provided in the MCR and remote shutdown center (RSC) for the RWSP, but that a high-temperature alarm is not necessary because the RWSP is inside containment which is a controlled environment that is kept below 120 °F. This implies that the potential for the RWSP to exceed a temperature of 120 °F is unlikely. The applicant further pointed out that according to NUREG-1431, Revision 3.1, standard technical specifications (STS), Surveillance Requirement 3.5.4.1 (same for US-APWR TS), surveillance of the RWSP temperature is required only if the ambient temperature exceeds 120 °F. The response to **RAI 464-3520, Question 05.04.07-7**, is therefore satisfactory. Accordingly, it is resolved and closed.

Regarding SFP gravity injection into the RHRS, **RAI 464-3520, Question 05.04.07-8**, was issued. The applicant was asked to provide the flow rate of the refueling recirculation pump used to refill the RWSP and the flow rate from the SFP into the RHRS to verify that sufficient capacity exists to maintain inventory of the RWSP during RHRS injection from the SFP. The response to **RAI 464-3520, Question 05.04.07-8**, dated November 4, 2009, specified SFP gravity injection capability to be 195 gpm and the refueling recirculation pump injection capability to be 200 gpm. Since the recirculation pump capacity exceeds the SFP gravity injection capability, the staff finds the stated flow rates to be sufficient to maintain RWSP inventory during RHRS injection from the SFP. Therefore, **RAI 464-3520, Question 05.04.07-8**, is resolved and closed.

RAI 464-3520, Question 05.0407-10, was issued because it was not clear that adequate ITAAC existed associated with verifying the CS/RHR relief valve capability. In its November 4, 2009, response to **RAI 464-3520, Question 05.04.07-10**, the applicant referenced DCD Tier 1, Table 2.4.5-5 8.e. The acceptance criterion for the referenced ITAAC states that the relief valve is to open at a pressure less than the set pressure required to provide low-temperature over-pressure protection (LTOP) for the RCS. The set pressure is to be determined by the LTOP system evaluation based on the pressure-temperature curves developed for the as-procured reactor vessel material. In the applicant's response and also in Table 5.2.2-2 in Tier 2 of the DCD, the approximate set pressure is given to be 470 psig. This is for a 6-inch inlet and outlet size providing 1,320 gpm of relief capacity per valve. Technical specification 3.4.12 requires the set pressure to be less than 484 psig and greater than 456 psig. Based on the reference provided in the response to **RAI 464-3520, Question 05.04.07-10**, the staff finds that sufficient ITAAC associated with verifying the CS/RHR relief valve capability do exist. Therefore, **RAI 464-3520, Question 05.04.07-10**, is resolved and closed.

To address generic staff concerns related to gas accumulation in safety-related systems, Interim Staff Guidance (ISG) DC/COL-ISG-019 (ML092360375) was issued to the public for comment on November 3, 2009. The purpose of the ISG is to revise and update SRP Sections 5.4.7, 6.3, and 6.5.2 for the next SRP revision in order to clarify the NRC staff guidance related to gas accumulation in safety-related systems. In accordance with the proposed ISG, **RAI 464-3520, Question 05.04.07-11**, was issued to address staff concerns regarding gas accumulation potential in the US-APWR design. The applicant was asked to determine if potential pathways for gas intrusion into the CS/RHRS have been evaluated and if so to discuss them. The applicant was also asked to discuss the design features and operating procedures that provide for detection and control of gas accumulation. Finally, a discussion of ITAAC test conditions for the CS/RHR pumps net-positive suction head (NPSH) test was requested.

The applicant's response to **RAI 464-3520, Question 05.04.07-11**, dated November 4, 2009, discussed two potential pathways for gas intrusion into the RHRS, which are from the nitrogen in the accumulator and from air entrainment during mid-loop operation. The applicant explained that the US-APWR design precludes nitrogen injection by means of two check valves between the accumulator and the RHRS and by means of a valve that is normally closed, which isolates the RHRS from the accumulator system because the RHRS is not used as a low-head injection system. The applicant explained that to preclude air entrainment during mid-loop operation, a high RCS water level is to be maintained (0.33 ft above the loop center) in addition to a flow rate between 1,550 and 2,650 gpm. To address the ITAAC test conditions for the CS/RHR pump NPSH test, the applicant stated that mid-loop-operation mode is the test condition used. The justification for mid-loop-operation mode being the most drastic case for introducing voids into the closed system is that the RCS is at its lowest water level during this state allowing for the possibility of an increased volume of gas to accumulate. The applicant also explains in DCD Section 6.2.2.2 that dynamic venting during in-service testing of the full-flow test lines that discharge to the RWSP provides periodic removal of voids that might accumulate. The staff finds the current treatment of the gas accumulation issue with respect to the US-APWR to be substantial and appropriate. However, important aspects of the issue that are potentially unexamined remain. This is identified as **Open Item 05.04.07-11**.

5.4.7.5 Combined License Information Items

There are no COL information items identified DCD Tier 2, Table 1.8-2, as applicable to the RHRS and none were identified as a result of the staff's review.

5.4.7.6 Conclusions

On the basis of its review of DCD Tier 2 Section 5.4.7, "Residual Heat Removal System," in accordance with SRP 5.4.7, "Residual Heat Removal (RHR) System," Revision 4, and with the exception of open item **OI-05.04.07-11** described in section 5.4.7.4 of this report, the staff concludes that the design of the US-APWR RHRS as described in Tier 2, Section 5.4.7, of the DCD is acceptable.

5.4.8 Reactor Water Cleanup System

This section is applicable to BWRs only; therefore, there is no information to be reviewed in this section.

5.4.9 Reserved per RG 1.206

Also, neither NUREG-0800; nor the DCD have a section 5.4.9; therefore there is no information in this section to be reviewed.

5.4.10 Pressurizer

5.4.10.1 Introduction

The pressurizer is a vertically oriented, cylindrical vessel with hemispherical top and bottom heads. It maintains liquid and vapor in equilibrium under saturated conditions for RCS pressure control. Electrical immersion heaters are installed vertically through the bottom head of the vessel while the nozzles such as the spray nozzle and safety valve nozzle are located in the top head of the vessel. The surge line, which is attached to the bottom of the pressurizer, connects to the hot leg of a reactor coolant loop. Pressurizer SRVs and SDVs provide overpressure protection for the RCS and the SDVs function as part of the RCS high-point vent system. The pressurizer is a standard component of PWR RCSs. It is essentially a large electric water heater or boiler, which pressurizes the RCS by means of a steam bubble in the upper portion of its volume. The pressurizer is connected by a surge line (pipe) to a hot leg pipe and situated higher in elevation than the RV and the coolant loops in order to keep the steam bubble segregated from the RV and coolant loops. As in all standard PWRs, except for pressure differences resulting from coolant flow head loss and pressure differences due to static head, the pressure throughout the RCS is relatively uniform and well above saturation pressure for the highest RCS (normally hot leg) temperature. However, the pressurizer is maintained at saturation temperature for the RCS pressure by means of electric immersion heaters, i.e., well above hot leg temperature, in order to maintain the steam bubble for RCS pressure control during power operation and also such that the steam bubble will be in the pressurizer and nowhere else in the RCS.

The pressurizer performs its primary function of controlling RCS pressure as follows: If RCS temperature rises, the coolant expands and the added volume passes up through the surge lines into the pressure causing the liquid level to rise. This compresses the steam bubble, some of which condenses under increased pressure and mitigates the pressure rise. If pressure gets high enough, a set of pressurizer heaters is turned off automatically to allow pressure to stabilize. If pressure gets even higher, for example in a transient, subcooled liquid coolant from the discharge side of an RCP is sprayed into the steam bubble by means of a spray line and spray nozzle in the upper volume of the pressurizer that is not unlike a shower head. This condenses some of the steam and lowers the pressurizer pressure, and hence, RCS pressure. As pressure falls, the spray is stopped and if pressure falls further, additional pressurizer heaters may be turned on. The pressurizer’s active safety functions are to provide means for RCS overpressure protection, for example in a transient, as well as helping to keep RCS pressure above saturation during a low-pressure transient, e.g., in the event of a LOCA. The pressurizer’s passive safety functions as part of the RCPB are to contain and maintain coolant inventory and form part of the second fission product barrier.

RG 1.206 reserves Section 5.4.10, but does not have a section for the pressurizer. NUREG-0800 does not have a section 5.4.10 and has no other section for the pressurizer. However, Section 5.4.10 of the US-APWR DCD contains the information on the pressurizer.

5.4.10.2 Summary of Application

DCD Tier 1: The Tier 1 information associated with this section is found in DCD Tier 1, Section 2.4.2, “Reactor Coolant System.” This section includes the applicant’s proposed ITAAC pertinent to the pressurizer in accordance with SRP Section 14.3, "Inspections, Tests, Analyses, and Acceptance Criteria."

DCD Tier 2: The applicant has provided a DCD Tier 2 system description in Section 5.4.10, summarized here as follows:

The pressurizer provides a point in the RCS where liquid and vapor are maintained in equilibrium under saturated conditions for pressure control of the RCS during steady-state operations and transients.

The pressurizer surge line connects the pressurizer to one reactor coolant hot leg. This allows continuous coolant volume and pressure adjustments between the RCS and the pressurizer.

The pressurizer is sized to meet the following requirements:

- The combined saturated water volume and steam expansion volume is sufficient to provide the desired pressure response to system volume changes.
- The water volume is sufficient to prevent uncovering of the heaters following reactor trip and turbine trip.
- The steam volume is large enough to accommodate the surge resulting from a step load reduction of 100% of full power without reactor trip, assuming automatic reactor control.
- The steam volume is large enough to prevent water relief through the safety valves following a feedwater line rupture.
- ECCS actuation signal on low pressurizer pressure will not be activated as a result of a reactor trip and turbine trip.

The pressurizer is sized to have sufficient volume to accomplish the preceding requirements without the need of power-operated relief valves. The safety valves provide overpressure protection for the RCS.

The pressurizer surge nozzle and the surge line between the pressurizer and one hot leg are sized to maintain the pressure drop between the RCS and the safety valves within allowable limits during a design discharge flow from the safety valves.

The RCS pressure is controlled by the pressurizer whenever a steam volume is present in the pressurizer. A design basis safety limit has been set so that the RCS pressure does not exceed the maximum transient value based on the design pressure as allowed under the ASME Code, Section III. Evaluation of plant conditions of operation considered for design indicates that this safety limit is not reached. During startup and shutdown, the rate of temperature change in the RCS is controlled by the turbine bypass valves. The heat up rate is controlled by energy input from the RCPs and by modulation of the turbine bypass valves. The pressurizer heat-up rate is controlled by the electrical heaters in the pressurizer.

The pressurizer heaters are powered from the 480-Vac system. During a loss of offsite power concurrent with a turbine trip, selected backup heaters powered from the onsite emergency power sources are capable of being manually energized. This permits use of the pressurizer for pressure control purposes during loss of offsite power. The power supplied by the emergency power sources is sufficient to establish and maintain natural circulation in hot standby condition in conformance with the requirement of 10 CFR 50.34(f)(2)(xiii). If loss of offsite power occurs and onsite power is available, the pressurizer heaters and emergency feedwater pumps can operate to provide natural circulation and cooling through the SGs.

The pressurizer is designed in accordance with the requirements of GDC 2 and GDC 4. The design and construction of the pressurizer is made to comply with the ASME Code, Section III. The pressurizer is designed to allow inspections as stipulated by ASME Code, Section XI.

ITAAC: The ITAAC associated with Tier 2 Section 5.4.10 are given in Table 2.4.2-5, RCS ITAAC, of DCD Tier 1 Section 2.4.2, "Reactor Coolant System."

Technical Specifications: The Technical Specifications associated with DCD Tier 2 Section 5.4.10 are given in DCD Tier 2, Chapter 16 Sections:

- 3.4.9, "Pressurizer"
- 3.4.10, "Pressurizer Safety valves"
- 3.4.11, "Safety Depressurization Valves"
- B3.4.9, "Pressurizer"
- B3.4.10, "Pressurizer Safety valves"
- B3.4.11, "Safety Depressurization Valves"

Cross-Cutting Issues: TMI Action Item II.E.3.1 (per 10 CFR 50.34(f)(2)(xiii) regarding pressurizer heater power during LOOP

Initial Plant Testing: DCD Tier 2 Chapter 14 identifies the pressurizer tests to be performed as part of the initial plant testing program including: 14.2.12.1.2, "Pressurizer Pressure and Water Level Control Preoperational Test"; 14.2.12.1.107, "Pressurizer Heater and Spray Capability and Continuous Spray Flow Verification Test"; and 14.2.12.2.3.8, "Pressurizer Heater and Spray Capability and Continuous Spray Flow Verification Test."

5.4.10.3 Regulatory Basis

The relevant requirements of the Commission's regulations for this area of review, and the associated acceptance criteria, are given in Sections 5.1.3 and 5.4.10.4 of this FSER.

5.4.10.4 Technical Evaluation

Although NUREG-0800 does not have a section 5.4.10, the pressurizer is identified in SRP Section 5.4, "Reactor Coolant System Component and Subsystem Design," Subsection I, "Area of Review," Part 5, "Pressurizer," and is classified within the various components and subsystems associated with the reactor coolant system (RCS). The specific areas of review of the other SRP sections for the pressurizer are provided below. The specific acceptance criteria and review procedures are contained in the specified SRP sections. Therefore, the review and evaluation of the pressurizer is contained in the sections listed below.

The staff's review of pressurizer design specifications as summarized in Subsection 5.4.10.2 of the report above, revealed that the applicant did not provide sufficient information to support the statement that the pressurizer volume is sufficient to prevent operation of the SRVs upon a feedwater line rupture. This issue is being evaluated under Chapter 15.of this SER.

Detailed information about the pressurizer and the staff's evaluation regarding the US-APWR's pressurizer design features and performance requirements, as well as the analytical methods used and its structural integrity, are discussed under the sections of this report related to the following SRP sections:

- SRP 3.9.1, "Special Topics for Mechanical Components"
- SRP 3.9.2, "Dynamic Testing and Analysis of Systems, Components, and Equipment"
- SRP 3.9.3, "ASME Code Class 1, 2, and 3 Components, Component Supports, and Core Support Structures"
- SRP 5.2.2, "Overpressure Protection Review Responsibilities."
- SRP 5.2.3, "Reactor Coolant Pressure Boundary Materials"
- SRP 5.2.4, "Reactor Coolant Pressure Boundary Inservice Inspection and Testing"
- SRP 15.6.1, "Inadvertent Opening of a PWR Pressurizer Pressure Relief Valve or a BWR Pressure Relief Valve,"

Cross-Cutting Issue: During its review of the DCD information on the pressurizer, the staff noted that 10 CFR 50.34(f)(2)(xiii), regarding TMI Action Item II.E.3.1, requires that the pressurizer heaters be provided with a sufficient power supply and associated motive and control power interfaces to establish and maintain natural circulation in hot standby conditions with only onsite power available. DCD Section 5.4.10.3.1 states that the US-APWR design conforms to this requirement because the power supplied by the emergency power sources during a LOOP is sufficient to establish and maintain natural circulation during hot-standby conditions. To evaluate 50.34(f)(2)(xiii) compliance, in **RAI 47-839, Question 05.04.10-01**, the staff asked the applicant to further describe the emergency power sources to be used to power the backup pressurizer heaters during a loss of offsite power and to state whether these emergency power sources are of a safety- or non-safety-grade design. In its response to **RAI 47-839, Question 05.04.10-01**, dated September 22, 2008, the applicant stated that the emergency power sources referenced in Section 5.4.10.3.1 are Class 1E power sources and, as listed in DCD Chapter 8, Table 8.3.1-4, pressurizer heater backup groups A, B, C and D are respectively powered from their own train of Class 1E gas turbine-generator power sources during a LOOP. The staff finds this response acceptable because it satisfactorily explains how the US-APWR pressurizer design complies with 10 CFR 50.34(f)(2)(xiii). Therefore, **RAI 47-839, Question 05.04.10-01**, is resolved and closed.

5.4.10.5 Combined License Information Items

There are no COL information items for this area of review.

5.4.10.6 Conclusions

The staff reviewed DCD Tier 2, Section 5.4.10, in accordance with SRP Section 5.4.0, "Reactor Coolant System Component and Subsystem Design," Revision 2. The staff's approval of the US-APWR pressurizer design is dependent upon approval of the DCD sections associated with the SRP sections listed in the evaluation section above that are related to the pressurizer, and also upon the applicant's demonstration that the pressurizer has sufficient steam volume to prevent SRV lifting upon a feedwater line rupture. The steam volume issue is being evaluated in Chapter 15 of this SER. Therefore, the staff cannot finalize its conclusions regarding the design of the US-APWR pressurizer until these issues are resolved. The staff will update this subsection upon satisfactory resolution of all pressurizer design issues.

5.4.11 Pressurizer Relief Tank

5.4.11.1 Introduction

The pressurizer relief tank (PRT) system collects and cools the steam and water discharged from various safety and relief valves in the containment. The four SRVs and two safety depressurization valves (SDVs) operate to discharge the steam from the pressurizer into the PRT in certain design-basis events to prevent over-pressurization of the RCS and pressurizer. The system consists of the PRT, the discharge piping connected to the PRT and the PRT internal spray header, and associated piping. The PRT is a horizontal, cylindrical vessel with elliptical dished heads. The vessel is constructed of austenitic stainless steel and is overpressure-protected by means of rupture disks. The system design, including the PRT design volume, is based on the requirement to condense and cool a discharge of steam equivalent to 100 percent of the full-power pressurizer steam volume.

5.4.11.2 Summary of Application

DCD Tier 1: The Tier 1 information associated with DCD Tier 2 Section 5.4.11 is found in DCD Tier 1, Section 2.4.2, "Reactor Coolant System."

DCD Tier 2: The applicant has provided a DCD Tier 2 description of the PRT system in Section 5.4.11, "Pressurizer Relief Tank," summarized here in part, as follows:

> The minimum volume of water in the PRT is determined by the energy content of the steam to be condensed and cooled. The assumed initial temperature of the quench water is 120 °F. This temperature was chosen because it corresponds to the design maximum expected containment temperature for normal conditions. If the PRT temperature rises above 120 °F during plant operation, cooler water from the primary makeup water system feeds into the PRT to reduce water temperature. The expected final temperature, following a design discharge to the PRT is 210 °F.

The steam and reactor-grade water discharged from the various safety and relief valves inside containment are routed to the PRT. DCD Table 5.4.11-2 provides an itemized list of the discharges to the PRT.

A nitrogen gas blanket is used to control the atmosphere in the PRT and to allow room for the expansion of the initial water volume. The PRT gas volume is sized such that the pressure following a design basis steam discharge does not exceed the rupture disks release pressure. Gas in the PRT is periodically analyzed to determine the concentration of hydrogen and/or oxygen. The internal spray and bottom drain of the PRT function to cool the water when the temperature exceeds 120 °F. The contents are cooled by a feed-and-bleed process, with cold primary makeup water entering the PRT through the spray water inlet and the warm mixture draining by the containment vessel (C/V) reactor coolant drain pump. The discharge piping and support arrangement is designed considering the effect of thrust forces on the piping system from valve operations. The design and location of the PRT rupture disks are such that they do not pose a missile threat to any safety-related equipment. The application describes PRT design data in DCD Table 5.4.11-1 and the components that discharge to the PRT in DCD Table 5.4.11-2.

ITAAC: The ITAAC associated with Tier 2 Section 5.4.11 are given in DCD Tier 1, Table 2.4.2-5, "Reactor Coolant System Inspections, Tests, Analyses, and Acceptance Criteria," of DCD Tier 1 Section 2.4.2, "Reactor Coolant System."

Technical Specifications: There are no Technical Specifications associated with this area of review.

5.4.11.3 Regulatory Basis

The relevant requirements of the Commission's regulations for this area of review and the associated acceptance criteria are given in Section 5.4.11 of NUREG-0800 and are summarized below. Review interfaces with other SRP sections can be found in Section 5.4.11 of NUREG-0800.

1. GDC 2, as it relates to the protection of essential systems from the effects of earthquakes. Acceptance is based on meeting the guidelines in Position C.2 of RG 1.29 regarding the location of the tank in relation to other plant systems (the design of the tank system should be such that the plant safety-related systems would continue to perform their safety functions in the event of a tank failure) and in Position C.3 regarding the extension of seismic Category I boundaries.

2. GDC 4, as it relates to a failure of the system that results in missiles or adverse environmental conditions that could produce unacceptable damage to safety-related systems or components.

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 is built 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. Acceptance, as it relates to the protection of essential systems from the effects of earthquakes, is based on meeting the guidelines in Regulatory Position C.2 of RG 1.29 regarding the location of the tank in relation to other plant systems (the design of the tank system should be such that the plant safety-related systems would continue to perform their safety functions in the event of a tank failure) and in Regulatory Position C.3 regarding the extension of Seismic Category I boundaries.

2. The staff uses the following specific criteria to determine whether the requirements of GDC 4 are met:

    A. The rupture disks have a relief capacity that at least equals the combined capacity of the pressurizer relief and safety valves, with sufficient allowance for rupture disk tolerance.

    B. The pressurizer relief tank volume and the quantity of water initially stored in the tank should be such that no steam or water will be released to containment under any normal operating conditions or AOOs. It should be assumed that the initial temperature of water inside the tank will be no lower than 49 °C (120 °F). Systems performing similar functions should also be shown to have no release to containment during normal operations and AOOs.

    C. The design of the pressurizer relief tank and rupture disk should accommodate full vacuum so that the tank will not collapse if the contents are cooled after a discharge of steam without the addition of nitrogen.

    D. Alarms for high temperature, high pressure, and high and low liquid levels for the pressurizer relief tank have been provided. Systems performing similar functions should also have appropriate instrumentation to inform the operator about the condition of the systems.

    E. The location of the tank should be such that the rupture discs do not pose a missile threat to safety-related equipment.

    F. At the interface between Seismic Category I and non-Seismic Category I SSCs, the Seismic Category I dynamic analysis requirements should be extended to either the first anchor point in the non-seismic system or a sufficient distance into the non-Seismic Category I system so that the Seismic Category I analysis remains valid

5.4.11.4 Technical Evaluation

The staff evaluated this section of the DCD based on the requirements of GDC 2 and GDC 4 as described in the acceptance criteria of RG 129, Positions C.2 and C.3 related to the PRT system. The review procedures recommended in Section 5.4.11, “Pressurizer Relief Tank,” of the SRP were followed.

Design-basis events (DBEs) could result in RCS overpressurization. To relieve the overpressurization, steam from the pressurizer is discharged through the pressurizer's four SRVs and relief piping into the PRT. In addition to the four SRVs, there are two SDVs which are used for severe accident mitigation. Both the SRFVs and SDVs share the same discharge header to the PRT. The PRT also limits the release of radioactive water and steam into the containment. The PRT also receives inventory from (1) the RCS through the reactor vessel head vent valves and (2) the chemical and volume control system (CVCS) through the seal water return line relief valve and letdown line relief valve.

The PRT is constructed of austenitic stainless steel and is protected from overpressure by means of rupture disks. The discharge piping connected to PRT is also constructed of austenitic stainless steel. The PRT and the discharge piping connected to PRT are designed to be in compliance with RG 1.29, Regulatory Positions C.2 and C.3 such that the system design will not adversely affect the performance of safety-related SSCs.

A spray header provides cooling water from the drain and vent recirculation cooling system and makeup water from the demineralized water system (DWS). The internal spray and bottom drain of the PRT function to cool the water when the temperature exceeds 120 °F. The gaseous waste processing system supplies nitrogen gas to the PRT through the pressurizer relief discharge lines and extracts gases from the PRT through an extraction line connected to the PRT. Gas in the PRT is periodically analyzed to determine the concentration of hydrogen and oxygen. For effective condensing and cooling of steam discharged from the pressurizer and other sources, the steam is discharged through a sparger pipe under the water level located near the bottom of the PRT.

To allow room for the expansion of the initial water volume, a nitrogen gas blanket is used to control the atmosphere in the PRT. The PRT is maintained under a slight vacuum with a gas volume sized such that the pressure following a design basis steam discharge does not exceed the rupture disks release pressure.

The minimum volume of water in the PRT is determined by the energy content of the steam to be condensed and cooled, the assumed initial temperature of the water, and by the desired final temperature of the water. The initial water temperature is assumed to be 120 °F, which corresponds to the design maximum expected containment temperature for normal conditions. If the PRT water temperature rises above 120 °F during plant operation, cooler water from the primary makeup water system (PMWS) feeds into the PRT to reduce water temperature while draining some of the PRT water using the reactor coolant drain pump. The expected final temperature, following a design discharge to the PRT, is 210 °F. The staff finds that the initial water temperature assumption, in the applicant's system design, is acceptable because it meets the acceptance criteria, as stated in Subsection 5.4.11.3 above, that the initial water temperature will be no greater than 120 °F.

The applicant states that the PRT design volume "is based on the requirement to condense and cool a discharge of steam equivalent to 100% of the full-power pressurizer steam volume." On the basis of its review of the design data provided in Section 5.4.10 and Section 5.4.11, including DCD Table 5.4.10-1, "Pressurizer Design

Data" and DCD Table 5.4.11-1, "Pressurizer Relief Tank Design Data," the staff finds that the data provided were sufficient to confirm that the PRT volume is adequate. The rupture disks are designed to open should the discharge result in a pressure that exceeds the PRT design pressure. In addition, the PRT rupture disks have a relief capacity greater than or equal to the combined capacity of the pressurizer safety valves. DCD Section 5.4.11 states: "The PRT and rupture disks are also designed for full vacuum to prevent PRT collapse, in the event that the contents are cooled following a discharge without nitrogen being added." However, during its review, the staff found no information or reference to support the applicant's assertion. Therefore, in **RAI 5688, Question 05.04.11-01**, the staff requested the applicant to provide a reference and discussion to support the information in regard to the PRT rupture disk design features stated above. This will be tracked as **Open Item 05.04.11-01**.

PRT instrumentation is provided in the main control room (MCR). It includes (1) pressure indication and high-pressure alarm, (2) water level indication with high-water-level alarm and low-water-level alarm, and (3) temperature indication with high-temperature alarm. Using the MCR PRT instrumentation and controls, an MCR operator can regulate the PRT level and temperature to maintain these parameters within the PRT design basis. The staff finds this acceptable because it meets the acceptance criteria, as stated in Subsection 5.4.11.3 of this report, that adequate alarms for high temperature, high pressure, and high and low liquid levels for the PRT have been provided in the system design.

Equipment Class 1 is assigned to the pressurizer piping upstream of and including the pressurizer SRVs, SDVs, and depressurization valve. Therefore, these SSCs are classified as Seismic Category I, and the codes and standards for the NRC Quality Group A are applied. Equipment Class 1 components are designed to meet ASME Code, Section III, Class 1 requirements and the QA criteria of 10 CFR Part 50, Appendix B. Section 3.2 of this SER describes the US-APWR equipment classification, seismic category and ASME Code classification in regard to the PRT and associated piping and valves. However, according to DCD Chapter 3, Table 3.2-2, "Classification of Mechanical and Fluid Systems, Components, and Equipment," the PRT itself and its associated equipment and piping other than that identified as Class 1, are Class 4.

The PRT discharge piping and support arrangement is designed considering the effect of thrust forces on the piping system from safety valve operations. The safety valve opening generates transient thrust forces at each change in flow direction or area. The analysis of the piping system and support considers the transient forces associated with valve opening. The effects of thrust forces on the piping system are described in Subsection 3.9.3 of this report.

The applicant states that the PRT rupture disks are designed and located in such a way "that they do not pose a missile threat to any safety-related equipment." However, after reviewing DCD Section 3.5.1, "Missile Selection and Description" and specifically Section 3.5.1.2, "Internally Generated Missiles (Inside Containment)," the staff found that the information provided in these sections was general in nature and the analyses were not related specifically to rupture disks. Therefore, in **RAI 5688, Question 05.04.11-02**, the staff requested the applicant to identify a reference and provide a description that shows that the PRT rupture disks do not pose a missile threat because of their design and location within the containment. This is being tracked as **Open Item 05.04.11-02**.

In its review of DCD Tier 2, Chapter 3, "Design of Structures, Systems, Components, and Equipment," and DCD Section 5.4.11, the staff could not find sufficient information to satisfy Acceptance Criterion 2.F, which states that at the interface between Seismic Category I and non-Seismic Category I SSCs, the Seismic Category I dynamic analysis requirements should be extended to either the first anchor point in the non-seismic system or a sufficient distance into the non-Seismic Category I system so that the Seismic Category I analysis remains valid. Therefore, in **RAI 5688, Question 05.04.11-03**, the staff requested the applicant to identify a reference and provide a description that supports its position with respect to RG 1.29, Position C.3. This is being tracked as **Open Item 05.04.11-03**.

Since the pressurizer relief tank system does not constitute part of the RCPB in accordance with 10 CFR 50.2, the requirements of GDC 14 and 15 are not applicable. In addition, any failure of the auxiliary systems serving the PRT will not compromise the capability for safe shutdown. Therefore, a failure modes and effects analysis is not necessary.

5.4.11.5 Combined License Information Items

There are no COL information items for this area of review.

5.4.11.6 Conclusion

The staff reviewed DCD Tier 2, Section 5.4.11, in accordance with SRP 5.4.11, "Pressurizer Relief Tank," Revision 4. The staff's approval of the US-APWR PRT design is dependent upon approval of the sections of DCD Tier 2, Chapter 3, identified in the evaluation section above that are related to the PRT. It is also dependent upon the satisfactory resolution of **Open Items 05.04.11-01, -02, and -03.** Therefore, although the PRT design as described in DCD Section 5.4.11 otherwise satisfies the regulatory requirements and acceptance criteria identified in Subsection 5.4.11.3 of this report (in particular RG 1.29 and hence GDC 2 and GDC 4), the staff cannot finalize its conclusions regarding the design of the US-APWR PRT until the supporting design aspects are approved and until the open item is resolved. The staff will update this subsection upon satisfactory resolution of all PRT design issues.

5.4.12 RCS High-Point Vents

5.4.12.1 Introduction

Reactor coolant system (RCS) high-point vents (HPVs) are provided to remove non-condensable gases accumulated in the primary system that could inhibit natural circulation core cooling. As required by 10 CFR 50.34(f)(2)(vi) and TMI Action Plan Item II.B.1, as described in NUREG 0737, the design of the RCS must include high-point vents to maintain adequate core cooling, and the systems available to achieve this must be capable of being operated from the MCR. In addition, the operation of these systems must not lead to an unacceptable increase in the probability of a LOCA, or an unacceptable challenge to containment integrity.

In the US-APWR design, non-condensable gases from the RCS are vented using either a reactor head vent system or, the SDVs and DVs connected to the pressurizer or all. The primary design safety function of the SDVs is reactor core cooling by feed-and-bleed operation when there is a loss of heat sink from the steam generators, but they also function as part of the RCS high-point vent system.

5.4.12.2 Summary of Application

DCD Tier 1: The pertinent Tier 1 information is found in Tier 1 Section 2.4.2. The RCSHPV arrangement is shown in Tier 1, Figure 2.4.2, Sheets 1 and 2.

DCD Tier 2: The applicant has provided a Tier 2 system description in DCD Section 5.4.12 and DCD Figure 5.1 shows the system arrangements. The Tier 2 description is summarized here as follows:

> The design of RCSHPV includes the reactor vessel head vent system (RVHVS), the SDVs, and the DVs connected to the pressurizer. The system is designed to enhance natural circulation of the reactor coolant by eliminating non-condensable gases in the RCS.
>
> The RVHVS arrangement consists of a single flow path (pipe) from a nozzle at the highest point on the RV head, which diverges into two redundant parallel flow paths each with two redundant 1-inch, motor-operated, remote manual valves MOVs that can be manually operated remotely) connected in series. The 1-inch vent pipe (single flow path) upstream of the two redundant vent flow paths is located near the center of the reactor vessel head. The system discharges to the PRT.
>
> The motor-operated remote manual valves in each redundant, parallel vent flow path are normally closed and powered by independent Class 1E power supplies. The RVHVS is operated from the MCR.
>
> The RVHVS is designed to assure at least one of the redundant flow paths is able to open or isolate assuming a single failure of the remotely operated vent valves, their power supplies, or their control system. The two redundant, normally closed isolation valves in series, which must be operated separately, minimize the possibility of inadvertently opening a vent path.
>
> A break of the vent line on the reactor vessel head would result in a small-break LOCA no greater than 1-inch diameter. Such a break is similar to the hot-leg break LOCA analyzed in Section 15.6.5 of the DCD and therefore, the analysis results apply to a RVHVS vent line break.
>
> The RVHVS consists of safety-grade equipment. The piping and equipment from the vessel head vent up to, and including, the second normally closed, MOVs constitute part of the RCPB, and are designed and fabricated to ASME Code Section III, Class 1, requirements. The RVHVS meets the acceptance criteria specified in Section II of SRP Section 5.4.12, 10 CFR 50.34(f)(2)(vi) and TMI Action Plan Item II.B.1 as described in NUREG 0737.

As described in DCD Section 5.4.12, the SDVs and DVs connected to the pressurizer could be used for high-point vents to remove non-condensable gases from the top of the pressurizer.

The SDVs' arrangement consists of two flow paths with motor-operated remote manual block valves located upstream of each SDV. The block valves are powered from Class 1E dc busses and SDVs are powered from independent 480-Vac Class 1E busses. These valves are operated from the MCR. The valves are seismically qualified to IEEE Standard 344.

The DVs arrangement consists of a single flow path with two redundant motor-operated, remote manual valves in series. The valves are powered by the safety-related Class 1E power supply system. These valves are operated from the MCR. The valves are qualified to IEEE-344. The primary design function for the DVs is to prevent high-pressure melt ejection at vessel failure and temperature-induced steam generator tube rupture (TI-SGTR) by depressurizing the RCS. Non-condensable gases and/or steam are directly discharged to the containment.

ITAAC: The ITAAC associated with Tier 2 Section 5.4.12 are given in Tier 1 Section 2.4.2, Table 2.4.2-5.

Technical Specifications: There are no Technical Specifications proposed for the RCS high-point vent system.

Cross-Cutting Issues: TMI Action Plan Item II.B.1 as described in NUREG 0737 and codified in 10 CFR 50.34(f)(2)(vi), regarding a RCS high-point vent system capable of being operated from the MCR

5.4.12.3 Regulatory Basis

The relevant requirements of the Commission's regulations for this area of review, and the associated acceptance criteria, are given in Section 5.4.12 of NUREG-0800 and are summarized below.

1. 10 CFR 50.34, as related to the to the RCS high-point vent system.

2. 10 CFR 50.46(b), as it relates to the long-term cooling of the core following any calculated successful initial operation of the emergency core cooling system (ECCS) to remove decay heat for an extended period of time.

3. 10 CFR 50.46a, as it relates to the provision of and requirements related to high-point vents for the RCS, the reactor vessel head, and other systems required to maintain adequate core cooling if the accumulation of non-condensable gases would cause the loss of function of these systems.

4. 10 CFR 50.49, as it relates to environmental qualification of electrical equipment necessary to operate the reactor coolant vent system.

5. 10 CFR 50.55a and GDC 1 and 30, as they relate to the vent system components that are part of the reactor coolant pressure boundary (RCPB) being designed, fabricated, erected, tested, and maintained to high quality standards.

6. GDC 1, as it relates to the quality standards and records applicable to the design, fabrication, erection and testing of the high-point vents.

7. GDC 2 as it relates to the capability of the HPVS to withstand the effects of natural phenomena and GDC 4 as it relates to the capability of the HPVS to withstand the effects of design-bases accidents and transients.

8. GDC 14, as it relates to the RCPB being designed, fabricated, erected, and tested to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture.

9. GDC 17 and 34, as they relate to the provision of normal and emergency power for the vent system components.

10. GDC 19, as it relates to the vent system controls being operable from the control room.

11. GDC 30, as it relates to providing means for detecting and, to the extent practical, identifying the location of the source of reactor coolant leakage with the high-point vents system.

12. GDC 32 as it relates to periodic inspection and testing of RCPB components

13. GDC 36, as it relates to the vent system being designed to permit periodic inspection.

14. 10 CFR 52.47(b)(1), which requires that a DC application include the proposed ITAAC that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the DC is built and will operate in accordance with the DC, the provisions of the Atomic Energy Act, and the NRC's regulations;

Acceptance criteria adequate to meet the above requirements include:

Determine that the high-point vents are capable of removing non-condensable gases from the RCS, in beyond-DBA environments, as part of the RCPB with a minimal probability of spurious actuation. In addition, the high-point vents should not cause or aggravate a design-basis LOCA. Other high-point vents and component-specific acceptance criteria are listed in NUREG-0800, Section 5.4.12:

5.4.12.4 Technical Evaluation

The staff performed its review of the US-APWR RCS high-point vent (HPV) system design in accordance with Section 5.4.12 of the SRP with the information provided in

DCD Tier 1, Section 2.4.2, DCD Tier 2, Section 5.4.12, and the applicant's responses to **RAI 48-840, Questions 05.04.12-1 through -6**, in its letter to the NRC, dated September 22, 2008.

As required by 10 CFR 50.34(f)(2)(vi), the US-APWR is designed with a RCSHPV system to remove non-condensable gases accumulated in the primary system that could inhibit natural-circulation core cooling. The RCSHPV system is capable of being operated from the MCR. In addition, the operation of this system would not lead to an unacceptable increase in the probability of a LOCA, or an unacceptable challenge to containment integrity. In this design, the non-condensable gases from the RCS can be vented using either the reactor vessel head vent system (RVHVS), or the SDVs and DVs connected to the pressurizer or all.

Section 5.4.12.2 of the DCD indicates that the RVHVS consists of a flow path that diverges into two parallel paths each with two redundant motor operated remote manual valves in series. Also, each valve connected in series is powered by an independent Class 1E power supply. In response to **RAI 48-840, Question 05.04.12-1**, regarding detailed arrangement of the power supplies to the MOVs in this system, the applicant stated that the redundant MOVs in the two parallel vent flow paths are each powered from one of four independent Class 1E power supplies. When one of the Class 1E power trains is out of service for on-line maintenance, the other Class 1E power train that supports the unaffected valve in the same flow path will be manually connected to the affected MOV (thus the two valves in the same flow path will be powered from the same Class 1E power supply during the maintenance period). This arrangement will assure that at least one of the redundant parallel vent flow paths will open (i.e., both MOVs will open) on demand. Having two normally closed MOVs in series, which, by the design, must be operated separately, satisfies the requirement of preventing an inadvertent operation. Having MOVs that can be operated from the MCR allows for prompt manual closing of one or more of the valves; thereby satisfying the prevention of irreversible actuation requirement. Therefore, the design of the RVHVS meets the requirements of 10 CFR 50.46(b), 10 CFR 50.46a, and GDCs 17 and 34 allowing for a single failure of an active, electrical or mechanical component. Accordingly, **RAI 48-840, Question 05.04.12-1**, is resolved and closed.

Similar to the RVHVS system, the SDVs of the pressurizer have two redundant flow paths, each with two MOVs in series, namely, SDV block valves and the SDVs themselves. Accordingly, in **RAI 48-840, Question 05.04.12-02**, the staff requested additional information from the applicant regarding the ability of the SDVs to perform their safety gas venting function as part of the RCSHPVS, while preventing inadvertent or irreversible actuation, given a single active mechanical or electrical failure. In its response to this RAI question, dated September 22, 2008, the applicant explained that each motor-operated SDV and each of their motor-operated SDV block valves is connected to a separate Class 1E power supply, thus ensuring operability with a failed power supply, even with one of the Class 1E busses out of service for maintenance (by manually connecting to an unaffected power supply as can be done for the MOVs in the RVHVS). Then, similar to the RVHVS, having two normally closed MOVs in series in each vent path, which must be operated separately, minimizes the probability of inadvertent actuation, Finally, the valves being MOVs allow for closing of the valves; thereby satisfying the prevention of irreversible actuation requirement. Thus, the design of the SDVs as part of the RCS high-point vent system meets the requirements of 10

CFR 50.46(b), 10 CFR 50.46a, GDC 17 and GDC 34, allowing for a single failure of an active, electrical or mechanical component. Accordingly, **RAI 48-840, Question 05.04.12-02**, is resolved and closed.

However, the pressurizer DVs are designed without a redundant flow path, thus prompting the staff to issue **RAI 48-840, Question 05.04.12-6**, which requested the applicant to justify this design feature. In response to **RAI 48-840, Question 05.04.12-6**, the applicant stated that since the DVs are primarily designed for severe accidents, system redundancy is not required. The staff finds this acceptable because the pressurizer SDVs are designed as a safety-grade system and because use of the DVs as a backup for or supplement to the RCSHPVS is considered an extra vent feature. Accordingly, **RAI 48-840, Question 05.04.12-6**, is resolved and closed.

In response to **RAI 48-840, Question 05.04.12-3** regarding potential other RCS high points that may require to providing vents, the applicant stated that based on its evaluation of the RCS isometric drawing Fig 4.4-1, the only high points in the RCS that could accumulate non-condensable gases are the reactor vessel head, pressurizer and the steam generator U tubes. The staff has reviewed the RCS isometric drawing and agrees with the applicant's assessment that these are the RCS high points. As it is impractical to vent the large number of individual U tubes, and not required according to 10 CFR 50.46a, and as operating procedures are to be established to remove non-condensable gases from the U tubes, no other RCS vent systems are required. However, the applicant needs to develop one or more COL information items as necessary for the COL applicant/licensee to develop operating procedures for the RCSHPV system, and for this issue, that specifically address removal of non-condensable gasses from the SG U-tubes. This RAI question is closed/unresolved, but will be tracked under **Open Item 05.04.12-5**.

In **RAI 48-840, Question 05.04.12-4**, the staff asked if an orifice in the RCS high-point vent system is needed to avoid that an inadvertent opening of vent flow path or a pipe break in the vent line that would unnecessarily challenge the ECCS. In its response to **RAI 48-840, Question 05.04.12-04**, the applicant stated that flow restrictors such as orifices are not provided in the RCS high-point vent system and the valve arrangement precludes the possibility of inadvertently opening a vent flow path. In addition, a postulated break of the vent line is bounded by the LOCA analyzed in Section 15.6.5. The staff finds this acceptable since the clarification for Item II.B.1 in NUREG 0737 does not require flow restrictors in the RCSHPVS if isolation valves can be arranged to preclude inadvertent opening of a flow path. The design of RCSHPVS satisfies this design criterion. Therefore, **RAI 48-840, Question 05.04.12-04**, is resolved and closed.

The RCSHPVS are part of the RCPB and are designed, fabricated, erected and tested to have an extremely low probability of abnormal leakage, a rapidly propagating failure, and of a gross rupture (GDC 14). The mechanical portions of the high-point vents are designed and tested to the requirements of ASME Code Section III Class 1 as indicated in DCD Section 5.4.12.1. The above features comply with the requirements of 10 CFR 50.55a and GDCs 1, 14 and 30 as they relate to components of the RCPB. Electrical equipment and components are seismic and dynamically qualified in accordance with IEEE-344. Since the high-point vent valve actuators are designed as IEEE Class 1E components and are specified by the design to be qualified for harsh environments in accordance with DCD Section 3.11, this design meets the environmental qualification

(EQ) requirements of 10 CFR 50.49. Design verification (EQ) and installation bounded by the qualification are to be verified as prescribed in the associated ITAAC. This RAI is being tracked under Open Item 05-04-12-5.

An SRP acceptance criterion of Section 5.4.12 specifies that procedures should be developed to use the vent paths to remove gases that may inhibit core cooling using the SG U-tubes. In addition, it specifies that the procedures to operate the vent system should consider when venting is needed and when it is not needed, taking into account a variety of initial conditions, operator actions, and necessary instrumentation). In response to **RAI 48-840, Question 05.04.12-5**, regarding the need for RCS high-point vent system operating procedures, the applicant provided discussion on the high-level guidelines for developing the required operating procedures. However, detailed operating procedures have not been made available for the staff review. If the applicant cannot provide detailed operating procedures in response to **RAI 48-840, Question 05.04.12-5**, the staff will accept a COL (licensee) information item specifying that COL applicant or licensee fulfill this acceptance criterion. This will be tracked as **Open Item 05.04.12-05.**

The high-point vents are operated from the main control room (MCR) where valve position displays are also available, thus meeting the requirements of 10 CFR 50.46a and GDC 19. RPV water level indication is also provided in the MCR to aid operators.

The design and installation arrangement of the high-point vents provides the capability for necessary periodic, inservice inspection and testing. DCD Section 3.9.6 and Table 3.9.2-14 provide the detailed requirements for the IST program for the high-point vent valves. DCD Section 6.6 describes the ISI requirements for the high-point vents and Section 5.2.4 provides the program descriptions for PSI and periodic ISI for all RCPB components and piping, including the high-point vents to meet the requirements of GDC 36.

The ITAAC associated with the high-point vents is presented in Tier 1 Section 2.4.2. The staff review of the ITAAC is addressed in Section 14.3.4 of this FSER.

No technical specifications were identified in the DCD applicable to the high-point vents. The staff find this consistent with the criteria set forth in 10 CFR 50.36, and it is therefore acceptable.

The staff performed its review of the US-APWR RCS high-point vent design in accordance with Section 5.4.12 of the SRP as discussed below.

5.4.12.4.1 Reactor Vessel Head Vent System

The reactor vessel head vent system (RVHVS) is designed to enhance natural circulation of the reactor coolant by eliminating non-condensable gases in the upper plenum of the reactor vessel. The RVHVS arrangement consists of a flow path that diverges into two parallel flow paths each with two redundant 1-inch motor-operated remote manual valves connected in series. A 1-inch vent pipe is located near the center of the reactor vessel head. The system discharges to the pressurizer relief tank (PRT). The motor-operated remote manual valves in the redundant vent flow path are normally closed and powered by the safety-related Class 1E power supply system. The

RVHVS is operated from the MCR. The valves are qualified to Institute of Electrical and Electronics Engineers (IEEE) Standard (Std) 344.

The RVHVS is designed so that a single failure of the remotely operated vent valves, power supply, or control system does not prevent isolation of the vent path. The two redundant isolation valves in series minimize the possibility inadvertently open a vent path. The staff notes that a break of the vent line on the reactor vessel head would result in a small-break LOCA. The DCD assets that such a small-break LOCA would be no greater than 1 inch diameter. The staff finds that such a break is similar to the hot-leg-break LOCA analyzed in Section 15.6.5 of the DCD. The staff finds that these analysis results also apply to a RVHVS vent line break and are therefore acceptable.

The RV head vent system consists of safety-grade equipment. The piping and equipment from the vessel head vent up to, and including, the second normally closed motor operated valves constitute part of the RCPB, and are designed and fabricated to ASME Code Section III, Class 1, requirements. The staff finds that, with the exception of open items, the RVHVS as described in the DCD meets the acceptance criteria specified in Section II of Section 5.4.12 of the SRP, 10 CFR 50.34(f)(2)(vi) and TMI Action Plan Item II.B.1 as described in NUREG 0737 and, therefore, is acceptable.

5.4.12.5 Combined License Information Items

The US-APWR DCD does not identify any COL action item for the RCS high-point vent system. If the applicant cannot provide detailed operating procedures for the RCS high-point vent system in response to **RAI 48-840, Question 05.04.12-5**, the staff requires that the applicant propose an action item for COL applications to provide the RCS high-point vent system operating procedures as discussed in the staff evaluation above.

5.4.12.6 Conclusions

On the basis of the staff's evaluation of the information in the DCD on the RCSHPVS and the applicant's satisfactory responses to the RAI questions discussed above, with the exception of **RAI 48-840, Question 05.04.12-5,** regarding operating procedures for the RCS high-point vent system to close out **Open Item 05.04.12-05**, the staff concludes that the design of the RCS high-point vent system as described in the DCD is acceptable and meets the relevant requirements of 10 CFR 50.34(f)(2)(vi), 10 CFR 50.46(b), 10 CFR 50.46a, 10 CFR 50.49, and 10 CFR 50.55a, and GDCs 1, 14, 17, 19, 30, and 36, and Appendix B to 10 CFR Part 50.