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No figures were included in this chapter.

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5 REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

5.0 Reactor Coolant System and Connected Systems

This chapter describes the staff's review of the U.S. EPR reactor coolant system (RCS). The RCS configuration is a conventional four-loop design. The reactor pressure vessel (RPV) is located at the center of the Reactor Building and contains the fuel assemblies. The reactor coolant flows from the RPV through the hot leg pipes to the steam generators (SGs) and returns to the RPV via the cold leg pipes, which contain the reactor coolant pumps (RCPs). The pressurizer (PZR) is connected to one hot leg via a surge line and to two cold legs by spray lines.

5.1 Summary Description

5.1.1 Introduction

The RCS is a closed, four-loop system designed to transfer heat generated by the reactor core, located in the RPV, to the secondary side of the SGs for plant power generation. The RCS is located in the Reactor Building.

5.1.2 Summary of Application

FSAR Tier 1: In U.S. EPR Final Safety Analysis Report (FSAR) Tier 1, Section 2.2.1, "Reactor Coolant System," the applicant states that the RCS is a closed, four-loop system. The RCS consists of one RPV, four SGs, four RCPs, one PZR, one pressurizer relief tank (PRT), and the piping that connects the components. The RCS is a safety-related system that is in continuous operation whenever irradiated fuel is in the reactor core. The RCS components (RPV, RCPs, SGs, and PZR) are supported by the RCS component supports. The RCS component supports maintain the integrity of the reactor coolant pressure boundary (RCPB). The supports are designed to account for the movement of the components due to thermal expansion and contraction. The functional arrangement of the RCS is shown in FSAR Tier 1, Figure 2.2.1-1, "RCS Functional Arrangement." The location of the RCS equipment is as listed in FSAR Tier 1, Table 2.2.1-1, "RCS Equipment Mechanical Design."

The RCS provides the following safety-related functions:

- The RCPB provides the second barrier against radioactive product leakage.
- The PZR regulates pressure and provides overpressure protection.
- The RCS transfers decay heat from the reactor core to the SGs or to the residual heat removal system.
- The RCS allows for safe depressurization down to residual heat removal (RHR) system operating pressures.

- The water of the RCS is used as a neutron moderator, neutron reflector, and solvent for concentrated boric acid solutions. The RCS receives borated water from the chemical and volume control system (CVCS) and from the extra borating system (EBS).

The RCS provides the following non-safety-related functions:

- The RCS provides forced circulation of reactor coolant between each SG and the reactor core.

- In case of a total loss of heat removal through the SGs, the RCS performs the bleed function in the feed and bleed mode of core cooling in concert with the low head safety injection and residual heat removal (LHSI/RHR) system.

- Primary depressurization system valves lower RCS pressure in the event of a severe accident.

Detailed mechanical, electrical, and instrumentation information associated with the reactor coolant system design is specified in FSAR Tier 1, Tables 2.2.1-1 through 2.2.1-3.

FSAR Tier 2: The applicant has provided an FSAR Tier 2 description of the RCS in Section 5.1, summarized here, in part, as follows:

The RCS transfers the heat generated in the core during power operation to the secondary side of the SGs, where steam is produced to drive the turbine generator. Borated, demineralized water is circulated in the RCS at a flow and temperature consistent with achieving reactor core thermal-hydraulic performance. The water also acts as a neutron moderator, neutron reflector, and solvent for the boron used as a chemical shim. During operation, a saturated water-steam mixture within the PZR is maintained in equilibrium by electrical heaters and water sprays and is adjusted as necessary to control RCS pressure. Steam can be formed by the heaters or condensed by the PZR spray to minimize pressure variations within the RCS due to contraction and expansion of the reactor coolant associated with normal plant operation or accident conditions. PZR safety-relief valves (PSRVs) allow for steam discharge from the RCS if necessary. Discharged steam is piped to the PRT, where the steam is condensed and cooled by mixing with water. The RCS includes the following major components: RPV and reactor vessel internals; control rod drive mechanisms (CRDMs); RCP, drive motor, and auxiliaries; SGs; reactor coolant piping; PZR; PRT; post-accident high point vents; PSRVs; primary depressurization valves; and component and pipe supports and restraints. The RCS also includes piping and valves connected to the major components, component insulation, and RPV closure head equipment.

ITAAC: Item 2.1 in FSAR Tier 1, Table 2.2.1-5, “RCS ITAAC,” states that an inspection will be conducted to verify that the as-built RCS conforms to the functional arrangement shown on FSAR Tier 1, Figure 2.2.1-1. Item 2.4 in FSAR Tier 1, Table 2.2.1-5 states that an inspection will be performed to verify that the equipment listed in FSAR Tier 1, Table 2.2.1-1, “RCS Equipment Mechanical Design,” is located as listed in FSAR Tier 1, Table 2.2.1-1. Item 3.9 in FSAR Tier 1, Table 2.2.1-5 states that a test of the RCS will be performed to ensure that the measured gaps meet the specification requirements for the necessary component supports. Additional inspections, tests, analyses, and acceptance criteria (ITAAC) will be performed to verify the detailed FSAR Tier 1 mechanical, electrical, and instrumentation information associated with the RCS.

Technical Specifications: The Technical Specifications (TS) associated with FSAR Tier 2, Section 5.1 are given in FSAR Tier 2, Chapter 16, "Technical Specifications," Sections 3.4, "Reactor Coolant System," and B 3.4 "Reactor Coolant System."

5.1.3 Technical Evaluation

A detailed description of the content of the application for this section is provided in FSAR Tier 2, Sections 5.2, "Integrity of the Reactor Coolant Pressure Boundary", 5.3, "Reactor Vessel"; and 5.4, "Component and Subsystem Design."

5.2 Integrity of the Reactor Coolant Pressure Boundary

This section evaluates the measures employed to provide and maintain the integrity of the reactor coolant pressure boundary for the plant design lifetime. Consistent with the definition in Title 10 of the *Code of Federal Regulations* (10 CFR) 50.2, "Definitions," the RCPB includes all pressure containing components, such as pressure vessels, piping, pumps, and valves which are part of the RCS or connected to the RCS, up to and including those stated below:

- The outermost containment isolation valve in system piping which penetrates primary reactor containment
- The second of two valves usually closed during normal reactor operation in system piping which does not penetrate primary reactor containment
- The RCS safety and relief valves

5.2.1 Compliance with Codes and Code Cases

5.2.1.1 *Compliance with 10 CFR 50.55a*

5.2.1.1.1 Introduction

This Safety Evaluation Report (SER) section addresses use of acceptable codes (i.e., American Society of Mechanical Engineers (ASME) *Boiler and Pressure Vessel Code* (B&PV Code) and the ASME *Code for Operation and Maintenance of Nuclear Power Plants* (OM Code)), code editions, and addenda required by 10 CFR 50.55a, "Codes and Standards," in the AREVA NP, Inc. (AREVA or the applicant), design certification for the U.S. EPR.

5.2.1.1.2 Summary of Application

FSAR Tier 1: The FSAR Tier 1 description of several of the Nuclear Island systems, for example the 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), CVCS, EBS, and the fuel-handling system (FHS), indicated that components in these systems will be designed, constructed, and tested in accordance with Section III, "Rules for Construction of Nuclear Facility Components," of the ASME B&PV. The system descriptions also indicated that a fatigue analysis would be done for Class 1 piping and components, as required by Section III of the ASME Code.

FSAR Tier 2: FSAR Tier 2, Section 5.2.1.1, "Compliance with 10 CFR 50.55a," states that the RCPB components are designed and fabricated as Class 1 components in accordance with Section III, of the ASME B&PV Code, except for components that meet the exclusion requirements of 10 CFR 50.55a(c), which are designed and fabricated as Class 2 components. FSAR Tier 2, Table 3.2.2-1, "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 Regulatory Guide (RG) 1.26, "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants," as specified in FSAR Tier 2, Section 3.2, "Classification of Structures, Systems, and Components."

FSAR Tier 2, Section 5.2.1.1 states that the code of record for the design of the U. S. EPR is the 2004 edition of the ASME B&PV Code (no addenda). The application of Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," of the 2004 edition of the ASME B&PV Code is described in FSAR Tier 2, Section 5.2.4, "Inservice Inspection and Testing of the RCPB," and FSAR Tier 2, Section 6.6, "Inservice Inspection of Class 2 and 3 Components." The application of the ASME OM Code is described in FSAR Tier 2, Section 3.9.6, "Functional Design, Qualification, and Inservice Testing Programs for Pumps, Valves, and Dynamic Restraints."

A combined license (COL) applicant that references the U.S. EPR design will need to identify any subsequent editions and addenda that are used (e.g., for construction, inspection, or examination) and reconcile any differences between the later version of the Code and the requirements in the 2004 edition.

ITAAC: There are no ITAAC associated with this area of review.

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

5.2.1.1.3 Regulatory Basis

The relevant requirements of U.S. Nuclear Regulatory Commission (NRC) regulations for compliance with 10 CFR 50.55a and the associated acceptance criteria are given in Section 5.2.1.1 of NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants − LWR Edition," (hereafter referred to as NUREG-0800 or 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. 10 CFR Part 50, Appendix A, General Design Criterion (GDC) 1, "Quality Standards and Records," as it relates to the requirement that systems, structures, and components (SSCs) be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety-function performed.

2. 10 CFR 50.55a, "Codes and Standards," as it relates to the establishment of minimum quality standards for the design, fabrication, erection, construction, testing, and inspection of RCPB components and other fluid systems of boiling- and pressurized-water reactor nuclear power plants by compliance with appropriate editions of published industry codes and standards. Pursuant to 10 CFR 50.55a, components important to safety are subject to the following requirements:

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

- 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.

Acceptance criteria adequate to meet the above requirements include:

1. RG 1.26, "Quality Group Classification and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants," 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.

2. Other system − specific acceptance criteria are listed in NUREG-0800 Section 5.2.1.1.

5.2.1.1.4 Technical Evaluation

The staff has reviewed the FSAR for compliance with 10 CFR Part 52, and has discussed the classification of U.S. EPR SSCs in Section 3.2 of this report. FSAR Tier 2, Table 3.2.2-1 specifies the 2004 Edition of the ASME B&PV Code that is incorporated by reference in 10 CFR 50.55a for Class 1, 2 and 3 components in the U.S. EPR design. For the reasons discussed in the following paragraphs, the staff finds that the FSAR meets the requirements of 10 CFR 50.55a and GDC 1 for the construction of SSCs important to safety to quality standards by ensuring that RCPB components, as defined in 10 CFR 50.55a, are classified properly in FSAR Tier 2, Table 3.2.2-1 as ASME Code Section III, Class 1 (Quality Group A) components, except for those which meet the 10 CFR 50.55a(c) exclusion requirements. For the same reasons, the staff also finds that the FSAR meets the 10 CFR 50.55a requirements by classifying properly other U.S. EPR components in FSAR Tier 2, Table 3.2.2-1 as ASME B&PV Code, Section III (Quality Group B) or Class 3 (Quality Group C).

In its review of FSAR Tier 2, Section 5.2.1.1, the staff noted a statement in the FSAR that the COL applicant would identify subsequent ASME Code editions or addenda that might be used and would determine the consistency of the U.S. EPR design with construction practices. In request for additional information (RAI) 41, Question 05.02.01.01-1 the staff requested that the applicant clarify 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 for the U.S. EPR Design Certification. In an August 15, 2008, response to RAI 41, Question 05.02.01.01-1, the applicant stated that the statement in question would be deleted from FSAR Tier 2, Section 5.2.1.1 and Table 1.8-2, "U.S. EPR Combined License Information Items." The applicant noted that if a COL applicant decides to use a different code edition than the U.S. EPR code of record, the COL applicant would have to take a departure from the FSAR and justify the use of the different code edition in the COL application. The staff has confirmed that Revision 1 of FSAR, dated May 29, 2009, Tier 2, Section 5.2.1.1 and FSAR Tier 2, Table 1.8-2 was revised as committed in the RAI response and clarifies the use of the U.S. EPR code of record. Accordingly, the staff finds that the applicant has adequately addressed this issue and, therefore, the staff considers RAI 41, Question 05.02.01.01-1 resolved.

A COL applicant referencing the U.S. EPR design must ensure that the design complies 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 a COL applicant proposes an ASME Code edition and addenda different from that specified in the design certification FSAR as the code of record, the COL applicant must identify in its application the later code editions and addenda for staff review and approval.

On December 11, 2008, the applicant requested an exemption for the U.S. EPR Standard Design Certification (DC) pursuant to 10 CFR 52.7, "Specific exemptions," and 10 CFR 50.12, "Specific exemptions," from 10 CFR 50.55a, which incorporates by reference the ASME Code through 2003 Addenda at the time of application, while the Code of record for the design of U.S. EPR is the 2004 Edition (no addenda) of the ASME Code. The requested exemption is unnecessary, since the revision of 10 CFR 50.55a was issued in September 2008 to incorporate by reference the 2004 Edition of the ASME Code.

FSAR Tier 2, Section 5.2.1.1 stated that the RCPB component classification complies with the requirements of GDC 1 and 10 CFR 50.55a. FSAR Tier 2 Table 3.2-1 also listed the RCPB components, including pressure vessels, piping, pumps, and valves, along with the applicable component codes. The applicant stated that other safety-related plant components are classified in accordance with RG 1.26, as specified in FSAR Tier 2, Section 3.2. However, in RAI 51, Question 05.02.01.01-2, the staff noted that Table 3.2-1 is not found in the applicant's submitted FSAR Tier 2. In a September 30, 2008, response to RAI 51, Question 05.02.01.01-2, the applicant stated that the correct table number should be FSAR Tier 2, Table 3.2.2-1 and FSAR Tier 2, Section 5.2.1.1 will be revised to make this correction. The staff has confirmed that Revision 1 of FSAR, dated May 29, 2009, Tier 2, Section 5.2.1.1 was revised as committed in the RAI response. Accordingly, the staff finds that the applicant has adequately addressed this issue and, therefore, the staff considers RAI 51, Question 05.02.01.01-2 resolved. The staff's review of the classification of these components is described in Section 3.2.2 of this report.

FSAR Tier 2, Section 5.2.1.1 states that a COL applicant that references the U.S. EPR design certification will identify subsequent ASME Code Editions or Addenda that may be used and will determine the consistency of the U.S. EPR design with construction practices (including inspection and examination methods) reflected within the subsequent code editions and addenda identified in the COL application. FSAR Tier 2, Section 5.2.1.1 also states that the Code of record for the design of U.S. EPR is the 2004 Edition (no addenda) of the ASME Code. In RAI 51, Question 05.02.01.01-3, the staff requested that the applicant explain how the U.S. EPR design for piping and components will meet 10 CFR 50.55a(b)(1)(ii) and (iii) by using the ASME 2004 Edition. In a September 30, 2008, response to RAI 51, Question 05.02.01.01-3, the applicant indicated that the description of how the U.S. EPR piping and components comply with 10 CFR 50.55a(b)(1)(ii) and (iii) is provided in AREVA Topical Report ANP-10264NP, "U.S. EPR Piping Analysis and Pipe Support Design Topical Report." However, the staff has determined that in a November 20, 2007, response to an RAI regarding the Topical Report (RAI EPR-3), the applicant did not address how 10 CFR 50.55a(b)(1)(ii) is satisfied for the U.S. EPR design. The staff noted that the code of record for U.S. EPR design is the 2004 Edition of ASME code, and the piping stress analysis and seismic design are performed in accordance with the 1993 Addenda to the 1992 Edition to meet the requirement of 10 CFR 50.55a(b)(1)(iii). However, the use of either the 2004 Edition or the 1993 Addenda is disallowed by 10 CFR 50.55a(b)(1)(ii). In a follow-up RAI 365, Question 05.02.01.01-5, the staff requested that the applicant provide the technical basis of how 10 CFR 50.55a(b)(1)(ii) is addressed while using the 2004 Edition and 1993 Addenda. **RAI 365, Question 05.02.01.01-5 is being tracked as an open item.**

FSAR Tier 2, Section 3.12.2, “Codes and Standards,” references Section 2.0 of AREVA Topical Report ANP-10264NP, Revision 0, September 2006, for applicable codes and standards for the design of piping and pipe supports. Section 2.1 of ANP-10264NP states that piping analysis and pipe support design for the U.S. EPR addressed in this topical report use the 2001 ASME Code, Section III, Division 1, with the 2003 addenda, as the base code with restrictions identified in 10 CFR 50.55a(b)(1). The staff determined that this was inconsistent with the code and standard cited in FSAR Tier 2, Section 5.2.1.1, that the code of record for the design of U.S. EPR is the 2004 Edition (no addenda) of the ASME Code. In RAI 51, Question 05.02.01.01-4, the staff requested that the applicant confirm whether the code of record for the design of the U.S. EPR is the 2004 Edition or the 2001 Edition through 2003 addenda of the ASME Code. In a September 30, 2008, response to RAI 51, Question 05.02.01.01-4, the applicant confirmed that the code of record for the design of the U.S. EPR is the 2004 Edition of the ASME Code. The applicant is requested to revise Section 2.1 of ANP-10264NP to state that the code of record is the 2004 Edition of the ASME Code and not the 2001 ASME Code, Section III, Division 1, 2003 addenda. **RAI 51, Question 05.02.01.01-4, which is associated with the above request, is being tracked as a confirmatory item.**

5.2.1.1.5 Combined License Information Items

No COL information items have been identified in FSAR Tier 2, Table 1.8-2, “U.S. EPR Combined License Information Items,” related to compliance with Codes. In view of the above discussion, the staff finds this acceptable.

5.2.1.1.6 Conclusions

Except for the open items and confirmatory items discussed above, the staff concludes that the information provided in the FSAR with respect to the use of codes and standards is acceptable and the applicant correctly identified the ASME Code of record to apply to the U.S. EPR design. With respect to the ASME Code of record, the application is sufficient to support compliance with the requirements of 10 CFR Part 50, Appendix A, 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.

5.2.1.2 *Compliance with Applicable Code Cases*

5.2.1.2.1 Introduction

This SER section discusses the use of Code Cases to the B&PV Code. In general, a Code Case is developed by the ASME based on inquiries from the nuclear industry associated with possible clarification or modification of the Codes or alternates to the Code. The ASME B&PV Standards Committee has eliminated Code Case expiration dates since March 11, 2005. Therefore, all Code Cases will be automatically reaffirmed and remain available for use unless annulled by the ASME B&PV Standards Committee. ASME Code Cases acceptable to staff are published in RG 1.84, “Design, Fabrication, and Materials Code Case Acceptability, ASME Section III”; RG 1.147, “Inservice Inspection Code Case Acceptability, ASME Section XI, Division 1”; and RG 1.92, “Operation and Maintenance Code Case Acceptability, ASME OM Code,” in accordance with the requirements of 10 CFR 50.55a(b)(4), (b)(5) and (b)(6).

5.2.1.2.2 Summary of Application

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

FSAR Tier 2: In FSAR Tier 2, Section 5.2.1.2, “Compliance with Applicable Code Cases,” the applicant states that Code Cases related to ASME B&PV Code, Section III, acceptable for use in the U.S. EPR design, subject to the limitations specified in 10 CFR 50.55a, are listed in RG 1.84. The applicant refers to FSAR Tier 2, Table 5.2-1, “ASME Section III Code Cases,” for the use of ASME B&PV Code, Section III Code Cases in the U.S. EPR design. FSAR Tier 2, Table 5.2-1 lists ASME Code Case N-60-5 (February 15, 1994), “Material for Core Support Structures Section III, Division I,” and ASME Code Case N-71-18 (December 8, 2000), “Additional Materials for Subsection NF, Class 1, 2, 3, and MC Supports Fabricated by Welding, Section III, Division 1.” FSAR Tier 2, Section 5.2.1.2 states that a COL applicant that references the U.S. EPR design certification will identify any additional ASME Code Cases to be used. Code Cases pertaining to ASME B&PV Code Section III, Division 2, are addressed in FSAR Tier 2, Section 3.8, “Design of Category I Structures.”

FSAR Tier 2, Section 5.2.1.2 states that Code Cases related to ASME B&PV Code, Section XI, “Rules for Inservice Inspection of Nuclear Power Plant Components,” acceptable for use are listed in RG 1.147. FSAR Tier 2, Section 5.2.1.2 refers to FSAR Tier 2, Section 5.2.4, “Inservice Inspection and Testing of the RCPB,” and FSAR Tier 2, Section 6.6, “Inservice Inspection of Class 2 and 3 Components,” for additional discussion of these code cases.

FSAR Tier 2, Section 5.2.1.2 states that the ASME OM Code Cases acceptable for use are listed in RG 1.192, “Operation and Maintenance Code Case Acceptability, ASME OM Code.” FSAR Tier 2, Section 5.2.1.2 refers to FSAR Tier 2, Section 3.9.6, “Functional Design, Qualification, and Inservice Testing Programs for Pumps, Valves, and Dynamic Restraints,” for additional discussion of ASME OM Code Cases.

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

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

5.2.1.2.3 Regulatory Basis

The relevant requirements of NRC regulations for compliance with applicable Code Cases and the associated acceptance criteria are given in Section 5.2.1.2 of NUREG-0800 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, as it relates to the requirement that 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. This requirement is applicable to both pressure-retaining and non-pressure-retaining SSCs that are part of the reactor coolant pressure boundary, 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 adequacy and applicability.

2. 10 CFR 50.55a, as it relates to the rule that 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 compliance with appropriate editions of specified published industry codes and standards.

Acceptance criteria adequate to meet the above requirements include:

1. RG 1.84 as it relates to ASME Section III Code Cases

2. RG 1.147 as it relates to ASME Section XI Code Cases

3. RG 1.192 as it relates to ASME OM Code Cases

5.2.1.2.4 Technical Evaluation

The staff has reviewed the FSAR for compliance with 10 CFR Part 52. Acceptable ASME Code Cases that may be used for the U.S. EPR standard plant are those either conditionally or unconditionally approved in applicable NRC regulatory guides, as incorporated into 10 CFR 50.55a(b)(4), (5), and (6), and that are in effect at the time of design certification.

FSAR Tier 2, Section 5.2.1.2 states that ASME Section III Code Cases acceptable for use in the U.S. EPR design, subject to the limitations specified in 10 CFR 50.55a, are listed in RG 1.84. FSAR Tier 2, Table 5.2-1, "ASME Section III Code Cases," lists the specific Code Cases used in the U.S. EPR design. However, there are only two code cases listed in FSAR Tier 2, Table 5.2-1. In RAI 51, Question 05.02.01.02-3, the staff requested that the applicant provide a complete list of Section III Code Cases used for the U.S. EPR design, as mentioned in FSAR Tier 2, Sections 3.8, "Design of Category 1 Structures"; 4.5, "Reactor Materials"; 5.4; and 10.3, "Main Steam Supply System," including those used for the design of piping and pipe supports in Topical Report ANP-10264NP.

In a September 30, 2008, response to RAI 51, Question 05.02.01.02-3, the applicant stated that Code Case N-284-1, "Metal Containment Shell Buckling Design Methods, Section III, Division 1, Class MC," which is identified in FSAR Tier 2, Section 3.8 will be added to FSAR Tier 2, Table 5.2-1. There are no additional code cases identified in FSAR Tier 2, Sections 4.5, 5.4, and 10.3. AREVA Topical Report ANP-10264NP identifies the following Code Cases: N-122-2, "Procedure for Evaluation of the Design of Rectangular Cross Section Attachments on Class 1 Piping, Section III, Division 1"; N-318-5, "Procedure for Evaluation of the Design of Rectangular Cross Section Attachments on Class 2 or 3 Piping, Section III, Division 1"; N-319-3, "Alternate Procedure for Evaluation of Stresses in Butt Welding Elbows in Class 1 Piping, Section III, Division 1"; N-391-2, "Procedure for Evaluation of the Design of Hollow Circular Cross Section Welded Attachments on Class 1 Piping, Section III, Division 1"; and N-392-3, "Procedure for Evaluation of the Design of Hollow Circular Cross Section Welded Attachments on Classes 2 and 3 Piping, Section III, Division 1." Four of these code cases (i.e., Code Cases N-122-2, N-318-5, N-391-2, and N-392-3) were incorporated in the 2004 Edition of the ASME Code, Division 1 Appendix Y, and have been approved since the 2004 Edition of the Code has been incorporated into 10 CFR 50.55a. The applicant indicated that the remaining Code Case (i.e., Code Case N-319-3) is an alternate procedure for evaluation of stresses in butt welded elbows in Class 1 piping and will be added to FSAR Tier 2, Table 5.2-1. RG 1.84 lists ASME Code Case N-319-3 as acceptable without conditions. The staff has confirmed that Revision 1 of FSAR Tier 2, Table 5.2-1, dated May 29, 2009, was revised as committed in the RAI response. Accordingly, the staff finds that the applicant has adequately addressed this issue and, therefore, the staff considers RAI 51, Question 05.02.01.02-3 resolved.

FSAR Tier 2, Section 5.2.1.2 indicated that ASME Section XI Code Cases acceptable for use for inservice inspection (ISI), subject to the limitations specified in 10 CFR 50.55a, are listed in RG 1.147 and described in FSAR Tier 2, Section 5.2.4, "Inservice Inspection and Testing of the

RCPB," and FSAR Tier 2, Section 6.6, "Inservice Inspection of Class 2 and 3 Components." ASME OM Code Cases acceptable for use for inservice testing (IST), subject to the limitations specified in 10 CFR 50.55a, are listed in RG 1.192 and described in FSAR Tier 2, Section 3.9.6, "Functional Design, Qualification, and Inservice Testing Programs for Pumps, Valves, and Dynamic Restraints." FSAR Tier 2 also indicated that a COL applicant that references the U.S. EPR design certification should identify additional ASME Code Cases to be used. In RAI 51, Question 05.02.01.02-4, the staff requested that the applicant provide a complete list of Code Cases used for U.S. EPR ISI and a complete list of OM Code Cases used for operation and maintenance associated with IST. The staff indicated that listed Code Cases should be listed as acceptable by RGs 1.147 or 1.192. If the code cases were not listed in RGs 1.147 or 1.192, the staff requested that the applicant provide justification to meet requirements in accordance with 10 CFR 50.55a(a)(3).

In a September 30, 2008, response to RAI 51, Question 05.02.01.02-4, the applicant stated that FSAR Tier 2, Section 5.2.4.1.6, "Code Exemptions," says, "No exceptions from code required examinations for Class 1 preservice inspection or inservice inspection are required for the U.S. EPR." FSAR Tier 2, Section 6.6 states that no exemption to or relief from code requirements are requested, or code cases invoked, for Class 2 or Class 3 pre-service or inservice inspection requirements. FSAR Tier 2, Section 3.9.6 does not invoke any code cases for the OM code. The applicant indicated that Code Case N-729-1, "Alternative Examination Requirements for PWR Reactor Vessel Upper Heads with Nozzles Having Pressure-Retaining Partial-Penetration Welds," will be added to FSAR Tier 2, Table 5.2-1. The NRC regulation in 10 CFR 50.55a(g)(6)(ii)(D) requires the use of ASME Code Case N-729-1 with conditions. The applicant's use of this code case is further discussed and evaluated in Section 5.2.4 of this report. The staff has confirmed that Revision 1 of FSAR, dated May 29, 2009, Tier 2, Table 5.2-1 was revised as committed in the RAI response. Accordingly, the staff finds that the applicant has adequately addressed this issue and, therefore, the staff considers RAI 51, Question 05.02.01.02-4 resolved.

FSAR Tier 2, Section 5.2.1.2 states the ASME B&PV Code, Section XI Code Cases acceptable for use are listed in RG 1.147 and described in FSAR Tier 2, Sections 5.2.4 and 6.6. FSAR Tier 2, Section 5.2.1.2 states that the ASME OM Code Cases acceptable for use are listed in RG 1.192 and described in FSAR Tier 2, Section 3.9.6. In RAI 41, Question 05.02.01.02-1, the staff requested that the applicant identify (or specify that the COL Applicant will identify) any ASME Code Cases applicable to the ASME B&PV Code, Section XI, and ASME OM Code to be used in the U.S. EPR design. In an August 15, 2008, response to RAI 41, Question 05.02.01.02-1, the applicant stated that FSAR Tier 2, Sections 5.2.4 and 6.6 indicate that no Code Cases applicable to Class 1, 2, or 3 preservice or inservice inspection requirements are requested for the U.S. EPR. The applicant stated that ASME OM Code Cases are described in FSAR Tier 2, Section 3.9.6, but did not identify any ASME OM Code Cases planned to be used for the U.S. EPR. The applicant addressed the application of any ASME OM Code Cases as part of its justification for the inservice testing provisions in FSAR Tier 2, Section 3.9.6. This issue is discussed as part of the NRC review of FSAR Tier 2, Section 3.9.6, and the staff considers RAI 41, Question 05.02.01.02-1 resolved.

FSAR Tier 2, Section 5.2.1.2 discusses the use of the two ASME Code Cases listed in FSAR Tier 2, Table 5.2-1. ASME Code Case N-60-5, "Material for Core Support Structures Section III, Division I," listed in FSAR Tier 2, Table 5.2-1 is unconditionally accepted in RG 1.84 and is, therefore, acceptable. 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," listed in FSAR

Tier 2, Table 5.2-1 is conditionally accepted in RG 1.84. In RAI 199, Question 05.02.01.02-2, the staff requested that the applicant indicate the components that will be fabricated using N-71-18 and the materials specifications and grades that will be used. In a September 30, 2008, response to RAI 199, Question 05.02.01.02-2, the applicant indicated that Code Case N-71-18 will be used for the reactor pressure vessel support shell, gussets (both on the inside diameter and outside diameter of the shell), special gusset (i.e., modified gusset for load handling features), gusset for access plate, key connecting plate, and the access plate, and that these components use materials specification SA-572, grade 50 (plate). The staff verified that the use of this material is acceptable for its intended application in accordance with Code Case N-71-18 and, therefore, RAI 199, Question 05.02.01.02-2 is resolved.

FSAR Tier 2, Table 5.2-1 notes that use of Code Case N-284-1 is addressed in FSAR Tier 2, Section 3.8. FSAR Tier 2, Section 3.8.2.4, "Design and Analysis Procedures," specified that ASME B&PV Code Case N-284-1, with additional guidance in RG 1.193, "ASME Code Cases Not Approved for Use," will be used to evaluate buckling of shells with more complex geometries and loading conditions than in ASME B&PV Code, Section III, Division 1, Subsection NE, for containment penetrations. The staff evaluated the use of Code Case N-284-1 as part of its review of FSAR Tier 2, Section 3.8.2, as discussed in Section 3.8 of this report.

Compliance with the requirements of these Code Cases will result in a component quality that is commensurate with the importance of the safety functions of the components. This satisfies the requirements of GDC 1 and 10 CFR 50.55a and, therefore, is acceptable.

5.2.1.2.5 Combined License Information Items

Table 5.2.1.2-1 below provides a list of code case related COL Information Item numbers and descriptions from FSAR Tier 2, Table 1.8-2:

Table 5.2.1.2-1 U.S. EPR Combined License Information Items

Item No.	Description	FSAR Tier 2 Section
5.2-1	A COL applicant that references the U.S. EPR design certification will identify additional code cases to be used.	5.2.1.2

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 U.S. EPR certified design. The COL information item is sufficient to alert a COL applicant who seeks to use additional code cases to identify them to the staff. No additional COL information items need to be included in FSAR Tier 2, Table 1.8-2 for Code Cases.

5.2.1.2.6 Conclusions

With the exception of the open item identified above, the staff finds that the ASME Code Cases identified in the FSAR are acceptable as specified in the applicable NRC regulatory guides, or have been reviewed and found acceptable by the staff for use in the EPR design, as discussed above. Except for the open item, the staff concludes that the information provided in the FSAR with respect to the use of ASME Code Cases is acceptable and sufficient to support compliance with the requirements of 10 CFR Part 50, Appendix A, 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.

5.2.2 Overpressure Protection

5.2.2.1 *Introduction*

Overpressure protection for both the primary and secondary sides of the RCPB must be provided with sufficient margins to assure the integrity of the RCS. The upper head of the pressurizer has four large nozzles, one for each of the three PSRVs and one for the PZR depressurization system (PDS) lines. The PSRVs protect the reactor coolant pressure boundary from overpressure on the primary side during all modes of operation including low temperature operation. The U.S. EPR design does not use power operated relief valves (PORVs) for overpressure protection for the RCPB on the primary side of the RCS.

Overpressure protection on the secondary side of the RCPB is provided by main steam safety valves (MSSVs) in conjunction with main steam relief trains (MSRTs) that limit challenges to the MSSVs and regulate cooldown rates to prevent overcooling of the RCS during transients and anticipated operational occurrences (AOOs). Secondary side overpressure protection is addressed in FSAR Tier 2, Section 10.3.

5.2.2.2 *Summary of Application*

FSAR Tier 1: The FSAR Tier 1 information associated with this section is found in FSAR Tier 1, Section 2.2.1, “Reactor Coolant System,” which describes the pressurizer and its function of regulating the RCS pressure during normal operational modes and, in conjunction with PSRVs, providing overpressure protection during AOOs and design-basis accidents (DBA).
FSAR Tier 1, Section 2.2.1 also states that PSRVs open below their maximum design setpoint to provide relief capacity.

FSAR Tier 1, Section 2.8.2, “Main Steam System,” provides a description of the main steam supply system (MSSS) and the features and equipment included for secondary side overpressure protection. FSAR Tier 1, Section 2.8.2 also includes information on the design features of the MSSVs and the MSRTs.

FSAR Tier 2: FSAR Tier 2, Section 5.2.2.1, “Design Bases,” the applicant describes the PSRVs as follows: The PSRVs are part of, and provide overpressure protection for, the RCPB. The opening set pressures and capacity of the PSRVs are sufficient to limit the RCS pressure to less than 110 percent of the RCPB design pressure during any condition of normal operation, including AOOs. The bounding design transient for RCPB overpressure is a turbine trip at full power. The PSRVs maintain the RCS pressure below brittle fracture limits when the RCPB is stressed under operating, maintenance, testing, and postulated accident conditions, including low temperature operation, so that the probability of rapidly propagating fracture in the RCPB is minimized. The PSRVs can perform their overpressure protection functions at power and low temperature operations assuming a single failure or malfunction of an active component.

At power, a spring-operated pilot valve actuates the main safety valve of the PSRV assembly. During low temperature conditions, two solenoid-operated pilot valves in series actuate the main safety valve. Direct indication of PSRV main disk position is provided in the main control room (MCR). A surge line allows unobstructed flow between the PZR and RCS. The surge line is sized to provide an allowable pressure drop between the RCS loops and the PZR during overpower transients. The PZR is sized to preclude actuation of the PSRVs, in conjunction with normal spray, during normal operational transients.

During reactor cool down, an alarm is generated from the wide range cold leg temperature instrumentation once low-temperature overpressure protection (LTOP) permissive conditions are met, allowing the operator to validate the permissive to enable LTOP. The LTOP system enables temperature and PSRV set points to be selected so that the peak RPV pressure does not exceed the 10 CFR Part 50, Appendix G, "Fracture Toughness Requirements," limit. Low temperature RCPB overpressure events include mass and heat input events.

FSAR Tier 2, Section 5.4.10, "Pressurizer," presents design-basis information on the PZR and states that the PZR will accommodate, among other things, RCS coolant expansion and contraction to limit RCS mass removal and addition. In addition, the PZR will preclude the opening of PSRVs during normal operational transients.

An integral part of overpressure protection of the RCPB is the steam generator secondary side's MSSS. The MSSS has two distinct features that comprise the secondary side overpressure protection capability. One feature is the MSSVs, and the second feature is the MSRTs. These two features complement each other and are described in detail in FSAR Tier 2, Section 10.3.

The MSSVs provide ultimate overpressure protection against the steam generators exceeding 110 percent of their ASME Section III design pressure. The MSRTs function to limit challenges to the MSSVs during normal operational transients.

ITAAC: The ITAAC associated with FSAR Tier 2, Section 5.2.2, "Overpressure Protection," are given in FSAR Tier 1, Section 2.2.1. The ITAAC includes testing the PSRVs to verify they open (within 0.70 seconds including pilot valve opening time), and testing each PSRV to ensure that it lifts below the maximum design setpoint 17.94 MPa (2,600.4 psia) and provides relief capacity (≥ 300,011 kg/hr (661,400 lbm/hr) at 17.58 MPa (2,535 psig)).

The ITAAC associated with FSAR Tier 2 Section 5.2.2 are given in FSAR Tier 1, Section 2.8.2. They include the MSSVs providing relief capacity ≥ 645,052 kg/hr (1,422,073 lbm/hr) at ≤ 10.54 MPa (1,504 psig) for the first valve and ≤ 10.69 MPa (1,535 psig) for the second valve. In addition, each MSRT provides relief capacity ≥ 1,290,104 (2,844,146 lbm/hr) at ≤ 9.86 MPa (1,414.7 psia).

Technical Specifications: U.S. EPR TS 3.4.10 would require that a licensee have three PSRVs operable, and that each PSRV have lift setting of ≥ 17.24 MPa (2,484.3 psig) and ≤ 17.94 MPa (2,585.7 psig) when in Modes 1, 2, and 3 and in Mode 4 when all RCS cold leg temperatures are greater than the low temperature overpressure protection arming temperature specified in the pressure-temperature limits report (PTLR). U.S. EPR TS 3.4.11 would require that low temperature overpressure protection be established when in Mode 5, in Mode 6 when the reactor vessel head is on, and in Mode 4 when any RCS cold leg temperature is less than or equal to the LTOP arming temperature specified in the PTLR. TS Bases B 3.4.10 and 3.4.11 provide the bases for TS 3.4.10 and 3.4.11.

TS 3.7.1 would require that two MSSVs per steam generator be operable for Modes 1, 2, and 3. TS 3.7.4 is the surveillance requirements (SR) section for the MSRTs and is of secondary concern since TS 3.7.1 is controlling for overpressure protection. However, TS 3.7.4 controls the SR for the MRST valves, which supplement the MSSVs. TS Bases B 3.7.1 and B 3.7.4 provide the bases for TS 3.7.1 and 3.7.4.

5.2.2.3 *Regulatory Basis*

The relevant requirements of NRC regulations for overpressure protection 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 the identification and evaluation of codes and standards to determine their applicability, adequacy, and sufficiency and be supplemented or modified as necessary to ensure a quality product in keeping with the required safety function.

2. GDC 15, “Reactor Coolant System Design,” 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 RCPB are not exceeded during any condition of normal operation, including AOOs.

3. GDC 30, “Quality of Reactor Coolant Pressure Boundary,” as it relates to material specifications of components which are part of the reactor coolant pressure boundary being designed, fabricated, erected, and tested to the highest quality standards practical.

4. GDC 31, “Fracture Prevention of Reactor Coolant Pressure Boundary,” 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 52.47(a)(8), “Contents of applications; technical information,” provides the requirement for design certification reviews to comply with the technically relevant portions of the TMI requirements in 10 CFR 50.34(f).

6. 10 CFR 50.34(f)(2)(x), “Contents of applications; technical information,” 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, “Clarification of TMI Action Plan Requirements.”

7. 10 CFR 50.55a as it relates to meeting material specifications of the ASME code and code cases as described in RG 1.84.

8. The applicable provisions of the ASME Code (e.g., ASME Section II in accordance with ASME Section III, NB-2000, NB 3500, NB-7000, and NB-7511.1).

Acceptance criteria adequate to meet the above requirements include:

1. Acceptable application of material Code Cases, as described in RG 1.84.

2. Branch Technical Position (BTP) 5-2, “Overpressurization Protection of Pressurized-Water Reactors While Operating at Low Temperatures,” as it relates to the LTOP system operability as defined in paragraph B.2.

5.2.2.4 *Technical Evaluation*

As described in more detail below, the primary side of the RCPB is protected from overpressure during all modes of operation including low temperature operation. This protection is provided by the PSRVs. The portion of the RCPB that is exposed to significant pressure from the secondary side, the steam generator tubes, is protected from overpressure by the MSSVs and the MSRTs. The overpressure protection provided by the MSSVs and the MSRTs is presented in the FSAR Tier 2, Section 10.3, "Main Steam Supply System."

The staff has reviewed FSAR Tier 2, Section 5.2.2 in accordance with the guidance, review procedures, and regulations outlined and identified in the NRC Standard Review Plan (SRP) 5.2.2 of NUREG-0800 and BTP 5-2. In addition, the staff reviewed FSAR Tier 2, Section 10.3 using the applicable guidance in SRP Section 10.3 that pertains to the overpressure protection features for the secondary side of the steam generators.

Design Bases

The applicant summarizes the design basis for the PZR as follows: The U.S. EPR RCS design includes the PZR and the three PSRVs as components of the RCPB. Three PSRVs are arranged on top of the PZR for overpressure protection of the RCPB. The PZR is specifically designed by its volumetric size and shape in conjunction with PZR spray capability to minimize pressure changes and challenges to the PSRVs during normal operations including AOOs. The PSRVs are not challenged during normal operational transients. In addition, the PZR is connected to the RCS loop 3 hot leg piping by a surge line approximately 9.14 m (30 ft) long. The surge line provides an increase in resistance to pressure surges for the PZR. Each PSRV inlet connects to a nozzle on the pressurizer upper head, and each outlet pipe connects to a 40.64 cm (16 in.) discharge header approximately 36.6 m (120 ft) long. The discharge header collects the discharge of the three PSRVs, and directs the flow to the pressurizer relief tank, as described in FSAR Tier 2, Section 5.4.11, "Pressurizer Relief Tank." The PSRVs are attached by welding to nozzles located on the upper head region of the PZR as shown in FSAR Tier 2, Figure 5.4-6, "Pressurizer Assembly." In addition, schematic representations of the PSRVs are illustrated in FSAR Tier 2, Figure 5.4-8, "Pressurizer Safety Relief Valve Schematic." Design parameters of the PSRVs are provided in FSAR Tier 2, Table 5.4-9, "Pressurizer Safety Relief Valve Design Parameters." Valve stem position indication sensors are provided on the PSRVs and readout in the MCR in accordance with 10 CFR 50.34(f)(2)(xi). In addition, down stream temperature, level and pressure instrumentation capability are available in the PRT to detect possible PSRV leakage.

The applicant described the PSRVs as follows: The PSRVs are pilot (spring and solenoid) operated relief valves designed to ASME Code Section III, NB-7520 and NB-7530 for overpressure protection at power and at low temperature operation. As such, they can accommodate the passage of steam, two phase mixtures and sub-cooled liquid. As part of the RCPB, the PSRVs are designed to meet the requirements of ASME Section III Class 1 components (thereby meeting GDC 1, GDC 30, and 10 CFR 50.55a) as outlined in FSAR Tier 2, Section 3.2, Table 3.2.2.1, "Classification Summary," and as shown in FSAR Tier 1, Table 2.2.1-1, "RCS Equipment Mechanical Design." Other components of the overpressure protection system are classified in accordance with RG 1.26. Also, in accordance with RG 1.29, FSAR Tier 2, Table 3.2.2.1 classifies the seismic category for the overpressure protection system related components of the RCS. For instance, the discharge piping from the PSRVs is designed to ASME Seismic Category II requirements. As described in the FSAR, the discharge piping is routed to the PRT which collects and condenses the steam during operational Modes 1, 2, 3, and 4. The staff reviewed the FSAR Tier 1 information including FSAR Tier 1, Table 2.2.1-5, "RCS ITAAC," for completeness and ITAAC compliance of the overpressure protection system related components and finds the information to be acceptable. In addition, the staff's evaluation of the capability of the overpressure protection system related components to perform their intended safety functions is in Sections 5.4.10, 5.4.11, and 5.4.13 of this report.

The opening pressure setpoints are chosen and relief capacities of the PSRVs are designed to be sufficient to limit pressure excursions in the RCS to less than 110 percent of the design pressure of 17.58 MPa (2,550 psia) during any condition of normal operation including AOOs, thereby meeting GDC 15. During RCS Modes 1, 2, 3, and 4, the PSRVs are considered as passive overpressure protection devices and do not rely on electrical power sources. The spring loaded pilot valves are designed in accordance with the requirements of ASME Code Section III, NB-7511.I. The PSRVs are subjected to a qualification and testing program in accordance with 10 CFR 50.34(f)(2)(x). This program is designed to demonstrate that the PSRVs perform properly under all fluid conditions expected under normal operating conditions including transients, accidents, and anticipated transients without scram (ATWS). In addition, the PSRVs will be subjected to the inservice testing program requirements as provided in FSAR Tier 2, Section 3.9.6.3.6, "Inservice Testing Program for Safety and Relief Valves," which the staff evaluates in Section 3.9.6.3 of this report.

The PSRVs provide overpressure protection to maintain the RCPB below brittle fracture limits when the RCS is stressed under conditions such as operation, maintenance, testing, and postulated accident conditions including low temperature operation. This overpressure protection prevents the RCPB from behaving in a brittle manner, and the probability of rapid propagation fracture is minimized or eliminated, thereby meeting GDC 31. FSAR Tier 2, Section 5.2.1, "Compliance with Codes and Code Cases," and FSAR Tier 2, Table 5.2-1 provides a listing of the applicable ASME Section 3 Code Cases used in the U.S. EPR and provided in RG 1.84.

The MSSS presented in FSAR Tier 2, Section 10.3 describes, in part, the overpressure protection provisions provided for the secondary side of the steam generators. The staff reviewed the provisions in FSAR Tier 2, Section 10.3 as they relate to the secondary side overpressure protection of the RCPB to assure complete, seamless, and consistent design-basis coverage for the entire RCPB.

The main steam line from each steam generator contains two MSSVs as described in FSAR Tier 2, Section 10.3.2.2, "Component Description," and design data presented in Table 10.3-2, "Design Data for Main Steam Safety Valves." The MSSVs are designed in accordance with the ASME Code Section III, seismic Category I as shown in FSAR Tier 1, Table 2.8.2-1, "MSS Equipment Mechanical Design." The MSSVs are designed to limit the peak pressure excursions in the secondary side of the steam generators to less than 110 percent of the design pressure of 10.00 MPa (1,450 psia) during any condition of normal operation including AOOs, thereby meeting GDC 15. The MSSVs discharge to the atmosphere through connected vent stacks.

The main steam line from each steam generator also contains, in series, one MSRT, which is composed of one ASME Section III designed fast opening main steam relief isolation valve (MSRIV), normally closed, and one main steam relief control valve (MSRCV) normally open. The MSRT is a part of the overpressure protection for the secondary side of the steam generators. The FSAR Tier 1 information for the MSSVs and MSRTs is located in Section 2.8.2. The staff reviewed this information and concluded that the ITAAC in FSAR Tier 1, Section 2.8.2 is satisfactory and sufficient to verify the attributes of the system as installed.

The MSRT relief setpoints and capacities of the MSRT are capable of preventing the steam generator from exceeding 110 percent of its design pressure from a full loss of load event in conjunction with a reactor trip. During mild pressure transient events, the MSRT controls the steam generator pressure to prevent the MSSVs from opening. The MSRT discharges to atmosphere if the turbine bypass system is not available. In addition, the MSRCVs also provide a safety function by being able to control the MSRT steam flow and preventing over cooling of the RCS through the steam generator and mitigating the effects of a stuck open MSRIV. The MSRCVs are automatically opened an increasing percentage as a function of core thermal power. The staff's evaluation of the capability of the MSRT to perform its safety functions is in Chapter 15 of this report.

The MSRIVs are designed to Subsection NC of the ASME Section III, Division 1, including Article NC-7000. The MSSVs and MSRTs are safety-related components and are located within the safety-related portion of the MSSS. FSAR Tier 2, Table 3.2.2-1 provides the classification summary for the components of the MSSS. Since the ITAAC in FSAR Tier 1, Table 2.8.2-3 will require verification of fabrication and installation of the pressure relief valves in accordance with the ASME Code, the staff finds this acceptable.

Overpressure Protection at Power

During normal power operation, all three PSRVs are in service on the primary side, and two MSSVs and one MRST for each steam generator are in service on the secondary side. These valves are of simple design and construction and have low failure rates in service, as demonstrated by operating experience. Accordingly, the staff considers this combination of overpressure protection to be sufficient to meet the requirements of GDC 15 related to providing an extremely low probability of failure of the RCPB during transients.

The staff verified that the PRT design described in FSAR Tier 2, Section 5.4.11 provides a tank volume large enough to accommodate the anticipated discharges from the PSRVs during the bounding AOO event, which is a turbine trip (TT) at full reactor power conditions as specified in FSAR Tier 2, Chapter 15, "Transient and Accident Analyses," FSAR Tier 2, Section 15.2.2, "Turbine Trip." The FSAR Tier 2, Table 3.9.1-1, "Summary of Design Transients," states that the event is expected to occur about 60 times over the 60-year planned life of a plant using the U.S. EPR design. The TT event was analyzed using the method described in the NRC-approved, "Codes and Methods Applicability Report for the U.S. EPR," (ANP-10263(P)) Reference 7 of FSAR Tier 2, Section 5.2. The staff reviewed the initial conditions and assumptions established for the beginning of the TT event and considered them to be conservative. The results of the TT analysis presented in FSAR Tier 2, Section 15.2.2, show that the three PSRVs can maintain pressure below the ASME limit; however, the pressure margin calculated is small, less than 6.9 kPa (1 psia), between the steam generator design pressure and the ASME 110 percent limit. Although the margin is small, the peak pressure is calculated using an approved methodology and conservative initial conditions; therefore, the staff finds the design acceptable in this regard.

The staff also reviewed the two limiting transients that increase RCS inventory to determine the margin available to prevent a solid liquid PZR, which would significantly challenge the RCPB. The two transients are presented in the FSAR Tier 2, Sections 15.5.1, “Inadvertent Operation of the Emergency Core Cooling System (ECCS) or Extra Borating System,” and 15.5.2, “Chemical Volume and Control System Malfunction that Increases Reactor Coolant Inventory.” The events were analyzed using the NRC-approved topical report, “Codes and Methods Applicability Report for the U.S. EPR.” The staff again reviewed the initial conditions and assumptions established for the beginning of the transients and has determined them to be conservative. The analysis results showed that the maximum PZR level reach in each transient was 98.7 percent for the EBS event and 98.6 percent for the CVCS event. In either case, the PZR did not completely fill. Since the maximum PZR level is calculated using an approved methodology and conservative initial conditions, the staff finds this result acceptable.

Low Temperature Overpressure Protection

LTOP features of the U.S. EPR for the RCPB are designed to be in accordance with the guidance provided in BTP 5-2. The U.S. EPR design uses the PSRVs for LTOP, and other low pressure systems connected to the RCS are not used. LTOP needs of the U.S. EPR only rely on two of the PSRVs to satisfy regulatory requirements including the loss of one PSRV as a single-active failure. LTOP is provided during startup and shutdown operations when the RCPB conditions are at or below the reactor vessel (RV) brittle fracture protection temperature limits.

During reactor cooldown operations, RCS cold leg temperature instrumentation is used to activate a permissive signal in the MCR indicating the need for LTOP. Operators in the MCR acknowledge the permissive alert alarm to activate the LTOP. The LTOP equipment is designed in accordance with Institute of Electrical and Electronics Engineers (IEEE) Std 603. FSAR Tier 2, Section 7.2.1.3.12, “P17 Permissive,” provides a description of the permissive controls. LTOP compliance with IEEE Std 603 is discussed in Chapter 7 of this report. The permissive temperature and setpoints for the PSRVs for LTOP are determined to meet the requirements of 10 CFR Part 50, Appendix G limits. FSAR Tier 2, Section 5.3.2, “Pressure-Temperature Limits, Pressurized Thermal Shock, and Charpy Upper-Shelf Energy Data and Analyses,” addresses the pressure-temperature (P-T) limits for the RV. The initial P-T limit curves are calculated based on the physical properties, chemical composition, and the expected life of the reactor vessel.

Two solenoid-operated pilot valves in series provide protection during LTOP against spurious opening or closure failure of the PSRVs. These pilot valves are powered from separate electrical sources that are backed up by uninterruptible power supplies.

FSAR Tier 2, Section 5.2.2.2.2, “Design Evaluation,” discusses bounding analyses performed for mass and heat input events and presents the applicant’s conclusion that the analysis demonstrates the adequacy of the LTOP features for the U.S. EPR over the range of expected conditions. The staff was unable to find the analyses demonstrating the adequacy of the LTOP. The staff issued RAI 117, Question 05.02.02-3 to address this concern.

In a December 15, 2008, response to RAI 117, Question 05.02.02-3, the applicant stated that for LTOP events, the low temperature reactor coolant pressure boundary overpressure events included mass input events and heat input events. The bounding events were selected from the list in FSAR Tier 2, Section 5.2.2.2.2 and are described in further detail as follows:

Mass Addition

The applicant stated as follows: To identify the limiting mass addition event, the potential flowrates of each source of injection were compared. The EBS flowrate is smaller than injection by either the medium head safety injection (MHSI) or CVCS charging pumps; therefore, EBS activation is not a limiting case. The accumulators have a fixed volume and their release into the RCS was determined to be within the 10 CFR Part 50, Appendix G limits even without PSRV actuation. Therefore, this case (i.e., accumulator injection) was not identified for analysis. The remaining cases in FSAR Tier 2, Section 5.2.2.2.2 were identified for analysis using the methodology described below.

The most limiting single failure for the charging pump injection case is a failure of the overpressure protection system (i.e., a failure of one PSRV to open) where full pump flow is entering the RCS and letdown is isolated. With the miniflow lines open, the MHSI pumps provide a minimal RCS injection flow rate at pressures near the LTOP setpoints. Therefore, to produce a mass addition for the MHSI pump case that has the potential to challenge the LTOP system, a conservative assumption was made that a MHSI miniflow line fails closed, resulting in more mass addition to the RCS.

Heat Addition

With respect to heat addition, the applicant stated as follows: To identify the limiting heat addition case for the LTOP event, the potential heating from each source of heat injection was evaluated. The most limiting event is the event that adds energy to the RCS at the greatest rate until the time of PSRV opening. The most rapid temperature rise will cause the greatest increase in pressure after the PSRVs open. The energy input of the pressurizer heaters is bounded by both the failure of RHR or a transient where the secondary system is initially 27.78 °C (50 °F) greater than the primary system coupled with the spurious activation of an RCP. A conservative calculation (i.e., highest RCS temperature of RHR initiation with two RCPs running) determined the rate of RCS temperature increase resulting from a failure of the RHR system. The LTOP analysis demonstrates that the spurious activation of an RCP while the secondary system remains at an elevated temperature results in a faster rise in RCS temperature and thus is the limiting case. For the heat addition cases, the most limiting single failure is a failure of the overpressure protection system (i.e., a failure of one PSRV to open).

Each selected mass addition or heat addition event was analyzed for initial RCS temperatures of 21.11 °C (70 °F), 37.78 °C (100 °F), 65.56 °C (150 °F), 93.33 °C (200 °F), and 121.11 °C (250 °F) to validate the LTOP setpoint over the entire temperature range at which the LTOP system is operable.

Analytical method

The LTOP transient analysis used RELAP5/MOD2-B&W to model the U.S. EPR nuclear steam supply system (NSSS) using the methodology in BAW-10169P-A, B&W Safety Analysis Methodology for Recirculating Steam Generator Plants (Reference 1). The model was modified to provide conservative results for LTOP scenarios.

Based on the applicant's response to RAI 117, Question 05.02.02-3, the staff issued a follow-up RAI 332, Question 05.02.02-11 with respect to the applicability to the U.S. EPR of the methodology, identified in BAW-10169P-A, "B&W Safety Analysis Methodology for Recirculating Steam Generator Plants," B&W Fuel Company, October 1989, to request that the applicant perform LTOP analyses for the U.S. EPR design. **RAI 332, Question 05.02.02-11 is being tracked as an open item.**

FSAR Tier 2, Section 5.2.2.10, “Testing and Inspection,” provides a discussion on the testing of the solenoid-operated valves prior to entering the LTOP mode of operation during plant shutdown. This testing mode and sequencing is acceptable to the staff and meets the guidance of BTP 5-2.

Initial Testing Program

Initial testing of the PSRVs is described in FSAR Tier 2, Section 14.2.12.3.14, “Pressurizer Safety Relief Values (Test No. 037)” and FSAR Tier 2, Section 14.2.12.12.5, “Primary Depressurization (Test No. 151).” The objectives of these tests are to verify the setpoints of the PSRVs and their flow capabilities. FSAR Tier 2, Test No. 037 tests the high setpoint operation of the PSRV; however, the staff observed that a test to verify solenoid actuator operation is also needed. Also, the reference to FSAR Tier 2, Section 5.4.11 for the acceptance criteria appeared to be incorrect. The correct reference appears to be FSAR Tier 2, Sections 5.2.2 and 5.4.13. In RAI 117, Question 05.02.02-4 and RAI 98, Question 14.02-39, the staff requested that the applicant address the need to include provisions to verify the solenoid actuator operation of the pressurizer safety valve in Test No. 37 and revise the references to be FSAR Tier 2, Sections 5.2.2 and 5.4.13, not FSAR Tier 2, Section 5.4.11.

In a December 15, 2008, response to these RAIs, the applicant indicated that FSAR Tier 2, Section 14.2.12, Test No. 037, which tests the PSRVs, will be revised to verify the remote manual function and to reference FSAR Tier 2, Section 5.2.2 and FSAR Tier 2, Section 5.4.13, instead of FSAR Tier 2, Section 5.4.11.

Also, the applicant indicated that FSAR Tier 2, Section 14.2.12, Test No. 037 will be revised to better match terminology in FSAR Tier 2, Section 5.4.13. The staff confirmed that Revision 1 of the U.S. EPR FSAR, dated May 29, 2009, contains the changes committed to in the RAI response. Therefore, based on these corrections, the tests described above are acceptable to the staff and RAI 117, Question 05.02.02-4 and RAI 98, Question 14.02-39 is resolved and closed. The staff determined that no additional PSRV initial testing is necessary.

Initial testing of the MSSVs and MSRTs is provided in FSAR Tier 2, Section 14.2, “Initial Plant Test Program,” Test No. 062 for the MSSVs, and Test No. 148 and Test No. 152 for the MSRTs. The purpose of Test No. 062 and Test No. 148, respectively, is to verify the proper operation of the MSSV and MRST designs as described in FSAR Tier 2, Sections 7.3 and 10.3, and to demonstrate their electrical independence and redundancy of the power supplies for safety-related functions. In addition, the purpose of Test No. 152 is to verify the flow path of the MSRT during partial cooldown, setpoint reduction upon receipt of a safety injection signal, and response of the MSRT (MSRCV and MSRIV) to simulated signals. The staff reviewed these tests and judged them to be acceptable for their intended purpose. No additional initial testing was deemed necessary by the staff.

ITAAC

The ITAAC associated with FSAR Tier 2, Section 5.2.2 is given in FSAR Tier 1, Section 2.2.1. The ITAAC associated with FSAR Tier 2, Section 10.3, as it relates to the overpressure protection features for the steam generator secondary side are given in FSAR Tier 1, Section 2.8.2. The staff reviewed both FSAR Tier 1, Sections 2.2.1 and 2.8.2 and finds them to be necessary and sufficient for their intended features for RCPB overpressure protection since the ITAAC includes checking of the valves locations, types of valves, valve calibration, among other things. No additional ITAAC items were identified by the staff.

Technical Specifications

The staff reviewed TS 3.4.10 and 3.4.11 for conditions applicable to the PSRVs and finds them adequate to meet the overpressure protection for the primary side of the RCPB, as described in Chapter 16, “Technical Specifications,” of this report. In addition, the staff also reviewed the TS Bases Sections B 3.4.10 and B 3.4.11, and agrees with the bases.

In addition, the staff reviewed TS 3.7.1 applicable to the MSSVs. TS 3.7.1 would require that two MSSVs per steam generator be operable for Modes 1, 2, and 3. Actions provided within TS 3.7.1 account for using MSRTs to supplement inoperable MSSVs, time frames for restoring inoperable MSSVs to an operable status, and requirements for reduced power limits or shutdown depending on the combination of factors involved. The standard used is to assure that through safety analyses, the RCPB pressure does not exceed 110 percent of the design limit on either the primary or secondary sides. The staff has determined that TS 3.7.1 is adequate to ensure that the plant is operated within the assumptions of the design basis analysis.

The staff also reviewed TS 3.7.4 applicable to the MSRTs for their function of providing a method for cooling the plant to RHR conditions should the condenser be unavailable. This method of cooling is performed in conjunction with the feedwater or EFWS. Since TS 3.7.1 is controlling with regard to overpressure protection for the RCPB, the only interface of concern with TS. 3.7.4 is the SR section for the MSRTs. The staff determined that SR 3.7.4 is adequate to ensure that MSRT surveillance will occur at intervals sufficient to ensure the plant is operated within the assumptions of the design basis analysis. In addition, the staff reviewed TS Bases B 3.7.1 and 3.7.4, agrees with the bases.

In addition to the evaluation of FSAR Tier 2, Section 5.2.2, the staff noted that the inlets to the PRSVs and PDS valves have water loop seals and, therefore, reviewed FSAR Tier 2, Section 3.9.3.2.1, “Class 1 Pressurizer Safety Relief Valves,” as it related to an analysis of the dynamic loads developed by the loop seals.. The staff was unable to identify a reference or discussion with respect to an analysis of the dynamic loads developed by the loop seals. The staff evaluated loop seal dynamic loading extensively in NUREG-0737 and found that the dynamic loads caused by loop seals were very significant. The staff issued RAI 332, Question 05.02.02-12 to request that the applicant provide a reference or discussion on the dynamic loading analysis of the loop seals during the discharge of the safety valves. **RAI 332, Question 05.02.02-12 is being tracked as an open item.**

During the staff’s review, no information was identified that should be considered as combined license information items.

5.2.2.5 *Combined License Information Items*

For overpressure protection, no COL information items have been identified in FSAR Tier 2, Table 1.8-2. The staff finds this acceptable, because the proposed ITAAC and initial plant test program assure that the overpressure protection features will be constructed in accordance with the certified design.

5.2.2.6 *Conclusions*

The overpressure protection features of the U.S. EPR reactor coolant system design were reviewed and evaluated by the staff, as described above. The scope of the review included the design bases of the overprotection of both the primary and secondary sides of the RCPB, the bounding AOOs analyses, analytical methods, codes and standards, testing and verification, instrumentation and technical specifications.

Except for open items discussed above, and for the reasons set forth above, the staff concludes that the overpressure protection design meets the requirements of GDC 1, GDC 15, GDC 30, and GDC 31; 10 CFR 50.34(f)(2)(x); 10 CFR 50.34(f)(2)(xi), or 10 CFR 52.47(a)(8) as applicable; and 10 CFR 50.55a.

5.2.3 Reactor Coolant Pressure Boundary Materials

5.2.3.1 *Introduction*

This section addresses the materials which make up the reactor coolant pressure boundary. RCPB materials are fabricated and selected to maintain pressure boundary integrity for the design life of the plant. RCPB materials are selected using approved ASME Codes and code cases (e.g., N-60-5, N-71-18). Ferritic low-alloy and carbon steel RCPB components have either austenitic stainless steel or nickel-based alloy overlay cladding on surfaces exposed to the reactor coolant.

5.2.3.2 *Summary of Application*

FSAR Tier 1: The FSAR Tier 1 information associated with this section is found in FSAR Tier 1, Section 2.2.1. It provides that the RCPB provide the second barrier against radioactive product leakage. Detailed FSAR Tier 1 mechanical, electrical, and instrumentation information associated with the RCPB are specified in Tables 2.2.1-1 through 2.2.1-3.

FSAR Tier 2: The applicant has provided an FSAR Tier 2 description of the materials used in the RCPB in Section 5.2.3, summarized here in part as follows:

The application specifies the grade or type and final metallurgical conditioning performed on the materials used for the RCPB. RCPB materials and reactor coolant chemistry are specified for compatibility to avoid degradation or failure in environmental conditions associated with normal operations, maintenance, testing, and postulated accidents. The RCS coolant chemistry is controlled to minimize negative impacts of coolant chemistry on materials integrity, fuel rod corrosion, fuel design performance, and radiation fields, and reactor coolant is routinely analyzed for verification. The application provides a description of the fabrication and processing of both ferritic and austenitic stainless steels used in the RCPB.

Ferritic low alloy and carbon steels used in principal pressure retaining applications have either austenitic stainless steel or nickel-base alloy corrosion resistant overlay cladding on all surfaces that are exposed to the reactor coolant. The overlay clad ferritic type base materials receive a post weld heat treatment. Low alloy steel pressure boundary forgings have limited sulfur content not exceeding 0.008 wt percent. Overlay clad low alloy steel pressure boundary materials have specified ASTM (American Society for Testing of Materials) grain size 5 or finer.

Austenitic stainless steel materials with primary pressure retaining applications are used in the solution annealed and water quenched (or rapidly cooled) condition. Austenitic stainless steel materials and weld metal have limited carbon content not to exceed 0.03 wt percent with the exception of stabilized austenitic stainless steel Grade 347 used for the CRDM pressure housing. Austenitic stainless steel material and weld filler metal in contact with RCS primary coolant has limited cobalt content not to exceed 0.05 wt percent. Austenitic stainless steel material in contact with RCS primary coolant has limited sulfur content not to exceed 0.02 wt percent. Austenitic stainless steel welds in RCS piping, including surge line piping, have delta ferrite content limited to a ferrite number (FN) between 5 and 10. Austenitic stainless steel weld materials for stainless steel weld joints in the balance of the RCPB system have delta content ferrite limited to a FN between 5 and 20. Because the concentration of oxygen, chlorides, and fluorides in the reactor coolant are controlled, stainless steel welds should not experience stress corrosion cracking during normal plant operation. Precipitation hardened stainless steel is used as a necked-down bolt for the control rod drive mechanism, and because of its location will not come in contact with reactor coolant. The RCP bolting is also external to the wetted pressure boundary.

NiCrFe Alloy 690 material has controlled chemistry, mechanical properties, and thermo-mechanical processing requirements that produce an optimum microstructure for resistance to intergranular corrosion. Alloy 690 materials used in primary pressure retaining applications are used in the solution annealed and thermally treated condition to optimize resistance to intergranular corrosion. NiCrFe Alloy 600 materials or Alloys 82/182 weld metal are not used in RCPB applications.

ITAAC: Items 3.12 and 3.13 in FSAR Tier 1, Table 2.2.1-5, “RCS Inspections, Tests, Analyses, and Acceptance Criteria,” state that inspections will be performed to verify that specifications exist for components and piping listed as ASME Section III in FSAR Tier 1, Table 2.2.1-1, “RCS Equipment Mechanical Design,” which includes the RCPB.

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

5.2.3.3 *Regulatory Basis*

The relevant requirements of NRC 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. 10 CFR Part 50, Appendix A, GDC 1, and GDC 30, as they relate to quality standards for design, fabrication, erection, and testing.

2. GDC 4, “Environmental and Dynamic Effects Design Bases,” as it relates to the compatibility of components with environmental conditions.

3. GDC 14, “Reactor Coolant Pressure Boundary,” and GDC 31, as they relate to minimizing the probability of rapidly propagating fracture and gross rupture of the RCPB.

4. 10 CFR Part 50, Appendix B, “Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants,” Criterion XIII, as it relates to onsite material cleaning control.

5. 10 CFR Part 50, Appendix G, 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. 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 properties, and the criteria for verifying conformance during production welding.

3. RG 1.36, “Nonmetalic Thermal Insulation for Austenitic Stainless Steel,” as it relates to acceptance criteria for compatibility of austenitic stainless steel with thermal insulation.

4. 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.

5. 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-allow steel safety-related components clad with stainless steel.

6. 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.

7. RG 1.71, “Welder Qualification for Areas of Limited Accessibility,” as it relates to welder qualification testing.

5.2.3.4 *Technical Evaluation*

The staff reviewed FSAR Tier 2, Section 5.2.3, “Reactor Coolant Pressure Boundary Materials,” in accordance with SRP Section 5.2.3, “Reactor Coolant Pressure Boundary Materials,” Revision 3, March 2007.

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, GDC 30, and 10 CFR 50.55a, 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, and by applying Code Cases included in RG 1.84, which is incorporated by reference in 10 CFR 50.55a.

The staff reviewed FSAR Tier 2, Section 5.2.3.1, “Materials Specifications,” and FSAR Tier 2, Table 5.2-2 “Materials Specifications for RCPB Components,” to determine the suitability of the

RCPB materials and compliance with ASME Code, Section III, GDC 1, GDC 30, and 10 CFR 50.55a.

FSAR Tier 2, Table 5.2-2 lists martensitic stainless steel materials SA-182 Grade F6NM and SA-479 (UNS S41500) for the control rod drive mechanism pressure housing. While SA-182 Grade F6NM is listed in ASME Code, Section II, Part D, Subpart I, Table 2A, for use in Class 1 systems, the staff noted that SA-479 UNS S41500 is not listed in Table 2A and, therefore, does not meet ASME Code, Section III, NB-2121 requirements and, thus, does not meet the requirements of 10 CFR 50.55a. In RAI 88, Question 05.02.03-1, the staff requested that the applicant delete SA-479 UNS S41500 material and, if necessary, provide an alternative material that meets ASME Code requirements. The staff noted that alternatively, the applicant may propose a Code Case to ASME Code to include SA-479 UNS S41500 in Table 2A. In a November 10, 2008, response, the applicant stated that it has submitted a request to ASME Code to extend the properties currently provided in Section II, Part D for SA-182 Grade F6NM (UNS S41500) to SA-479 (UNS S41500) material. The applicant expects the Code Case to be issued in the near future. This issue will remain an open item until ASME issues the applicant's Code Case, and the staff reviews its acceptability. **RAI 88, Question 05.02.03-1, which is associated with the above request, is being tracked as an open item.**

In RAI 88, Question 05.02.03-3, the staff requested, in part, that the applicant list the material specifications for pressurizer safety-relief valves. In addition, the staff requested that the applicant identify weld filler materials used to weld the various material types and combinations.

In a December 17, 2008, response, the applicant provided a proposed revision to FSAR Tier 2, Table 5.2-2. The applicant's proposed revision to FSAR Tier 2, Table 5.2-2 indicated that the materials specifications for the pressurizer safety-relief valves are not available, because a vendor has not been selected for these components. The staff also noted that the applicant's proposed revision to FSAR Tier 2, Table 5.2-2 did not list weld filler material specifications and classifications to be used to weld various material types and combinations in the RCPB as requested by the staff. In order for the staff to verify the applicant's compliance with ASME Code Section III, the staff issued RAI 199, Question 05.02.03-18.

In RAI 199, Question 05.02.03-18, the staff requested, in part, that the applicant modify FSAR Tier 2, Table 5.2-2 to include material specifications and grades for all valves, piping, fittings, or other components that form part of the reactor coolant pressure boundary, or provide a reference in FSAR Tier 2, Table 5.2-2 to the location of this information in the FSAR. In addition, the staff requested that the applicant modify FSAR Tier 2,Table 5.2-2 to list weld filler metal specifications and classifications used to weld various material types and combinations in the RCPB. In an April 23, 2009, response to RAI 199, Question 05.02.03-18, the applicant provided a proposed revision to FSAR Tier 2, Table 5.2-2 that lists the materials specifications and grades for pressurizer safety-relief valves and other RCPB valves. The staff reviewed the applicant's proposed revision of FSAR Tier 2, Table 5.2-2, related to valves, and finds it acceptable because the materials listed meet the requirements of ASME Code, Section III, Paragraph NB-2121, "Permitted Materials Specifications," which requires the use of materials listed in ASME Code, Section II, Part D, Subpart 1, Tables 2A and 2B. The staff verified that the materials identified by the applicant are acceptable materials for use in ASME Code, Section III, Class 1 valves. The staff will verify that the appropriate modifications are made in a revision to the FSAR. **RAI 199, Question 05.02.03-18 is being tracked as a confirmatory item.**

In an April 23, 2009, response to RAI 199, Question 05.02.03-18 the applicant also stated that weld filler material specifications are listed in FSAR Tier 2, Section 5.2.3.1 and that no revision to FSAR Tier 2, Table 5.2-2 is required. The staff notes that FSAR Tier 2,Table 5.2-2 lists weld filler material specifications and classifications for fabrication of the reactor coolant pumps and CRDMs but does not provide weld filler material specifications and classifications for fabrication of the RCPB piping, steam generators, or pressurizer. In addition, in a November 10, 2008, response to RAI 88, Question 05.02.03-12, the applicant indicated that F347 material will be used to fabricate part of the CRDM pressure housing. However, FSAR Tier 2, Table 5.2-2 does not list a forging specification for Grade 347 material. The staff issued RAI 278, Question 05.02.03-20 to address this concern. **RAI 278, Question 05.02.03-20 is being tracked as an open item.**

Except for the open items and confirmatory items above, the staff finds that the materials selected by the applicant for use in the RCPB as listed in FSAR Tier 2, Table 5.2-2 and described in FSAR Tier 2, Section 5.2.3.1, are suitable for their intended use and meet the requirements of the ASME Code, Section III, GDC 1, GDC 30, and 10 CFR 50.55a.

Compatibility of Materials with Reactor Coolant

The staff reviewed FSAR Tier 2, Section 5.2.3, in accordance with SRP Section 5.2.3, Revision 3, March 2007, as related to compatibility of materials with reactor coolant. The staff acceptance of the materials is based on meeting the requirements of GDC 14 and Appendix B to 10 CFR Part 50, Criterion XIII.

The staff reviewed the materials of construction of the RCPB exposed to reactor coolant identified in FSAR Tier 2, Table 5.2-2 for their compatibility with the coolant, contaminants, and radiolytic products. The listed materials are austenitic stainless steels and a nickel-base alloy (Alloy 690) which is used in the steam generator. Ferritic steel RCPB components are overlay clad on wetted surfaces with austenitic stainless steel weld metal. Austenitic stainless steel weld metal is used to join austenitic stainless steel components. Nickel-base weld filler metal is used to join dissimilar metals. As discussed above, these materials are identified by ASME specifications that are listed within Section II of the ASME B&PV Code.

For many years, pressurized water reactors have used nickel-base alloys which have demonstrated negligible general corrosion. However, operating experience has shown that nickel-base alloys, particularly Alloy 600, are susceptible to stress-corrosion cracking in primary coolant water. FSAR Tier 2, Section 5.2.3.5, "Prevention of Primary Water Stress-Corrosion Cracking for Nickel-Base Alloys," notes that Alloy 600 is not used in any RCPB component. Research has shown that increasing the chromium content of Alloy 600 reduces the potential for stress-corrosion cracking. The alloy implementing this concept is Alloy 690, which has a nominal chromium content of 28 percent compared to 16 percent chromium in Alloy 600. Therefore, Alloy 690 is an acceptable replacement for Alloy 600 for use in the reactor coolant environment. FSAR Tier 2, Table 5.2-2 specified Alloy 690 for steam generator tubing exposed to the primary coolant.

Austenitic stainless steels, used for many years in pressurized water reactors, have experienced negligible general corrosion. The only concern has been stress corrosion cracking. The staff guidance provided in RG 1.44 and RG 1.37, to avoid stress corrosion cracking, is addressed by the FSAR as follows:

- FSAR Tier 2, Section 5.2.3.4.2, “Cleaning and Contamination Protection Procedures,” adequately addresses the fabrication and process controls to minimize contamination with chloride and fluoride ions, cleaning prior to heat treatment of components, and cleanliness controls on flush water.

- FSAR Tier 2, Table 5.2-3, “Reactor Coolant Water Chemistry - Control Parameters,” adequately addresses the control of detrimental ions in the reactor coolant. Specifically, chloride less than 0.15 parts per million (ppm), fluoride less than 0.15 ppm, and at elevated temperatures dissolved oxygen less than 0.10 ppm.

- FSAR Tier 2, Section 5.2.3.4.1, “Prevention of Sensitization and Intergranular Corrosion of Austenitic Stainless Steels,” satisfies the controls to prevent sensitization of stainless steel materials by limiting carbon content to less than 0.03 percent, and by control of fabrication and welding processes.

The staff concludes the fabrication and cleaning controls follow the recommendations of RG 1.44, RG 1.37 and, therefore, satisfy Appendix B to 10 CFR Part 50 Criterion XIII with respect to cleaning RCPB components, so as to avoid stress corrosion cracking in RCPB components constructed of austenitic stainless steels.

The staff reviewed the materials of construction of the RCPB for compatibility with external insulation. Certain types of thermal insulation contain leachable chloride and fluoride ions that can cause accelerated corrosion of RCPB components if the insulation is wetted. FSAR Tier 2, Section 5.2.3.4.3, “Compatibility of Construction Materials with External Insulation and Reactor Coolant,” notes that reflective stainless steel insulation is used on RCPB components wherever clearances permit. The applicant stated further as follows: Reflective stainless steel insulation has no leachable chloride or fluoride ions. In areas of little clearance, reflective stainless steel insulation cannot be used. In these areas, only insulating materials that follow the guidance of RG 1.36 are used.

The main element of RG 1.36 is that the insulating materials have low leachable chloride and fluoride ion concentrations. By its nature, reflective stainless steel insulation contains no leachable chloride or fluoride ions. Accordingly, the staff concludes the materials of construction for the RCPB components are compatible with the thermal insulation used in these areas and are in conformance with the recommendations of RG 1.36.

Some RCPB components are constructed of ferritic materials. Surfaces of these components wetted by the primary coolant are clad with austenitic stainless steel. However, in the event of a primary coolant leak, the unclad portion of the ferritic materials may be wetted and as a result will show increased general corrosion rates. The boric acid corrosion control program (BACCP), described in FSAR Tier 2, Section 5.2.4.4.9, calls for periodic inspection of RCPB components to check for primary coolant leakage and corrosion resulting from such leakage. The staff finds the BACCP acceptable as documented in Section 5.2.4.4.9 of this report, because it conforms to the recommendations of Generic Letter (GL) 88-05, “Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components in PWR Plants,” and ASME Section XI requirements. Compliance with these documents ensures the BACCP will detect and correct primary coolant leakage before significant degradation of the RCPB can occur, and allow corrective action if necessary.

The applicant proposed to use SA-479 (UNS S41500) or SA-182 Grade F6NM (UNS S41500), in the quenched and tempered condition, for a portion of the control rod drive mechanism pressure housing. UNS S41500 is a martensitic stainless steel. Since there has been a history of environmentally induced cracking (stress-corrosion cracking or service-induced hydrogen embrittlement) of martensitic stainless steels in operating reactors, (reference NRC Bulletin 89-02, "Stress Corrosion Cracking of High-Hardness Type 410 Stainless Steel Internal Preloaded Bolting in Anchor Darling Model S350W Swing Check Valves or Valves of Similar Design," Information Notice 94-55, "Problems with Copes-Vulcan Power-Operated Relief Valves," and Information Notice 95-26, "Valve Failure During Patient Treatment with Gamma Stereotactic Radiosurgery Unit"), the staff requested additional information. In RAI 199, Question 05.02.03-19, the staff requested that the applicant identify: (1) Measures to be taken during material processing and fabrication to ensure the probability of stress corrosion cracking is minimized; (2) operating experience or other information the applicant used as a basis for concluding that UNS S41500 stainless steel will be compatible with the reactor coolant system environment, particularly in regard to its resistance to environmentally induced cracking (either stress-corrosion cracking or service-induced hydrogen embrittlement); and (3) the resistance of the material to thermal aging embrittlement. In a June 5, 2009, response to RAI 199, Question 05.02.03-19, the applicant described measures to be taken to minimize the possibility of stress corrosion cracking of the UNS S41500 stainless steel, including limits on tempering temperature, and identified mechanical properties and chemical composition more stringent than those specified in SA-479 or SA-182. The applicant also described controls on welding to ensure the hardness of the heat-affected zone (HAZ) is not excessive. Further, the applicant cited 19 years operating experience with this material in CRDM pressure housings in German pressurized water reactors (PWRs) with no incidence of cracking, which has been verified through nondestructive examination. In addition, with respect to intergranular stress corrosion cracking (IGSCC), the applicant indicated that the low carbon content of Type-415 material makes it less likely to form carbide precipitates, resulting in a reduced susceptibility to IGSCC compared to other martensitic stainless steels.

With respect to thermal aging embrittlement, the applicant used a model developed for French PWRs to predict the shift in upper shelf energy and transition temperature of the material due to the anticipated temperature exposure and time over the life of the plant. The model showed the upper shelf energy decrease to be insignificant and an acceptable transition temperature shift.

The staff finds the applicant's basis for concluding that UNS S41500 will adequately resist environmentally assisted cracking (EAC) is acceptable, because it has specified appropriate controls on fabrication and processing of the material to control the material properties such that the resistance to EAC will be adequate. The applicant has also cited representative operating experience from German PWRs that supports its conclusion that the material is not susceptible to EAC. Therefore, the staff concludes the applicant's proposed use of UNS S41500 stainless steel is acceptable and meets GDC 4 with respect to compatibility with the RCS operating environment.

Fabrication and Processing of Ferritic Materials

The staff reviewed FSAR Tier 2, Section 5.2.3.3, "Fabrication and Processing of Ferritic Materials," to ensure that the RCPB components satisfy the requirements regarding prevention of RCPB fracture, control of welding, and nondestructive examination (NDE).

The acceptance criteria for fracture toughness properties of ferritic materials are specified in 10 CFR Part 50, Appendix G, "Fracture Toughness Requirements." Ferritic materials meeting these criteria satisfy the requirements of GDC 14 and GDC 31.

Appendix G to 10 CFR Part 50 requires that the pressure-retaining components of the RCPB made of ferritic materials meet specified requirements for fracture toughness, including those of the ASME Code, during system hydrostatic tests and any condition of normal operation, including anticipated operational occurrences. These requirements are met by complying with ASME Code, Section III, Subarticle NB-2300. The fracture toughness requirements of GDC 14 and GDC 31 are also met through compliance with the acceptance standards in Subarticle NB-2300 of ASME Code, Section III. The U.S. EPR design meets these Code requirements, as stated in FSAR Tier 2, Section 5.2.3.3.1, "Fracture Toughness," which states that the fracture toughness properties of the RCPB components including pumps, piping, and valves comply with the requirements of 10 CFR 50, Appendix G and ASME Section III, NB-2300. Therefore, the staff finds that the fracture toughness of materials used to fabricate RCPB components satisfies the requirements of 10 CFR Part 50, Appendix G. The staff's evaluation of the fracture toughness of the RPV is discussed in Section 5.3.1 of this report.

The acceptance criteria for control of ferritic steel welding are met through compliance with the applicable provisions of ASME Code, Section III and following the recommendations stated in the regulatory positions of RG 1.34; RG 1.43; RG 1.50, "Control of Preheat Temperature for Welding of Low-Alloy Steel"; RG 1.71; and ASME Code Section III, Appendix D, "Nonmandatory Preheat Procedures."

The assurance of satisfactory electroslag welds for low-alloy steel and stainless steel can be increased by maintaining a weld solidification pattern with a strong intergranular bond in the center of the weld. RG 1.34 provides staff guidance on the control of electroslag weld properties. FSAR Tier 2, Sections 5.2.3.3.2, "Control of Welding," and 5.2.3.4.4, "Control of Welding," provide a description of the applicant's use of electroslag welding. The staff reviewed the applicant's description of its use of electroslag welding, which includes electroslag weld procedure qualification and testing, and determined that it conforms to the guidance provided in RG 1.34. In addition, the applicant states, in the aforementioned FSAR sections, that it conforms to the guidance in RG 1.34. Therefore, the staff finds this acceptable.

The amount of specified preheat for the welding of ferritic steels should be in accordance with the recommendations listed in ASME Code, Section III, Appendix D, Article D-1000. Appendix D provides preheat temperature recommendations that the staff considers sufficient to prevent delayed hydrogen cracking in ferritic steel welds. Appendix D is supplemented by positions described in RG 1.50 which recommends that preheating of low-alloy steel welds be maintained until PWHT. In an April 23, 2009, response to RAI 199, Question 05.02.03-15, the applicant stated that FSAR Tier 2, Section 5.2.3.3.2 would be modified to indicate that procedure qualification records and welding procedure specifications performed to support welding of carbon and low alloy steel welds in the RCPB conform to RG 1.50 and ASME Code, Section III, Division 1, Nonmandatory Appendix D. The staff finds this acceptable pending confirmation that modifications to the FSAR, described in the applicant's response are made. **RAI 199, Question 05.02.03-15 is being tracked as a confirmatory item.**

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. Stainless steel corrosion resistant weld overlay clad low alloy steel components in the RCPB have an ASTM grain size of 5 or finer. Per RG 1.43, the low alloy steel weld overlay clad materials used in the U.S. EPR RCPB, which are made using a fine grain practice (ASTM grain size of 5 or finer), are not considered susceptible to underclad cracking. Therefore, the staff finds this acceptable because the applicant has appropriately addressed the potential for underclad cracking in weld overlay clad components as described in RG 1.43.

In FSAR Tier 2, Section 5.2.3.3.2, the applicant stated that controls to limit underclad cracking of susceptible materials in the U.S. EPR design conform to the requirements of RG 1.43.

ASME Code, Section III requires adherence to the requirements of ASME Section IX for welder qualification for production welds. However, there is a need for supplementing this section of the Code, because the assurance of providing satisfactory welds in locations of restricted direct physical and visual accessibility can be increased significantly by qualifying the welder under conditions simulating the space limitations under which the actual welds will be made. RG 1.71 provides the necessary supplement to ASME Code, Section IX, in this respect. FSAR Tier 2, Sections 5.2.3.3.2 (ferritic materials) and 5.2.3.4.4 (stainless steel materials) state that welders and welding operators are qualified in accordance with ASME Code, Section IX and RG 1.71.

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, which requires that tubular products be examined over the entire volume of the material. The aforementioned paragraphs include examination techniques and acceptance requirements. In FSAR Tier 2, Section 5.2.3.3.3, the applicant states that NDE performed on ferritic steel tubular products will comply with the requirements of ASME Code Section III, Paragraphs NB-2550 through NB-2570.

Fabrication and Processing of Austenitic Stainless Steels

Process controls 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 controls prevent sensitization of the material by minimizing reactor coolant system and connected systems exposure of the stainless steel to contaminants that could lead to stress-corrosion cracking (SCC), and reduce the likelihood of component degradation or failure through contaminants.

The requirements of GDC 1, GDC 4, and 10 CFR Part 50, Appendix B, Criterion XIII, are met through compliance with the applicable provisions of the ASME Code, Section III 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 ferritic steel tubular products, compliance with the 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 which requires that tubular products be examined over the entire volume of the material. The aforementioned paragraphs include examination techniques and acceptance requirements. In FSAR Tier 2, Section 5.2.3.4.5, the applicant states that NDE performed on stainless steel tubular products will comply with the requirements of ASME Code Section III, Paragraphs NB-2550 through NB-2570.

The U.S. EPR design conforms with RG 1.34, RG 1.36, RG 1.37, RG 1.44, and RG 1.71 for austenitic stainless steel components as discussed above under fabrication and processing of ferritic materials and compatibility of materials with reactor coolant. However, FSAR Tier 2, Section 5.2.3.2.2, “Compatibility of Construction Materials with Reactor Coolant,” states that unstabilized austenitic stainless steels are not heated above 626.7 °C (800 °F), other than locally heated by welding operations, after the final heat treatment. FSAR Tier 2, Section 5.2.3.4.1 states that utilization of materials in the solution annealed plus rapidly cooled condition and the prohibition of subsequent heat treatments in the 626.7 °C (800 °F) to 815.6 °C (1,500 °F) temperature range is one of five methods used to avoid intergranular attack in austenitic stainless steel. The staff determined that these statements appeared to be inconsistent with the applicant’s process for joining low alloy steel nozzles to austenitic stainless steel safe-ends which includes subjecting the safe-ends to post weld heat treatment, as is required by the ASME Code. In RAI 278, Question 05.02.03-23, the staff requested that the applicant modify the FSAR to address these inconsistencies by: (1) Discussing those components used in the solution annealed and rapidly cooled condition, and (2) discussing those components that will be used in the solution annealed and rapidly cooled condition followed by post weld heat treatment after welding. In addition, in order to make the FSAR clear as to the testing of post weld heat treated stainless steel safe-ends, the staff requested that the applicant modify FSAR Tier 2, Section 5.2.3 to state that for post weld heat treated austenitic stainless steel safe-ends, non-sensitization of the safe-ends will be verified in accordance with RG 1.44. **RAI 278, Question 05.02.03-23, which is associated with the above request, is being tracked as an open item.**

RG 1.31 contains staff guidance pertaining to the delta ferrite content in austenitic stainless steel welds to minimize the presence of microfissures, which could have an adverse effect on the integrity of components. Austenitic stainless steel welds in RCS piping, including surge line piping, have delta ferrite content limited to a ferrite number (FN) between 5 and 10. Austenitic stainless steel weld materials for stainless steel welds joints in the balance of the RCPB have delta content ferrite limited to an FN between 5 and 20. The applicant’s ferrite content in RCPB welds of between FN 5 and 20, conforms to what was specified in RG 1.31. In addition, FSAR Tier 2, Section 5.2.3.4.4 states that welding on RCPB components conforms to the guidance contained in RG 1.31. 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.

FSAR Tier 2, Section 3.6.3.4.2 and FSAR Tier 2, Table 5.2-2 indicate that main coolant loop (MCL) and pressurizer surge line (SL) piping will be fabricated from SA-336 F304LN or SA-182 F304LN forged austenitic stainless steel material. These specifications do not contain limitations on grain size. The staff noted that grain size can affect the ability to perform ultrasonic examination. In an April 23, 2009, response to RAI 199, Question 05.02.03-16, the applicant stated that FSAR Tier 2, Section 5.2.3.4.2 will be revised to indicate that forged stainless steel components within the RCPB that are subject to ASME Section XI volumetric examinations have a grain size that allows inspection by ultrasonic methods. The staff finds this acceptable, because the applicant has included supplementary requirements on grain size to facilitate the ultrasonic examination of forged stainless steel piping. The above acceptance of the applicant’s grain size controls is pending staff confirmation that modifications to the FSAR, as described in the applicant’s response, are made. **RAI 199, Question 05.02.03-16 is being tracked as a confirmatory item.**

FSAR Tier 2, Section 5.2.3.4.6, "Cast Austenitic Stainless Steel Materials used in the RCPB," states that the ferrite content of cast austenitic stainless steel (CASS) components in the RCPB will be limited to a ferrite content of less than 20 percent. For CASS material used in the RCPB, the percent ferrite is calculated using Hull’s equivalent factors as indicated in

NUREG/CR-4513, "Estimation of Fracture Toughness of Cast Stainless Steels During Thermal Aging in LWR Systems," Revision 1, May 1994. The staff finds this acceptable for low molybdenum content CASS, because it meets staff guidance as documented in a letter from Christopher I. Grimes of the NRC to Douglas J. Walters of the Nuclear Energy Institute, May 19, 2000. However, in an April 23, 2009, response to RAI 199, Question 05.02.03-18, the applicant provided a proposed revision to FSAR Tier 2, Table 5.2-2. The staff noted that the applicant's proposed revised table now includes CASS grades CF3M and CF8M for the fabrication of RCPB valves. These two grades of CASS contain Molybdenum ranging between 2.0-3.0 percent, which increases their susceptibility to thermal aging embrittlement. To conform to the staff guidance, these materials should have a ferrite content of ≤14 percent in order to be considered not susceptible to thermal aging embrittlement. In RAI 278, Question 05.02.03-21, the staff requested that the applicant modify the FSAR to limit the ferrite content of high molybdenum RCPB CASS components, such as CF3M and CF8M, to ≤14 percent. **RAI 278, Question 05.02.03-21 is being tracked as an open item.**

In FSAR Tier 2, Section 5.2.3.4.1, the applicant states that stabilized grades of austenitic stainless steels have a stabilizing heat treatment above 426.7 °C (800 °F). The staff identified only one stabilized stainless steel material (Grade 347) used to fabricate components in the RCS pressure boundary, and it is used to fabricate the CRDM pressure housing. In a June 5, 2009, response to RAI 199, Question 05.02.03-17, the applicant stated that no stabilizing heat treatment will be performed for Grade 347 material used for the CRDM pressure housing. The staff determined that there appeared to be an inconsistency between the FSAR, which references stabilizing heat treatments, and the applicant's response to RAI 199, Question 05.02.03-17, which indicates that Grade 347 material used to fabricate the CRDM will not receive a stabilizing heat treatment. In RAI 278, Question 05.02.03-22, the staff requested that the applicant address this inconsistency and modify the FSAR accordingly. **RAI 278, Question 05.02.03-22 is being tracked as an open item.**

Dissimilar Metal Welds

Cold work and residual stress imparted on components fabricated from austenitic materials such as austenitic stainless steels and nickel based alloys have contributed to stress corrosion cracking in currently operating PWRs, particularly in dissimilar metal welds. In RAI 88, Question 05.02.03-13, the staff requested that the applicant describe special fabrication processes 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 order to minimize the susceptibility of components to stress-corrosion cracking for the design life of the plant. In a December 17, 2008, response, the applicant provided a table that lists its fabrication processes to limit cold work and residual stresses in nuclear steam supply system components such as the reactor pressure vessel, pressurizer, reactor coolant pump, and reactor coolant piping. The applicant stated that weld repairs are limited to the extent practical. Where necessary, weld repairs of dissimilar metal welds in contact with primary fluids are performed using repair methods selected to achieve compressive (or as low as reasonably achievable tensile) stress conditions on the wetted inner surface. Alloy 52/52M/152 partial penetration welds in high-stress joints, such as RPV head penetrations, are polished with a flapper wheel after grinding to reduce roughness, remove the surface with the highest cold work induced by grinding, and leave compressive residual stresses at the surface. The applicant also stated that weld repair of forgings by the forging supplier is prohibited. The staff finds this acceptable because the applicant has provided adequate process controls to limit the effects of cold work and residual stress imparted on austenitic materials in the RCPB as a result of welding and welding repairs.

Degradation in dissimilar metal welds due to primary water stress-corrosion cracking (PWSCC) in currently operating plants has resulted in numerous repairs and replacement of components due to the use of Alloy 600 and Alloy 82/182 weld filler metal, which are susceptible to PWSCC. Although the U.S. EPR dissimilar metal welds (DMWs) utilize Alloy 52/52M/152, which is resistant to PWSCC, these welds are challenging to make and further, the use of such material results in welds that contain different values of the coefficient of thermal expansion, thus making them more susceptible to degradation than similar metal welds. The DMWs in the U.S. EPR RCPB are the RPV, pressurizer, and steam generator nozzle to safe-end welds. These welds involve the joining of low alloy steel to austenitic stainless steel. Alloy 52/52M/152 is also used to join Alloy 690 CRDM adapter tubes to RPV closure head.

FSAR Tier 2, Sections 5.3, “Reactor Vessel,” and 3.6.3, “Leak-Before-Break Evaluation Procedures,” indicate that dissimilar metal welds joining low alloy steel nozzles and stainless steel safe-ends will use the gas tungsten arc weld (GTAW) process with a narrow groove weld joint design with no weld buttering on low alloy steel nozzles. The applicant’s method to perform full penetration DMWs in RCS piping is substantially different than methods used to fabricate currently operating PWRs. Nozzle to safe-end DMW joints using Inconel weld filler material in currently operating U.S. plants were completed by applying a buttering layer of Alloy 82/182 weld filler material to the nozzle followed by post-weld heat treatment (PWHT). After completion of PWHT, the safe-end would be welded to the nozzle using a traditional groove type weld joint configuration using Alloy 182/82 weld filler material. These welds were typically made using the GTAW process and the shielded metal arc welding (SMAW) process. Thus, the safe-end was not subject to PWHT. The use of Alloy 82/182 resulted in several instances of primary water stress-corrosion cracking in currently operation plants. Alloy 52 is selected for U.S. EPR DMWs because of its resistance to PWSCC and its successful use in fabrication of replacement components and repair welds in currently operating plants. The GTAW narrow gap welding process is used because of its proven ability to produce high quality welds and the reduction in weld size when compared to groove weld type joint designs.

The staff conducted an audit at AREVA’s Rockville, Maryland office on June 25, 2009, to review documents associated with the research and testing that AREVA has performed in support of its use of narrow gap welding without buttering for the fabrication of full penetration dissimilar metal welds in the U.S. EPR design. The development process for GTAW narrow gap DMW joints included several full scale mock-ups which received PWHT. Metallographic examination and chemical analysis were performed to determine the microstructure of the weld and heat-affected zones and the amount of weld metal dilution. The applicant conducted several mechanical tests including fracture toughness testing. In addition, the applicant conducted testing to determine the susceptibility of DMWs to hydrogen embrittlement and PWSCC. The staff reviewed a sample of test results that were provided with a technical presentation, which the applicant submitted to the NRC by letter dated July 17, 2009, made by the applicant during the audit and found them acceptable, because the test results showed an acceptable weld microstructure which did not exhibit any characteristics that could be identified as being detrimental and acceptable mechanical test results which are consistent with the acceptance criteria in ASME Code, Section III. In addition, hydrogen embrittlement test results documented in this presentation were satisfactory, with no embrittlement detected in the weld metal. PWSCC susceptibility testing data reviewed by the staff was also acceptable.

FSAR Tier 2, Section 5.3 indicates that Alloy 690 CRDM adapters are welded to the RPV head using Alloy 52/52M/152. The staff noted that recently fabricated RPV replacement heads have undergone extensive welding repairs during fabrication. Further, the partial penetration j-groove joint design can be difficult to weld given the highly restrained nature of the joint design. The highly restrained joint design also makes these welds more susceptible to ductility dip cracking (sometimes referred to as hot cracking). Given the susceptibility of Alloys 52, 52M, and 152 to ductility dip cracking and other types of welding flaws, in partial penetration j-groove welds, in RAI 88, Question 05.02.03-11, the staff requested that the applicant discuss its welding process controls to minimize welding flaws in CRDM adapter to RPV head welds and any other partial penetration welds that involve dissimilar materials within the RCPB. In addition, the staff requested that the applicant discuss welding process controls employed to reduce weld metal dilution in order to retain the maximum percentage of chromium possible in order to decrease the susceptibility of components to stress corrosion cracking. In a December 17, 2008, response to RAI 88, Question 05.02.03-11, the applicant stated the following:

> To reduce the risk of weld cracking, the applicant developed a specific method to test welding products for different type of welding flaws (labeled "hot cracks"). This method consists of building up a multi-pass gas tungsten arc welding or manual metal arc welding (MMAW) weld in circular groove, which has a high degree of mechanical constraint. The degree of constraint and the nature of the base metal of the test block are varied depending on the severity of the representative case. The test is analyzed for the number and types of cracks. It is thus possible to test batches of welding products in a specific condition representative of (or even more severe in terms of mechanical constraint, base metal, and dilution) specific applications, such as control rod drive mechanism J-groove welds and reactor pressure vessel DMWs. At the research and development stage, this test method allows for sorting welding products and suppliers and is a guide for selection of new welding products. At the implementation stage, the product specification includes a specific test of sensitivity to hot cracking that is used as a reception test. If the test acceptance criteria are not met, the product is rejected.
>
> The level of chromium in dissimilar metal weld[s] in contact with primary water is sufficient to prevent PWSCC crack initiation, as demonstrated by AREVA testing. The reproducibility of the welding process, and thus of the dilution of chromium, is addressed by the procedure qualification of the welding process.

The staff finds the applicant's response acceptable, because the applicant has taken into account the susceptibility of Alloy 52/52M/152 weld filler materials to different types of welding flaws during fabrication by employing supplementary weld filler metal testing, in excess of ASME Code requirements, to screen out heats of weld filler metal that may be more susceptible to ductility dip cracking and other types of fabrication flaws. This screening process will minimize weld flaws and welding repairs during fabrication, thus reducing the potential susceptibility of Alloy 52/52M/152 welds to service induced degradation. In addition, the applicant has considered weld metal dilution, and the applicant maintains a sufficient level of chromium in DMWs to prevent PWSCC crack initiation. The staff finds this acceptable, because the applicant will take the appropriate measures to ensure the weld quality of Alloy 690 CRDM nozzle to RPV head DMWs.

Fabrication of the CRDM pressure housing involves DMWs joining Grade 347 stabilized austenitic stainless steel to Type 415 martensitic stainless steel using Alloy 52/52M weld filler material. In a June 5, 2009, response to RAI 199, Questions 05.02.03-17 and 05.02.03-19, the applicant provided additional information related to the pressure housing DMWs, as follows:

The CRDM housing is fabricated with solution annealed Grade 347 and quenched and tempered type 415 materials. Welding is performed with the GTAW process in a rotating fixture resulting in all welding being performed in the flat position. Following welding, the welds are cooled to less than 80 °C (176 °F) to ensure full transformation to martensite has occurred in the HAZ prior to PWHT. Corrosion testing will be performed on Grade 347 to ensure that the fabrication process used by the applicant will not sensitize the material. Weld procedure qualification testing will include micro-hardness testing to verify a maximum hardness of 350 HV (Vickers Hardness), which will ensure that the final weld will not be susceptible to hydrogen cracking. In addition, impact testing in accordance with ASME Code Section III will also be performed. All other ASME Code, Section III requirements will also be met. The staff considers the applicant's process for using a rotating fixture to be optimal with respect to reducing the potential for welding discontinuities and defects. The staff also considers that the applicant's micro-hardness testing and corrosion testing of welding procedure qualification coupons provides reasonable assurance that production welds will not be susceptible to hydrogen cracking or stress corrosion cracking. In addition, the staff notes that all ASME Code, Section III requirements will be met. Therefore, the staff finds that the applicant's process to perform DMWs in the CRDM pressure housing is acceptable.

Technical Specifications and Surveillance

There are no Technical Specifications associated with FSAR Tier 2, Section 5.2.3.

ITAAC

The ITAAC for the reactor coolant system are located in FSAR Tier 1, Table 2.2.1-5. The staff reviewed the ITAAC associated with RCPB piping and components excluding the RPV, which is addressed in Section 5.3 of this report.

FSAR Tier 1, Table 2.2.1-5, "RCS ITAAC," lists ITAAC for the reactor coolant system. The staff reviewed the ITAAC related to fabrication, welding, NDE, and hydrostatic testing and noticed some inconsistencies. In RAI 88, Question 05.02.03-14, the staff requested that the applicant modify FSAR Tier 1, Table 2.2.1-5 to provide clear and consistent ITAAC associated with the design, fabrication (including verification that welding was performed in accordance with ASME Code, Section III), weld inspection (NDE), and hydrostatic testing for ASME Code Section III piping and components. In addition, the staff noted that the type of report that is required under the acceptance criteria should be listed in FSAR Tier 1, Table 2.2.1-5 (i.e., N-5 data report or other type of report). The staff also noted that this format should be applied to all FSAR Tier 1 ITAAC for ASME Code Section III, Class 1, 2, or 3 systems. The staff is still in the process of reviewing the applicant's responses to the above RAI and other staff RAIs pertaining to ITAAC. **RAI 88, Question 05.02.03-14, which is associated with the above request, is being tracked as an open item.**

Preoperational Testing

There is no preoperational testing related to FSAR Tier 2, Section 5.2.3.

5.2.3.5 *Combined License Information Items*

Table 5.2.3-1 provides a list of RCPB materials related COL information item numbers and descriptions from FSAR Tier 2, Table 1.8-2:

For the RCPB, no COL information items have been identified in FSAR Tier 2, Table 1.8-2. The staff finds this acceptable, 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 open items and 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 necessary so that aging effects or other incipient degradation phenomena may be identified and preventive measures may be promptly taken to preclude potential loss of reactor coolant or impairment of reactor core cooling. 10 CFR 50.55a requires that SSCs be designed, fabricated, erected, constructed, tested, and inspected to quality standards that are commensurate with the importance of the safety functions they are intended to perform. Therefore, 10 CFR 50.55a incorporates by reference Sections III and XI of the ASME B&PV Code, as well as the Operations and Maintenance Code. Section XI defines, for each component Code Class, the specific inservice inspection 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*

FSAR Tier 1: The FSAR Tier 1 information associated with this section is found in FSAR Tier 1, Section 2.2.1. It provides that the piping and equipment, indicated on Figure 2.2.1-1 as ASME Code Section III, be designed, welded, and tested in accordance with ASME Code Section III.

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

The application states that the U.S. EPR design provides ready access to SSCs to accommodate comprehensive inspection using currently available inspection equipment and techniques. The components and welds requiring ISI have design features that allow ready inspection, including clearances for personnel and favorable materials, weld-joint simplicity, elimination of geometrical interferences, and proper weld surface preparation. Removable insulation is used on piping and components requiring volumetric and surface inspection. Pipe hangers and supports are positioned to accommodate weld inspection. The surfaces of welds within the inspection boundary are finished to permit effective examination. A readily accessible configuration facilitates flaw detection and characterization, and tends to lower occupational radiation exposure through reduced inspection times.

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; and (4) system pressure tests, and references the applicable ASME Code requirements. The application states that no exemptions are required from Code Class 1 PSI or ISI examination requirements. Consequently, no NRC relief is required from code-specified PSI or ISI requirements. Furthermore, no ASME Code cases applicable to Class 1 PSI or ISI requirements are invoked for the U.S. EPR design. However, the application states that "a COL applicant's compliance with NRC order EA-03-009 and first revised order EA-03-009 may be accomplished with conditional implementation of Code Case N-729-1" [sic].

ITAAC: With respect to this area of review, Item 3.4b in FSAR Tier 1, Table 2.2.1-5, indicates that inspections will be performed of the as-built piping, and that a report exists which concludes that the piping as indicated on FSAR Tier 1, Figure 2.2.1-1 as ASME Code Section III has been: (1) Welded in accordance with ASME Code Section III welding requirements and (2) hydrostatically tested in accordance with ASME Code Section III requirements.

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

5.2.4.3 *Regulatory Basis*

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

1. GDC 32, "Inspection of Reactor Coolant Pressure Boundary," 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.

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, "Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components in PWR Plants," 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*

FSAR Tier 2 states that preservice and inservice inspections are required for Quality Group A components of the U.S. EPR. 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 U.S. EPR components meet the definition for Quality Group A components presented in RG 1.26. Subsection NB of Section III of the ASME Code presents the construction requirements for Class 1 components, and Subsection IWB of Section XI presents preservice and inservice inspection requirements. FSAR Tier 2, Section 3.2.2, "System Quality Group Classification" includes a list of the ASME Code Class 1 pressure retaining components and addresses the application of the 10 CFR 50.55a regulatory and ASME Code, Section III, criteria to their classification.

The applicant states that the U.S. EPR design standards include provisions for placement of Class 1 piping and components while 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.

The ASME Code of record (edition) for the design of the U.S. EPR is the 2004 edition of the ASME Boiler and Pressure Vessel Code, as stated under FSAR Tier 2, Section 5.2.1.1.

5.2.4.4.1 Arrangement and Accessibility of Systems and Components

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

FSAR Tier 2, Section 5.2.4.1.1 describes accessibility for inspection and states that accessibility incorporated into the design complies with IWA-1500 and 10 CFR 50.55a(g)(3)(i). The FSAR continues as follows: Factors such as examination requirements, techniques, accessibility, component geometry, and material selection are used in evaluating component designs for ease of inspection, including clearances for personnel and favorable materials, weld-joint simplicity, elimination of geometric interferences, and proper weld surface preparation. Removable insulation is used on piping and components requiring volumetric and surface inspection. Pipe hangers and supports are positioned to accommodate weld inspection. The staff had concerns on whether the design provided for two-sided access to dissimilar metal welds and austenitic welds to enable the performance of inservice examinations; therefore, the staff issued RAI 65, Question 05.02.04-1.

In an October 21, 2008, response to RAI 65, Question 05.02.04-1, the applicant stated that nickel-based alloys used in the U.S. EPR are protected from primary water stress corrosion cracking because the Alloy 690 materials are not susceptible to PWSCC, as described in FSAR Tier 2, Section 5.2.3.5, "Prevention of Primary Water Stress-Corrosion Cracking for Nickel-Base Alloys." The staff notes that Alloy 690 is the most recent, state-of-the-art material in use during construction. The staff position is that the most robust design will be achieved when inspections can be performed throughout the life of the components. Therefore, in order to detect new, unforeseen failure mechanisms prior to loss of structural integrity, the applicant must assure that interferences to preservice and inservice examinations are eliminated during the design stage. The applicant stated that no relief from inservice examinations is requested for welds susceptible to PWSCC on the basis of design, geometry, or materials selection. The applicant reiterated that FSAR Tier 2, Section 5.2.4.1.1 states that the accessibility of the design complies with IWA-1500 and the requirements of 10 CFR 50.55a(g)(3)(i), which requires that components be designed to enable the performance of inservice examinations. The staff concludes that the design eliminates obstructions to inservice examinations due to design, geometry, and materials of construction, enabling the performance of inservice examinations, meeting the access provisions under the SRP acceptance criteria and, therefore, finds that the response to RAI 65, Question 05.02.04-1 is acceptable and that the design requirement of 10 CFR 50.55a(g)(3)(i) has been met.

5.2.4.4.2 Examination Categories and Methods

The examination categories and methods specified in the FSAR 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 components and parts of the pressure-retaining boundary.

The applicant's examination techniques and procedures used for preservice inspection or inservice inspection 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 for training and experience, among other things, regarding qualification of nondestructive examination personnel are in accordance with Article IWA-2300, "Qualification of Nondestructive Examination Personnel," of ASME Code, Section XI.

- The methods, procedures, and requirements for training and experience, among other things, 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.

FSAR Tier 2, 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 specified in FSAR Tier 2, Section 5.2.4.1.2, in connection with the components and supports listed there comply with the requirements of Subarticle IWA-2200 and Table IWB-2500-1 of ASME Code, Section XI. Examination categories and other examination requirements are established according to Subarticle IWB-2500 and Table IWB-2500-1 of ASME Code, Section XI. Qualification of the NDE personnel is in compliance with Subarticle IWA-2300 of ASME Code, Section XI. The penetrant testing (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. FSAR Tier 2, Section 5.2.1.1, indicates that the baseline Code used for the U.S. EPR design certification is the 2004 Edition 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. Because the examination methods and categories applied to Class 1 components comply with the requirements of ASME Code, Section XI, as discussed above, the staff finds examination categories and methods for Class 1 components to be acceptable.

5.2.4.4.3 Inspection Intervals

The required examinations and pressure tests must be completed during a prescribed interval of service as defined in the ASME Code, Section XI, paragraph IWA-2430, designated as the "inspection interval." FSAR Tier 2, Section 5.2.4.4, "Inspection Intervals," discusses inspection intervals. The inspection intervals specified for the U.S. EPR components comply with the definitions in Section XI of the ASME Code and, therefore, are acceptable. However, the applicant states that it is not necessary that the inspection intervals for the Class 1 portions of the ISI program conform to the same inspection programs as those for the Class 2 and Class 3 inspections. Similarly, FSAR Tier 2, Section 6.6.4, "Inspection Intervals," states that "it is not necessary that the inspection intervals be the same for the IWC (Class 2) and the IWD (Class 3) portions of the ISI program." The staff has concerns that this interpretation of the ASME Code could allow changes to component inspection intervals contrary to the interval start date specified under 10 CFR 50.55a(g)(4). As a result, RAI 65, Question 05.02.04-2 was issued to address the above.

In an October 21, 2008, response to RAI 65, Question 05.02.04-2, the applicant stated that the regulations do not preclude a licensee from applying different inspection intervals for the Class 1, 2, and 3 components. The applicant also quoted ASME Section XI, paragraph IWA-2430, which states: "It is not required that the inspection intervals of IWB, IWC, IWD, IWE, and IWF conform to the same inspection program." The staff does not agree with the applicant's interpretation of the regulations and IWA-2430, and is uncertain of the intent of the applicant's statement in the FSAR. The statement in the FSAR implies that the COL holder can arbitrarily change the intervals by switching from Inspection Program A or B. Furthermore, IWA-2430(b) states that the interval shall be determined by calendar years following placement of the plant into commercial service. IWA-2430(g) states that the inspection program of IWA-2431 may be replaced by the inspection program of IWA-2432, and vice versa, during the first 3 years of the service lifetime of the plant. Finally, the statement made in FSAR Tier 2, Section 6.6.4, due to its abbreviation, is inaccurate, because it incompletely states the ASME Code requirement. In follow-up RAI 208, Question 05.02.04-06, the staff requested that if it is the intent of the applicant to meet the ASME Code, Section XI requirements for inspection intervals, then the statement in the FSAR should be removed, since the applicant is already required to meet ASME Code, Section XI requirements. If the applicant intends to deviate from Code requirements and is proposing an alternative approach, then it should clearly describe the alternative approach and demonstrate that the alternative approach provides an acceptable level of quality and safety pursuant to 10 CFR 50.55a(a)(3)(i). In a May 8, 2009, response to RAI 208, Question 05.02.04-06, the applicant indicated that it would remove the statement "It is not necessary that the inspection intervals be the same for the IWC (Class 2) and the IWD (Class 3) portions of the ISI program." The staff concludes that the proposed changes, as shown in the FSAR mark-up, will bring the FSAR into compliance with ASME Section XI in this respect, and, is therefore, acceptable. **RAI 208, Question 05.02.04-6, which is associated with the above request, is being tracked as a confirmatory item.**

5.2.4.4.4 Evaluation of Examination Results

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

FSAR Tier 2, Section 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. Accordingly, the staff finds the method of evaluating examination results, and the use of the appropriate ASME Code rules for repair are in conformance with ASME Section XI requirements and 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.

FSAR Tier 2, Section 5.2.4.1.5, “System Pressure Tests,” states that Class 1 systems and components are pressure tested in accordance with Articles IWA-5000 and IWB-5000 of the ASME Code. In addition, a hydrostatic test and visual examination may be performed in lieu of the system pressure test and visual examinations. The applicant states that the test pressure shall not exceed the limiting conditions specified in the technical specifications. Since the applicant’s methodology for performing pressure testing of the Class 1 boundary and components meets the acceptance criteria of the ASME Code and the technical specification requirements (which may represent more severe conditions than otherwise specified in 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. Such code exemptions are listed under IWB-1220 and do not require any further NRC approval. FSAR Tier 2, Section 5.2.4.1.6, “Code Exemptions,” states that no exceptions from code required examinations for Class 1 PSI or ISI are required for the U.S. EPR. However, certain Class 1 components are exempt from surface and volumetric examination in accordance with Subarticle IWB-1220, which include:

- Components that are connected to the RCS and are part of the RCPB, and that are of such a size and shape so that upon postulated rupture the resulting flow of coolant from the RCS under normal plant operating conditions is within the capacity of makeup systems that are operable from on-site emergency power. The emergency core cooling systems are excluded from the calculation of makeup capacity.

- Components and piping segments of nominal pipe size (NPS) 1 and smaller, except for steam generator tubing, including those:

    - That have only one inlet and only one outlet both of which are NPS 1 and smaller

    - Those that have multiple inlets or multiple outlets whose cumulative cross-sectional area does not exceed the cross-sectional area defined by the outside diameter of NPS 1 pipe

    - Reactor vessel head connections and associated piping, NPS 2 and smaller, made inaccessible by control rod drive penetrations

The staff determined that all of the exemptions proposed by the applicant are allowed by IWB-1220. The staff noted that the proposed design for buried components exceeds IWB-1220(d). FSAR Tier 2, Section 5.2.4 states:

> 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.

The staff notes that U.S. EPR buried piping is designed for accessibility to perform inservice examinations, under 10 CFR 50.55a(g)(3)(i) and this is more stringent than the ASME Code, because the FSAR eliminates an exemption listed under IWB-1220(d) pertaining to buried components. The staff's review of the exemptions listed in the FSAR Tier 2, is in accordance with Subarticle IWB-1220 and 10 CFR 50.55a(g)(3)(i), conforms to the acceptance criteria in the SRP and is, therefore, acceptable.

5.2.4.4.7 Relief Requests

The FSAR Tier 2, Section 5.2.4.1.7, "Relief Requests," states that no relief from Class 1 PSI or ISI requirements is required for the U.S. EPR. The staff concludes that because the applicant has identified no Code ISI requirements that are impractical when using the 2004 Edition of Section XI, the SRP acceptance criterion has been met. Nonetheless, should a COL application referencing the U.S. EPR design be granted, 10 CFR 50.55a(g)(4) will require updating the ISI program in accordance with the ASME Code 12 months prior to fuel load. Relief requests may be identified with respect to the updated ISI program.

5.2.4.4.8 Code Cases

The SRP acceptance criteria state that ASME Code Cases are reviewed for acceptability and compliance with Regulatory Guide 1.147. FSAR Tier 2, Section 5.2.4.1.8, "Code Cases," states that no code cases applicable to Class 1 PSI or ISI requirements are invoked for the design. However, the reactor pressure vessel head inspections will 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. Furthermore, it states that, "COL applicants that reference the U.S. EPR design may invoke Code Case N-729-1, with conditions cited in the two orders, until subsequent NRC requirements supersede the order."

The NRC has amended its regulations in 10 CFR 50.55a(g)(6)(ii)(D),which now requires implementation of ASME Code Case N-729-1, which is related to reactor vessel head inspections, with specified modifications. At the time of the review, the staff could not determine if the applicant would take into consideration the most recent amended regulations affecting inspection of upper heads; therefore, the staff issued RAI 65, Question 05.02.04-3.

In an October 21, 2008, response to RAI 65, Question 05.02.04-3, the applicant stated that FSAR Tier 2, Section 5.2.4.1.8 will be revised to indicate that the subject Code Case would be implemented in accordance with the conditions specified in the final amended rule for 10 CFR 50.55a. Because the design certification applicant is revising its FSAR to reflect the most recent requirements under the final rule, the change meets 10 CFR 50.55a and is, therefore, acceptable. **RAI 65, Question 05.02.04-3 is being tracked as a confirmatory item.**

5.2.4.4.9 Augmented ISI to Protect Against Postulated Piping Failures

FSAR Tier 2, Section 5.2.4.1.9, "Augmented ISI to Protect Against Postulated Piping Failures," states that no Class 1 piping penetrates the Reactor Building and that no augmented ISI is required to protect against postulated piping failures of Class 1 piping between containment isolation valves. The SRP recommendation to discuss this program is not applicable to the U.S. EPR design and the applicant's approach is therefore acceptable.

5.2.4.4.10 Other Inspection Programs

The SRP indicates that PWR plants should be subject to an inspection program to detect and correct potential RCPB corrosion caused by boric acid leaks as described in NRC Generic Letter 88-05. FSAR Tier 2, Section 5.2.4.1.10, "Other Inspection Programs," states that the ISI program includes provisions to detect and correct potential RCPB corrosion caused by boric acid leaks, as described in NRC GL 88-05. The information provided by the applicant did not provide a sufficient level of detail describing the specific provisions of the program and how they detect and correct potential RCPB corrosion caused by boric acid leaks. To inquire into this information, the staff issued RAI 186, Question 05.02.04-4.

In a March 11, 2009, response to RAI 186, Question 05.02.04-4, the applicant proposed to add the following sentence to the FSAR Tier 2, Section 5.2.4.1.10: "The ISI program for the U.S. EPR regarding potential RCPB corrosion due to boric acid leaks includes regular visual inspections as specified in Sections 5.2.4.1.1 through 5.2.4.1.4 and use of RCPB leakage detection systems as described in Section 5.2.5." The staff determined that the applicant's added description does not provide a level of detail sufficient to describe its boric acid corrosion program and requested the applicant, as a minimum, to describe how the program addresses the four key elements of the program as discussed in NRC GL 88-05. The staff issued follow-up RAI 208, Question 05.02.04-5 to address this concern.

In a May 26, 2009, response to RAI 208, Question 05.02.04-5, the applicant stated that the U.S. EPR ISI program complies with ASME Section XI and NRC GL 88-05. FSAR Tier 2, Section

5.2.4.1.10 will be revised to provide a description of the basic boric acid corrosion control program and to describe how the four key elements of the program in NRC GL 88-05 are implemented. The staff reviewed the proposed changes to FSAR Tier 2, Section 5.2.4.1.10 to determine conformance with GL 88-05. The staff found that the proposed changes addressed determination of principal locations where leaks may occur, methods for locating leaks smaller than the allowable technical specification limit, methods of examination, evaluation to determine impact of leakage, and determination of corrective actions to prevent recurrences of such corrosion. In addition, the applicant stated that the boric acid control program is a site specific program to identify, evaluate, and correct small borated water leaks in the primary system that could cause corrosion damage to reactor coolant pressure boundary components or other auxiliary system components. The staff concludes that the proposed changes conform to ASME Section XI and contain the essential elements discussed in GL 88-05, and, is therefore, acceptable. **RAI 208, Question 05.02.04-5, which is associated with the above request, is being tracked as a confirmatory item.**

5.2.4.4.11 Preservice Inspection and Testing Program

The SRP states that the techniques and procedures for visual, surface, or volumetric examination should be in accordance with Article IWA-2000, “Examination and Inspection,” and Article IWB-2000, “Examination and Inspection,” of Section XI of the ASME Code, with the acceptance criteria for the examination results under IWB-3000, “Acceptance Standards.” The applicant states that the PSI program for Class 1 components complies with NB-5280 of Section III, Division 1 of the ASME Code, for Class 1 components initially selected for the ISI program described in Section 5.2.4.1, “Inservice Inspection and Testing Program,” of the FSAR Tier 2. NB-5280 of the ASME Code, Section III, provides requirements for the conduct of the preservice inspection program, referencing IWA-2000 and IWB-2000. In addition, the acceptance criteria for the examination results stated in the application conform to the criteria under Table IWB-3410-1, consistent with the SRP guidance.

The SRP states that for PSI or ISI examinations, the methods, procedures, and requirements (personnel qualifications) for the ultrasonic examination of reactor-vessel-to-flange welds, closure-head-to-flange welds, and integral attachment welds should incorporate the regulatory positions provided in RG 1.150, “Ultrasonic Testing of Reactor Vessel Welds During Preservice and Inservice Examinations,” unless qualified by performance demonstration in accordance with the requirements of Appendix VIII of Section XI of the ASME Code. FSAR Tier 2, Section 5.2.4.1.2, “Examination Categories and Methods,” states that the methods, procedures, and requirements of personnel ultrasonic testing comply with the guidance provided in Appendix VIII of Section XI of the ASME Code. The FSAR also states that performance demonstration for ultrasonic examination procedures, equipment, and personnel used to detect and size flaws is in accordance with the requirements of Appendix VIII of ASME Section XI. The staff notes that at the time of this writing, RG 1.150 has been withdrawn and replaced by 10 CFR 50.55a(g)(6)(ii)(C)(1). RG 1.150 represents a method that is no longer acceptable to the staff. See 73 Federal Regulation 7766, 7767, February 11, 2008. 10 CFR 50.55a(g)(6)(ii)(C)(1) requires both preservice and inservice inspection activities to be performed using personnel, equipment, and procedures qualified in accordance with the ASME Code, Appendix VIII. The staff finds that the ultrasonic examination procedures, equipment, and personnel comply with the regulations and are, therefore, acceptable.

5.2.4.5 *Combined License Information Items*

Table 5.2.4-1 provides a list of inservice inspection and testing of the RCPB related COL Information Item numbers and descriptions from FSAR Tier 2, Table 1.8-2:

Table 5.2.4-1 U.S. EPR Combined License Information Items

Item No.	Description	FSAR Tier 2 Section
5.2-3	A COL applicant that references the U.S. EPR design certification will identify the implementation milestones for the site-specific ASME Section XI pre-service and inservice inspection program for the reactor coolant pressure boundary, consistent with the requirements of 10 CFR 50.55a (g). The program will identify the applicable edition and addenda of the ASME Code Section XI, and will identify additional relief requests and alternatives to Code requirements.	5.2.4

The staff finds the above listing to be complete. Also, the list adequately describes actions necessary for the COL applicant. No additional COL information items need to be included in FSAR Tier 2, Table 1.8-2 for inservice inspection and testing of the RCPB considerations.

5.2.4.6 *Conclusions*

As described above, the design of the RCS incorporates provisions for access to enable the performance of ISI examinations in accordance with 10 CFR 50.55a(g)(3) and the 2004 Edition of the ASME Code, Section XI. The final ISI program is required to meet the latest ASME Code, Section XI Edition/Addenda incorporated by reference 12 months before the date scheduled for initial loading of fuel. 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. The periodic inspections and pressure testing of pressure-retaining components of the reactor coolant pressure boundary are performed in accordance with the requirements in applicable subsections of Section XI of the ASME Code and provide 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 ISIs required by the ASME Code constitutes an acceptable basis for satisfying, in part, the requirements of GDC 32.

Except for the open items and confirmatory items discussed above, the staff concludes the description of the ISI program is acceptable and meets the inspection and testing requirements of GDC 32 and 10 CFR 50.55a. This conclusion is based on the applicant meeting the requirements of the ASME B&PV 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 intended 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 include RCS inventory balance and localized humidity and temperature monitoring.

5.2.5.2 *Summary of Application*

FSAR Tier 1: FSAR Tier 1, Section 2.4.8, "Leakage Detection System," indicates that there are no FSAR Tier 1 entries for this system.

FSAR Tier 2: Reactor coolant leakage is categorized as either identified leakage or unidentified leakage. Identified leakage includes leakage from any one of three sources: (1) Leakage into closed systems (e.g., pump seal or valve packing leaks) which is captured, quantified, and directed to a sump or collection tank; (2) leakage into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of unidentified leakage monitoring systems or not to be from a flaw in the RCPB; and (3) intersystem leakage into connected systems, including leakage through steam generator tubes. All other leakage is categorized as unidentified leakage.

The U.S. EPR application specified the following design leakage rates which are reflected in the U.S. EPR Technical Specifications:

- No pressure boundary leakage (Seal and gasket leakage is not "pressure boundary leakage" in this context.)
- 3.8 liters per minute (lpm) (1 gallon per minute (gpm)) unidentified leakage
- 38 lpm (10 gpm) identified leakage
- 568 lpd (150 gpd) primary to secondary leakage through any one steam generator

The U.S. EPR design certification application identified methods for detecting, monitoring, and collecting unidentified leakage; methods for detecting, monitoring, and collecting identified leakage; methods for detecting and monitoring inter-system leakage; TS, inspection and testing requirements; and instrumentation function and design.

ITAAC: There are no inspections, tests, analyses, and acceptance criteria specific to the RCPB leakage detection systems.

Technical Specifications: The U.S. EPR TS would require (TS 3.4.12) that a licensee verify that RCS operational leakage is within limits by performance of an RCS water inventory balance. The U.S. EPR TS would also require (TS 3.4.14) that at least one containment sump (level or discharge flow) monitor, one containment atmospheric radioactivity (particulate) monitor, and one containment air cooler condensate flow rate monitor be OPERABLE while in Modes 1, 2, 3, and 4.

5.2.5.3 *Regulatory Basis*

The relevant requirements of NRC 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, 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.

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. Other system-specific acceptance criteria are listed in NUREG-0800, Section 5.2.5.

5.2.5.4 *Technical Evaluation*

The staff reviewed the reactor coolant pressure boundary leakage detection systems described in the FSAR, Revision 0, in accordance with NUREG-0800, Section 5.2.5, "Reactor Coolant Pressure Boundary Leakage Detection," Revision 2, March 2007.

FSAR Tier 2, Section 5.2.5, "RCPB Leakage Detection," states that the RCPB leakage detection systems are designed to detect and, to the extent practical, identify the source of reactor coolant leakage. FSAR Tier 2, Section 5.2.5, describes the RCPB leakage detection systems as follows: Diverse measurement methods include monitoring of sump level and flow, containment airborne radioactivity, and containment air cooler condensate flow. Additional methods include RCS inventory balance, localized humidity, and temperature monitoring. The RCPB leakage detection systems are sufficient reliable, redundant, and sensitive to support the application of leak-before-break (LBB) analyses to eliminate the need to consider the dynamic effects of main reactor coolant loop and pressurizer surge line ruptures from the design basis. LBB analyses are addressed in FSAR Tier 2, Section 3.6.3, "Leak-Before-Break Evaluation Procedures." The staff's review of the LBB analyses is documented in Section 3.6.3 of this report.

The following is the staff's evaluation against the acceptance criteria of SRP Section 5.2.5, Subsection II, using the review procedures of SRP Section 5.2.5, Subsection III.

GDC 2, "Design Basis for Protection Against Natural Phenomena"

GDC 2 may be satisfied based on the guidance of RG 1.29, Regulatory Position C.1 for the safety-related portions of system, and Position C.2 for the non-safety-related portion of system.

The staff reviewed FSAR Tier 2, Section 5.2.5, and FSAR Tier 2, Table 3.2.2-1 against RG 1.29. The staff was not able to find the safety-classification information in FSAR Tier 2, Section 5.2.5, and the RCPB leakage detection system was not identified in FSAR Tier 2, Table 3.2.2-1. As a result, the staff issued RAI 154, Question 05.02.05-1. The staff requested that the applicant clarify the safety classification of the RCPB leakage detection system in the FSAR. In a February 11, 2009, response to RAI 154, Question 05.02.05-1, the applicant did not provide the requested clarification. Therefore, the staff issued follow-up RAI 244, Question 05.02.05-5. In a July 14, 2009, response to follow-up RAI 244, Question 05.02.05-5, the applicant stated that the containment sump level and discharge flow monitors are part of the Nuclear Island drain/vent system (NIDVS) and are identified in FSAR Tier 2, Table 3.2.2-1 as safety-related (see FSAR markups for the Response to RAI 163, Question 09.03.03-5, dated July 14, 2009). Additionally, the applicant indicated that the containment sump level and discharge flow monitoring system is seismically qualified (Seismic Category I) in accordance with RG 1.29. The applicant indicated further that the containment atmosphere radiation monitors are part of the sampling activity monitoring system (SAMS) and are identified in FSAR Tier 2, Table 3.2.2-1 as having a safety classification of NS-AQ (Supplemented Grade). This classification, as defined in FSAR Tier 2, Section 3.2 is assigned to those non-safety-related SSCs that are classified as supplemented grade and will be included in the 10 CFR Part 50, Appendix B quality assurance program. The applicant stated that these containment atmosphere radiation monitors have a seismic classification of Seismic Category I.

The staff finds these radiation monitors meet RG 1.45 as related to the seismic classification. The containment air cooler condensate monitoring system is identified in FSAR Tier 2, Table 3.2.2-1. The applicant indicated that the condensate level and flow sensors are part of the balance of the leak detection system and have a safety classification of NS-AQ and a seismic classification of Seismic Category II in accordance with RG 1.29. RG 1.45 states that at least one of the leakage monitoring systems should be capable of performing its function following any seismic event that does not require plant shutdown. Since the containment sump level and discharge flow monitoring system and containment atmosphere radiation monitors are Seismic Category I monitoring systems, the staff finds the RCPB leakage detection system meets the guidance of RG 1.45 and RG 1.29 in terms of seismic design for the RCPB leakage detection system. Therefore, the RCPB leakage detection meets GDC 2, and RAI 154, Question 05.02.05-1 and RAI 244, Question 05.02.05-5 are resolved. The staff has confirmed that Revision 1 of FSAR, dated May 29, 2009, Tier 2, Section 5.2.5.1.3 was revised as committed in response to RAI 154, Question 05.02.05-1. Accordingly, the staff finds that the applicant has adequately addressed this issue and, therefore, the staff considers RAI 154, Question 05.02.05-1 resolved.

GDC 30, “Quality of Reactor Coolant Pressure Boundary”

GDC 30 may be satisfied based on meeting the guidance of RG 1.45. The staff reviewed FSAR Tier 2, Section 5.2.5 in accordance with SRP Section 5.2.5, Subsection III.1 and III.2, “Review Procedures,” to determine if the RCPB leakage detection system can separately monitor and collect both identified and unidentified leakage. The staff reviewed the applicable information in the following FSAR sections. FSAR Tier 2, Section 5.2.5 describes identified leakage, which is captured, quantified, and directed to a sump or collection tank, as separate from unidentified leakage. FSAR Tier 2, Section 5.2.5.2, “Detecting, Monitoring and Collecting Identified Leakage,” states that provisions are incorporated into the design to isolate, capture, and quantify leakage from known potential sources, such as flanges and relieve valves, so that such leakage may be monitored separately from unidentified leakage. FSAR Tier 2, Section 5.2.5.1, “Detecting, Monitoring and Collecting Unidentified Leakage,” describes the detecting, monitoring, and collecting of unidentified leakage. FSAR Tier 2, Section 5.2.5.5.1, “RCDT Indications,” identifies that the reactor coolant drain tank (RCDT) indications are the instruments for the identified leakage. In addition, FSAR Tier 2, Section 5.2.5.5.2, “Reactor Building Sump Level,” states that Reactor Building sump level and automatic pump operation for sumps are indicated in the MCR for unidentified leakage. The staff determined the FSAR has demonstrated adequately that the RCPB leakage detection system can separately monitor and collect leakage from both identified and unidentified leakage without masking between the two types. These design features meet the regulatory positions in RG 1.45 that flow rates from identified sources are monitored separately from the flow rates from unidentified sources, and are, therefore, acceptable.

The staff reviewed the intersystem leakage detection system in accordance with SRP Section 5.2.5, Subsection III.3. FSAR Tier 2, Section 5.2.5.3, “Detecting and Monitoring Intersystem Leakage,” states that intersystem leakage is identified by increasing level, temperature, flow, or pressure in the connected systems, and intersystem leakage is also detected through relief valve actuation or increasing radioactivity in the connected systems. Further, FSAR Tier 2, Section 5.2.5.3.1 through Section 5.2.5.3.3 describe the intersystem leakage detection system for SIS/RHRS including accumulators, steam generator tubes, and component cooling water system. Based on the FSAR description, the staff finds that the design of intersystem leakage detection meets the regulatory positions in RG 1.45 that the plant should monitor intersystem leakage for systems connected to the RCPB, and is, therefore, acceptable.

The staff reviewed the alarms and readouts in the control room associated with leakage in accordance with SRP Section 5.2.5, Subsection III.5. FSAR Tier 2, Section 5.2.5.5, “Instrumentation Requirements,” states that leakage detection systems provide data to the instrumentation and control systems for indication, alarm, and archival purposes. Operators in the MCR are provided with the leakage rate, from each detection system and a common leakage equivalent (in gpm) from both identified and unidentified sources. Alarms indicate when leakage has exceeded predetermined limits. In reviewing the above, however, the staff could not find in the FSAR procedures, charts, or graphs for the operator to convert the instrument indications of various detected leakage (e.g., from containment radioactivity monitors, containment sump monitors, containment air cooler condensate flow rate monitors) into common leakage rate terms (gpm). Therefore, the staff issued RAI 154, Question 05.02.05-2, and requested that the applicant provide the following information:

- Identify a COL information item for the COL applicant to describe how it will provide operators the procedures, charts, or graphs that permit rapid conversion of instrument indications from various leakage detection instruments into a common leak rate (gpm).

- Define the alarm setpoints and demonstrate the setpoints are sufficiently low to provide an early warning for operator actions prior to Technical Specification limits being exceeded.

In a February 11, 2009, response to RAI 154, Question 05.02.05-2, the applicant stated that the requested information will be provided in site-specific plant operating procedures, which are the responsibility of the COL applicant as described in FSAR Tier 2, Section 13.5, “Plant Procedures.” The staff reviewed FSAR Tier 2, Section 13.5, and determined that there is no COL information item that would require the COL applicant to provide information related to conversion of instrument indications and alarm setpoints to a common unit for early leakage warning. Therefore, the staff, issued follow-up RAI 244, Question 05.02.05-6. In a July 14, 2009, response to RAI 244, Question 05.02.05-6, the applicant restated in the revised FSAR Tier 2, Section 5.2.5 that the requested procedures will be prepared as operating and emergency operating procedures, which are the responsibility of the COL applicant, but maintained its position that no specific COL information item for these procedures is needed. The staff determined the applicant's response to be inadequate. Even though the responsibility for the operating and emergency operating procedures is with the COL applicant, the instrumentation could be designed to give indications in consistent units or a COL applicant can write procedures to perform the appropriate unit conversions. Since the design does not provide for instrument indication in consistent units, a COL information item is necessary to require the COL applicant to address this issue. Therefore, the design certification application is incomplete without identifying a COL information item specifically for the procedures relating to the conversion of instrument indicators and alarm setpoint. The staff issued follow-up RAI 365, Question 05.02.05-9 to address this concern. **RAI 365, Question 05.02.05-9 is being tracked as an open item.**

The staff reviewed detection system sensitivity and response in accordance with SRP Section 5.2.5, Subsection III.6. FSAR Tier 2, Section 5.2.5.1.1, "Containment Sump Level and Discharge Flow Monitoring," states that the containment sump level and discharge flow monitoring system can detect a leakage rate of 1.9 lpm (0.5 gpm) in one hour. FSAR Tier 2, Section 5.2.5.1.2, "Containment Atmosphere Radiation Monitoring," states that the airborne particulate radiation monitors can detect a 3.8 lpm (1.0 gpm) leakage rate within 1 hour at full power operation. FSAR Tier 2, Section 5.2.5.1.3, "Containment Air Cooler Condensate Monitoring," states that the containment air cooler condensate monitoring system can detect a 3.8 lpm leak (1.0 gpm) within 1 hour. The sensitivity and response times as described in the FSAR meet the regulatory positions in RG 1.45 that the leakage detection systems should be adequate to detect a leakage rate of one gpm in less than 1 hour and are, therefore, acceptable. The applicant needs to verify the sensitivity and response times by testing in the ITAAC and Initial Testing Program, which will be discussed below.

In FSAR Tier 2, Section 5.2.5.1, the applicant identifies the methods in the technical specification for detecting unidentified leakage inside containment, which include containment sump level and discharge flow monitoring, containment atmosphere radiation monitoring, and containment air and cooler condensate monitoring. Additional methods include RCS inventory balancing, localized humidity and temperature monitoring. These diverse leakage detection methods satisfy the regulatory positions in RG 1.45 that, in addition to monitoring systems identified in the technical specifications, diverse leakage monitoring systems should be used, and are, therefore, acceptable.

FSAR Tier 2, Section 5.2.5.4, "Inspection and Testing Requirements," states that the leakage detection systems are designed to permit operability testing and calibration during plant operation. Surveillance requirements are in the proposed plant TS under SR 3.4.14. The floor drainage system is subject to periodic testing to verify that it is free of blockage. This provision satisfies the regulatory positions in RG 1.45 that the leakage detection systems should have provisions to permit testing and calibration during plant operation, and is, therefore, acceptable.

In FSAR Tier 2, Chapter 16, the applicant has provided technical specification limited condition for operation (LCO) 3.4.12 and LCO 3.4.14 for the leakage detection system that specify allowable leakage limits and operability requirements for instruments of diverse monitoring principles during plant operating modes 1, 2, 3, and 4. This meets the guidelines in

Regulatory Position of RG 1.45 that plant technical specifications should include the limiting conditions for identified, unidentified, and RCPB leakage, and should address the availability of various types of instrumentations to ensure adequate coverage during all phases of plant operation. In addition, the limits proposed by the applicant in LCO 3.4.13 and LCO 3.4.15 and the associated SRs are in agreement with the recommended set of Standard Technical Specifications for a PWR as provided in NUREG-1431, "Standard Technical Specifications Westinghouse Plants," Volume 1, Revision 3.0, June 2004.

COL Information Items

The staff reviewed FSAR Tier 2, Table 1.8-2, and did not identify any COL information items relative to RCPB leakage detection. The staff's review of the RCPB leakage detection systems identified a potential COL information item to provide operators the procedures, charts, or graphs to permit rapid conversion of indications from various leakage detection instruments into a common leak rate (lpm or gpm). This was discussed above relating to RAI 154, Question 05.02.05-2 and RAI 244, Question 05.02.05-6.

In addition to the above, there is another potential COL information item that needs to be addressed. An important safety lesson learned from the Davis Besse reactor vessel head leakage event indicated that small RCS leakage rates and boron corrosion, if it lasts for a long time, can be a significant safety concern. This concern was not recognized in Revision 0 of RG 1.45, because Revision 0 was published in 1973. However, in Revision 1 of RG 1.45, dated May 2008, the concern was identified. The FSAR does not state that procedures will be developed to provide operators an early warning mechanism and guidance in response to a low-level leakage (well below TS limits) event such as the one that occurred at Davis Besse. In RAI 154, Question 05.02.05-3, the staff requested that the applicant provide a COL information item regarding the development of procedures for determining operator responses (identifying, monitoring, trending, locating) to prolonged low-level leakage conditions.

In a February 11, 2009, response to RAI 154, Question 05.02.05-3, the applicant replied that the requested information will be provided in site-specific plant operating procedures, which are the responsibility of the COL applicant as described in FSAR Tier 2, Section 13.5. The staff reviewed FSAR Tier 2, Section 13.5 and determined that there is no COL information item that would require the COL applicant to provide information as related to operator actions to manage long-term low-level RCS leakage. The staff issued a follow-up RAI 244, Question 05.02.05-7. In a July 14, 2009, response to RAI 244, Question 05.02.05-7, the applicant acknowledges the need to establish procedures that specify operator actions in response to leakage rates less than the limits set forth in the plant technical specifications. The applicant indicated that the procedures to specify operator actions for abnormal conditions will be prepared as operating- and emergency-operating procedures described in FSAR Tier 2, Section 13.5.2.1, "Operating and Emergency Operating Procedure," and are the responsibility of the COL applicant. However, the applicant maintained its position that no specific COL information item for the long-term low-level RCS leakage management is needed. The staff determined the applicant's response to be inadequate. Even though the responsibility for the operating- and emergency-operating procedures is with the COL applicant, the Davis Besse event demonstrated the importance of monitoring long term low level RCS leakage, and warrant a COL information item to ensure that the procedures for performing this function are appropriately addressed. Therefore, the design certification application is incomplete without identifying a COL information item specifically for the procedures relating to operator actions to manage the long-term low-level RCS leakage. The staff issued follow-up RAI 365, Question 05.02.05-10 to address this concern. **RAI 365, Question 05.02.05-10 is being tracked as an open item.**

Based on the above, and except for the open items discussed above, the staff concludes that 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

FSAR Tier 2, Table 14.3-8, “ITAAC Screening Summary,” (Sheet 7 of 8) shows that the leakage detection system is within the scope of FSAR Tier 1, and has an ITAAC in FSAR Tier 1, Section 2.4.8. The staff determined, however, that the leak detection systems were inadequately described, and there were no associated ITAAC. Accordingly, the staff requested that the applicant provide a more detailed design description and appropriate ITAAC table to verify the design features of the leakage detection systems as described in RAI 154, Question 05.02.05-4. The ITAAC should verify the design of RCPB leakage detection sensitivity, response time, and alarm limits for the RCPB leakage detection instrument. In a February 11, 2009, response to RAI 154, Question 05.02.05-4, the applicant indicated that it used its screening process to determine that the RCPB leakage detection system need not be in the ITAAC. The staff reviewed the applicant’s response to RAI 154, Question 05.02.05-4, and determined that it is not acceptable. Therefore, the staff issued a follow-up question in RAI 244, Question 05.02.05-8. In a July 14, 2009, response to RAI 244, Question 05.02.05-8, the applicant agreed to include the RCPB leakage detection system in the ITAAC. The staff reviewed the marked-up pages for the revised ITAAC and determined that the verification of the RCPB leakage detection sensitivity, response time, and alarm limits for the RCPB leakage detection instrument is not included in the proposed ITAAC. The staff issued follow-up RAI 365, Question 05.02.05-11 to address this concern. **RAI 365, Question 05.02.05-11 is being tracked as an open item.**

Initial Testing Program

FSAR Tier 2, Section 14.2.12.12.1, “Leak Detection Systems (Test No. 137),” verifies the proper operation of the leak detection systems, provides for adjustment of the alarm setpoints, and verifies the automatic calibration features. FSAR Tier 2, Section 14.2.12.14.9, “Post-Core Reactor Coolant System Leak Rate Measurement (Test No. 187),” measures the RCS leakage and distinguishes between identified and unidentified leakage. FSAR Tier 2, Section 14.2.12.14.11, “Leak Detection Systems (Test No. 189),” obtains baseline data on the leak detection system and provides for the adjustment of leak detection alarm setpoints as necessary. The initial test program for U.S. EPR is evaluated in Section 14.2 of this report, 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 is appropriate, since it is to demonstrate the capability of the leak detection systems. However, the staff determined that the sensitivity and response time of the leak detection systems did not appear to be included in scope of the above tests. In RAI 158, Question 14.02-89, dated January 28, 2009, the staff requested that the applicant identify tests to demonstrate the sensitivity and response time of the leak detection systems. In a February 27, 2009, response to RAI 158, Question 14.02-89, the applicant stated that FSAR Tier 2, Section 14.2.12, Test No. 137 will be revised to verify the design bases described in FSAR Tier 2, Section 5.2.5, and FSAR Tier 2, Section 7.1.1.5.12 and will be modified to clarify that the acceptance criteria are based on RG 1.45. The applicant also replied that the sensitivity and response time will depend on the selection of equipment from a particular vendor. Thus, this type of detail will be available later in the design process. The staff finds that the applicant's response adequately addressed the staff's concern identified in the RAI since the sensitivity and response time of the leak detection systems will be verified. The staff has confirmed that Revision 1 of FSAR, dated May 29, 2009, was revised as committed in the RAI response. Accordingly, the staff finds that the applicant has adequately addressed this issue and, therefore, the staff considers RAI 158, Question 14.02-89 resolved.

5.2.5.5 *Combined License Information Items*

The evaluation of the COL information items is discussed above in Section 5.2.5.4, "Technical Evaluation." **RAI 365, Question 05.02.05-9 and RAI 365, Question 05.02.05-10 are being tracked as open items.**

5.2.5.6 *Conclusions*

The staff concludes that the design of the RCPB leakage detection system follows the guidelines of SRP Section 5.2.5 and applicable regulatory guides and, therefore, meets the requirements of GDC 2 and GDC 30, except for the open items discussed above.

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 and four outlet 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 of the vessel shell and then upward through the core, acquiring thermal energy. The reactor coolant leaves the RPV through the outlet nozzles. The RPV closure head contains penetrations for control rod drive mechanism adapters, in-core instrumentation adapters, and a high point vent. There are no penetrations in the RPV lower head.

5.3.1 Reactor Vessel Materials

5.3.1.1 *Introduction*

This section addresses material specifications, special processes used for manufacture and fabrication of components, special methods for nondestructive examination, special controls and special processes used for ferritic steels and austenitic stainless steels, fracture toughness, material surveillance (which will be referred to as reactor vessel surveillance capsule program (RVSP) to avoid confusion with material surveillance programs that exist in other parts of a nuclear power plant), and reactor vessel fasteners. This section of the FSAR should contain pertinent data in sufficient detail to provide assurance that the materials (including weld materials), fabrication methods, and inspection techniques used for the reactor vessel and applicable attachments and appurtenances conform to all applicable regulations. Reactor coolant system components are addressed separately in Section 5.2.3 of this report.

5.3.1.2 *Summary of Application*

FSAR Tier 1: The FSAR Tier 1 information associated with this section is found in FSAR Tier 1, Section 2.2.1. Tier 1 provides that the RPV internals be designed to withstand the effects of flow-induced vibration. Detailed FSAR Tier 1 mechanical, electrical, and instrumentation information associated with the RPV design is specified in FSAR Tier 1, Tables 2.2.1-1 through 2.2.1-3.

FSAR Tier 2: The applicant has provided a FSAR Tier 2 description of the materials used in the RPV in Section 5.3.1, "Reactor Vessel Materials," summarized here in part as follows:

The RPV provides support for internal reactor components and is designed to accommodate the effects of environmental conditions associated with normal operations, maintenance, testing, postulated accidents, and anticipated operational occurrences. The ferritic materials provide sufficient margin to account for uncertainties associated with flaws and the effects of service and operating conditions, while allowing the vessel to behave in a non-brittle manner and minimizing the probability of rapidly propagating fracture. The RPV is a vertically mounted cylindrical vessel consisting of forged shells, heads, and nozzles joined by circumferential welds. The surfaces of the RPV that come into contact with the reactor coolant are clad in austenitic stainless steel or Ni-Cr-Fe alloy.

The RPV is made of low-alloy steel due to its mechanical and physical properties, toughness, availability in the required sizes and thicknesses, satisfactory prior service in neutron fields, fabrication requirements, and weldability. The low-alloy steel is also compatible with the stainless steel cladding used for corrosion resistance. The austenitic stainless steels and non-ferrous materials used for RPV appurtenances are used for their corrosion resistance, acceptable mechanical properties, and fabrication requirements. Limits have been placed on the amount of certain elements in the material composition of the shell forgings of the RPV beltline and in the associated weld filler material. The phosphorous, nickel, and copper content is limited to reduce sensitivity to radiation embrittlement of the vessel. Stainless steel normally in contact with the reactor coolant has a maximum cobalt content of 0.05 wt percent. The Ni-Cr-Fe Alloy 600 base metal and Alloy 82/182 weld filler metal are not used in Ni-Cr-Fe applications. Alloy 690 base metal and Alloy 52/52M/152 weld filler metal are used in Ni-Cr-Fe applications. The Ni-Cr-Fe base metal in contact with the reactor coolant has a limited sulfur content not exceeding 0.02 wt percent. The initial Charpy V-notch minimum upper-shelf fracture energy level for the RPV beltline materials, including welds, is 101.7 N-m (75 ft-lbs). The maximum initial nil-ductility reference temperature (RT_{NDT}) of the RPV is -20 °C (-4 °F). Materials are evaluated with regard to the effects of chemistry (copper content), initial upper-shelf energy, and neutron fluence to assure that 67.8 N-m (50 ft-lbs) upper-shelf energy is maintained throughout the life of the vessel. The RPV studs are manufactured of high-strength bolting steel. The closure studs are the necked-down type and are screwed into tapped blind holes in the RPV flange.

An RPV material surveillance program monitors the RPV beltline materials for changes in fracture toughness resulting from exposure to neutron irradiation and the thermal environment. The materials selected for the reactor vessel surveillance program are those that are adjacent to the active height of the core. Using the maximum initial RT_{NDT} values, maximum nickel and copper contents allowed in the RPV, and a 60 effective full power year (EFPY) fluence, the limiting RPV beltline material for the U.S. EPR is predicted to be Weld No. 2 (i.e., weld between upper and lower core shells).

There are four surveillance capsules located in guide baskets bolted to the outside of the core barrel and positioned directly opposite the center portion of the core. All four irradiated capsules contain the same type and number of mechanical test specimens, neutron dosimeter, and temperature monitors. Data from the tested material samples are used to predict the material property changes to the RPV. RPV specimen welds are made of the same weld wire heat, flux, and procedure as the respective RPV weld. The following materials are included in the reactor vessel surveillance program:

- Weld No. 2
- Weld No, 3 (weld between lower core shell and the transition ring)
- Upper core shell forging
- Lower core shell forging
- Heat affected zone from a core shell forging and RPV Weld No. 2

The neutron fluence on the vessel material test specimens and the vessel itself are determined based on core-follow calculations of the cycle-by-cycle operation. Non-destructive examinations of the RPV and its appurtenances are also performed. A COL applicant that references the U.S. EPR design will identify the implementation milestones for the material surveillance program.

ITAAC: Item 2.2 in FSAR Tier 1, Table 2.2.1-5 indicates that inspections of the as-built system will be conducted to verify that the as-built RPV and heavy reflector conform to the functional arrangement shown in FSAR Tier 1, Figure 2.2.1-2, "Reactor Pressure Vessel Functional Arrangement." Item 3.8, in FSAR Tier 1, Table 2.2.1-5 indicates that type tests, tests, analyses, or a combination of tests and analyses will be performed, for the first plant only, to verify that the RPV internals can withstand the effects of flow-induced vibration.

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

5.3.1.3 *Regulatory Basis*

The relevant requirements of NRC 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. 10 CFR Part 50, Appendix A, GDC 1, and GDC 30, 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.
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, "Handling, Storage, and Shipping," as it relates to onsite material cleaning control.

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

10. 10 CFR Part 50, Appendix H, "Reactor Vessel Material Surveillance Program Requirements," as it relates to the determination and monitoring of fracture toughness.

Acceptance criteria adequate to meet the above requirements include:

1. RG 1.31, as it relates to the control of welding in fabricating and joining safety-related austenitic stainless steel components and systems.

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

3. RG 1.36, as it relates to acceptance criteria for compatibility of austenitic stainless steel with thermal insulation.

4. RG 1.37, as it relates to the quality of water used for final cleaning or flushing of finished surfaces during installation.

5. RG 1.43, as it relates to criteria to limit the occurrence of under-clad cracking in low-allow steel safety-related components clad with stainless steel.

6. RG 1.44, as it relates to the compatibility of RCPB materials with the reactor coolant and the avoidance of stress corrosion cracking.

7. RG 1.50, as it relates to controlling preheat temperature when welding low-alloy steel.

8. RG 1.65, "Materials and Inspections for Reactor Vessel Closure Studs," October 1973.

9. RG 1.71, as it relates to welder qualification.

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

11. 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 FSAR Tier 2, Section 5.3.1, "Reactor Vessel Materials," in accordance with SRP Section 5.3.1.

In order to evaluate reactor pressure vessel functionality and ensure standardization of the vessel, the staff requested in RAI 64, Question 05.03.01-1 that the applicant provide key RPV dimensions (e.g., thickness, height, width, and location of nozzles) in the FSAR. In an October 22, 2008, response, the applicant stated that FSAR Tier 2, Section 5.3.3.1 and FSAR Tier 2, Figure 5.3-4, "Reactor Pressure Vessel," will be revised and that FSAR Tier 2, Table 5.3-7, "Reactor Pressure Vessel Design Data," will be added to incorporate these dimensions. The staff confirmed that Revision 1 of the U.S. EPR FSAR, dated May 29, 2009, contains the changes committed to in the RAI response. In RAI 64, Question 05.03.01-2, the staff requested that the applicant provide the thickness for the cladding. In an October 22, 2008, response, the applicant stated that the nominal cladding thickness was 7.5 millimeters (0.295 in.) and that this information would be added to FSAR Tier 2, Table 5.3-7. The staff confirmed that Revision 1 of the U.S. EPR FSAR, dated May 29, 2009, contains the changes committed to in the RAI response. On this basis, the staff finds the applicant responses to RAI 64, Question 05.03.01-1 and to the question on cladding thickness in RAI 994, Question 05.03.01-2 to be acceptable.

In RAI 64, Question 05.03.01-2, the staff requested that the applicant clarify the material used for the vessel cladding and describe the process for applying it. In an October 22, 2008, response, the applicant stated that cladding would be deposited by weld metal overlay and that FSAR Tier 2, Sections 1.2.3.2.1, 5.3.1.1, and 5.3.1.2 would be revised to clarify the use of either stainless steel or Ni-Cr-Fe alloy. The applicant incorporated this information in Revision 1 to the FSAR dated May 29, 2009. However, the staff determined that additional clarification was needed to make a safety finding regarding the materials used for cladding the RPV. In RAI 202, Question 05.03.01-7, the staff requested that the applicant specify the material types used to clad the RPV or provide a COL information item requiring an applicant that references the U.S. EPR standard plant design to select the specific material used for cladding. In an April 15, 2009, response, the applicant stated that the cladding materials were primarily Alloy 308L/309L but that Alloy 52/52M/152 would be used in areas where Alloy 690 attachments are to be welded to the clad surface. In an April 25, 2009, response to RAI 202, Question 05.03.01-7, the applicant did not commit to revise the FSAR. In RAI 232, Question 05.03.01-10a, the staff further requested that the applicant identify the specific areas of the reactor vessel that are clad with Alloy 308L/309L or Alloy 52/52M/152 and to include this information in the FSAR. In a July 7, 2009, response, the applicant stated that the internal surfaces of the RPV are clad with Alloy 309L/308L except for the areas where the Alloy 690 radial keys are welded. The staff finds the applicant's response to RAI 232, Question 05.03.01-10a to be acceptable. The staff's evaluation of the adequacy of the cladding material is discussed below. The applicant's responses to RAI 64, Question 05.03.01-2 and RAI 202, Question 05.03.01-7 are acceptable as supplemented by the applicant's response to RAI 232, Question 05.03.01-10a. The staff will verify that FSAR Tier 2, Section 5.3.1.2 is modified as shown in the markup copy provided in response to RAI 232, Question 05.03.01-10a. **RAI 232, Question 05.03.01-10a is being tracked as a confirmatory item.**

In RAI 213, Question 05.03.01-8 the staff requested that the applicant delete filler material type "304L" from FSAR Tier 2, Table 5.3-2, "Reactor Pressure Vessel Weld Material Specifications," and in Question 05.03.01-9 list filler material specifications and classifications for each filler material listed in FSAR Tier 2,Table 5.3-2. In a May 14, 2009, response, the applicant stated that FSAR Tier 2, Table 5.3-2, would be revised to delete type 304L filler metal but that the description of weld filler specifications would be added to FSAR Tier 2, Section 5.3.1.1 instead of FSAR Tier 2, Table 5.3-2. The staff finds the applicant's response to RAI 213, Questions 05.03.01-8 and 05.03.01-9 to be acceptable. The staff will verify that FSAR Tier 2, Section 5.3.1.1 and FSAR Tier 2, Table 5.3-2 are modified as shown in the markup copy provided by the applicant. **RAI 213, Questions 05.03.01-8 and 05.03.01-9 are being tracked as confirmatory items.**

In RAI 232, Question 05.03.01-10b, the staff requested that the applicant discuss the sequencing of the weld overlay cladding to ensure that defects are not introduced in the weld overlay cladding or adjacent reactor vessel material. In a July 7, 2009, response, the applicant stated that either 309L/308L cladding or 52/52M/152 cladding would be deposited for the full cladding thickness and applied directly to the low alloy steel base metal. The applicant also responded that the cladding at interfaces between the 309L/308L cladding and the Alloy 52/52M/152 cladding would be deposited with Alloy 52/52M/152 weld filler metal. The applicant's proposed sequencing of cladding deposition would not be expected, in and of itself, to result in cracking. Accordingly, the staff finds the applicant response to RAI 232, Question 05.03.01-10b to be acceptable and will confirm that the FSAR is revised as shown in the markup copy provided by the applicant. **RAI 232, Question 05.03.01-10b is being tracked as a confirmatory item.**

In RAI 232, Question 05.03.01-10c, the staff requested that the applicant clarify the cladding thickness based on the sequence used to deposit the cladding. In a July 7, 2009, response, the applicant stated that the nominal cladding thickness of 7.5 millimeters (0.295 in.) applies to the total cladding thickness regardless of cladding material. However, the applicant also indicated that where the cladding is also qualified as weld buttering, additional thicknesses of cladding may be warranted, as addressed in the response to RAI 232, Question 05.03.01-11b. The applicant's response did not address the minimum qualified thickness of cladding qualified as buttering, and because cladding thickness has implications for heat input to the RPV during subsequent welding operations. In a follow-up RAI 365, Question 05.03.01-14, the staff requested that the applicant specify the minimum thickness of the cladding when qualified as weld buttering. **RAI 365, Question 05.03.01-14 is being tracked as an open item.**

In RAI 232, Question 05.03.01-11a, the staff requested that the applicant confirm that the weld procedures and welders for the Alloy 52/52M/152 weld overlay are qualified for welding both corrosion resistant cladding and structural welds. In a July 7, 2009, response, the applicant stated that the weld deposits directly below the radial keys are considered part of the structural weld and are qualified in accordance with ASME Boiler and Pressure Vessel Code, Sections III and IX. The staff finds the applicant's response to RAI 232, Question 05.03.01-11a to be acceptable and will confirm that the FSAR is revised as shown in the markup copy provided by the applicant. **RAI 232, Question 05.03.01-11a is being tracked as a confirmatory item.**

In RAI 232, Question 05.03.01-11b, the staff requested that the applicant confirm that the attachment weld of the radial key to the cladding would be made with a low-heat input process to prevent under-clad cracking of the reactor vessel. This is critical if the RPV is not subsequently post-weld heat treated. In a July 7, 2009, response to RAI 232, Question 05.03.01-11b, the applicant stated that qualification of the welding procedures and processes in accordance with ASME Sections III and IX provides acceptable controls to ensure the integrity of the RPV. The applicant clarified that the radial-key attachment weld would be post-weld heat-treated, unless the cladding is qualified as buttering, in which case the subsequent weld is not required by the ASME Code to be post-weld heat-treated. The applicant further clarified that in a manufacturing sequence used for an EPR RPV currently under fabrication, the cladding is qualified as buttering with a thickness of 9.9 mm (0.39 in.), and the radial key attachment welds are not post-weld heat-treated. However, the transition ring with the radial-key attachment welds is post-weld heat-treated in the RPV construction sequence. Therefore, the radial-key attachment welds receive an indirect, post-weld heat-treatment as a result of the heat treatment of the RPV. Because the option exists for the applicant to fabricate the radial-key attachment welds without a subsequent post-weld heat treatment, and because the applicant did not confirm that a low-heat-input weld process will be used, the staff requested that the applicant confirm that a low-heat-input weld process will be used for sequences where the radial-key attachment welds are made without subsequent post-weld heat treatment. Therefore, the staff issued follow-up RAI 365, Question 05.03.01-15 to address this concern. **RAI 365, Question 05.03.01-15 is being tracked as an open item.**

In RAI 232, Question 05.03.01-11c, the staff requested that the applicant provide the joint design for the radial-key-to-cladding attachment weld, discuss whether the joint design is full penetration, and discuss the welding processes to be used. In a July 7, 2009, response, the applicant stated that the attachment welds to the reactor vessel are required by ASME Section III, NB-4433, to be full penetration welds. The applicant responded to the question on welding processes in a July 7, 2009, response to RAI 232, Question 05.03.01-11b. The staff finds the joint design as discussed in applicant's response to RAI 232, Question 05.03.01-11c meets the requirements of the ASME Code, and is therefore acceptable.

In RAI 357, Question 05.03.01-13, the staff requested that the applicant discuss operating experience with the use of bolted connections that attach the RVSP specimen guide baskets to the outside of the core barrel and how these bolted connections will maintain their structural integrity for the life of the plant for anticipated degradation mechanisms. This question is important since this attachment is typically made with welds in U.S. operating experience and this is a new and different approach with no precedents. GDC 4 requires that structures, systems and components important to safety to be designed to accommodate the effects of and to be compatible with the environmental conditions associated with operations. **RAI 357, Question 05.03.01-13 is being tracked as an open item**.

The staff reviewed the U.S. EPR RPV materials to ensure they meet relevant requirements of GDC 1 and GDC 30 and 10 CFR 50.55a(a)(1) related to material specifications, fabrication, and non-destructive examination. Compliance with these requirements will determine whether the RPV materials are adequate to ensure a quality product commensurate with the importance of the safety function to be performed. The material specifications for the EPR design are in accordance with the requirements of the ASME Code, Section III to 10 CFR Part 50, as discussed in Section 5.2.3 of this report. In addition, the staff's evaluation of whether the design of the RPV meets ASME Code, Section III, Class 1 requirements, is in Chapter 3 of this report. Furthermore, the RPV and its appurtenances are fabricated and installed in accordance with ASME Code, Section III, NB-4000. The NDE of the RPV and its appurtenances is conducted in accordance with ASME Code, Section III requirements. Examination of the RPV and its appurtenances by NDE complies with ASME Code, Section III, NB-5130. The staff finds this acceptable, because compliance with ASME Code, Section III to 10 CFR Part 50 constitutes an adequate basis for satisfying the requirements of GDC 1 and 30 and 10 CFR 50.55a(a)(1) as they relate to the material specifications, fabrication, and NDE of RPV materials.

The staff's evaluation of the special controls and special processes for welding of ferritic and austenitic stainless steels is provided in Section 5.2.3 of this report.

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 (Also, see the discussion on fracture toughness below). The applicant indicates in

FSAR Tier 2, Table 5.3-3, “Maximum Limits for RPV and Appurtenances Material Composition,” that restrictive, maximum-content limits would be imposed on the critical residual elements (copper, nickel, phosphorous, etc.) in the materials of the RPV beltline. Specified limits for RPV materials used in the core-beltline region are as follows:

- Base Materials: 0.06 percent maximum copper, 0.008 percent maximum phosphorus, 0.005 percent maximum sulfur, and 0.8 percent maximum nickel (forging)

- Weld Materials: 0.06 percent maximum copper, 0.012 percent maximum phosphorus, 0.015 percent maximum sulfur, and 1.2 percent maximum nickel

The staff finds the applicant’s limits are acceptable, because the chemical content controls imposed on the RPV materials meet the guidelines of RG 1.99, Revision 2.

The tests for fracture toughness of RPV materials specified in the FSAR are in accordance with the requirements of the ASME Code, Section III, NB-2300 and Appendix G to 10 CFR Part 50. Specifically, the staff verified that the applicant's initial Charpy V-notch minimum upper-shelf, fracture-energy level for the RPV beltline base-metal transverse direction and welds is 101.7 N-m (75 ft-lbs). FSAR Tier 2, Table 5.3-4, “60 EFPY RPV Fluence, Upper Shelf Energy, ART, and RT_{NDT} Projections” indicates that the end-of-life (EOL) values for the upper-shelf energy (USE) are greater than 67.8 N-m (50 ft-lbs) for the beltline forgings and welds. The staff confirmed this by using calculations following RG 1.99 guidelines for the beltline forgings and welds.

FSAR Tier 2, Section 5.3.1.5, “Fracture Toughness,” states that the vessel fracture toughness data are calculated in accordance with RG 1.99, Revision 2, but use of the procedures in RG 1.99, Revision 2 is only valid for a nominal irradiation temperature of 287.8 °C (550 °F). This is because irradiation below 273.9 °C (525 °F) produces greater embrittlement. In RAI 64, Question 05.03.01-3, the staff requested that the applicant confirm that operating temperature of the RPV is above 273.9 °C (525 °F). In an October 22, 2008, response, the applicant confirmed that operating temperatures exceed the 273.9 °C (525 °F) threshold value as specified in the FSAR Tier 2, Table 5.1-1, “RCS Design and Operating Parameters.” The staff verified that the operating temperatures specified in FSAR Tier 2, Table 5.1-1, are such that the reactor will be operated in temperatures to preclude any adverse effects caused by embrittlement, therefore, the response to RAI 64, Question 05.03.01-3 is acceptable.

As described above, the predicted EOL Charpy USE throughout the life of the vessel and adjusted reference temperature for the RPV materials meet 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 RPV. This methodology will provide adequate safety margins 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 GDC 31 and 10 CFR 50.55a regarding the prevention of fracture of the RPV. Therefore, the staff finds that the applicant has adequately met the requirements of GDC 14, GDC 31, and 10 CFR 50.55a for the RPV.

The design of an RPV must consider the potential embrittlement of RPV 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 RPV. Appendix H to 10 CFR Part 50 details the requirements of such a program.

To meet the requirements of GDC 32, the U.S. EPR design includes provisions for a material surveillance program to monitor changes in the fracture toughness caused by exposure of the RPV beltline materials to neutron radiation. Appendix H to 10 CFR Part 50 requires that the surveillance program for the U.S. EPR RPV meet ASTM Std E-185. The U.S. EPR surveillance capsule program includes four specimen capsules, with archive materials available for additional replacement 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 RPV. 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 U.S. EPR design are in accordance with those specified in ASTM Standard 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 RPV. Thus, the staff finds the U.S. EPR design meets the requirements of GDC 32.

FSAR Tier 2, Section 5.3.1.6, "Material Surveillance," discusses the RPV material surveillance program based on a 60-year design life. FSAR Tier 2, Section 5.3.1.6 also states that the program will use four specimen capsules. In RAI 202, Question 05.03.01-6 the staff requested that the applicant confirm that the predicted transition temperature shift (ΔRT_{NDT}) was calculated using RG 1.99, Revision 2. In an April 15, 2009, response, the applicant stated that the calculated transition temperature shift is 31.67 °C (89 °F) per RG 1.99, Revision 2. The staff performed a confirmatory calculation of the temperature shift which is consistent with the applicant's calculation and supports the basis for using four capsules. On this basis, the staff finds the applicant response to RAI 202, Question 05.03.01-6 to be acceptable.

In FSAR Tier 2, Section 5.3.1.6, the applicant did not discuss locations of capsules or lead factors. In Part (a) to RAI 64, Question 05.03.01-5, the staff requested that the applicant discuss the locations and associated lead factors of the surveillance capsules in the FSAR. The staff confirmed that Revision 1 to the U.S. EPR FSAR, dated May 29, 2009, contains the revised FSAR Tier 2, Section 5.3.1.6 to address Part (a) of RAI 64, Question 05.03.01-5 by describing the locations and associated lead factors of the surveillance capsules (including specimen guide basket fabrication materials, attachment, and capsule position). On this basis, the staff finds the FSAR revision acceptable.

FSAR Tier 2, Section 5.3.1.6 states that the material surveillance program has been fully described. In Part (b) to RAI 64, Question 05.03.01-5, the staff requested that the applicant provide additional information (e.g., capsule environment and capsule preparation) to evaluate the adequacy of a fully described surveillance capsule program. The staff also requested that any information on the surveillance program that will not be included in the FSAR should be identified as COL information items. In an October 22, 2008, response, the applicant stated that FSAR Tier 2, Section 5.3.1.6 constitutes a fully described program, but also provided a markup copy of the FSAR with additional information on capsule environment and capsule preparation. The staff confirmed that Revision 1 of the U.S. EPR FSAR, dated May 29, 2009, contains the changes committed to in the RAI response. With this additional information, the staff finds the surveillance capsule program is fully described, and that the applicant's response to RAI 64, Question 05.03.01-5b is acceptable.

The RPV studs, nuts, and washers for the main closure flange are manufactured using ASME SA-540 Grade B24V (4340V Mod) Class 3 material, and their testing conforms to guidance in RG 1.65. The applicant indicated that the material used to fabricate the closure studs meets the fracture toughness requirements of Section III of the ASME Code and Appendix G to 10 CFR Part 50. NDE of the studs will be performed in accordance with Section III of the ASME Code, Subarticle NB-2580. The integrity of the U.S. EPR RPV closure studs is assured by conformance with the recommendations of RG 1.65, thus satisfying the quality standards requirements of GDC 1 and GDC 30 and 10 CFR 50.55a. Conformance with the recommendations of RG 1.65 also satisfies the requirement in GDC 31 for the prevention of fracture of the RCPB and the requirements of Appendix G to 10 CFR Part 50, as detailed in the provisions of Section III of the ASME Code. Conformance with the recommendations of RG 1.65 is further discussed in Section 3.13 of this report.

5.3.1.5 *Combined License Information Items*

The following is a list of item numbers and descriptions from FSAR Tier 2, Table 1.8-2

Table 5.3.1-1 U.S. EPR Combined License Information Items

Item No.	Description	FSAR Tier 2 Section
5.3-1	A COL applicant that references the U.S. EPR design certification will identify the implementation milestones for the material surveillance program.	5.3.1.6

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

5.3.1.6 *Conclusions*

Except for the open items discussed above, and for the reasons set forth above, the staff concludes that the U.S. EPR RPV material specifications, RPV manufacturing and fabrication processes, NDE methods of the RPV and its appurtenances, fracture toughness testing, material surveillance, and RPV 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 corresponding requirements of GDC 1, GDC 14, GDC 30, GDC 31, and GDC 32.

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

5.3.2.1 *Introduction*

Radiation embrittlement causes a reduction in the ductility of the RPV beltline materials. This reduction is measured in terms of the adjusted reference nil-ductility 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 RPV. Pressure-temperature limits, derived using linear-elastic fracture mechanics principles, provide margins of safety to prevent nonductile fracture during normal operation, heat-up, cooldown, anticipated operational occurrences, and system hydrostatic, preservice, and inservice leakage tests.

Pressurized thermal shock (PTS) events are potential transients in a pressurized-water RPV that can cause severe overcooling of the vessel wall, followed by immediate repressurization. The thermal stresses, caused when the inside surface of the RPV 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 those in the RPV beltline region where neutron radiation gradually embrittles the material over time.

5.3.2.2 *Summary of Application*

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

FSAR Tier 2: The applicant provided an FSAR Tier 2 description of how it addresses P-T limits, PTS, and Charpy upper-shelf energy in Section 5.3.2, “Pressure-Temperature Limits, Pressurized Thermal Shock, and Charpy Upper-Shelf Energy Data and Analyses,” summarized here in part as follows:

RCS pressure and temperature limits for heatup, cooldown, low temperature operation, criticality, and hydrostatic testing, low temperature overpressure protection settings, and pressurizer safety-relief valve lift settings, as well as heatup and cooldown rates will be established and documented for LCO 3.4.3, “RCS Pressure and Temperature Limits”; and LCO 3.4.11, “Low Temperature Overpressure Protection.” Topical Report ANP-10283P, “U.S. EPR Pressure-Temperature Limits Methodology for RCS Heatup and Cooldown,” contains the detailed methodology for developing the P-T limit curves. Revised P-T limits will be provided to the NRC for each reactor vessel fluence period. A COL applicant that references the U.S. EPR design will provide plant-specific P-T limits consistent with an approved methodology. Plant operating procedures will be used to ensure that the P-T limits are not exceeded during conditions of normal operation, abnormal operational occurrences, and system hydrostatic tests.

The FSAR states that the RPV design provides protection against unstable crack growth under faulted conditions. An analysis was performed by the applicant to determine the RPV pressurized thermal shock reference temperatures (RT_{PTS}) applicable to 60 EFPYs. The RT_{PTS} values were calculated for various RPV materials over 60 EFPYs with the most limiting core design. The FSAR contains a table which provides the projected fluence, initial and End of Life upper-shelf energy, and adjusted reference temperature (ART) values for the RPV beltline materials.

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

Technical Specifications: The TS associated with Section 5.3.2 of this report are given in FSAR Tier 2, Chapter 16, Sections 3.4.3, B.3.4.3, 3.4.11, and B 3.4.11. In addition, TS 5.6.4 specifies the content of the RCS pressure temperature limits report.

5.3.2.3 *Regulatory Basis*

The relevant requirements of NRC 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 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. 10 CFR Part 50, Appendix A, GDC 1, 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.

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

Acceptance criteria adequate to meet the above requirements include:

1. RG 1.99 as it relates to RPV beltline material properties.

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

5.3.2.4 *Technical Evaluation*

Pressure-Temperature Limits

FSAR Tier 2, Section 5.3.2 states that the COL applicant will provide a plant-specific pressure temperature limits report. In RAI 64, Question 05.03.02-5, the staff requested that the applicant discuss their rationale for not including the PTLR in the FSAR and how the adequacy of TS requirements can be assured at the design certification stage if generic P-T limits or a PTLR is not provided at the design certification stage. In an April 30, 2009 response, the applicant stated that ANP-10283P, which contains the detailed methodology for developing the P-T limit curves, will be revised to include the complete methodology to support the PTLR in compliance with GL 96-03. The revised technical report, ANP-10283P, Revision 1, was submitted to the NRC in a letter dated April 30, 2009. The applicant indicated that the revised technical report contains bounding P-T limit curves based on bounding material properties in the design specifications, and a generic PTLR. The staff's evaluation of ANP-10283P, Revision 1, is not yet complete. The staff will update this report to reflect the final disposition of the technical report. **RAI 64, Question 05.03.02-5, which is associated with the above request and technical report, is being tracked as an open item.**

In RAI 278, Question 05.03.02-7, the staff requested that the applicant provide a table of the data points (reactor coolant temperature vs. pressure) for each P-T curve displayed in Technical Report ANP-10283, Revision 1. **RAI 278, Question 05.03.02-7 is being tracked as an open item.**

In RAI 278, Question 05.03.02-8, the staff requested that the applicant clarify the thickness value (including vessel thickness and cladding thickness) used to calculate the fluence at the 1/4t and 3/4t locations for all materials provided in Technical Report ANP-10283, Revision 1. **RAI 278, Question 05.03.02-8 is being tracked as an open item.**

In RAI 278, Question 05.03.02-9, the staff requested that the applicant provide all values (i.e., chemistry factors, fluence factors, margins, ΔRT_{NDT}, etc.) used to calculate the ART at the 1/4t and 3/4t locations for all applicable materials provided in Technical Report ANP-10283, Revision 1. **RAI 278, Question 05.03.02-9 is being tracked as an open item.**

In RAI 278, Question 05.03.02-10, the staff requested that the applicant address PTLR Criterion 4 (GL 96-03) and clearly identify both the limiting adjusted reference temperature (ART) values and limiting materials at the 1/4t and 3/4 t locations (t= vessel thickness) used in the development of the P-T limits. **RAI 278, Question 05.03.02-10 is being tracked as an open item.**

Pressurized Thermal Shock

PTS events are potential transients in a pressurized-water RPV that can cause severe overcooling of the vessel wall, followed by immediate re-pressurization. The thermal stresses, caused when the inside surface of the RPV 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 RPV beltline where neutron radiation gradually embrittles the material over time.

The PTS rule established screening criteria to serve as a limiting level of RPV 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 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 U.S. EPR reactor beltline design consists of two forgings and one circumferential weld. The U.S. EPR beltline forging material will contain a maximum of 0.06 weight percent copper and 0.80 weight percent nickel, and weld metal will contain a maximum of 0.06 weight percent copper and 1.20 weight percent nickel. The initial RT_{NDT} is -20 °C (-4 °F) for the reactor vessel. FSAR Tier 2, Table 5.3-4 provides the EOL RT_{NDT}, RT_{PTS}, USE and the projected fluence at 60 EFPY for the RPV beltline materials. The EOL RT_{NDT} is based on the 60 EFPY fluence, rather than the EOL fluence, which 10 CFR 50.61 defines as the fluence protected to the license expiration date. Since the EOL fluence for the license expiration date would be calculated for 40 EFPY, the use of 60 EFPY is conservative. Using the projected fluence values at 60 EFPY, the staff verified that the limiting RT_{PTS} value for the forgings is 21.28 °C (70.3 °F), and for the circumferential welds, 60.61 °C (141.1 °F), both of which satisfy the PTS screening criteria of 10 CFR 50.61. In addition, FSAR Tier 2, Section 5.3.2.3, “Pressurized Thermal Shock,” states that a COL applicant that references the U.S. EPR design will provide plant-specific RT_{PTS} values in accordance with 10 CFR 50.61 for vessel beltline materials.

Upper-Shelf Energy

FSAR Tier 2, Section 5.3.1.5 states that the fracture toughness data for the U.S. EPR vessel is calculated in accordance with the requirements of RG 1.99, Revision 2, and provided in FSAR Tier 2, Table 5.3-4. The tests for fracture toughness of RPV materials specified in the FSAR are in accordance with ASME Code, Section III, Paragraph NB-2300, and 10 CFR Part 50 Appendix G. The applicant confirmed that the initial Charpy V-notch USE level for the RPV are 101.7 N-m (75 ft-lbs). Also, the projected EOL USE values for the RPV materials are greater than 67.8 N-m (50 ft-lbs). The staff verified that the applicant used the methodology recommended in RG 1.99, Revision 2 to calculate USE given the 60 EFPY fluence. Therefore, the staff finds that the applicant’s USE values meet the requirements of Appendix G to 10 CFR Part 50 and are, therefore, acceptable.

5.3.2.5 *Combined License Information Items*

The following is a list of item numbers and descriptions from FSAR Tier 2, Table 1.8-2.

Table 5.3.2-1 U.S. EPR Combined License Information Items

Item No.	Description	FSAR Tier 2 Section
5.3-2	A COL applicant that references the U.S. EPR design certification will provide plant-specific pressure and temperature limits consistent with an approved methodology.	5.3.2.1
5.3-3	A COL applicant/holder that references the U.S. EPR design certification will provide plant-specific RT_{PTS} values in accordance with 10 CFR 50.61 for vessel beltline materials	5.3.2.3

The staff finds the above listing to be complete. Also, the list adequately describes actions necessary for the COL applicant or holder. No additional COL information items need to be included in FSAR Tier 2, Table 1.8-2 for P-T limits, PTS, or Charpy USE considerations.

5.3.2.6 *Conclusions*

Pressure-Temperature Limits

Except for the open items discussed above, the staff, for the reasons set forth above, concludes that the P-T limits for the RCS for operating and testing conditions to ensure adequate safety margins against non-ductile and rapidly propagating failure are in compliance with the fracture toughness criteria of Appendix G to 10 CFR Part 50. Further, the change in fracture toughness properties of the RPV beltline materials during operation will be verified through a material surveillance program developed in compliance with Appendix H to 10 CFR Part 50. The use of operating limits, as determined by the criteria defined in SRP Section 5.3.2 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 G to 10 CFR Part 50, and GDC 1, 14, 31, and 32, with respect to P-T limits.

Pressurized Thermal Shock

For the reasons set forth above, the staff concludes that the U.S. EPR RPV meets the relevant requirements of 10 CFR 50.61, because calculations show that the RPV beltline materials will be substantially below the PTS screening criteria at the license expiration date. The COL applicant will provide RT_{PTS} values based on plant-specific material properties and projected neutron fluence for the end of plant life.

Upper Shelf Energy

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. As described above, the staff concludes that the U.S. EPR RPV 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 report, 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*

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

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

The reactor vessel is positioned and supported in the containment by a steel structure ring embedded in the concrete of the reactor pit. The RV and closure head form the enclosure which contains the reactor core. The vessel holds the internals that support the fuel assemblies and that direct the reactor coolant flow through the reactor core. Eight nozzles provide inlet and outlet connections to the four RCS loops. The reactor vessel closure head is attached to the RV with stud-nut-washer sets. The joint between the RV and the closure head is sealed by two seals located in concentric, circular recesses on the head flange. A seal leak-off line drains from the space between the two head flange seals. The closure head can be removed for refueling and vessel maintenance. The semi-hemispherical upper head contains penetrations to accommodate the adapters for the CRDMs, in-core instrumentation, thermocouple tubes and vent piping. Eighty nine CRDMs are installed on top of the closure head. They are affixed to adaptors welded to the RV head. Instrumentation adaptors are mounted to the vessel head via welded adapter penetrations to monitor the core temperature and neutron flux. The semi-hemispherical lower head does not contain any penetrations. The internal and external surfaces of the RV are accessible for periodic inspection using visual and NDE techniques. Shop ultrasonic examinations are performed on internally clad surfaces to confirm an adequate cladding bond and to facilitate later volumetric testing of the base metal from the inside surface.

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 NRC 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. 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 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, 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, as it relates to the control of welding in fabricating and joining safety-related austenitic stainless steel components and systems.

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

3. RG 1.38, as it relates to quality assurance requirements for packaging, shipping, receiving and handling of items for water-cooled nuclear power plants.

4. RG 1.43, as it relates to criteria to limit the occurrence of under-clad cracking in low-allow steel safety-related components clad with stainless steel.

5. RG 1.44, as it relates to the compatibility of RCPB materials with the reactor coolant and the avoidance of stress corrosion cracking.

6. RG 1.71, as it relates to welder requalification.

7. RG 1.99, as it relates to RV fracture toughness.

8. RG 1.190, as it relates to the RV material surveillance program.

5.3.3.4 *Technical Evaluation*

The staff reviewed the fracture toughness of the ferritic materials for the RV, the pressure and temperature 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 following information which is discussed in other sections of this report.

- RCPB materials (Section 5.2.3)
- RCS pressure boundary inservice inspection and testing (Section 5.2.4)
- RV materials (Section 5.3.1)
- P/T limits, pressurized thermal shock, and upper-shelf energy (Section 5.3.2)

The staff has reviewed the following areas related to the integrity of the U.S. EPR RV: 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 Boiler and Pressure Vessel 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 RAI 64, Question 05.03.03-1, the staff requested that the applicant describe in the FSAR the provisions that are in place for the U.S. EPR design if thermal annealing of the RV should be necessary. In an October 22, 2008, response, the applicant stated that the fracture toughness evaluations performed to date for thermal and radiation aging indicate that a thermal annealing of the vessel will not be necessary to maintain the required material properties over the life of the plant or even into life extension. The applicant also stated that any instance arising during plant operation that would necessitate this action will be addressed by the licensee in accordance to 10 CFR 50.66, "Requirements for thermal annealing of the reactor pressure vessel." The staff finds that the response to RAI 64, Question 05.03.03-1 adequately describes provisions for thermal annealing and is acceptable. In follow-up RAI 180, 05.03.03-2, the staff requested that the applicant revise the FSAR to provide the response to RAI 64, 05.03.03-1. In a February 20, 2009, response, the applicant stated that FSAR Tier 2, Section 5.3.3.7 will be revised to include the response to RAI 64, Question 05.03.03-1. **RAI 64, Question 05.03.03-1 is being tracked as a confirmatory item.**

The special considerations relating to fracture toughness and radiation effects limit the basic Code-approved materials that are currently acceptable for most parts of reactor vessels to SA 533 Grade B Class 1, SA 508 Grade 2 Class 1, and SA 508 Grade 3 Class 1. The U.S. EPR design utilizes SA 508 Grade 3 Class 1 for the RV, and thus the materials of construction are acceptable to the staff. The acceptability of fabrication methods and inspection requirements has been addressed in FSAR Tier 2, Section 5.3.1.

FSAR Tier 2, Revision 1, Section 5.3.3.5 states that protective measures taken during shipment of the reactor vessel and its installation at the site verify that the as-built characteristics of the reactor vessel are not degraded by improper handling. Vessel openings are sealed to prevent the entrance of moisture and debris. External surfaces are painted with a strippable coating before shipment. Coatings are removed during the installation of the components. This is acceptable to the staff because proper cleanliness and freedom from contamination during all stages of shipping, storage and installation of the RV is ensured. The RV will operate under conditions, procedures, and protective devices that ensure that the vessel design conditions will not be exceeded during normal reactor operation, maintenance, testing, and anticipated transients. Section 5.3.2 addresses how the acceptance criteria in Appendix G to 10 CFR Part 50 and 10 CFR 50.61 for PTS has been adequately addressed. The staff finds the design acceptable to ensure that the vessel remains leak-tight enough to support adequate core cooling.

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 U.S. EPR RV are 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 necessary to enable these inspections have been provided. Access is also provided to RV nozzle safe ends. The insulation covering the nozzle-to-pipe welds can be removed. Reactor cavity and thermal insulation allows access to the outside surface of the vessel. Thus, the staff finds that the U.S. EPR design appropriately enables access for inspections of the RV.

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.

5.3.3.5 *Combined License Information Items*

For the reactor vessel integrity, no COL information items have been identified in FSAR Tier 2, Table 1.8-2. The staff finds this acceptable, because the proposed ITAAC and initial plant test program assure that the reactor vessel will be constructed in accordance with the certified design.

5.3.3.6 *Conclusions*

In view of the foregoing (in Sections 5.3.1 – 5.3.3 of this report), the staff concludes that the structural integrity of the U.S. EPR RV meets the requirements of GDC 1, GDC 4, GDC 14, GDC 30, GDC 31, and GDC 32 of Appendix A to 10 CFR Part 50; Appendices G and H to 10 CFR Part 50; and 10 CFR 50.55a, except for the open items identified above. Therefore, the staff finds the structural integrity of the U.S. EPR RV to be acceptable. The basis for this conclusion is that the design, materials, fabrication, inspection, and quality assurance measures of the U.S. EPR conform to the regulatory guides discussed above, and, accordingly comply with the applicable NRC regulations as well as the rules of Section III of the ASME Code. The U.S. EPR design meets the fracture toughness requirements of the regulations and Section III of the ASME Code, including requirements for surveillance of vessel 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 Nonductile Failure," of ASME Code, Section III, and Appendix G to 10 CFR Part 50.

5.4 Component and Subsystem Design

This section discusses the performance requirements and design features of the RCP, SG, reactor coolant piping, RHRS, PZR, PRT, RCS high point vents, SRV, and component supports for the U.S. EPR design.

The RCS components are designed to operate in an environment when the reactor is at power. Components important to safety are designed to perform their safety functions in an environment degraded by a design-basis accident. The environmental qualifications (EQ) program for electrical, mechanical, instrument, and control components designated as safety-related or important to safety is addressed in FSAR Tier 2, Section 3.11, "Environmental Qualification of Mechanical and Electrical Equipment." The equipment addressed by the EQ program is identified in FSAR Tier 2, Appendix 3D.

5.4.1 Reactor Coolant Pumps

5.4.1.1 *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 RV 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.

Each RCP flywheel has a large mass and rotates at 1,200 revolutions per minute (rpm) during normal reactor operation. A loss of flywheel integrity could result in high energy missiles and excessive vibration of the reactor coolant pump assembly. The safety consequences could be significant because of possible damage to the reactor coolant system, containment, or engineered safety features. Reactor coolant pump flywheel failure can also result in reduction or loss of forced coolant flow.

5.4.1.2 *Summary of Application*

FSAR Tier 1: In FSAR Tier 1, Section 2.2.1, "Reactor Coolant System," the applicant states that the RCPs provide flow. The RCP motors will include a device to prevent reverse rotation. The RCPs will also have rotational inertia to provide coastdown flow of reactor coolant on loss of power to the pump motors. Minimum flow (percent of initial flow) during four pump coastdown (for times ranging from 0 to 20 seconds) is specified in Table 2.2.1-4, "Minimum Flow (% of Initial Flow) During Four Pump Coastdown." The RCP standstill seal system (SSSS) can be closed or engaged when the RCP is stopped. Detailed FSAR Tier 1 mechanical, electrical, and instrumentation information associated with the reactor coolant pump design is specified in FSAR Tier 1, Tables 2.2.1-1 through 2.2.1-3.

FSAR Tier 2: The applicant has provided a FSAR Tier 2 description of the U.S. EPR RCPs in Section 5.4.1, summarized here in part as follows:

There are four identical RCP assemblies in the U.S. EPR design, one in each reactor coolant loop. The RCPs are centrifugal, single-stage pumps with mechanical shaft seals driven by asynchronous squirrel-cage induction motors. The motors are open and self-ventilated. Each RCP assembly has one common vertical shaft line for the pump and motor with main and auxiliary bearings, one single double thrust bearing, and a flywheel located at the top of the motor shaft. The flywheel consists of a sandwich of two circular steel discs mounted on the end of the motor shaft of each RCP. It provides sufficient inertia, in addition to the inertia of the RCP rotating assembly, for the RCPs to maintain departure from nucleate boiling (DNB) margin during the gradual loss of forced RCS flow that occurs during RCP coastdown. An anti-rotation device is mounted on the lower face of the flywheel to prevent the RCP from rotating in the reverse flow direction. The guide bearings are oil lubricated pad bearings. During normal motor operation and coastdown, no external oil pump is needed. Bearing lubrication is accomplished by the pumping action of the oil within the bearing. All parts of the pump in contact with the reactor coolant are austenitic stainless steel, except for seals, bearings, and gaskets. The shaft seal system is made up of a series of three seals and a standstill seal. The shaft seal design provides redundancy so that a failure of a single-seal stage will not result in an uncontrolled loss of reactor coolant. The standstill seal is a metal-to-metal contact seal that prevents leakage when the RCP has stopped and the three seal leak-off lines have been isolated. The RCPs receive a safety-related trip from the protection system during a loss-of-coolant accident (LOCA) or a stage two containment isolation signal.

ITAAC: Item 3.6 in FSAR Tier 1, Table 2.2.1-5 states that an inspection will be performed to verify that a device to prevent reverse rotation is installed on each RCP motor. Item 7.2 in FSAR Tier 1, Table 2.2.1-5 indicates that tests will be performed to verify that the RCPs provide the minimum coastdown flow as listed in FSAR Tier 1, Table 2.2.1-4. Item 7.3 in FSAR Tier 1, Table 2.2.1-5 indicates that testing and analysis will be performed to verify that (1) the RCP provides greater than the minimum required flow rate of 453,084 lpm/loop (119,692 gpm/loop), and (2) the RCP provides less than the maximum required flow rate of 509,751 lpm/loop

(134,662 gpm/loop). Item 7.4 in FSAR Tier 1, Table 2.2.1-5 indicates that testing will be performed to verify that the SSSS can be engaged when the RCP is stopped. Additional ITAAC will be performed to verify the detailed FSAR Tier 1 mechanical, electrical, and instrumentation information associated with the RCPs.

Technical Specifications: The TS related to the RCPs can be found in FSAR Tier 2, Chapter 16, Sections 3.4 and B 3.4. In addition, TS 5.5.6 requires a RCP flywheel inspection program. This program provides for the inspection of each reactor coolant pump flywheel per the recommendations of Regulatory Position C.4.b of RG 1.14, "Reactor Coolant Pump Flywheel Integrity," Revision 1, August 1975.

5.4.1.3 *Regulatory Basis*

The relevant requirements of NRC 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 Part 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 structures, systems, and components of nuclear power plants from the effects of missiles that might result from reactor coolant pump 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.

2. RG 1.189, "Fire Protection for Nuclear Power Plants," as it relates to the fire protection requirements for the RCP oil collection system.

5.4.1.4 *Technical Evaluation*

5.4.1.4.1 RCP Design

Detailed information about the RCPs and the staff's evaluation and conclusion regarding U. S. EPR RCP design features and performance requirements are discussed in the following sections of this report:

- 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”
- 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”

5.4.1.4.2 RCP Flywheel Integrity

The staff reviewed FSAR Tier 2, Section 5.4.1.6, “Pump Flywheel Integrity,” which describes the materials used in the fabrication of the reactor coolant pump flywheel to ensure 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 SRP Section 5.4.1.1, “Pump Flywheel Integrity.” The following evaluation addresses the acceptance criteria outlined in SRP Section 5.4.1.1.

Material Selection, Fabrication, and Fracture Toughness

FSAR Tier 2, Section 5.4.1.6.3, “Material Selection and Fabrication,” provides the material specifications for the reactor coolant pump flywheel. The flywheel is forged from ASME Code, SA-508, Grade 4N, Class 1 steel that is quenched and tempered. The material is vacuum treated to minimize impurities and flaws in the material, thereby improving its fracture toughness properties. RG 1.14 states that past evaluations have shown that ASME SA-533-B Class 1 and SA-508 Classes 2 and 3 materials generally have suitable toughness for typical flywheel applications, but other materials may be considered if the strength and toughness properties are evaluated and justified for the application. FSAR Tier 2, Section 5.4.1.6.4, “Fracture Toughness,” states that the reference nil-ductility transition temperature of the flywheel material is no greater than -45.6 °C (-50 °F), and is 32.2 °C (90 °F) below the reactor minimum operating service temperature of 4.4 °C (40 °F). This meets the fracture toughness guidelines provided in

Criteria B of RG 1.14; therefore, the fracture toughness of the ASME Code, SA-508, Grade 4N, Class 1 steel flywheel is sufficient for this application. In addition, ASME Code, SA-508, Grade 4N, Class 1 steel has a higher yield strength than SA-508 Classes 2 and 3 (85 ksi versus 50 ksi, respectively).

The ASME Code, SA-508, Grade 4N, Class 1 steel will be vacuum treated to degas the steel, thereby removing detrimental gases such as hydrogen, in accordance with SA-788 which meets the guidelines of RG 1.14. In addition, FSAR Tier 2, Section 5.4.1.6.3 incorporates the guidance of RG 1.14 with regards to prohibiting welding on the flywheel, and that all flame-cut surfaces of the flywheel are removed by machining to a depth of 12.7 mm (0.50 in.) minimum below the flame-cut surface to minimize any loss of fracture toughness during fabrication. 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.

Pre-service Inspection

The flywheel is inspected prior to being placed into service, which includes a dimensional inspection, a 100 percent ultrasonic inspection in accordance with ASME Code, Section III, NB-2540, and a surface inspection using liquid-penetrant or magnetic-particle examination of areas of high-stress concentrations and the adjacent surfaces. However, RG 1.14, Section C.4.a, states that following the spin test, each finished flywheel should receive a check of critical dimensions and a non-destructive examination. The non-destructive examination should include 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. In RAI 341, Question 05.04.01.01-2, the staff requested that FSAR Tier 2, Section 5.4.1.6.5, “Preservice Inspection,” specify that the surface and volumetric examinations will be performed after the spin test so that any flaws that have initiated or grown during the spin test can be detected. Also, the FSAR should specify that the flywheel will be inspected for critical dimensions after the spin test so that any dimensional changes can be detected. **RAI 341, Question 05.04.01.01-2, which is associated with the above request, is being tracked as an open item.**

Except for the open item indicated above, 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 discussed below, 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, design-basis loss-of-coolant accident, safe-shutdown earthquake without loss of structural integrity, and the potential of generating a missile. FSAR Tier 2, Section 5.4.1.6.1, “Design Basis,” stated that the flywheel is designed to minimize the possibility of generating high-energy missiles in the event of a design overspeed transient or postulated accident condition consistent with the requirements of GDC 4 of 10 CFR Part 50 and the guidelines of RG 1.14 and SRP Section 5.4.1.1.

FSAR Tier 2, Section 5.4.1.6 does not provide a reactor coolant pump flywheel analysis topical report as recommended by RG 1.14, Paragraph D.3 (which is referenced in SRP Section 5.4.1.1). RG 1.14 states that the recommendations of this regulatory guide will be used in evaluating all topical reports on flywheel integrity after January 1, 1976. In addition, RG 1.14 states that several analyses (critical flywheel speed for ductile fracture, nonductile fracture, and excessive deformation) should be submitted in a topical report to the NRC for evaluation. Therefore, in RAI 18, Question 05.04.01.01-1, the staff requested that a detailed technical or topical report concerning the reactor coolant pump flywheel analysis be provided for review in accordance with RG 1.14 and SRP Section 5.4.1.1 in order to meet the requirements of GDC 4. In a letter dated April 1, 2009, the applicant provided AREVA Report No. ANP-10294P, "U.S. EPR Reactor Coolant Pump Motor Flywheel Structural Analysis Technical Report," Revision 1, which is a detailed analysis for the RCP flywheel in accordance with the recommendations of RG 1.14 and SRP Section 5.4.1.1 in order to meet the requirements of GDC 4. In addition, FSAR Tier 2, Section 5.4.1.6, Revision 1, references the AREVA report for the flywheel analysis. AREVA Report No. ANP-10294P, Revision 1, 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 LOCA overspeed. The staff compared the evaluation in the report to the regulatory position of RG 1.14 for the flywheel design based on these critical speeds.

FSAR Tier 2, Section 5.4.1.6.2, "Design Description," and AREVA Report No. ANP-10294P, Revision 1, state that the flywheel assembly consists of two steel discs that are clamped together. The flywheel is fitted on the top part of the motor shaft by shrink fitting and vertically retained by a large nut at the top of the shaft. There are three sets of keys that transmit the torque from the shaft to the flywheel. For the ductile fracture analysis, the AREVA Report No. ANP-10294P, Revision 1, uses two methods for conservatism. First, it uses the elastic stress analysis method of the ASME Code, Section III, Appendix F-1331.1, and then uses the plastic deformation method using the Tresca criterion 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 ASME Code, Section III, Appendix F-1331.1 produced the most conservative result of 3,831 rpm at the outer radial edge of the keyway. The staff verified that the minimum calculated limiting speed (3,831 rpm), assuming a 12.7 mm (0.50 in.) crack, is at least twice the normal operating speed (1,200 rpm). The staff finds that the critical speed for ductile fracture meets the criteria in RG 1.14. In addition, the staff confirmed that the critical speed for ductile fracture (3,831 rpm) is greater than the LOCA overspeed, which is less than 1,500 rpm, assuming a LBB scenario.

For the non-ductile fracture analysis, the AREVA Report No. ANP-10294P uses the linear elastic stress analysis method with a plastic zone crack size correction factor in accordance with ASME Code, Section XI, to predict the critical speed for non-ductile fracture of the flywheel. The analysis assumed an axial crack emanating from the bore which has the largest applied force. The analysis used the more conservative fracture toughness of 165 MPa√mm (150 ksi√in) as recommended by SRP Section 5.4.1.1, since the flywheel material will be tested in accordance with ASTM E1820 to have a fracture toughness greater than 165 MPa√mm (150 ksi√in). The analysis calculated that at 2,400 rpm (twice the operating speed), the critical crack size is 58.93 mm (2.32 in.). Assuming an initial crack depth of 6.35 mm (0.25 in.), the analysis calculated that the critical speed for non-ductile failure is 2,693 rpm. The staff considers this acceptable, since it meets the criteria in RG 1.14 that half of the critical speed (1,346 rpm) is greater than the operating speed (1,200 rpm).

The AREVA Report No. ANP-10294P, Revision 1 documents a fatigue crack growth analysis in accordance with Appendix A, Subsection A-4300, of Section XI to the ASME Code. An initial crack length of 6.35 mm (0.25 in.) was assumed, with an assumed duty cycle of 4,000 starts and stops for a 60 year design life. A crack growth of 0.836 mm (0.034 in.) was calculated with a final crack size of 7.21 mm (0.284 in.), which is less than the critical crack size of 58.93 mm (2.32 in.). The staff finds this acceptable, since the final crack size of 7.21 mm (0.284 in.) is less than the critical crack size of 58.93 mm (2.32 in.) as determined by the ASME Code.

The AREVA Report No. ANP-10294P, Revision 1 states the excessive deformation of the flywheel due to overspeed conditions can result in separation of the flywheel from the shaft. Calculations show that during operation, the shrink fit of the flywheel to the shaft is lost. However, the flywheel is attached to the shaft by a large retaining nut and the torque of the flywheel is transmitted by three sets of keys. In addition, a circular collar at the top of the thrust runner is positioned into a ring groove in the bottom of the flywheel, so that when shrink fit is lost, the collar rides along the ring groove, to prevent separation of the flywheel from the shaft. Therefore, since the collar is the limiting part concerning the effects of excessive deformation, the critical speed for excessive deformation of the flywheel is determined by the allowable stress intensity in the collar, caused by contact with the flywheel. These stress intensities were compared to the allowable stress criteria in Appendix F, Subsection F-1331 of Section III of the ASME Code. The critical speed for the collar due to excessive deformation is 3,438 rpm. The staff finds that the critical speed due to excessive deformation meets the criteria of RG 1.14, in that half of the critical speed (1,719 rpm) is greater than the normal speed (1,200 rpm). The staff notes that the AREVA Report No. ANP-10294P, Revision 1 uses material properties of the thrust runner in the excessive deformation analysis. However, the material specification for the thrust runner was not provided in the report or in FSAR Tier 2, Section 5.4.1.6. To ensure that the analysis in AREVA Report No. ANP-10294P, Revision 1, bounds the material that will be used for the thrust runner, the material specification should be included in FSAR Tier 2, Section 5.4.1.6. The staff issued RAI 341, Question 05.04.01.01-3 to address this concern. **RAI 341, Question 05.04.01.01-3 is being tracked as an open item.**

Except for the open item identified above, the staff finds that the reactor coolant pump flywheel analysis was performed and meets the requirement in GDC 4 and the acceptance criteria in RG 1.14 and SRP Section 5.4.1.1 and, therefore is designed, in combination with the inservice inspection discussed below, to provide reasonable assurance that the flywheel failure is sufficiently small.

Overspeed Testing

The design overspeed of the flywheel is determined to be 1,500 rpm based on 125 percent of the normal operating speed of 1,200 rpm. The flywheel will be tested at 1,500 rpm. The staff finds this acceptable, since the flywheel 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. However, there is no ITAAC in FSAR Tier 1, Table 2.2.1-5, Chapter 2, specified for performing this test to ensure that the flywheel assembly can withstand a design overspeed (125 percent of normal operating speed) condition and preclude the generation of missiles, as required by GDC 1 and GDC 4. The staff issued RAI 341, Question 05.04.01.01-4 to address this concern. **RAI 341, Question 05.04.01.01-4 is being tracked as an open item.**

Inservice Inspection

Inservice inspection of the flywheel will be performed every 10 years, coinciding with the ASME Code Section XI inservice inspection schedule. Since the inservice inspection will consist of either an ultrasonic inspection of areas of high stress concentrations or a surface examination by liquid penetrant or magnetic particle examination of all exposed areas, this should detect flaws in order to ensure the basis for safe operation of the reactor coolant pump. The staff finds the inservice inspection of the flywheel is acceptable, since it meets the guidelines of RG 1.14 and SRP Section 5.4.1.1, to ensure that the flywheel integrity is maintained to preclude the generation of missiles, as required by GDC 1 and 4 of 10 CFR Part 50, Appendix A.

5.4.1.5 *Combined License Information Items*

For the RCPs, no COL information items have been identified in FSAR Tier 2, Table 1.8-2. **RAI 341, Question 05.04.01.01-4 has been issued and is being tracked as an open item to assure that the RCPs will be constructed in accordance with the certified design**.

5.4.1.6 *Conclusions*

RCP Flywheel Integrity

Except for open items identified above, and based on the above evaluation, the staff finds 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, except for the Open Item associated with RAI 341, Question 5.4.1.1-3, the flywheel analysis results along with the corresponding inservice inspection, will ensure that flaws do not propagate beyond those predicted for the service life of the flywheel. The staff finds that FSAR Tier 2, Section 5.4.1.6 follows the guidance of RG 1.14, and therefore is acceptable, because it meets the requirements of Section 50.55a, GDC 1 and GDC 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.

5.4.2 Steam Generators (PWR)

5.4.2.1 *Introduction*

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

5.4.2.2 *Summary of Application*

FSAR Tier 1: In FSAR Tier 1, Section 2.2.1, the applicant states that the steam outlet nozzles on the SGs include flow-limiting devices. Detailed FSAR Tier 1 mechanical, electrical, and instrumentation information associated with the steam generator design is specified in Tables 2.2.1-1 through 2.2.1-3.

FSAR Tier 2: The applicant has provided a FSAR Tier 2 description of the U.S. EPR SG in Section 5.4.2; summarized here in part as follows:

The U.S. EPR SGs are vertical shell, U-tube heat exchangers with integral moisture separating equipment. Each is fitted with an axial economizer to provide increased steam pressure by preheating the inlet feedwater by directing it to the cold leg side of the tube bundle. On the primary side, the reactor coolant flows through 5,980 inverted U-tubes, entering and leaving through nozzles located in the hemispherical bottom head of the SG. The tubes are made of thermally-treated Alloy 690. The bottom head is divided into hot and cold channel heads by a primary partition plate extending from the apex of the head to the tubesheet. A manway provides access to each side of the channel head for inservice inspection of the tubes, tube plugging, and maintenance operations. The hot side of the channel head is connected to the reactor vessel outlet, and the cold side of the channel head is connected to the reactor vessel inlet via the reactor coolant pump.

The lower part of the vessel secondary side is formed by a cylindrical shell and a conical shell. It is fitted with eight handholes in the lower part of the cylindrical shell for maintenance operations and removal of accumulated sludge. To support the tubes, there are nine tube support plates (TSPs) spaced over the height of the tube bundle. The TSPs have trifoil broached holes with flat lands to eliminate dryout and to allow free flow of the secondary steam-water mixture. The trifoil geometry of the broached holes of the TSPs limits the formation of aggressive environments which may corrode the TSP or the tubing. Corrosion of a TSP can lead to denting of the adjacent tubing. This geometry also results in a reduced fluid pressure drop across the TSPs, thereby increasing the recirculation ratio and fluid velocities in the tube bundle. A flow distribution baffle increases the cross-flow velocity immediately above the tubesheet to sweep sludge to the center of the tube bundle where the intakes to the blowdown pipes are located. The SGs are designed to promote flow into the central regions of the tube bundle to minimize sludge build up. The upper portion of the secondary side of the SG is accessible through two manways located on either side of the component between the moisture separator and dryer equipment in the steam drum.

Steam is generated on the shell side of the SG, flows upward, and exits through the steam outlet nozzle at the top of the vessel. Feedwater enters the SG at an elevation above the top of the U-tubes. The steam-water mixture from tube bundle rises to the steam drum and continues to the dryer assembly which removes moisture. The dry steam exits from the SG through the steam outlet nozzle which has a steam flow restriction. The RCS operational primary-to-secondary leakage is limited to 567.81 L (150 gal) per day through any one SG.

A SG program is provided to monitor and manage tube degradation and degradation precursors and to provide prompt preventive and corrective actions to maintain the structural and leak-tight integrity of the SG tubes. The major SG program elements are assessment of degradation, inspection requirements for the tubes (including plugging), tube integrity assessment, tube plugging, primary-to-secondary leak monitoring, maintenance of SG secondary side integrity, water chemistry, foreign material exclusion (including loose parts management), contractor oversight, self assessment and reporting.

ITAAC: Item 3.5 in FSAR Tier 1, Table 2.2.1-5 states that an inspection will be performed to verify that the flow area through each flow-limiting device is less than 0.13 m^2 (1.39 ft^2). Additional ITAAC will be performed to verify the detailed FSAR Tier 1 mechanical, electrical, and instrumentation information associated with the SGs.

Technical Specifications: The U.S. EPR Technical Specifications related to the SGs can be found in FSAR Tier 2, Chapter 16, Sections 3.4 and B 3.4. For example, in certain modes, SG operability is verified by ensuring that the secondary side narrow range water level is ≥ 20 percent for required RCS loops. If the SG secondary side narrow range water level is < 20 percent, the tubes may become uncovered and the associated loop may not be capable of providing the heat sink for removal of the decay heat. TS related to SG tube integrity are located in TS 3.4.12 (operational leakage), 3.4.16, B 3.4.12, and B 3.4.16. In addition, TS 5.5.8 requires a SG Program. The provisions specified in TS 5.5.8 for the SG Program are aimed at ensuring that SG tube integrity is maintained. TS 5.5.9 provide controls for monitoring secondary water chemistry to inhibit SG tube degradation and low pressure turbine disc stress corrosion cracking. TS 5.6.7 establish requirements for a SG tube inspection report and its contents.

5.4.2.3 *Regulatory Basis*

The relevant requirements of NRC regulations for this area of review, and the associated acceptance criteria, are given in Sections 5.4.2.1 and Section 5.4.2.2 of NUREG-0800 and are summarized below. Review interfaces with other SRP sections can be found in Section Sections 5.4.2.1 and Section 5.4.2.2 of NUREG-0800.

1. GDC 1 of Appendix A to 10 CFR Part 50 requires, in part, that SSCs important to safety shall be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed. 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 should 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, 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.

4. GDC 15 requires that the reactor coolant system and associated auxiliary control and protection systems should 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 should be designed, fabricated, erected, and tested to the highest quality standards practical.

6. GDC 31 requires, in part, that the RCPB should 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. GDC 32 , "of Appendix A to 10 CFR Part 50 requires, in part, that the designs of all components that are part of the RCPB permit periodic inspection and testing of critical areas and features to assess their structural and leak tight integrity.

8. 10 CFR 50.36, "Technical Specifications," applies to the steam generator program in the TS.

9. 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 ASME Code.

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

11. 10 CFR 50.65, "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," requires that licensees monitor the performance or condition of SSCs against goals to provide reasonable assurance that such SSCs are capable of fulfilling their intended functions.

12. 10 CFR Part 50, Appendix B applies to the steam generator materials. Of particular note is Criterion XIII, which requires, in part, that measures shall be established to control the cleaning of material and equipment in accordance with work and inspection procedures to prevent damage or deterioration. Appendix B to 10 CFR Part 50 also applies to the implementation of the steam generator program. Of particular note are Criteria IX, XI, and XVI. Criterion IX requires, in part, that measures shall be established to ensure that special processes, including nondestructive testing, are controlled and accomplished by qualified personnel using qualified procedures. Criterion XI requires, in part, the establishment of a test program to ensure that all testing required to demonstrate that SSCs will perform satisfactorily in service is identified and performed in accordance with written test procedures that incorporate the requirements and acceptance limits in applicable design documents. Criterion XVI requires, in part, that measures shall be established to ensure the prompt identification and correction of conditions that are adverse to quality.

13. 10 CFR Part 50, Appendix G 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.

Acceptance criteria adequate to meet the above requirements include:

1. RGs 1.31, RG 1.34, RG 1.43, RG 1.50, and RG 1.71 as they relate to the welding of SG components.

2. RG 1.44 as it relates to the RCS water chemistry program.

3. NUREG-1431, as it relates to the secondary side water chemistry program and the steam generator program.

4. RG 1.121, “Bases for Plugging Degraded PWR Steam Generator Tubes,” as it relates to the evaluation of SG tube integrity and determining tube plugging criteria.

5. RG 1.65, as it relates to the selection of suitable steam generator bolting material.

6. RG 1.36 as it relates to the selection and use of thermal insulation.

7. BTP 5-1, “Monitoring of Secondary Side Water Chemistry in PWR Steam Generators,” as it relates to the monitoring of secondary side water chemistry in pressurized water reactor SGs.

5.4.2.4 *Technical Evaluation*

5.4.2.4.1 Steam Generator Design

Detailed information about the SG and the staff’s evaluation and conclusion regarding U.S. EPR’s SG design features and performance requirements are discussed in the following sections of this report:

- 3.9.1, “Special Topics for Mechanical Components”
- 3.9.2, “Dynamic Testing and Analysis of Systems, Components, and Equipment”
- 3.9.3, “ASME Code Class 1, 2, and 3 Components, Component Supports, and Core Support Structures”
- 5.4.2.1, “Steam Generator Materials”
- 5.4.2.2, “Steam Generator Program”
- 5.2.4, “Reactor Coolant Pressure Boundary Inservice Inspection and Testing”
- 15.1.1-15.1.4, “Decrease in Feedwater Temperature, Increase in Feedwater Flow, Increase in Steam Flow, and Inadvertent Opening of a Steam Generator Relief or Safety Valve”
- 15.1.5, “Steam System Piping Failures Inside and Outside of Containment (PWR)”
- 15.2.8, “Feedwater System Pipe Break Inside and Outside Containment (PWR)”

- 15.0.3, “Design Basis Accident Radiological Consequence Analyses For Advanced Light Water Reactors”

5.4.2.4.2 Steam Generator Materials

The staff reviewed FSAR Tier 2, Section 5.4.2.4, “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 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 provisions of the ASME Code and RGs, by specifying design features shown to preserve SG tube integrity, and by specifying water chemistry practices that limit degradation of SG materials.

Selection, Processing, Testing, and Inspection of Materials

The steam generator materials proposed for the U.S. EPR are listed in, Section 5.4.2.4.1, “Selection and Fabrication of Materials,” FSAR Tier 2, Section 5.2.3.1, “Material Specifications,” and FSAR Tier 2, Table 5.2-2. 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 the steam generators is designed and fabricated to comply with ASME Code Class 1 criteria. The secondary side of the steam generator is designed and fabricated to comply with ASME Class 2 criteria. These design criteria are identified in FSAR Tier 2, Table 3.2.2-1.

The applicant proposes to use thermally treated Alloy 690 (Alloy 690 TT, ASME SB-163) for its 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 because Alloy 690 TT tubes were first used in U.S. operating steam generators in 1989 and have thus far resisted degradation by corrosion mechanisms. The tubes have a nominal outside diameter of 1.91 cm (0.75 in.) and a nominal wall thickness of 0.109 cm (0.043 in.), which is typical for Alloy 690 SG tubes at operating plants. The tubes are arranged in a triangular pattern with a tube spacing (pitch) of 2.74 cm (1.08 in.). In a November 14, 2008, response to RAI 63, Question 05.04.02.01-1, the applicant also indicated, that the tubes are thermally treated in accordance with Electric Power Research Institute (EPRI) guidelines for steam generator tubes (TR-016743). This thermal treatment produces the metallurgical structure characteristic of the tubing used in operating U.S. PWRs with Alloy 690 SG tubes.

Alloy 690 TT is also used for the channel head divider plate. Welding consumables matched to Alloy 690 (Alloy 52/152) are used for the cladding on the primary face of the tubesheet and for welds to the Alloy 690 TT components. In a November 14, 2008, response to RAI 63, Question 05.04.02.01-7, the applicant also clarified that the minimum tubesheet thickness of 62.00 cm (24.41 in.) listed in FSAR Tier 2, Table 5.4-2, “Steam Generator Design Data,” corresponds to the unclad tubesheet. The response also stated the maximum unclad tubesheet thickness is 62.41 cm (24.57 in.) and the minimum tubesheet cladding thickness is 0.800 cm (0.315 in.).

Austenitic stainless steels are used for primary inlet and outlet nozzle safe ends and the emergency feedwater nozzle thermal sleeve/safe end. These are the only stainless steel components that are part of the steam generators and associated with the pressure boundary. The staff finds this application of austenitic stainless steels acceptable because it is listed in Section II of the ASME Code and is, thus, permitted by 10 CFR 50.55a. Requirements associated with stainless steel pressure boundary material are discussed further in FSAR Tier 2, Section 5.2 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. The staff finds these bolting materials acceptable because they are listed in Section II of the ASME Code and are, thus, permitted by 10 CFR 50.55a. The staff's full review of threaded fasteners under SRP Section 3.13 is discussed in Section 3.1.3 of this report.

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

Steam Generator Design

The staff reviewed the adequacy of the design and fabrication process proposed for the U.S. EPR 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, respectively, 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 U.S. EPR design includes expansion of the tubes into the tubesheet for the entire tube sheet length, and the tube support plates and flow distribution baffle are designed with broached trifoil holes. Experience with operating steam generators has shown that full-depth hydraulic expansion limits the crevice, 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 410 martensitic stainless steel to limit corrosion and the 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 410 stainless steel does not corrode in the secondary water environment (e.g., NUREG-1841).

In order to reduce the stress formed in small-diameter U-bend sections of tubes, a thermal stress relief heat treatment will be applied to the U-bend sections of tubes with a bend radius of up to 29.21 cm (11.5 in.). This goes beyond the EPRI guidelines for steam generator tubes (TR-016743), which requires stress relief for a bend radius less than 19.05 cm (7.5 in.) for this tubing diameter, as was stated in the November 14, 2008, response to RAI 63, Question 05.04.02.01-2.

The staff reviewed the design of the feedwater inlet ring with respect to integrity and loose parts. Feedwater enters the secondary side of the steam generator above the tubes through a feedwater nozzle with a thermal sleeve connection. The feedwater piping inside the steam generator slopes upward to the feeding to limit the potential for thermal stratification. The material for the main feedwater ring is Type 316L stainless steel. 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.

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, will promote flow and limit crevice areas between tubes and tube supports, use appropriate corrosion-resistance 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 report.

Fabrication and Processing of Ferritic Materials

To comply with GDC 14, GDC 15, and GDC 31, the fracture toughness of the RCPB (Class 1) ferritic materials for the steam generators 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 steam generators are acceptable with respect to fracture toughness if they (1) comply with Appendix G to 10 CFR 10 CFR 50, 10 CFR 50.55a(c), 10 CFR 50.55a(d), and 10 CFR 50.55a(e), and (2) follow the provisions of Appendix G to Section III of the ASME Code. For Class 1 and 2 steam generator components, the 10 CFR 50.55a requires 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 U.S. EPR design complies with these Code requirements, as stated in FSAR Tier 2, Section 5.4.2.4.1 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), 10 CFR 50.55a(d), and 10 CFR 50.55a(e). Ferritic steel RCPB welding must also meet the requirements of paragraph D-1210 of Appendix D to 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 EPR design follows these requirements and RGs, as stated in FSAR Tier 2, Section 5.4.2.4.1 and, therefore, satisfies the requirements related to SG welding. The staff's review of welding of RCPB materials is discussed in more detail in Section 5.3.2 of this report.

Fabrication and Processing of Austenitic Stainless Steel

To comply with GDC 1, GDC 14, GDC 15, GDC 30, and GDC 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 RGs 1.31, RG 1.34, RG, 1.36, RG 1.37, RG 1.44, and RG 1.71. The U.S. EPR steam generator pressure boundary design includes Type F316 austenitic stainless steel forgings (SA-336 or SA-182) for the primary water inlet and outlet nozzle safe ends and the emergency feedwater nozzle thermal sleeve/safe end. The U.S. EPR meets the requirements and the positions in the RGs listed above, as discussed in FSAR Tier 2, Sections 5.4.2.4.1, 5.2.3.3.2, and 5.2.3.4, "Fabrication and Processing of Austenitic Stainless Steels," 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 5.3.2 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. The U.S. EPR bases primary water chemistry control on the EPRI, "PWR Primary Water Chemistry Guidelines," which is acceptable as discussed in Section 9.3.4, "Chemical and Volume Control System (Including Boron Recovery System)," of this report. Secondary water chemistry for the U.S. EPR 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 Nuclear Energy Institute's (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.

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-6932, 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, which 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-6932. In a November 14, 2008, response to RAI 63, Question 05.04.02.01-3, the applicant stated that the carbon and low-alloy steels for the steam generator areas susceptible to flow accelerated corrosion (FAC) will be specified to contain 0.15 percent chromium. Studies such as EPRI TR-1008047 indicate this level of chromium reduces FAC in steam and feedwater systems by more than a factor of ten compared to chromium-free steel.

Onsite cleaning and cleanliness controls for the steam generators are acceptable, because they meet the guidance in RG 1.37, which provides an acceptable way to meet 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.

Provisions for Accessing the Secondary Side of the Steam Generator

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 FSAR related to secondary-side accessibility. The lower internals and blowdown system are designed to facilitate sludge lancing operations and are accessible through eight handholes distributed around the SG shell. In a November 14, 2008, response to RAI 63, Question 05.04.02.01-12, the applicant provided additional details about the location of these openings. The eight openings have an inside diameter of approximately 19.05 cm (7.5 in.) and are located 50.8 cm (20 in.) above the top of the tubesheet. The spacing between the openings varies from about 30 degrees to 70 degrees. The design also includes two inspection openings at approximately 685.8 cm (270 in.) above the tubesheet. These openings have an inside diameter of 5.08 cm (2 in.) and are located on the tube lane axis. Upper internals inside the steam drum are accessible by two manways. The tube bundle antivibration bars are accessible via a hatch through the wrapper roof. The staff finds this level of access acceptable for inspection, cleaning, and foreign object removal, because it closely resembles configurations used successfully in steam generators at operating plants.

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.

5.4.2.4.3 Steam Generator Program

The staff reviewed FSAR Tier 2, Section 5.4.2.5, “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 the SG program complies with the following NRC regulations and requirements: GDC 32 of Appendix A to 10 CFR Part 50; 10 CFR 50.55a(g); 10 CFR 50.36, Appendix B of 10 CFR 50; and 10 CFR 50.65. 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. However, the applicant stated that if the plant technical specifications differ from the requirements in Article IWB-2000 of Section XI of the ASME 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 TSs and Bases sections related to the Steam Generator Program are described in sections 3.4.12, 3.4.16, 5.5.8, 5.5.9, 5.6.7, B 3.4.12, and B 3.4.16 of FSAR Tier 2, Chapter 16.

The applicant proposed a Steam Generator Program that is consistent with the latest revision of the Standard Technical Specifications (STS), (NUREG-1430, NUREG-1431, and NUREG-1432), which is Revision 3.1, dated December 2005. Several RAIs resulted in clarification and wording changes in the FSAR and TS for consistency with the STS. The staff also identified an instance in which a TS requirement must differ from the STS. This involves STS 5.5.9.d.1, which 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 a November 14, 2008, response to RAI 63, Question 05.04.02.02-11, the applicant stated it would take an exception to the STS and modify the corresponding U.S. EPR TS 5.5.8.d.1 to require 100 percent inspection "during the first refueling outage (RFO) and following SG replacement." The staff has not accepted this wording. The staff issued follow-up RAI 364, Question 05.04.02.02-16 to revise the TS to clarify the requirement to inspect 100 percent of the tubes in newly installed original and replacement steam generators during the first refueling outage following installation. **RAI 364, Question 05.04.02.02-16 is being tracked as an open item.**

In a November 14, 2008, response to RAI 63, Question 05.04.02.02-4, the applicant clarified that the preservice inspection of the tubes will be conducted by examining the full length of each tube after fabrication and prior to placing the SGs in service. In the response, the applicant stated it would modify FSAR Tier 2, Section 5.4.2.5.2.2 to state that the PSI is conducted after field hydrostatic testing of the reactor coolant system and before placing the SGs in service. The staff finds the response and FSAR modification acceptable, because PSI will be performed after SG fabrication. This supports the objective of discriminating between service-related degradation and manufacturing imperfections and is consistent with industry practice as expressed in the EPRI Steam Generator Examination Guidelines. In RAI 63, Question 05.04.02.02-5, the staff requested that the applicant remove the reference to eddy current bobbin probes in FSAR Tier 2, Section 5.4.2.5.2.2 (Tube Inspection) to allow for the possibility that other inspection methods may be used in the PSI. In the same RAI, the staff requested that the applicant define the "abnormal conditions" referred to in the FSAR section on preservice inspection. In a November 14, 2008, response to RAI 63, Question 05.04.02.02-5, the applicant stated that reference to eddy current probes would be removed from the FSAR, and that the FSAR would be modified to clarify that abnormal conditions (e.g., manufacturing burnish marks, dings, etc.) detected with bobbin probes in the PSI would be examined by other methods. This is acceptable, because it is consistent with industry guidelines and practice at operating plants. The staff confirmed that Revision 1 of the FSAR, dated May 29, 2009, contains the changes committed to in the RAI response. Accordingly, the staff finds that the applicant has adequately addressed this issue and, therefore, the staff considers RAI 63, Question 05.04.02.02-5 resolved.

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 U.S. EPR, 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 FSAR.

As discussed in Section 5.4.2.1 above, the U.S. EPR steam generators are designed to be accessible for inspection. All of the 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 U.S. EPR steam generators as it relates to providing access to allow inservice inspection to be acceptable.

The staff concludes that the SG Program is acceptable and meets the requirements of GDC 32, 10 CFR 50.55a, 10 CFR 50.36, Appendix B to 10 CFR 50, and 10 CFR 50.65. This conclusion is based on the design of the steam generators being accessible for inspection and cleaning, and the implementation of a steam generator program to maintain the structural and leakage integrity of the steam generator tubes.

5.4.2.5 *Combined License Information Items*

Table 5.4.2-1 provides a list of steam generator related COL information item numbers and descriptions from FSAR Tier 2, Table 1.8-2:

Table 5.4.2-1 U.S. EPR Combined License Information Items

Item No.	Description	FSAR Tier 2 Section
5.4-1	A COL applicant that references the U.S. EPR design certification will identify the edition and addenda of ASME Section XI applicable to the site-specific Steam Generator inspection program.	5.4.2.5.2.2

The staff finds the above listing to be complete. Also, the list adequately describes actions necessary for the COL applicant. No additional combined license information items need to be included in FSAR Tier 2, Table 1.8-2 for steam generator considerations.

5.4.2.6 *Conclusions*

The staff concludes that the U.S. EPR steam generator materials satisfy the acceptance criteria for materials selection, design, fabrication, compatibility with the service environments, and secondary-side accessibility. Therefore, the staff finds these materials are acceptable and meet the requirements of GDC 1, GDC 4, GDC 14, GDC 15, GDC 30, and GDC 31, as well as the requirements of 10 CFR Part 50, Appendices B and G.

Except for the open item identified above, the staff concludes that the U.S. EPR Steam Generator Program satisfies the acceptance criteria for accessibility for periodic inspection and testing of critical areas for structural and leakage integrity, and that the proposed technical specifications are consistent with the standard technical specifications for domestic PWRs.

Until the open item is resolved, the staff cannot find that the SG Program is acceptable and meets the requirements of GDC 32, 10 CFR 50.55a.10 CFR 50.36, and 10 CFR 50.65.

5.4.3 Reactor Coolant Piping

The RCS piping includes the main coolant lines for the four loops; the post-accident high point vent line; and the pressurizer surge, spray, and relief lines. Portions of other systems such as the safety-injection system piping (refer to FSAR Tier 2, Section 6.3), extra borating system piping (refer to FSAR Tier 2, Section 6.8), and chemical and volume control system piping (refer to FSAR Teir2, Section 9.3.4) constitute a part of the reactor coolant pressure boundary, but are not part of the RCS piping.

Detailed information about the reactor coolant piping and the staff's evaluation and conclusion regarding U.S. EPR's reactor coolant piping design features and performance requirements are discussed in the following sections of this report:

- 3.12, "ASME Code Class 1, 2, and 3 Piping Systems, Piping Components and Their Associated Supports"
- 5.2.3, "Reactor Coolant Pressure Boundary Materials"
- 5.2.4, "Reactor Coolant Pressure Boundary Inservice Inspection and Testing"
- 5.2.5, "Reactor Coolant Pressure Boundary Leakage Detection"
- 5.4.7, "Residual Heat Removal System"
- 6.3, "Emergency Core Cooling System"
- 3.9.1, "Special Topics for Mechanical Components"
- 3.9.2, "Dynamic Testing and Analysis of Systems, Components, and Equipment"
- 3.9.3, "ASME Code Class 1, 2, and 3 Components, Component Supports, and Core Support Structures"
- 3.9.6, "Functional Design, Qualification, and Inservice Testing Programs for Pumps, Valves, and Dynamic Restraints"
- 3.10, "Seismic and Dynamic Qualification of Mechanical and Electrical Equipment"
- 10.3.6, "Steam and Feedwater System Materials"
- 6.1.1, "Engineered Safety-Features Materials"
- 6.6, "Inservice Inspection of Class 2 and 3 Components"

5.4.4 Main Steamline Flow Restrictors (Not Used in U.S. EPR Design)

5.4.5 Main Steamline Isolation System (Not Used in U.S. EPR Design)

5.4.6 Reactor Core Isolation Cooling System (Not Used in U.S. EPR Design)

5.4.7 Residual Heat Removal System

5.4.7.1 *Introduction*

The residual heat removal system (RHRS) is designed to remove heat from the reactor system during normal shutdown and accident conditions. The RHRS shares equipment and piping with the safety-injection system.

This section of the FSAR describes the design basis, system operation, and testing and instrumentation requirements of the RHRS. FSAR Tier 2, Section 6.3, "Emergency Core Cooling System," provides additional description of the RHRS equipment.

5.4.7.2 *Summary of Application*

FSAR Tier 1: The FSAR Tier 1 information associated with this section is found in FSAR Tier 1, Section 2.2.3, "Safety Injection System and Residual Heat Removal System."

FSAR Tier 2: The applicant has provided a description of the U.S. EPR residual heat removal system in FSAR Tier 2, Section 5.4.7 summarized here in part, as follows:

The RHRS is comprised of four physically separate and independently powered trains of safety-related equipment. Each train utilizes a low head safety-injection pump to draw RCS water from a reactor coolant system hot leg. The RCS water passes through the tube side of a LHSI heat exchanger where it is cooled by component cooling water system (CCWS) water and returned to the RCS cold leg.

The RHRS is designed to cool the RCS following initial cooldown to approximately 121 °C (250 °F) utilizing the steam generators. The application includes RHRS performance evaluations for conditions of all RHRS components operable and assuming the most limiting single failure with either offsite power or onsite power unavailable. With all components operable, the total time to cool the RCS down to 55 °C (131 °F) following a reactor trip is calculated by the applicant to be approximately 15 hours, which is well within the design requirement of 40 hours with all components available.

The RHRS is reported to be designed, and its components fabricated, to the quality standards applicable to the safety-related functions to be performed by the system. The system is designed to remain functional following a safe-shutdown earthquake.

The controls required to operate the RHRS are provided in the main control room, with control room indications for parameters such as system flow, pressure, and temperature. Controls and displays are also available in the remote shutdown station.

ITAAC: The ITAAC associated with FSAR Tier 2, Section 5.4.7 are given in Table 2.2.3-3, "SIS/RHRS ITAAC," of FSAR Tier 1, FSAR Tier 2, Section 2.2.3.

Technical Specifications: The Technical Specifications associated with FSAR Tier 2, Section 5.4.7 are given in FSAR Tier 2 Chapter 16, Sections 3.5.2, 3.9.4, and 3.9.5.

Initial Plant Testing: FSAR Tier 2, Chapter 14, "Initial Test Program and ITAAC," identifies the RHRS tests to be performed as part of the initial plant testing program including: Test No. 016, "Residual Heat Removal" Test No. 017, "Mid-Loop Operations Verification" and Test No. 161, "Hot Functional Sequencing Document." Also, system accessibility will be provided for periodic testing and inservice inspection of the RHRS.

5.4.7.3 *Regulatory Basis*

The relevant requirements of NRC regulations for this area of review, and the associated acceptance criteria, are given in Section 5.4.7 of NUREG-0800 and are summarized below. Review interfaces with other SRP sections also 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 RHR system.

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

3. GDC 5, "Sharing of Structures, Systems, and Components," as it relates to the 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 including necessary instrumentation and controls for the RHR system in the control room and at a location outside the control room.

5. GDC 34, "Residual Heat Removal," as it relates to requirements for an RHR system.

6. 10 CFR 50.34(f)(2)(xxvi), as it relates to the provisions for a leakage detection and control program 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.

Acceptance criteria adequate to meet the above requirements include:

1. The requirements of GDC 2 can be satisfied through conformance with RG 1.29.

2. One way to meet the requirements of GDC 34 is to follow the applicable portions of the system or systems must satisfy the functional, isolation, pressure relief, pump protection, and test-related criteria as well as the control room-related criteria specified in BTP 5-4.

3. 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.

4. Interfaces between the RHR system and 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.

5. When the RHR system is used to control or mitigate the consequences of an accident, it must meet the design requirements of an engineered safety-feature system. One way to accomplish this is by meeting the guidelines of RG 1.82, "Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident."

6. RG 1.68, "Initial Test Programs for Water Cooled Nuclear Power Plants" as it relates to the RHR system.

5.4.7.4 *Technical Evaluation*

The staff has reviewed the FSAR Tier 2, Section 5.4.7, "Residual Heat Removal System," including applicable Technical Specifications, and related ITAAC and initial test requirements to determine the acceptability of the design. Portions of FSAR Tier 2, Section 6.3 were also reviewed by the staff, as necessary, for the evaluation of RHRS. The staff's evaluation of the FSAR information was performed against the requirements of 10 CFR Part 50, GDC 2, GDC 4, GDC 5, GDC 19, and GDC 34, and 10 CFR 50.34(f)(2)(xxvi). The acceptance criteria of BTP 5-4, "Design Requirements of the Residual Heat Removal System"; NRC GL 88-17, "Loss of Decay Heat Removal"; and RG 1.82 were also addressed in the staff's review.

There are no COL information items identified in FSAR Tier 2, Table 1.8-2 as applicable to the RHRS, and none were identified as a result of the staff's review.

FSAR Tier 1, Section 2.2.3 identified above in Section 5.4.7.2 was also reviewed by the staff. The information contained in FSAR Tier 1, Section 2.2.3 is consistent with the design descriptions provided in FSAR Tier 2, Section 5.4.7. The FSAR Tier 1 section identifies 28 ITAAC items applicable to the RHRS, including verifications of as-built configuration, inspections of ASME code test reports, verifications of equipment operability, verifications of parameter displays in the main control room and the remote shutdown station, and component functional performance checks. The staff has reviewed the ITAAC and found them to be consistent with the certification description provided in FSAR Tier 1, Section 2.2.3.

The TS identified in FSAR Tier 1, Section 5.4.7.2 were reviewed by the staff and found to be consistent with the descriptions and requirements provided in FSAR Tier 2, Section 5.4.7. In addition, TS 3.9.4, "Refueling Operations – RHR Loops-High Water Level," and TS 3.9.5, "Refueling Operations – RHR Loops-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."

Based on the discussion above, the descriptions provided in FSAR Tier 1 pertaining to RHR System and the associated ITAAC are adequate.

Since the U.S. EPR is a single-unit plant design, GDC 5 is not applicable.

The acceptance criteria used in this evaluation are contained in SRP Section 5.4.7, "Residual Heat Removal System," as summarized above in Section 5.4.7.3 of this report. Completeness of the content of the FSAR Tier 2 section was reviewed against SRP Section 5.4.7 and the applicable portions of Regulatory Guide 1.206.

SRP Section 5.4.7 identifies a number of review interfaces with other SRP sections which are accomplished as part of the review of those other SRP sections.

5.4.7.4.1 Design Bases

FSAR Tier 2, Section 5.4.7.1, describes the bases for the RHRS, including the applicability of 10 CFR Part 50, GDC 2, GDC 4, GDC 5, GDC 19, and GDC 34, and 10 CFR 50.34(f)(2)(xxvi). The GDC identified as applicable by the applicant are consistent with the requirements listed in SRP Section 5.4.7.

The RHRS is designed to cool the RCS down using only safety-related equipment, after heat removal by the steam generators has been completed, and to maintain RCS temperature within specified limits during refueling and maintenance activities, including mid-loop operation, such that specified acceptable fuel design limits (SAFDLs) and the design conditions of the RCS pressure boundary are not exceeded. In addition, the U.S. EPR RHRS is designed to include sufficient redundancy in components and features, and suitable interconnections, leak detection, and isolation capabilities such that system operation is assured with either only onsite power or only off-site power available, assuming a single failure. This design basis meets the requirements of GDC 34.

The U.S. EPR RHRS is designed with a full capacity to reduce the RCS temperature from a power operating temperature of approximately 302 °C (575 °F) to approximately 55 °C (131 °F) in 40 hours.

GDC 2 requires that structures, systems, and components important to safety be designed to withstand the effects of natural phenomena, such as earthquakes, without loss of their safety function. In accordance with the guidelines of RG 1.29, and as described in FSAR Tier 2, Section 6.3, the safety-related RHRS is designated Seismic Category I and is required to withstand the effects of a safe-shutdown earthquake and remain functional. Accordingly, the U.S. EPR RHRS is designed to remain functional following a safe-shutdown earthquake, thus meeting the requirements of GDC 2. The seismic load on RHR component is evaluated in Section 3.7.

In addition, the U.S. EPR RHRS is designed to detect and control leakage outside of the containment structure following an accident. The FSAR also states that plant programs and procedures will detect, monitor, and control RHRS leakage. FSAR Tier 2, Section 5.2.5.3.1, “Safety Injection System / Residual Heat Removal System,” describes the RHRS leakage detection instrumentation, and FSAR Tier 2, Section 14.2.12.14.11, “Leak Detection Systems (Test No. 189) describes the initial testing of the leakage detection system. The design basis meets the design requirements of 10 CFR 50.34(f)(xxvi). The requirements for a leakage control program, including schedule for re-testing, are addressed in FSAR Tier 2, Chapter 16, TS 5.5.2, “Programs and Manuals – Primary Coolant Sources Outside Containment.”

FSAR Tier 2, Section 5.4.7.1, “Design Basis,” states that the U.S. EPR RHRS may be operated under all operating conditions from the main control room, with similar operating controls in the remote shutdown station. FSAR Tier 1, Section 2.2.3 identifies the specific controls available in both the main control room and the remote shutdown station, and the associated ITAAC 4.1 and ITAAC 4.2 require inspection and testing of the required controls. This design basis meets the requirements of GDC 19.

GDC 4 requires that structures, systems, and components important to safety be appropriately protected against dynamic effects associated with system flow instabilities and loads. FSAR Tier 2, Section 5.4.7.1, states that the RHRS is designed to be self-venting to ensure that piping system voids will not occur and that the system remains water solid when connected to the RCS such that the LHSI pumps are protected from cavitation, and piping is protected from water hammer. The staff requested that the applicant provide proof that system voids will not occur as part of its review. This issue is addressed in Section 6.3 of this report (RAI 32, Question 06.03-3). Since the RHRS pumps are also the LHSI pumps, the requested information also applies to the LHSI pumps.

Except for the issues addressed in Section 6.3 of this report, the design bases for the RHRS described by the applicant in FSAR Tier 2, Section 5.4.7 meet GDC 4 and 10 CFR 50.34(f)(2)(xxvi) requirements as defined in SRP Section 5.4.7.

5.4.7.4.2 System Description

FSAR Tier 2, Section 5.4.7.2 , “System Description,” describes the RHRS as four physically separate and independently powered RHRS trains, each comprised of an LHSI pump, LHSI heat exchanger, a heat exchanger bypass line, a heat exchanger discharge line with a temperature control valve, and an RCS cold leg return line. The staff has reviewed the FSAR Tier 2, Section 6.3 piping and instrument diagrams referred to in FSAR Tier 2, Section 5.4.7, and the functional arrangement diagrams contained in FSAR Tier 1, Section 2.2.3, and has found the diagrams consistent with the system description provided in FSAR Tier 2, Section 5.4.7.2. ITAAC 2.1 in FSAR Tier 1, Table 2.2.3-3, requires that the as-built system conform to the functional arrangement diagrams contained in FSAR Tier 1, Section 2.2.3. The physical locations and separation of the RHRS equipment is stipulated in FSAR Tier 1, Section 2.2.3, Table 2.2.3-1, “SIS/RHRS Equipment Mechanical Design,” and will be verified per ITAAC 2.2 and ITAAC 2.3 in FSAR Tier 1, Table 2.2.3-3. Each of the four safety-related RHRS trains are located in separate Seismic Category I Safeguard Building structures. In addition, the safety-related Class 1E divisional power sources for each component are listed in FSAR Tier 1, Table 2.2.3-1 and will be verified per ITAAC 5.1 in FSAR Tier 1, Table 2.2.3-3.

With regard to GDC 34, the staff has reviewed the U.S. EPR RHRS design against the staff positions provided in BTP 5-4. BTP 5-4 includes guidance on: RHRS function; system isolation; pressure relief; pump protection; and testing. The functional guidance in BTP 5-4 are met, as described in the design basis description above.

The system isolation positions of BTP 5-4 call for at least two motor-operated valves in series on the RHRS suction line with interlocks to prevent valve opening when RCS pressure is above the RHRS design pressure. According to FSAR Tier 2, Section 5.4.7.2, a pair of in-series manual isolation valves is installed in the RHRS suction line. The staff has confirmed existence of two motor-operated isolation valves based on review of the RHRS piping diagram, Figure 6.3-2, “Safety Injection / Residual Heat Removal System Train (Typical).” The two RHRS suction line isolation valves are interlocked to prevent opening when RCS pressure is above approximately 3.20 MPa (464 psia) and RCS temperature is above approximately 180 °C (356 °F), which is described further in FSAR Tier 2, Section 7.2.1.3.9, “P14 Permissive,” as the P14 Permissive. The P14 Permissive utilizes 3-out-of-4 logic based on independent and diverse divisional pressure and temperature inputs, as described in FSAR Tier 2, Figure 7.2-33, “P14 Permissive Logic,” and FSAR Tier 2, Section 7.6.1.2.1, “RHR Suction Valve Interlocks,” thus meeting the interlock guidance in BTP 5-4.

The applicant stated that the automatic isolation of RHRS is not provided for in the design; however, spring-loaded safety-relief valves provided on the RHRS suction line protect the RHRS from over-pressurization when connected to the RCS. The setpoints and flow capacity of the safety-relief valves are designed to limit the RHRS to 110 percent of its design pressure, based on a spurious startup of a medium head safety-injection pump with the large miniflow line isolated, as described in FSAR Tier 2, Section 6.3.2.2, “Equipment and Component Descriptions.” Discharge from the safety-relief valves is piped to the in-containment refueling water storage tank. On the RHRS discharge line, two in-series check valves provide isolation of RHRS from the RCS.

Protection of the LHSI pumps against cavitation damage is provided by an automatic stop upon detection of low loop water level or low Δp_{sat} (difference between RCS hot leg temperature and hot leg saturation temperature). An evaluation of LHSI net positive suction head (NPSH) during DBAs is provided in FSAR Tier 2, Section 6.3.3, “Performance Evaluation.” The evaluation includes consideration of IRWST water temperature, suction sump screen blockage, and uncertainty in hydraulic resistances, and concludes there is sufficient NPSH during DBAs. The staff has not completed review of the applicant’s response to a request for additional information issued under Section 6.3 of this report (RAI 212, Question 06.03-6) regarding this issue. This issue will be addressed in Section 6.3 of this report.

BTP 5-4 also provides guidance on RHRS testing. The design and installation guidance on the RHRS provide for accessibility for periodic testing and ISI of the RHRS components. FSAR Tier 1, Section 2.2.3 ITAAC 4.4 requires testing of the interlock circuits, and ITAAC 7.7 and ITAAC 3.2 require testing of valve functionality, as called for in BTP 5-4. The periodic testing of the in-series RHRS discharge line check valves for leak tightness as called for in BTP 5-4 is provided for via test connections shown in FSAR Tier 2, Figure 6.3-2 and as described in FSAR Tier 2, Section 3.9.6.3.3, “Inservice Testing Program for Check Valves.”

Additional details on RHRS periodic testing and inservice inspection are provided and evaluated in FSAR Tier 2, Sections 3.9.6, 5.2.4, 6.3.2.7, 6.3.4, and 6.6.

In RAI 129, Question 05.04.07-5a, the staff requested that the applicant describe how the overpressure protection guidance in BTP 5-4 are fulfilled. In a December 11, 2008, response, the applicant stated that the minimum ultimate pipe rupture strength of SIS/RHRS was analyzed for piping outside of the containment. The analysis determined that the minimum ultimate pipe rupture strength was 17.7 MPa (2,568 psig), which exceeds the RCS operating pressure of 15.5 MPa (2,250 psig). The applicant also explained that the RHRS safety-relief valve is designed to provide overpressure protection for the limiting event of a spurious actuation heat exchanger of the MHSI pump with its miniflow line closed. Fluid discharged from the RHRS safety-relief valve is directed to the IRWST, thus preventing flooding of safety-related equipment, avoiding reduction in the safety-injection capability, and preventing discharge outside of the containment as per BTP 5-4. The staff finds the applicant’s response acceptable because it conforms to the guidance in BTP 5-4.

RAI 129, Question 05.04.07-5b also requested that the applicant describe the testing that will be performed to confirm adequate mixing of borated water that may be added before or during cooldown under natural circulation conditions. In a December 11, 2008, response, the applicant stated that adequate mixing will be assured by sampling the reactor coolant either via the sampling lines attached to the RCS crossover legs in loops 1 and 3, via the sampling line located in the CVCS letdown line, or via the sampling line located downstream of each LHSI.

The applicant's response is acceptable to the staff because the sampling of boron concentration at two locations will enable the confirmation of proper boron mixing.

A mini-flow and test line equipped with flow measurement is connected between the RHRS return injection line upstream of the outboard isolation valve to the RHRS hot leg line. FSAR Tier 1, Section 2.2.3, ITAAC 7.8 requires testing of the mini-flow and test line. NRC Bulletin 88-04, "Potential Safety-Related Pump Loss," and NRC GL89-04, "Guidance on Developing Acceptable Inservice Testing Programs," identify concerns involving the design of mini-flow test lines, particularly with regard to the potential for pump damage while running pumps in mini-flow configuration. In RAI 129, Question 05.04.07-6 the staff requested that the applicant do the following: (1) Confirm the miniflow test line drawing in the FSAR; (2) confirm adequate sizing of the line and the presence of a flow measurement device; (3) describe the design function of the flow restrictor; and (4) address the single failure which might occur during RHRS/SIS pump testing. In a December 11, 2008, response, the applicant described the exact location of the LHSI miniflow line in FSAR Tier 2, Figure 6.3-2 (Sheet 2 of 2) and stated that the line is sized such that a sufficient minimum flow (more that 15 percent of the estimated maximum run-out of the LHSI pump at shutoff head conditions) will be recirculated to the IRWST when the LHSI pump is taking suction from the IRWST. This prevents the pump from dead-heading in the event that the downstream pressure is higher than the maximum LHSI injection pressure. The flow restrictor installed along the LHSI miniflow line is designed to maintain minimum flow to the IRWST for LHSI pump protection during both testing and safety injection, while permitting sufficient flow to the RCS when the system is in the safety-injection mode. Failure (spurious closing) of the LHSI heat exchanger main control valve is considered the worst single failure. In this case, one LHSI pump would be in a dead-head no flow condition. However, as noted in FSAR Tier 2, Table 6.3-7, "Safety Injection System Failure Modes and Effects Analysis," failure of one control valve would cause unavailability of only one RHRS train, leaving three other trains to provide residual heat removal. The staff finds the applicant's response to RAI 129, Question 05.04.07-6 acceptable.

The application specifically addresses the design of the U.S. EPR RHRS relative to mid-loop operations. Mid-loop operating conditions exist whenever the RCS water level is below the top of the flow area of the hot legs at the junction of the reactor vessel. During mid-loop operations, the likelihood of losing RHRS cooling, due to drop in reactor vessel water level, is increased.

GL 88-17, "Loss of Decay Heat Removal," and GL 87-12, "Loss of Residual Heat Removal While the Reactor Coolant System is Partially Filled," address loss of decay heat removal during mid-loop operation, and include the following RHRS design-related recommendations:

- the availability of continuous core exit coolant temperature indications in the main control room
- the availability of continuous RCS water level indication in the main control room
- the capability to continuously monitor RHRS performance in the main control room
- an alternate means of adding inventory sufficient to keep the core covered, in addition to the RHRS pumps
- adequate equipment for personnel communications that involve mid-loop operation activities

With regard to the above recommendations pertaining to mid-loop operation, the application states that sensors provide temperature measurement of each hot leg during mid-loop operation and that the reactor vessel water level is continually monitored during an outage utilizing a level sensor. FSAR Tier 2, Section 7.7.2.2.3, “RCS Loop Level Control,” describes the loop level control function during mid-loop operation, stating that RCS loop level is continuously monitored. In RAI 129, Question 05.04.07-7, the staff requested that the applicant provide additional information regarding the design features pertinent to mid-loop operation: (1) confirmation of the availability of redundant hot leg temperature indications in the main control room; (2) additional description of the instrumentation and procedures used to ensure reliable level indication in the RCS hot legs upstream of the RHRS line; and (3) a description of the communications available between control room personnel and field personnel during mid-loop operations.

In a December 11, 2008, response to RAI 129, Question 05.04.07-7a, the applicant stated that redundant hot leg temperature indications are available to the operators in the main control room via the process information and control system (PICS) and the safety information and control system (SICS). The hot leg WR temperature measurement is located between the RPV outlet and the RHRS suction nozzle. The instrument well, located at 135 or 225 degrees (with zero degrees being at the top of the pipe), permits the instrument to be fully submerged during mid-loop operation. The hot leg WR temperature measurement is used to determine the margin to saturation in the hot leg so that the RHRS/LHSI pump can be tripped in order to avoid cavitation. With regard to RAI 129, Question 05.04.07-7b, the applicant referenced its response to RAI 26, Question 19-184. That response states that each of the four loops of the RCS contains a level measurement sensor. These sensors are located in the hot leg between the SG and the RHRS suction nozzle. These sensors are permanently installed with metal tubing to avoid the use of tygon tubing, which is susceptible to collapse during vacuum sweeping and to improper routing that can trap air causing an erroneous level indication. Since Loop 3 contains the surge line, an additional level sensor is placed between the bottom of the hot leg and the top of the pressurizer. This provides WR level indication so that the RCS level is known during the process of draining to mid-loop. RCS loop level indication is available to the operator through PICS and SICS. When RCS loop level control is in the manual control mode, alarms are generated when the loop level deviates from the desired range. When RCS loop level control is in the automatic control mode, alarms are generated when the loop level limitation functions are actuated. The RCS loop level measurement is used to trip the RHRS/LHSI pump in case of low loop level, in order to prevent air ingestion.

In a December 11, 2008, response to RAI 129, Question 05.04.07-7c, the applicant noted that the plant’s communications systems are described in FSAR Tier 2, Section 9.5.2.

Based on the above reviews, the applicant’s response to RAI 129, Question 05.04.07-7 is acceptable.

The capability to continuously monitor RHRS performance in the main control room is described in FSAR Tier 2, Section 5.4.7.5, “Instrumentation Requirements,” which states that main control room displays provide information on RHRS functions such as flow, pressure, and temperature.

FSAR Tier 2, Section 5.4.7.1 describes the availability of the MHSI pumps during mid-loop operation to ensure availability of the RHRS function. FSAR Tier 2, Section 6.3.3 states that the MHSI pumps are maintained in standby for RCS makeup during RHRS operation. Additionally, RCS level is controlled by the chemical and volume control system low pressure reducing valve to ensure sufficient RCS inventory for the LHSI pumps during mid-loop RHRS operation, as described and evaluated further in FSAR Tier 2, Section 9.3.4.2.2, “Component Description.”

On the basis of the above evaluation, the staff finds that there is reasonable assurance that design of the U.S. EPR RHRS adequately addresses mid-loop operation.

FSAR Tier 2, Section 5.4.7.2.2, “Design Features Addressing Intersystem LOCA,” also describes the design features that address intersystem LOCA. An intersystem LOCA is a class of loss-of-coolant accidents in which a break occurs in a system that is connected to the RCS, hypothetically in the case of an over-pressurization of the RHRS, resulting in the discharge of coolant outside the containment building and unavailability for long-term recirculation cooling. The applicant states that portions of the RHRS that interface with the RCS are designed, manufactured, installed, and inspected in accordance with ASME B&PV Code, Section III, Class 2 Components, which are verified per the applicable ITAAC listed in FSAR Tier 1, Section 2.2.3. Further, the RHRS from the RCS to the second isolation valve are designed to RCS pressure, with the remaining portions designed for an ultimate rupture strength exceeding that of full RCS operating pressure. Interlocks preventing inadvertent opening of the RHRS isolation valves and installed pressure safety-relief valves provide protection against RHRS over-pressurization, as evaluated above.

On the basis of the above evaluation, the staff finds that the design of the U.S. EPR RHRS adequately addresses intersystem LOCA.

The LHSI and MHSI pumps provide a safety-injection function during a design-basis LOCA, taking suction from the IRWST. The IRWST provides a reliable source of water for long-term cooling, in conformance with RG1.82, as described and evaluated in FSAR Tier 2, Section 6.3.

5.4.7.4.3 Performance Evaluation

The application includes an evaluation of the performance of the RHRS assuming that all components are operable, plus assuming the worst single failure with either offsite or onsite power available. The single failure, the complete loss of an RHRS train, is in addition to one RHRS train being inoperable due to maintenance activities.

Initial RCS cooldown following reactor trip is accomplished with all four reactor coolant pumps running and utilizing the steam generators to remove decay heat. Two trains of RHRS are placed in service once the RCS pressure and temperature decrease below approximately 2.69 MPa (390 psia) and 121 °C (250 °F), respectively. The maximum allowed cooldown rate of 50 °C/hr (90 °F/hr) is assumed in the analysis. The applicant’s calculations show a total time to cool down to 55 °C (131 °F) for refueling operations to be about 15 hrs after reactor trip, or well within the 40 hours design basis given above in FSAR Tier 2, Section 5.4.7.3. The applicant’s calculated cooldown time from 121 °C (250 °F) to 55 °C (131 °F) is 7.7 hours.

With two RHRS trains and only offsite power available, the cooldown from 121 °C (250 °F) to 93 °C (200 °F) is stated to be approximately 10 hours. With only onsite power available, the RCPs immediately coast down, and main feedwater flow is lost. Cooldown to the RHRS conditions is achieved through natural circulation of the RCS using main steam relief and the emergency feedwater system. Various cooldown scenarios are presented by the applicant, showing the fastest cooldown from the highest RHRS starting temperature of 180 °C (356 °F) to cold shutdown of 93 °C (200 °F) to be about 3 hours.

RAI 129, Question 05.04.07-8 noted an error in the FSAR description of the time required for two RHR trains to cool the plant from approximately 121 °C (250 °F) to 93 °C (200 °F). The applicant proposed a revision to FSAR Page 5.4-33 to correct the error. The staff has confirmed that Revision 1 of FSAR, dated May 29, 2009, includes the correction as committed in the RAI response. Accordingly, the staff finds that the applicant has adequately addressed this issue and, therefore, the staff considers RAI 129, Question 05.04.07-8 resolved.

The applicant's analyses sufficiently demonstrate the design bases of the RHRS are met. In addition, based on the staff's review and evaluation of FSAR Tier 2, Section 10.4.9, "Emergency Feedwater System," the staff finds that the emergency feedwater system meets the guidance of BTP 5-4 for having sufficient inventory available to cool down the RCS to the RHRS cut-in point, assuming either onsite or offsite power is available with an assumed single failure.

5.4.7.4.4 Inspection and Testing Requirements

Initial plant testing described in FSAR Tier 2, Chapter 14 includes three tests, as described and evaluated below.

- Test No. 016, "Residual Heat Removal," provides testing to demonstrate functionality of the RHRS prior to fuel load, including pump flow verifications, verification of protective devices, indications, alarms, and interlocks, valve operability, heat exchanger performance, and electrical power supply capabilities.

- Test No. 017, "Mid-Loop Operations Verification," includes testing of RCS mid-loop level indicators and alarms and verification of pump performance at mid-loop level conditions.

- Test No. 161, "Hot Functional Sequencing Document," to demonstrate that RHRS can be used to achieve cold shutdown conditions within the cooldown rates permitted by Technical Specifications.

The staff has evaluated the above-listed initial tests against the RHRS design described in FSAR Tier 2, Section 5.4.7, and RG 1.68, "Initial Test Programs for Water Cooled Nuclear Power Plants." The staff finds that there is reasonable assurance that the initial testing will sufficiently demonstrate that the RHRS will meet its design performance requirements as described in FSAR Tier 2, Section 5.4.7, "Instrumentation Requirements." These tests are discussed further in Sections 14.2 and 14.3 of this report.

As described above in Section 5.4.7.4.2, displays in the main control room provide the information required for operation of the RHRS, including system flows, pressure, and temperature, as are equipment parameters such as high motor winding temperature, high bearing oil temperature, and high motor air temperature. Operation of the RHRS may be performed from either the main control room or the remote shutdown station. FSAR Tier 1, Section 2.2.3 identifies the controls and associated displays available in both the main control room and remote shutdown station, and the corresponding ITAAC 4.1 and 4.2 provide verification of those controls and displays.

There are no automatic operations of the RHRS, other than the equipment protection interlocks described above and other parameter range controls such as the temperature-dependent heat exchanger bypass flow control. The RHRS function must be started and stopped by manual action of the operator, and automatic isolation of the RHRS is not provided, except as a protective action in instances such as a low loop level or low Δp_{sat} as described above in

Section 5.4.7.4.2. The staff finds the RHRS instrumentation requirements described by the applicant acceptable because important RHRS equipment parameters, such as system flows, pressure and temperature, are available in the main control room and remote shutdown stations.

5.4.7.5 *Combined License Information Items*

For the RHRS, no COL information items have been identified in FSAR Tier 2, Table 1.8-2. The staff finds this acceptable because the proposed ITAAC and initial plant test program assure that the RHRS will be constructed in accordance with the certified design.

5.4.7.6 *Conclusions*

The U.S. EPR RHRS was reviewed and evaluated by the staff. The scope of the review included the design bases, system description and performance, inspections and tests, and instrumentation requirements.

Compliance with the regulatory requirements and acceptance criteria described above in Section 5.4.7.3 is summarized in the following paragraphs.

Except for the open items discussed above, on the basis of the Technical Evaluation presented in Section 5.4.7.4 above, the staff concludes that there is reasonable assurance that the U.S. EPR RHRS will operate as intended, with or without offsite power and given any single-active component failure.

The regulatory basis for the staff's conclusions is summarized in the paragraphs below.

1. The application meets GDC 2 with respect to Regulatory Position C-2 of RG 1.29 concerning the seismic design of SSCs whose failure could cause an unacceptable reduction in the capability of the RHRS. The RHRS is designated Seismic Category I and is required to withstand the effects of a safe-shutdown earthquake.

2. The application meets GDC 4 with respect to dynamic effects associated with flow instabilities and loads. The entire RHRS is designed to remain water solid when connected to the RCS, thus precluding the possibility of piping water hammer events and pump cavitation.

3. The application meets GDC 19 with respect to MCR requirements for normal operation and shutdown conditions in so far as minimum RHRS controls in both the MCR and the remote shutdown station are delineated in FSAR Tier 1, and ITAAC are specified to ensure proper functionality of the RHRS controls and displays.

4. The application meets GDC 34 by meeting the regulatory positions contained in BTP 5-4.

5. The application meets the requirements of 10 CFR 50.34(f)(2)(xxvi) by providing a leakage detection and control capability.

5.4.8 Reactor Water Cleanup System (Not Used in U.S. EPR Design)

5.4.9 Main Steamlines Feedwater Piping (Not Used in U.S. EPR Design)

5.4.10 Pressurizer

The PZR regulates the RCS pressure during steady-state operation and system transients by maintaining a saturated water-steam mixture in the PZR. The PZR connects to the RCS through a surge line via the hot leg of RCS loop 3. The surge line allows continuous coolant volume and pressure adjustments between the PZR and the RCS. The PZR uses sprays to reduce pressure by quenching the PZR steam bubble, and uses electrical heaters to heat the water and maintain a saturated condition. For overpressure protection of the RCS, the PZR is equipped with three PSRVs. Refer to FSAR Tier 2, Section 5.2.2 for a description of overpressure protection.

Detailed information about the PZR and the staff's evaluation and conclusion regarding U.S. EPR's PZR design features and performance requirements is discussed in the following sections of this report:

- 5.2.2, "Overpressure Protection Review Responsibilities"
- 15.6.1, "Inadvertent Opening of a PWR Pressurizer Pressure Relief Valve or a BWR Pressure Relief Valve"
- 3.9.1, "Special Topics for Mechanical Components"
- 3.9.2, "Dynamic Testing and Analysis of Systems, Components, and Equipment"
- 3.9.3, "ASME Code Class 1, 2, and 3 Components, Component Supports, and Core Support Structures"
- 5.2.3, "Reactor Coolant Pressure Boundary Materials"
- 5.2.4, "Reactor Coolant Pressure Boundary Inservice Inspection and Testing"

5.4.11 Pressurizer Relief Tank

5.4.11.1 *Introduction*

Design-basis events (DBE) can lead to overpressurization of the RCS and PZR. Depressurization is accomplished through the discharge of the steam in the PZR into the PRT through three PSRVs. The purpose of the PRT is to contain and condense the steam. If the RCS over pressurizes during a severe accident, the steam in the pressurizer is discharged into the PRT through two primary depressurization system valves. Two rupture disks protect the PRT against exceeding its design limits.

5.4.11.2 *Summary of Application*

FSAR Tier 1: The FSAR Tier 1 information associated with this section is found in FSAR Tier 1, Section 2.2.1.

FSAR Tier 2: The applicant has provided a FSAR Tier 2 system description in Section 5.4.11, summarized here in part, as follows:

The PRT, pressurizer and its associated relief lines, PSRVs and PDS valves are shown in FSAR Tier 2, Figure 5.1-4, “RCS Piping and Instrumentation Diagram,” Sheet 3 of 7. Additional piping used for filling, cooling, venting, draining and gas extraction is also shown in this figure. The general configuration of the PRT is shown in FSAR Tier 2, Figure 5.4-7, “Pressurizer Relief Tank.”

The PRT is normally maintained at a slight vacuum with a blanket of nitrogen gas over the water. A spray header, in the upper portion of the PRT, provides cooling water from the drain and vent recirculation cooling system and makeup water from the demineralized water system.

The PRT and associated piping and valves downstream from the PSRVs and PDS valves are non-safety-related. They are designed to Seismic Category II requirements, and their quality classification is group D.

The PRT is equipped with two rupture disks, which protect the PRT against exceeding its design limits. Each of the rupture disks can accommodate the flow from all three PSRVs. Both the PRT and rupture disks are located in areas where their failure would not cause failure of any safety-related SSCs or create adverse environmental conditions.

The primary design parameters of the PRT are provided in FSAR Tier 2, Table 5.4-8, “Pressurizer Relief Tank Design Parameters.”

ITAAC: The ITAAC associated with FSAR Tier 2, Section 5.4.11 are given in FSAR Tier 1, Section 2.2.1.

Technical Specifications: There are no FSAR Tier 2 Technical Specifications for this area of review.

5.4.11.3 *Regulatory Basis*

The relevant requirements of NRC regulations for this area of review, and the associated acceptance criteria, are given in Section 5.4.11, “Pressurizer Relief Tank,” of NUREG-0800 and are summarized below. Review interfaces with other SRP sections can also be found in Section 5.4.11 of NUREG-0800. These regulations include:

1. GDC 2, as it relates to the protection of essential systems from the effects of earthquakes.

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.

Acceptance criteria adequate to meet the above requirements include:

1. Positions C.2 and C.3 of RG 1.29 which recommends that the rupture disk have a relief capacity that at least equals the combined capacity of the pressurizer relief and safety valves.

2. As provided in the SRP 5.4.11, the PRT 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 anticipated operational occurrences. The location of the tank should be such that the rupture disks do not pose a missile threat to safety-related equipment.

5.4.11.4 *Technical Evaluation*

The staff evaluated this section of the FSAR against GDC 2 and GDC 4. The review procedures recommended in SRP Section 5.4.11 were followed.

DBEs could result in RCS overpressurization. To relieve the overpressurization during a DBE, the three PSRV open. The steam from the PZR is discharged through the PZR relief piping into the PRT. The PRT is located in the Reactor Building. The PSRVs are discussed in FSAR Tier 2, Section 5.4.13, "Safety and Relief Valves." In addition to the three PSRVs, there are two primary depressurization system valves which are used for severe accident mitigation. This is discussed in FSAR Tier 2, Section 19.2, "Severe Accident Evaluations." Both the PSRVs and the PDS valves share the same 40.64 cm (16 in.) discharge line. The PRT also limits the release of radioactive water and steam into the containment.

The PRT and associated piping is shown in FSAR Tier 2, Figure 5.1-4 - RCS Piping and Instrumentation Diagram, Sheet 3 of 7. A schematic of the PRT is provided in FSAR Tier 2, Figure 5.4-7. The PRT contains water and a nitrogen gas blanket, and it is maintained under slight vacuum. A spray header provides cooling water from the drain and vent recirculation cooling system and makeup water from the demineralized water system. The gaseous waste processing system supplies nitrogen gas to the PRT through the PZR relief discharge lines and extracts gases from the PRT through an extraction line connected to the PRT. There are two rupture discs that protect the PRT from overpressurization.

The PRT and associated piping are not safety-related, they are Quality Group D, and are designed to Seismic Category II requirements to prevent damage to safety-related systems in the event of an earthquake. This meets GDC 2. These requirements are further addressed in FSAR Tier 2, Section 3.2. The PRT and associated piping and valve classification is provided in FSAR Tier 2, Table 3.2.2-1, "Classification Summary," Sheet 5 of 182, "Pressurizer Relief Tank and Associated Piping & Valves (downstream of PSRV and PDS Valves)."

The PRT, rupture disks, and associated piping are situated so that they cannot damage safety-related SSCs resulting from missile generation or adverse environmental conditions. This meets GDC 4. The discharge of the rupture disks is directed towards an opening in the floor of the cubicles for RCPs two and three (RCP bunker). The discharge is routed such that any flow will not impact any safety-related components in the cubicle.

SRP Section 5.4.11, "SRP Acceptance Criterion," Item 2 criterion E provides guidance for the reviewer to verify that the location of the PRT should be such that the rupture disks do not pose a missile threat to safety-related equipment, to comply with GDC 4. FSAR Tier 2, Section 3.5, "Missile Protection," addresses protection against internally generated missiles for safety-related SSCs. The staff reviewed FSAR Tier 2, Section 3.5 and could not identify any reference in the context of the impact of missile generation from the PRT and its associated piping failures on SSCs or the missile behavior from a rupture disk. In RAI 117, Question 05.04.11-2, the staff requested that the applicant provide additional information to address this issue. In a

December 15, 2008, response to RAI 117, Question 05.04.11-2, the applicant stated that the PRT rupture disks have a bolted flange connection on the PRT discharge nozzles and the heavy walled discharge piping to provide protection against missile generation. The PRT, discharge nozzles and piping are located inside a concrete enclosure that protects surrounding equipment from potential missiles.

The applicant also stated that FSAR Tier 2, Section 3.5.1.2, "Internally Generated Missiles Inside Containment," describes the general methodology used for missile prevention and protection inside containment. The general arrangement drawings of the Reactor Building are provided in FSAR Tier 2, Section 3.8, "Design of Category I Structures," and Section 3.8.1, "Concrete Containment." FSAR Tier 2, Figure 3.8-4, "Reactor Building Plan at Elevation +5 Feet," FSAR Tier 2, Figure 3.8-12, "Reactor Building Section B-B," and FSAR Tier 2, Figure 3.8-13, "Reactor Building Section C-C," show the location of the PRT room and the surrounding concrete enclosure.

The applicant further stated that a failure modes and effects analysis (FMEA) for the PRT, performed by the applicant, determined that any single failure of the PRT and its associated piping will have no adverse consequence on plant safety. Additionally, as described in FSAR Tier 2, Table 3.2.2-1, "Classification Summary," Sheet 5 of 182, "Pressurizer Relief Tank and Associated Piping & Valves (downstream of PSRV and PDS Valves)," the PRT is classified as Seismic Category II in accordance with RG 1.29, so that its failure would not impair the ability of the surrounding components to perform their safety functions. The staff finds the applicant has adequately addressed this issue and, therefore, the staff considers RAI 117, Question 05.04.11-2 resolved.

The staff could not identify a discussion of environmental effects on safety-related systems due to a failure of the PRT and its associated piping. In RAI 117, Question 05.04.11-3, the staff requested that the applicant provide additional information to address this issue, specifically, "Has the harsh environmental conditions (such as temperature, humidity and radiation) that would be created by the discharge from a rupture disk of the PRT been considered as a potential impact on any safety-related equipment or components in the affected area?" In a December 15, 2008, response to RAI 117, Question 05.04.11-3, the applicant stated that FSAR Tier 2, Section 5.4.11.2, "System Description," states that the discharge of a rupture disk is routed such that any flow will not impact any safety-related components; therefore, there is no need to consider the environmental effects of the discharge from a PRT rupture disk. The applicant also concluded that compliance with GDC 4 does not require a need to address the potential environmental impact from the PRT created by the discharge from a rupture disk; none of SRP Section 5.4.11, "SRP Acceptance Criterion," Item 2 criteria provide guidance for the reviewer to verify that adverse environmental conditions from a rupture disk failure do not pose a threat to safety-related equipment, to comply with GDC 4.

The applicant also stated that FSAR Tier 2, Appendix 3D, "Methodology for Qualifying Safety-Related Electrical and Mechanical Equipment," identifies the methodology, parameters, and service conditions for environmental qualification of equipment. As shown in FSAR Tier 2, Appendix 3D, Figure 3D-1, "Typical Combined LOCA/SLB Inside Containment Temperature Service Conditions Envelope," the maximum EQ temperature inside the Reactor Building exceeds 204.4 °C (400 °F) which is greater than the operating temperature range for the PRT, 15 °C to 126.7 °C (59 °F to 260 °F).

Since the maximum operating temperature of the PRT is below the maximum EQ temperature inside the Reactor Building and the discharge from a rupture disk is routed so that the flow does not impact any safety related components, the staff concludes that the discharge from the rupture disk will not have an adverse environmental impact on PRT. Therefore, the staff concludes that the applicant has adequately addressed this issue and RAI 117, Question 05.04.11-3 is resolved.

In a December 15, 2008, response to RAI 117, Question 05.04.11-5 regarding FSAR Tier 2, Section 3.10, “Seismic and Dynamic Qualification of Mechanical and Electrical Equipment,” the applicant described the qualification of mechanical and electrical equipment. The mechanical and electrical equipment in the Reactor Building are designed for harsh environmental conditions and a harsh radiation environment. The applicant also referred to a December 15, 2008, response to RAI 117, Question 05.04.11-2, whereby, the applicant stated that FMEA for the PRT, performed by the applicant, determined that any single failure of the PRT and its associated piping will have no adverse consequence on plant safety.

The applicant stated it is also reasonable to conclude, should a PRT rupture disk fail, the safety-related SSCs in the Reactor Building, which are designed to post-accident conditions from LOCAs and SLBs, would not be adversely affected from the discharge from the PRT, since the mass, stored energy, and radioactivity released from the PRT would be bounded by LOCAs.

FSAR Tier 2, Revision 0 stated that Section 1.7, “Drawings and Other Detailed Information,” contains general arrangement and layout drawings for structures and systems. The staff reviewed Revision 0, Section 1.7 and could not identify any general arrangement and layout drawings pertaining to the PRT. In RAI 117, Question 05.04.11-5, the staff requested that the applicant provide drawings and additional information to address this issue. The staff also requested that the applicant discuss how the drawings were used to assess whether other SSCs inside containment are protected from the effects of high-energy line breaks and moderate-energy leakage cracks in the PZR relief system. In a December 15, 2008, response, the applicant stated that FSAR Tier 2, Section 5.4.11.3 will be revised to change the reference from FSAR Tier 2, Section 1.7 to FSAR Tier 2, Section 3.8.1. The applicant also sated that the method of analysis for piping is described in US EPR FSAR Tier 2, Section 3.6.2, which also describes the criteria for break postulation in piping. After performing a stress analysis of the PRT piping system to determine possible pipe break locations, the general arrangement and layout drawings were reviewed to determine if SSC's will be impacted by jet impingement from the postulated breaks. The general arrangement and layout drawings pertaining to the PRT are found in FSAR Tier 2, Section 3.8.1. FSAR Tier 2, Revision 1 has been updated to refer to this FSAR Tier 2 section 3.8.1. The staff finds the applicant has adequately addressed this issue and, therefore, the staff considers RAI 117, Question 05.04.11-5 resolved.

The PRT is designed to handle 110 percent of the steam in the PZR at full-power without exceeding the design pressure of the rupture disks or the design temperature of the PRT. The limiting scenario analyzed was a PZR discharge resulting from the turbine trip AOO described in FSAR Tier 2, Section 15.2.2. The tank sizing calculation assumes an initial PRT water temperature of 54.4 °C (130 °F) and an initial PRT pressure of 101.35 kPa (0 psig). The calculated pressure is less than or equal to 1.0 MPa (145 psia), and the final temperature is below the saturation temperature, 180 °C (356 °F). Therefore, the FSAR Tier 2, Table 5.4-8 design pressure of 2.51 MPa (350 psig) and design temperature of 223.9 °C (435 °F) provide substantial margin.

The flow area of one rupture disk is larger than required to handle the full flow rate from the three PSRVs. The rupture disk prevents the PRT pressure from exceeding the design pressure differential 2,068.4 kPa (300 psid). In RAI 117, Question 05.04.11-4, the staff requested that the applicant state the purpose of the second rupture disk. In a December 15, 2008, response, the applicant stated that the second rupture disk is provided for redundancy and is not credited for the prevention or mitigation of any design basis events. The staff finds the applicant has adequately addressed this issue and, therefore, the staff considers RAI 117, Question 05.04.11-4 resolved.

The PRT and rupture disks are designed for a full vacuum, and the PRT will not collapse if the contents are cooled following a discharge of steam from the pressurizer without the addition of nitrogen. The PSRV and PDS valve discharge piping is designed for pressures and temperatures anticipated during design-basis events. The PSRV discharge piping is sized so that the backpressure in the discharge system does not impede the overpressure protection function. A vacuum breaker on the pressurizer relief system piping prevents water from the PRT being drawn through the sparger and up the relief line when steam in the relief line is condensed after a relief valve actuation. Based on the evaluation discussed above and the resolution of RAI 117, Questions 05.04.11-2, 3, 4, and 5, the staff concludes that the design feature meets the relevant requirements of GDC 4.

The PRT is subject to testing during construction in accordance with the ASME Section VIII, Division 1. This is consistent with the ASME code indicated in FSAR Tier 2, Table 3.2.2-1, "Classification Summary," Sheet 5 of 182, "Pressurizer Relief Tank and Associated Piping & Valves (downstream of PSRV and PDS Valves)." The staff identified an initial test in FSAR Tier 2, Section 14.2.12.12.15, "Primary Depressurization (Test No. 151)." The objectives of the test are to verify (1) the depressurization flow paths of the DPS, (2) the PSRVs and PDS valves and associated piping can perform as designed, and (3) the proper operation of the reactor coolant gas vent system. The staff reviewed Test No. 151 and found the test to be adequate because it conforms to the guidance described in RG 1.68.

The FSAR Tier 2 states that the PRT is designed with instrumentation nozzles for pressure, level and temperature measurements. The main control room alarms indicate high pressure, temperature, and high and low water levels. The instrumentation nozzles are located to allow measurements in both the liquid and gaseous phases.

The staff determined the PRT isolation valves in FSAR Tier 1, Table 2.2.1-2, "Equipment and Valve Actuator Power Supplies and Controls," contained MCR/RSS (remote shutdown station) displays and controls and valve fail positions. The isolation valves are on the PRT cooling supply and returns, PRT fill, PRT gaseous waste processing line, and PRT degassing line. The staff finds that the FSAR Tier 1 information on these isolation valves is adequate because it conforms to the guidance in RG 1.68. For these isolation valves, the staff identified ITAAC in FSAR Tier 1, Table 2.2.1-5. These ITAAC pertain to retrievability of the displays in the MCR and RSS. They test whether the RCS system equipment controls are provided in the MCR and RSS. Finally, they test whether following loss of power, the valves fail in the position indicated (Fail-As-Is). The staff finds these ITAAC adequate because they conform to the guidance in RG 1.68.

The staff could not readily identify any additional references to the PRT instrumentation in FSAR Tier 1 and Tier 2, including the initial testing and the ITAAC related to the MCR alarms. In RAI 117, Question 05.04.11-6, the staff requested that the applicant identify any additional references to PRT instrumentation. In a December 15, 2008, response to RAI 117, Question 05.04.11-6, the applicant provided the following information.

In addition to FSAR Tier 2, Chapter 5, the PRT instrumentation is addressed in the following sections of the FSAR:

- FSAR Tier 1, Section 3.4, "Human Factors Engineering," provides the design ITAAC for human factors engineering. FSAR Tier 1, Section 3.4 also provides the process for selecting the minimum inventory of alarms, displays, and controls for the U.S. EPR.

- FSAR Tier 2, Section 3.10 describes the qualification of mechanical and electrical equipment. PRT instrumentation sensors for pressure, level, and temperature are listed in FSAR Tier 2, Table 3.10-1, "List of Seismically and Dynamically Qualified Mechanical and Electrical Equipment," Sheets 37 and 38 of 161.

- FSAR Tier 2, Section 7.5, "Information Systems Important to Safety," describes information systems important to safety. PRT instrumentation for pressure, level, and temperature is listed in FSAR Tier 2, Table 7.5-1, "Initial Inventory of Post-Accident Monitoring Variables," Sheet 2 of 2.

- FSAR Tier 2, Section 14.2.12.12.5, "Primary Depressurization (Test No. 151)," Section 4.0, "Data Required," Item 4.4 identifies data for the PRT pressure, level, and temperature.

- FSAR Tier 2, Section 18.7, "Human System Interface Design," describes the human system interface (HSI) design process. PRT instrumentation for pressure, level, and temperature is listed in FSAR Tier 2, Table 18.7-1, "Minimum Inventory of Main Control Room Fixed Alarms, Displays, and Controls," Sheet 1 of 2.

The staff finds the applicant has adequately addressed this issue and, therefore, the staff considers RAI 117, Question 05.04.11-5 resolved.

As stated in the above section, "Summary of Application," there are no TS for this area of review. The PRT and associated piping and valves downstream of PSRVs and PDS valves are non-safety-related. Based on the FSAR Tier 1 and Tier 2 descriptions, the design satisfies GDC 2 and GDC 4. Therefore, there is no need for TS.

The staff did not identify any additional COL information items that should be included.

One minor error and one minor inconsistency were identified in the review of FSAR Revision 0 Tier 2, Section 14.2.12.3.14, "Pressurizer Safety Valve (Test No. 037)." First, Section 5.0, "Acceptance Criteria," states that safety-valves perform as described in Section 5.4.11. The staff concluded that even though the PSRVs are mentioned in FSAR Tier 2, Section 5.4.11, their performance is addressed in FSAR Tier 2, Section 5.4.13. Second, the test procedure refers to the PZR safety and relief valves as the PZR safety valves. To assure completeness and accuracy of the plant design and licensing basis, in RAI 117, Question 05.04.11-7, the staff requested the applicant address these issues.

In a December 15, 2008, response, the applicant stated that FSAR Revision 1, FSAR Tier 2, Section 14.2.12.3.14 has been renamed "Pressurizer Safety Relief Valves (Test No. 037)" and has been revised to refer to the valves as the PZR safety-relief valves, and the Acceptance

Criteria section refers to FSAR Tier 2, Section 5.4.13. The staff has confirmed that Revision 1 of FSAR, dated May 29, 2009, was revised as committed in the RAI response. Accordingly, the staff finds that the applicant has adequately addressed this issue and, therefore, the staff considers RAI 117, Question 05.04.11-7 resolved.

5.4.11.5 *Combined License Information Items*

For the PRT, no COL information items have been identified in FSAR Tier 2, Table 1.8-2. The staff finds this acceptable, because the proposed ITAAC and initial plant test program assure that the PRT will be constructed in accordance with the certified design.

5.4.11.6 *Conclusions*

The PRT system includes components and piping such as the PSRV and PDS valve connections to the tank, tank spray system piping, PRT cooling supply and return, PRT fill, PRT gaseous waste processing, and PRT degassing. The design of the PRT system conforms to nonnuclear safety and Quality Group D requirements, because the system is not necessary for safe-shutdown, accident prevention, or accident mitigation. Based on the review performed in the Technical Evaluation section, the staff has concluded the following:

- The applicant's design meets the requirements of GDC 2 as it relates to protection against the effects of earthquakes. Failure of non-safety-related systems does not have any adverse effects on safety-related systems.

- The applicant's design meets the requirements of GDC 4 as it relates to the protection of safety-related equipment from adverse environmental effects and from missiles generated by rupture disk failure. This criterion is met, because the system design prevents steam or water release to containment under any normal operating conditions or anticipated operational occurrences. In addition, the tank is orientated such that the rupture disks do not pose a missile hazard to safety-related equipment.

The FSAR Tier 1 and Tier 2 documentation provides sufficient information to support the conclusion that the PRT will perform its intended function of condensing the steam from the PZR following a DBE without activating the rupture disks or exceeding the design pressure and temperature of the RPT, and without causing damage or creating unwanted environmental effects on safety-related SSCs.

The applicant has provided sufficient information in the FSAR Tier 2, Section 5.4.11, Revision 1 for satisfying the applicable regulations described in SRP Section 5.4.11.

5.4.12 Reactor Coolant System High Point Vents

5.4.12.1 *Introduction*

The U.S. EPR RCS design includes provisions for venting non-condensable gases from the reactor pressure vessel and the RCS through high point vent locations. This high point venting capability may be needed to mitigate conditions during beyond-design-basis accidents to maintain long term core cooling. The high point venting capability is not required for the mitigation of any DBA.

5.4.12.2 *Summary of Application*

FSAR Tier 1: The FSAR Tier 1 information associated with this section is found in FSAR Tier 1, Section 2.2.1. The high point vents arrangement is shown in FSAR Tier 1, Figure 2.2.1-1, "Reactor Coolant System Functional Arrangement," Sheet 1 of 6.

FSAR Tier 2: The applicant has provided a FSAR Tier 2 system description in FSAR Tier 2, Section 5.4.12, "Reactor Coolant System High Point Vents," summarized here in part, as follows:

The configuration of the high point vent is shown schematically in FSAR Tier 2, Figure 5.1-4, "RCS Piping and Instrumentation Diagram," Sheet 2 of 7. The high point vent system is comprised of two systems, the normal vent and purge system and the post-accident high point vent system. The post-accident high point vents are designed and constructed in accordance with applicable industry codes and standards. The seismic and environmental qualifications of the high point vent components and piping are presented in FSAR Tier 2, Section 3.10, Table 3.10-1, Sheet 1 of 161, and in FSAR Tier 2, Section 3.11, Table 3.11-1, "List of Environmentally Qualified Electrical/I&C Equipment," Sheet 1 of 101, respectively. The two parallel post-accident high point vent paths are connected through a tee branch to the single high point vent line from the top of the RPV closure head.

The post-accident high point vent components and piping, from the branch tee to the discharge orifice, are designed to operate at a differential pressure between the RCS design pressure and the containment atmospheric pressure. In addition, the vents are designed to withstand the design transients identified in FSAR Tier 2, Section 3.9.1, "Special Topics for Mechanical Components," FSAR Tier 2, Section 3.9.1.1, "Design Transients." The vent piping design and analysis is performed as described in FSAR Tier 2, Section 3.9.3, "ASME Code Class 1, 2, and 3 Components, Component Supports, and Core Support Structures."

Each of the two parallel post-accident high point vent paths contain two solenoid-operated isolation valves in series to ensure isolation of the vent path in the event that one valve fails to close. Each isolation valve is powered from a separate Class 1E division source. Both flow paths merge into a common line, which discharges to the nearest steam generator cubicle. This common line contains a restricting flow orifice sized to prevent the possible discharge flow rate from exceeding the capacity of one chemical and volume control system pump should the vent valves fail to close.

The high point vents form part of the RCPB and are designed and fabricated in accordance with ASME Code, Section III, Class 1 requirements. Post-accident high point vent component classifications are identified in FSAR Tier 2, Section 3.2, "Classification of Structures, Systems, and Component," Table 3.2.2-1, "Classification Summary," Sheets 3, 8, and 182.

ITAAC: The ITAAC associated with FSAR Tier 2, Section 5.4.12 are given in FSAR Tier 1, Section 2.2.1, Table 2.2.1-5.

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

5.4.12.3 *Regulatory Basis*

The relevant requirements of NRC regulations for this area of review, and the associated acceptance criteria, are given in Section 5.4.12, "Reactor Coolant System High Point Vents," of NUREG-0800 and are summarized below. Review interfaces with other SRP sections can be found in Section 5.4.12 of NUREG-0800.

1. GDC 1, as it relates to the quality stands and records applicable to the design, fabrication, erection, and testing of the high point vents.

2. 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.

3. GDC 17, "Electrical Power Systems," and GDC 34, as they relate to the provision of normal and emergency power for the vent system components.

4. GDC 19, as it relates to the vent system controls being operable from the control room.

5. 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 points system.

6. GDC 36, "Inspection of Emergency Core Cooling System," as it relates to the vent system being designed to permit periodic inspection.

7. 10 CFR 50.46a, "Acceptance Criteria for Reactor Coolant System Venting Systems," 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.

8. 10 CFR 50.46(b)(5), as it relates to the long-term cooling of the core following any calculated successful initial operation of the ECCS to remove decay heat for an extended period of time.

9. 10 CFR 50.49, "Environmental Qualification of Electrical Equipment Important to Safety for Nuclear Power Plants," as it relates to environmental qualification of electrical equipment necessary to operate the reactor coolant vent system.

10. 10 CFR 50.55a and GDC 30, as they relate to the vent system components that are part of the RCPB being designed, fabricated, erected, tested, and maintained to high quality standards.

The objective of the review is to determine whether the post-accident high point vent system is capable of removing non-condensable gases from the primary coolant system with a minimal probability of inadvertent or spurious actuation. The post-accident high point vents should not cause or aggravate a design-basis LOCA. Additional post-accident high point vent and component-specific review guidance is provided in NUREG-0800 Section 5.4.12.

5.4.12.4 *Technical Evaluation*

The RCS post-accident high point vents in the EPR design are provided to meet the requirements of 10 CFR 50.46a for provisions to exhaust noncondensable gases from the RCS. This capability will assist in mitigating conditions that could impact long term core cooling after beyond-design-basis events, thereby fulfilling the regulatory requirement of 10 CFR 50.46(b). This high point venting capability is not required to mitigate any DBAs. The PZR also has high point venting capability as a part of the RCPB, but noncondensable gas build up in the PZR is not considered a detriment to long term core cooling functions in beyond-design-basis events. Routine venting of non-condensable gases from the PZR is a common practice during normal operations. In addition, the primary depressurization system is used for severe accident mitigation purposes (and meets the requirements of 10 CFR 50.46a), and also has the capability to vent noncondensable gases from the PZR should that be necessary.

A high point vent system is also used during cold startup operations to vent the RCS, including the steam generators by brief intermittent operation of the reactor coolant pumps in each RCS loop to sweep gases in the U-tubes into the RPV and activating the RPV high point vents to release the gases. The tops of the steam generator U-tubes are located at an elevation above the reactor vessel. Venting of non-condensable gases directly from the U-tubes is not compulsory, since it is impractical to install vents on the large number of U-tubes. It is unlikely that they would accumulate sufficient non-condensable gases to affect adequate long term core cooling by interrupting natural circulation in the RCS loops.

The staff used the review procedures and other guidance in SRP Section 5.4.12 to evaluate the post-accident high point vents in the U.S. EPR design.

There are two parallel path post-accident high point vents, consisting of valves and associated piping, that branch from a piping tee section connected to the RPV vent line. These high point vents are designed to operate at the differential pressure between the RCS design pressure and the containment atmosphere pressure. In addition, the vents are designed to withstand the design transients identified in FSAR Tier 2, Section 3.9.1.1. The vent piping design and analysis is performed as described in FSAR Tier 2, Section 3.9.3.

Each of the two parallel post-accident high point vent paths contain two solenoid-operated isolation valves in series to allow for isolation of the vent path in the event that one valve fails to close and to ensure one vent path is available should one valve fail to open. The vent valves fail in the closed position. In FSAR Tier 2, Section 5.4.12.1, the applicant refers two "vent valves." The staff reviewed FSAR Tier 2, Figure 5.1-4, Sheet 2, and FSAR Tier 2, Table 3.9.6-2, "Inservice Valve Testing Program Requirements," Sheet 6, and determined that these "vent valves" refer to the "isolation valves" listed on the figure as 30JAA10AA508-AA511.

Each isolation valve is powered from a separate Class 1E division power source that can be supplied by an alternative safety-related source as described in FSAR Tier 2, Section 8.3.1.1.1, "Emergency Power Supply System." The isolation valve power source is controlled by Technical Specification 3.8, "Electrical Power Systems," and 3.8.1, "AC Sources – Operating," and meets the requirements of GDC 17 and GDC 34, because the above design features minimize the possibility and potential impact of inadvertent actuation of the high point vents. In addition, the separate power supply sources provide the capability of testing the high point vent valves without actuation of their flow function.

The parallel flow paths merge into a discharge common line, which discharges to the nearest SG cubicle. The top of the SG cubicle is designed with special features, rupture and convection foils, to allow for mixing of combustible and noncombustible gases and steam and water mixtures with the containment atmosphere. The control of combustible gases is discussed in FSAR Tier 2, Section 6.2.5, “Combustible Gas Control in Containment.”

The common discharge line also contains a restricting orifice sized to prevent the possible discharge flow rate from exceeding the capacity of one CVCS charging pump (the definition of a LOCA) should the vent valves fail to close. The restricting orifice is located downstream of the vent valves to minimize any flow induced shock that may occur should the vent valves fail to close.

FSAR Tier 2 did not provide the size of the restricting orifice, the conditions used to determine the maximum coolant discharge flow rate, or the CVCS minimum charging pump flow rate used to assure a LOCA event would not be caused and activate the ECCS. The staff issued RAI 154, Question 05.04.12-2 to address this concern. In a February 11, 2009, response to RAI 154, Question 05.04.12-2, the applicant did not identify any changes to the FSAR to address this request.

The applicant’s response was, “In the event that the vent flow path does not close, flow is restricted by an orifice such that the normally operating chemical and volume control system can make up the loss of coolant mass with the assumption that only one high pressure charging pump is available. The makeup flow is based on flow from one charging pump at system pressure and temperature minus the flow diverted to the RCP seals. Additionally, the redundancy of the vent design (i.e., two valves in series for each vent line powered from separate electrical divisions) precludes the possibility of a stuck open flow path. Breaks upstream of the flow-restricting orifice are bounded by the small break loss of coolant accident (SBLOCA) analysis in FSAR Tier 2, Section 15.6.5.2.”

The staff finds the response to be inadequate. Additional information is needed to confirm that the CVCS can provide adequate makeup if the high point vent system fails open and to confirm that this failure would not be classified as a LOCA.

The staff is also concerned with this response, because FSAR Tier 2, Revision 1, Section 9.3.4, “Chemical and Volume Control System (Including Boron Recovery System),” Page 9.3-62 states: “The CVCS is not a safety system and is not required to supply reactor coolant makeup to the RCS in the event of small breaks or leaks in the RCPB. Also, the CVCS is not designed to perform the safety function of the ECCS during a DBA. Therefore, GDC 33 and GDC 35 are not applicable to the CVCS.”

GDC 33 requires that a reactor coolant makeup system be provided for protection against small breaks in the RCPB. The applicant has not identified such a makeup system that complies with GDC 33. Until the applicant can adequately explain why GDC 33 is not applicable to the CVCS system and then relies on the CVCS charging pumps for protection against small breaks in the RCPB, the staff considers this an open item pending further information from the applicant.

In FSAR Tier 2, Revision 1, Section 5.2, on Page 5.2-23 the applicant states: “Components that are connected to the RCS and are part of the RCPB, and that are of such a size and shape so that upon postulated rupture the resulting flow of coolant from the RCS under normal plant operating conditions is within the capacity of makeup systems that are operable from on-site emergency power. The emergency core cooling systems are excluded from the calculation of makeup capacity." The applicant does not describe the makeup systems required to perform this function.

In FSAR Tier 2, Revision 1, Section 9.3.4, "Chemical and Volume Control System (Including Boron Recovery System)," on Page 9.3-48 the applicant states: "Even though the CVCS is not required to perform any DBA mitigation functions and is only an operational system, emergency buses, backed by the emergency diesel generators (EDGs), power the CVCS charging pumps and [motor-operated valves] MOVs." GDC 33 states, in part: "The system shall be designed to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished using the piping, pumps, and valves used to maintain coolant inventory during normal reactor operation." In addressing the makeup system issue, the applicant should clarify the FSAR, as appropriate, to ensure that the required makeup system is identified and how the system conforms to the guidance in GDC 33.

The staff issued follow-up RAI 342, Question 05.04.12-5 to address the deficiencies in the FSAR and RAI responses related to the unidentified makeup systems and/or the CVCS makeup in the event the high point vent system failed to open. **RAI 342, Question 05.04.12-5, which is associated with the above request, is being tracked as an open item.**

The post-accident high point vents are part of the RCPB and are designed, fabricated, erected and tested to have an extremely low probability of abnormal leakage, rapidly propagating failure, and of gross rupture (GDC 14). In addition, the design quality standards of the high point vents are in conformance with AREVA report ANP-10266 "AREVA NP Inc. Quality Assurance Plan (QAP) for Design Certification of the U.S. EPR, Topical Report," which has been approved by the staff and complies with 10 CFR Part 50 Appendices A and B including the guidance in SRP Chapter 17, "Quality Assurance."

FSAR Tier 2, Section 5.4.12 provides a description of the post-accident high point vents. FSAR Tier 2, Figure 5.1-4, Sheet 2 of 7 provides the configuration of the high point vents. The FSAR Tier 1 information associated with this section is found in FSAR Tier 1, Section 2.2.1. FSAR Tier 1, Figure 2.2.1-1, Sheet 1 of 6, "RSC Functional Arrangement," illustrates the high point vents arrangement. 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 FSAR Tier 1, Table 2.2.1-1. The high point vents are designed to ASME Seismic Category 1 as described in FSAR Tier 2, Section 3.10, Sheet 1 of 161, and can withstand a design seismic load without loss of function. The above features comply with the requirements of 10 CFR 50.55a and GDC 1, GDC 14, and GDC 30 as they relate to components of the RCPB. Electrical equipment and components are seismic and dynamically qualified in accordance with IEEE Std 344 and IEEE Std 382.

The staff's review of the FSAR Tier 1 information could not verify the adequacy of the information related to the high point vents. FSAR Tier 1, Section 2.2.1 does not provide adequate inspection to verify the "as-built" high point vent system conforms to the analysis used to determine, should the vent system fail open, the discharge flow rate can be offset by the CVCS. The staff issued RAI 342, Question 05.04.12-5 to address this issue. **RAI 342, Question 05.04.12-5, which is also associated with this request, is being tracked as an open item.**

The high point vent valves actuators are designed as IEEE Class 1 components and are qualified for harsh environments in accordance with FSAR Tier 2, Section 3.11, Table 3.11-1, Sheet 1 of 101 and, therefore, satisfy the requirements of 10 CFR 50.49.

The high point vents are operated from the main control room and the remote shutdown station (RSS) where valve position displays are also available meeting the requirements of 10 CFR 50.46(a) and GDC 19. RPV water level indication is also provided in the MCR and RSS to aid operators. In addition, the high point vent valve actuators are controlled by the safety-related priority actuation and control system which is discussed in Section 7 of this report.

The displays and controls in the MCR and RSS for the high point vents should not increase the potential for operator error when interfacing with the high point vents is necessary. Human-factor analyses applicable to the MCR and RSS including procedures and training are discussed in Chapter 18, "Human Factors Engineering," of this report.

The design and installation arrangement of the high point vents provides the capability for necessary periodic inservice inspection and testing. FSAR Tier 2, Section 3.9.6, "Functional Design, Qualification, and Inservice Testing Programs for Pumps, Valves, and Dynamic Restraints," and FSAR Tier 2, Table 3.9.2-2, "Inservice Valve Testing Program Requirements," Sheet 6 of 91, provide the detailed requirements for the inservice testing program for the high point vent valves. FSAR Tier 2, Section 6.6 describes the inservice inspection requirements for the high point vents and FSAR Tier 2, Section 5.2.4 provides the program descriptions for preservice inspections and periodic inservice inspection for all RCPB components and piping, including the high point vents that the applicant states meets the requirements of GDC 36.

The applicant did not provide any discussion about the design of the high point vent system at the top of the PZR. In RAI 154, Question 05.04.12-3, the staff requested that the applicant provide a detailed discussion regarding the RCS high point vents on the top of the PZR relative to the design requirements and demonstrate that the vent system at the top of the PZR meet all design requirements specified for the RCS high point vents including the requirements of 10 CFR 50.46a. In a February 11, 2009, response the applicant stated that the design does not include high point vents on top of the PZR. The applicant further stated that the design description of the high point vents, which meets 10 CFR 50.46a, is provided in FSAR Tier 2, Section 5.4.12.2, "Design Description." Additionally, FSAR Tier 2, Section 5.4.12, states:

> In accordance with the requirements of 10 CFR 50.34(f)(2)(vi) and 10 CFR 50.46a, the U.S. EPR is provided with high point vents to remove non-condensable gases from the RPV. High point venting is not necessary to provide safety-related core cooling following any postulated design basis accident.

The applicant also stated that the vent system provided at the top of the PZR is a continuous vent to maintain the PZR gas space free of noncondensable gases and is not needed to satisfy the requirements of 10 CFR 50.46a. This continuous vent is used for operating evolutions, such as degasification during plant cooldown and depressurization. Noncondensable gas accumulation in the PZR will not prevent adequate core cooling during any design basis event. For beyond-DBE, the design includes redundant large vents that rapidly reduce RCS pressure to the point the accumulator inventory is added to the RCS prior to the occurrence of inadequate core cooling.

The acceptance criteria in SRP Section 5.4.12 state, in part: "For reactors with U-tube steam generators, procedures should be developed to remove sufficient gas from the U-tubes to ensure continued core cooling, since it is impractical to individually vent the thousands of U-tubes.” The procedures to operate the high point 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. Detailed operating procedures are not available for the staff review. In RAI 154, Question 05.04.12-4, the staff requested that the applicant address the high point vent system operational procedures. In a February 11, 2009, response, the applicant stated that instrumentation requirements for the high point vents are described in FSAR Tier 2, Section 5.4.12.5, “Instrumentation Requirements.” Operating procedures are the responsibility of the COL applicant as described in FSAR Tier 2, Section 13.5 and in FSAR Tier 2, Table 1.8-2, COL Information Item 13.5-1. The staff accepts a COL information item to require COL applicants to provide detailed operating procedures in their application to fulfill this acceptance criterion.

No technical specifications were identified in the FSAR applicable to the high point vents and the staff agrees that none are necessary.

In addition, no initial testing requirements are specified for the high point vents in FSAR Tier 2, Section 14.2. The staff agrees that initial testing of the high point vent system is not necessary during plant start up operations.

5.4.12.5 *Combined License Information Items*

For the RCS high point vents, no COL information items have been identified in FSAR Tier 2, Table 1.8-2. The staff finds this acceptable, because the proposed ITAAC and initial plant test program assure that the RCS high point vents will be constructed in accordance with the certified design.

5.4.12.6 *Conclusions*

The staff verifies that the applicant has provided sufficient information, and their review supports the following conclusions.

Except for the open items discussed above, the staff concludes that the design of the RCS high point vents is acceptable and meets the relevant requirements of 10 CFR Part 50, 10 CFR 50.44(c), 10 CFR 50.46, 10 CFR 50.46a, 10 CFR 50.49, and 10 CFR 50.55a, and GDC 1, GDC 14, GDC 17, GDC 19, GDC 30, GDC 34, and GDC 36, and Appendix B to 10 CFR Part 50.

This conclusion is based on the staff’s determination that RCS high point vents include components and piping to remotely exhaust non-condensable gases from the primary coolant system and vent the gases to the containment atmosphere. The review included the applicant's proposed design criteria and design bases, and the results of the applicant's analyses of the vent system design. In addition, the basis for acceptance in the staff review is conformance of the applicant's designs, design criteria, and design bases, including resolution of the open items, for the RCS vents and supporting systems to applicable regulatory guides, branch technical positions, and industry standards.

5.4.13 Safety and Relief Valves

Three SRVs provide overpressure protection for the RCPB during power and low temperature operation. Overpressure protection for the RCPB is addressed in Section 5.2.2 of this report.

Detailed information about the safety and relief valves and the staff's evaluation and conclusion regarding U.S. EPR's PSRV design features and performance requirements are discussed in the following sections of this report:

- 3.9.1, "Special Topics for Mechanical Components"
- 3.9.2, "Dynamic Testing and Analysis of Systems, Components, and Equipment"
- 3.9.3, "ASME Code Class 1, 2, and 3 Components, Component Supports, and Core Support Structures"
- 3.9.6, "Functional Design, Qualification, and Inservice Testing Programs for Pumps, Valves, and Dynamic Restraints"
- 3.10, "Seismic and Dynamic Qualification of Mechanical and Electrical Equipment"
- 5.2.4, "Reactor Coolant Pressure Boundary Inservice Inspection and Testing"
- 6.6, "Inservice Inspection of Class 2 and 3 Components"
- 5.2.3, "Reactor Coolant Pressure Boundary Materials"
- 6.1.1, "Engineered Safety-Features Materials"
- 10.3.6, "Steam and Feedwater System Materials"
- 15.6.1, "Inadvertent Opening of a PWR Pressurizer Pressure Relief Valve or a BWR Pressure Relief Valve"

5.4.14 Component Supports

The RCS supports and restraints control relative displacement of system components due to normal thermal and pressure expansion, and restrict displacement during seismic events and design basis accidents. The component supports also provide deadweight support for RCS components. The RCS piping is supported and restrained by the major RCS components. This section describes the supports and restraints for the primary RCS components.

Detailed information about the component supports and the staff's evaluation and conclusion regarding U.S. EPR component supports design features and performance requirements are discussed in the following sections of this report:

- 3.9.1, "Special Topics for Mechanical Components"
- 3.9.2, "Dynamic Testing and Analysis of Systems, Components, and Equipment"
- 3.9.3, "ASME Code Class 1, 2, and 3 Components, Component Supports, and Core Support Structures"
- 3.9.6, "Functional Design, Qualification, and Inservice Testing Programs for Pumps, Valves, and Dynamic Restraints"