5 Reactor Coolant System and Related Systems

5.0 Reactor Coolant System and Related Systems

5.1 Summary Description

Chapter 5, “Reactor Coolant System and Related Systems,” of this safety evaluation report (SER) describes the results of the review by the staff of the U.S. Nuclear Regulatory Commission (NRC or Commission), hereinafter referred to as the staff, of Chapter 5 of the Design Control Document (DCD), for the design certification (DC) of the Advanced Power Reactor 1400 (APR1400), submitted by Korea Electric Power Corporation (KEPCO) and Korea Hydro & Nuclear Power Co., Ltd (KHNP), hereinafter referred to as the applicant.

5.1.1 Introduction

The reactor coolant system (RCS) includes the reactor vessel, steam generators (SGs), reactor coolant pumps (RCPs), pressurizer, and associated piping. Two parallel heat transfer loops, each containing one SG and two RCPs, are connected to the reactor vessel, and one pressurizer is connected to one of the reactor vessel hot legs. The RCS circulates water in a closed cycle, removing heat from the reactor core and internals and transferring it to a secondary system. The RCS is located in the Containment Building.

5.1.2 Summary of Application

DCD Tier 1: In APR1400 DCD Tier 1 Section 2.4.1, “Reactor Coolant System,” the applicant stated that the RCS is located in the Containment Building and consists of a reactor vessel, two vertical U-tube SGs, four RCPs, one pressurizer, four pressurizer pilot operated safety relief valves, ninety three control element drive mechanisms, piping, heaters, controls, instrumentation, and valves. The RCS is a safety-related system that removes the heat generated in the reactor core and transfers the heat to the SGs. The RCS forms part of the pressure and fission product boundary between the reactor coolant and the Containment Building atmosphere.

The safety-related functions of the RCS are as follows:

- To form a barrier against the uncontrolled release of reactor coolant and radioactive materials to the containment.
- In conjunction with other systems, to provide cooling during all plant evolutions and anticipated operational occurrences (AOOs) to preclude significant reactor core damage.
- To provide protection of the RCS from overpressure by pressure relief devices for all design basis events (DBEs).

DCD Tier 2: The applicant has provided a DCD Tier 2 description of the RCS in Section 5.1, “Summary Description,” summarized here, in part, as follows:

The reactor is a pressurized water reactor (PWR) with two coolant loops. The RCS circulates water in a closed cycle, removing heat from the reactor core and internals and transferring it to a secondary system. The reactor vessel, SGs, RCPs, pressurizer, and associated piping are the major components of the RCS. Two parallel heat transfer loops, each containing one SG and two RCPs, are connected to the reactor vessel, and one pressurizer is connected to one of the reactor vessel hot legs. All RCS components are located inside the Containment Building.

Table 5.1.1-1, “Reactor Coolant System Design Parameters,” shows the principal parameters of the RCS.

Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC): DCD Tier 1 Table 2.4.1-4, “Reactor Coolant System ITAAC,” specifies the inspections, tests, analyses, and associated acceptance criteria for the RCS.

DCD Tier 1 Table 2.4.1-4, “Reactor Coolant System ITAAC,” states that an inspection will be conducted to verify that the as-built RCS conforms to the functional arrangement shown on DCD Tier 1, Figures 2.4.1-1, “Reactor Coolant System,” and 2.4.1-2, “Reactor Coolant System (Pressurizer).”

Additional ITAAC will be performed to verify the detailed DCD Tier 1 mechanical, electrical, and instrumentation information associated with the RCS.

Technical Specifications (TS): The TS associated with DCD Tier 2 Section 5.1, “Summary Description,” are given in DCD 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 DCD Tier 2, Sections 5.2, “Integrity of the Reactor Coolant Pressure Boundary,” 5.3, “Reactor Vessel”; and 5.4, “Reactor Coolant System Component and Subsystem Design.”

5.2 <u>Integrity of the Reactor Coolant Pressure Boundary</u>

This section evaluates the measures employed to provide and maintain the integrity of the reactor coolant pressure boundary (RCPB) for the facility’s design lifetime. Consistent with 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 that 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 that penetrates containment
- the second of two valves usually closed during normal reactor operation in system piping that does not penetrate the containment

5.2.1 Compliance with Codes and Code Cases

5.2.1.1 *Compliance with the Codes and Standards Rule, 10 CFR 50.55a*

5.2.1.1.1 *Introduction*

This SER section addresses use of acceptable codes (i.e., American Society of Mechanical Engineers [ASME] *Boiler and Pressure Vessel Code* (BPV 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 DC for the APR1400.

5.2.1.1.2 *Summary of Application*

DCD Tier 1: The DCD Tier 1 description of several of the Nuclear Island systems, including the RCS, indicates 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 BPV Code. The system description for DCD Tier 1 Section 2.3, "Piping Systems and Components," also indicates that a fatigue analysis would be done for Class 1 piping and components, as required by Section III of the ASME BPV Code.

DCD Tier 2: DCD Tier 2 Section 5.2.1.1, "Conformance with 10 CFR 50.55a," states that the RCPB components are designed and fabricated as Class 1 components in accordance with ASME BPV Code, Section III, except for the components that meet the exclusion requirements of 10 CFR 50.55a(c)(2). RCPB components that meet the exclusion requirements are classified as Quality Group B in accordance with U.S. Nuclear Regulatory Commission (NRC) Regulatory Guide (RG) 1.26, "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants," Revision 4, and are fabricated as Class 2 components in accordance with ASME BPV Code, Section III. The applicant stated that the classification of RCPB components conforms with the requirements of 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," Appendix A, General Design Criterion (GDC) 1, "Quality Standards and Records." The remaining safety-related components are classified as Quality Group C in accordance with RG 1.26 and are fabricated as Class 3 components in accordance with ASME BPV Code, Section III.

DCD Tier 2 Section 5.2.4, "Inservice Inspection and Testing of the Reactor Coolant Pressure Boundary," and Section 6.6, "In-service Inspection of Class 2 and 3 Components," provide a description of the application of ASME BPV Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plants," to the RCPB. DCD Tier 2 Section 3.9.6, "Functional Design, Qualification, and Inservice Testing Programs for Pumps, Valves, and Dynamic Restraints," provides a description of the application of the ASME OM Code to the RCPB.

The footnote in DCD Tier 2 Table 5.2-1, "Reactor Coolant System Pressure Boundary Code Requirements," states that the ASME BPV Code, 2007 Edition with 2008 Addenda is applicable to the APR1400 for construction. In addition, the components listed in this table are designed and constructed to meet the test and inspection requirements of the ASME OM Code and ASME BPV Code, Section XI, 2007 Edition with 2008 Addenda.

DCD Tier 2 Table 3.2-1, "Classification of Structures, Systems, and Components," provides the component classifications of pressure vessels, piping, pumps, valves, and storage tanks, along with the applicable component codes.

ITAAC: The proposed ITAAC, as required by 10 CFR 52.47(b)(1), are addressed in DCD Tier 1 based on the selection criteria in DCD Tier 2 Section 14.3, "Inspections, Tests, Analyses, and Acceptance Criteria."

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

5.2.1.1.3 *Regulatory Basis*

The relevant requirements of the NRC regulations for compliance with 10 CFR 50.55a and the associated acceptance criteria are given in NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition" (referred to as the Standard Review Plan (SRP)), Section 5.2.1.1, "Compliance with the Codes and Standards

Rule, 10 CFR 50.55a." and are summarized below. Review interfaces with other SRP sections can be found in SRP Section 5.2.1.1.

1. Title 10 CFR Part 50, Appendix A, 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. Title 10 CFR 50.55a, 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 by compliance with appropriate editions of published industry codes and standards. Under to 10 CFR 50.55a, components important to safety are subject to the following requirements:

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

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

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

1. RG 1.26, as it relates to determining quality standards acceptable to the staff for satisfying GDC 1 for other (i.e., nonreactor 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 SRP Section 5.2.1.1, "Compliance with the Codes and Standards Rule, 10 CFR 50.55a."

5.2.1.1.4 *Technical Evaluation*

Code of Record

The staff has reviewed the DCD for compliance with the requirements presented above. DCD Tier 2 Table 3.2-1 specifies the 2007 edition of the ASME BPV Code with 2008 addenda that is incorporated by reference in 10 CFR 50.55a as the code of record for Classes 1, 2, and 3 components in the APR1400 design. Classification of components into Quality Groups A, B, and C, which are associated with ASME Classes 1, 2, and 3, respectively, is reviewed for acceptability in Section 3.2.2, "System Quality Group Classification," of this report.

DCD Tier 2 Section 5.2.1.1, states that the code of record for the APR1400 design is the 2007 edition with 2008 addenda, which is incorporated by reference in 10 CFR 50.55a and is therefore acceptable. However, the staff noted some inconsistencies as discussed below.

The staff initially observed that codes and standards were not consistently stated throughout the DCD. The control element drive mechanism (CEDM), for example, was listed as "ASME Section III Class 1" in DCD Tier 2 Table 5.2-1. In DCD Tier 2 Table 3.2-1, item 11a, it was listed as "ASME Section III NB-2007 with 2008 addenda." Additionally, DCD Tier 2 Section 5.2.1.1 initially included the following confusing statement:

The components and code classes that are listed in Table 5.2-1 are in accordance with the provisions of 10 CFR 50.55a with this exception: the applicable ASME Code edition for the APR1400 is the 2007 Edition with 2008 Addenda.

Also, it was not initially clear to the staff which edition and/or addenda of the ASME OM Code is referenced for the APR1400, based on the footnote in DCD Tier 2 Table 5.2-1. To enhance the ease of use of the DCD, the applicant was asked in a public meeting from June 23-24, 2015 (ML15181A111), to check designations across the DCD and cite them with a consistent level of specificity, such as ASME BPV Code, Section III or ASME BPV Code, Section XI and to clarify the confusing statement and identify the edition and/or addenda of the AMSE OM Code.

DCD Tier 2 Section 5.2.1.1 and Table 5.2-1, Revision 0, as modified by a letter dated July 6, 2015 (ML15181A100), describes the ASME Code of record for the APR1400 design and RCPB requirements. The staff reviewed this information to determine whether it addressed the issues identified above, and concluded that the revised content is acceptable because it conforms with the guidance of SRP Section 5.2.1.1, Acceptance Criterion 2 by establishing minimum quality standards for the design, fabrication, erection, construction, testing, and inspection of RCPB components and other safety-related fluid systems by compliance with appropriate editions of published industry codes and standards. This satisfies GDC-1, and therefore is acceptable. The staff confirmed incorporation of these changes in the DCD; therefore, this item was resolved and closed.

Pressure Boundary Definition

DCD Tier 2 Section 5.2, "Integrity of the Reactor Coolant Pressure Boundary," initially stated that the RCPB is defined in accordance with American National Standards Institute/American Nuclear Society (ANSI/ANS) 51.1, "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants." It was unclear why DCD Tier 2 Section 5.2 referenced ANSI/ANS 51.1 for the RCPB definition instead of 10 CFR 50.2. Also, the list of RCPB components in DCD Tier 2 Section 5.2 did not include RCS safety and relief valves in accordance with the 10 CFR 50.2 definition. The staff issued RAI 40-7958, Question 05.02.01.01-3 (ML15174A380), requesting the applicant to address these issues and make associated changes to the DCD.

In its response to RAI 40-7958, Question 05.02.01.01-3 (ML15197A277), the applicant stated that DCD Tier 2 Section 5.2 will be revised to update the RCPB definition.

The staff determined that the applicant's response to RAI 40-7958, Question 05.02.01.01-3, is acceptable because the applicant addressed the deficiency in the application by proposing a revision to DCD Tier 2 Section 5.2, referencing 10 CFR 50.2 and including the RCS safety and relief valves in the referenced list to meet the requirements of 10 CFR 50.2.

The staff confirmed the incorporation of changes in DCD Tier 2 Section 5.2, with a reference to 10 CFR 50.2 and included the RCS safety and relief valves in the RCPB list to meet the requirements of 10 CFR 50.2. Therefore, RAI 40-7958, Question 05.02.01.01-3, is closed and resolved.

ASME Boiler Pressure Vessel Code, Section III Conditions

Requirements in 10 CFR 50.55a(b)(1), state conditions on the NRC's incorporation by reference of ASME BPV Code, Section III. Of particular note for this review area are the conditions regarding weld leg dimensions and seismic design of piping.

The requirement in 10 CFR 50.55a(b)(1)(ii) states the following:

> When applying the 1989 Addenda through the latest edition and addenda, applicants or licensees may not apply subparagraphs NB-3683.4(c)(1) and NB-3683.4(c)(2) or Footnote 11 from the 1989 Addenda through the 2003 Addenda, or Footnote 13 from the 2004 Edition through the 2008 Addenda to Figures NC-3673.2(b)-1 and ND-3673.2(b)-1 for welds with leg size less than 1.09 t_n.

DCD Tier 2 Section 3.12.2.1, "ASME Boiler and Pressure Vessel Code," provides some information related to this requirement, stating that "for socket weld leg dimensions, ASME BPV Code, Section III, Footnote 11 to Figure NC/ND-3673.2(b)-1 in the 1989 Edition is used for socket welds with leg size less than 1.09 t_n, instead of Footnote 13 from 2007 Edition and 2008 Addenda to Figures NC/ND-3673.2(b)-1." However, this footnote was not initially specified in DCD Tier 2 Section 5.2.1.1, so it was unclear, within the scope of this review section, how the APR1400 design meets the requirements of 10 CFR 50.55a(b)(1)(ii) regarding weld leg dimensions for ASME Classes 1, 2, and 3 piping. The staff issued RAI 40-7958, Question 05.02.01.01-2 (ML15174A380), requesting the applicant to address this issue.

In its response to RAI 40-7958, Question 05.02.01.01-2 (ML15197A277), the applicant stated that DCD Tier 2 Section 5.2.1.1 will be revised to refer to DCD Tier 2 Section 3.12.2.1 for the application of ASME BPV Code, Section III for safety-related piping systems. Additionally, DCD Tier 2 Section 3.12.2.1 will be revised to address the requirement in 10 CFR 50.55a(b)(1)(ii) regarding ASME Class 1 weld leg dimensions.

The staff determined that the applicant's response to RAI 40-7958, Question 05.02.01.01-2, is acceptable because the applicant stated how they are meeting the requirements in 10 CFR 50.55a(b)(1)(ii) by proposing to add the following language to DCD Tier 2 Section 3.12.2.1:

> For ASME Class 1 weld leg dimensions, the requirements of subparagraphs NB–3683.4(c)(1) and NB–3683.4(c)(2) are not applied for welds with leg size less than 1.09t_n."

This language is consistent with the referenced regulatory requirement. The applicant also proposed a revision to DCD Tier 2 Section 5.2.1.1 pointing to Section 3.12.2.1, where this discussion relevant to 10 CFR 50.55a will be included. The additional staff's evaluation, related to this topic, is provided in Section 3.12, "ASME Code Class 1, 2, and 3 Piping Systems, Piping Components and their Associated Supports," of this SER.

DCD Tier 2 Sections 3.12.2.1 and 5.2.1.1 were updated in the DCD with Section 5.2.1 referencing Section 3.12.2.1 for the application of ASME Section III safety-related piping in accordance with 10 CFR 50.55a. DCD Tier 2 Section 3.12.2.1 was revised to address the requirement in 10 CFR 50.55a(b)(1)(ii) regarding ASME Class 1 weld leg dimensions. Based on the review of the DCD, the staff has confirmed incorporation of the changes described above; therefore, RAI 40-7958, Question 05.02.01.01-2, is resolved and closed.

The requirement in 10 CFR 50.55a(b)(1)(iii) states that applicants or licensees may use Subarticles NB-3600, NC-3600, and ND-3600 for the seismic design of piping in the 2006 Addenda through the 2008 Addenda, subject to two conditions listed in 10 CFR 50.55a(b)(1)(iii)(A) and (B). DCD Tier 2 Section 3.12.2.1 states that the safety-related piping system design and analysis for the APR1400 are performed in accordance with the 2007 Edition with 2008 Addenda of the ASME Section III. However, in DCD Tier 2 Section 3.12.2.1, Revision 0, only the second provision of 10 CFR 50.55a(b)(1)(iii) was initially discussed, regarding the requirements of NB-3656(b). Both provisions need to be applied in the design and analysis of safety-related piping. The staff issued RAI 31-7931, Question 05.02.01.01-1 (ML15167A283), requesting the applicant to address this issue.

In its response to RAI 31-7931, Question 05.02.01.01-1 (ML15197A273), the applicant stated that DCD Tier 2 Section 3.12.2.1 will be revised to address the first provision of 10 CFR 50.55a(b)(1)(iii).

The staff determined that the applicant's response to RAI 31-7931, Question 05.02.01.01-1, is acceptable because the applicant proposed in its RAI response a revision to DCD Tier 2 Section 3.12.2.1 to meet the requirements of 10 CFR 50.55a(b)(1)(iii), stating that when applying Note (1) of Figure NB-3222-1, "Stress Categories and Limits of Stress Intensity for Level A and Level B Service Limits," for Level B service limits, the calculation of $P_{b,}$ primary bending stress, stresses includes reversing dynamic loads (including inertia earthquake effects) if evaluation of these loads is required by NB–3223(b). Additional staff evaluation related to this topic is provided in Section 3.12 of this report.

DCD Tier 2 Section 3.12.2.1 was updated to meet the requirements of 10 CFR 50.55a(b)(1)(iii), by stating that when applying Note (1) of Figure NB-3222-1 for Level B service limits, the calculation of $P_{b,}$ primary bending stress, stresses includes reversing dynamic loads (including inertia earthquake effects) if evaluation of these loads is required by NB–3223(b). Therefore, RAI 31-7931, Question 05.02.01.01-1, is resolved and closed.

5.2.1.1.5 *Combined License Information Items*

No COL information items have been identified in DCD Tier 2 Table 1.8-2, "Combined License Information Items," related to compliance with codes. In view of the above discussion, the staff determined that this is acceptable.

5.2.1.1.6 *Conclusions*

The staff concluded that the information provided in the DCD with respect to the use of codes and standards is acceptable. With respect to the ASME code of record, the application is sufficient to support the staff's finding of 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 because the applicant correctly identified the ASME Code of Record to apply to the APR1400 design.

5.2.1.2 *Applicable Code Cases*

5.2.1.2.1 *Introduction*

This SER section discusses the use of Code Cases associated with the ASME BPV Code and OM Code. In general, a Code Case is developed by ASME based on inquiries from the nuclear industry associated with possible clarification or modification of the Codes or alternates to the Code. The ASME BPV 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 BPV Standards Committee. ASME Code Cases acceptable to the 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 1.192, "Operation and Maintenance Code Case Acceptability, ASME OM Code," as incorporated by reference in 10 CFR 50.55a(a)(3).

5.2.1.2.2 *Summary of Application*

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

DCD Tier 2: In DCD Tier 2 Section 5.2.1.2, "Compliance with Applicable Code Cases," the applicant referred to certain Code Cases related to ASME BPV Code, Section III that have been previously approved by the NRC. DCD Tier 2 Table 5.2-4, "ASME Section III Code Cases," lists the following Code Cases:

- ASME Code Case N-4-13, "Special Type 403 Modified Forgings or Bars, Section III, Division 1, Class 1 and CS," dated February 12, 2008.
- ASME Code Case N-60-5, "Material for Core Support Structures, Section III, Division 1," dated February 20, 2004.
- 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," dated May 9, 2003.
- ASME Code Case N-249-14, "Additional Materials for Subsection NF, Class 1, 2, 3, and MC Component Supports Fabricated without Welding, Section III, Division 1," dated February 20, 2004.
- ASME Code Case N-759-2, "Alternative Rules for Determining Allowable External Pressure and Comprehensive Stress for Cylinders, Cones, Spheres, and Formed Heads, Section III, Division 1," dated January 4, 2008.

DCD Tier 2 Section 5.2.1.2 states that a COL applicant that references the APR1400 DC is to address the addition of ASME Code Cases that are approved in RG 1.84, ASME Code Cases invoked for the inservice inspection (ISI) program of a specific plant, and ASME Code Cases invoked for operation and maintenance activities.

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

TS: There are no TS 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 SRP Section 5.2.1.2 and are summarized below. Review interfaces with other SRP sections can be found in SRP Section 5.2.1.2.

1. Title 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 RCPB, as well as other systems important to safety. Where generally recognized codes and standards are used, they must be identified and evaluated to determine their adequacy and applicability and shall be supplemented or modified as necessary to assure a quality product in keeping with the required safety function.

2. Title 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 BPV Code, Section III Code Cases.

2. RG 1.147, as it relates to ASME BPV Code, 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 DCD for compliance with the requirements presented above. Acceptable ASME Code Cases that may be used for the APR1400 standard plant are those either conditionally or unconditionally approved in applicable NRC RGs, as incorporated into 10 CFR 50.55a(a)(3), and that are in effect at the time of DC. DCD Tier 2 Section 5.2.1.2 states that ASME BPV Code, Section III Code Cases acceptable for use in the APR1400 design, subject to the limitations specified in 10 CFR 50.55a, are listed in RG 1.84, RG 1.147, and RG 1.192. The ASME BPV Code, Section III Code Cases used in the APR1400 design are listed in Section 5.2.1.2.2 above.

The staff reviewed DCD Tier 2 Section 5.2.1.2 and determined that Table 5.2-4 lists ASME Code Cases that have been conditionally and unconditionally approved in accordance with RG 1.84. The staff determined that the references to unconditionally approved Code Cases are consistent with 10 CFR 50.55a and RG 1.84 and, are therefore, acceptable.

DCD Tier 2 Table 5.2-4 initially listed certain Code Cases that have been conditionally accepted by the staff in RG 1.84, but did not explicitly state that the Code Cases will be used with the conditions as mentioned in RG 1.84, Table 2, "Conditionally Acceptable Section III Code Cases." The applicant also did not describe how the Code Cases are implemented in accordance with these conditions. For example, N-71-18 Condition 4 states that Paragraph 15.2.2 is not acceptable as written and must be replaced with the following:

> *When not exempted by 15.2.1 above, the post weld heat treatment must be performed in accordance with NF-4622 except that ASTM A-710 Grade A Material must be at least 1000°F (540°C) and must not exceed 1150°F (620°C) for Class 1 and Class 2 material and 1175°F (640°C) for Class 3 material.*

The staff issued RAI 26-7948, Question 05.02.01.02-2 (ML15174A374), requesting the applicant to address this issue.

In its response to RAI 26-7948, Question 05.02.01.02-2 (ML15196A670), the applicant stated that among the conditionally acceptable ASME BPV Code, Section III Code Cases listed in Table 2 of RG 1.84, Code Cases N-60-5, N-71-18, and N-249-14 are used for the APR1400 design. The applicant also described how the materials of Code Cases N-60-5, N-71-18, and N-249-14 are applied in conformance with RG 1.84, Revision 36. The applicant provided the information specific to the conditions in RG 1.84, Revision 36 related to these Code Cases, as quoted below.

N-60-5

> The minimum yield strength for SA-193 Gr.B8M Cl.2 (Strain Hardened) of Code Case N-60-5 and ASME BPV Code, Section II Part D is 65,000 psi. The maximum yield strength does not exceed 90,000 psi.

N-71-18

1. The ultimate tensile strengths of the materials do not exceed 170 ksi.

2. This condition is not applicable.

3. Paragraph 5.5 in Regulatory Guide 1.84 Rev.36 would be paragraph 4.2. The paragraph 4.2 of Code Case N-71-18 describes the condition of using SMAW [shielded metal arc welding] electrode (E70XX, E80XX, E90XX, and E100XX). There are two weld joints between middle cooling shroud shell and seismic ring beam of IHA [integrated head assembly]. However, these two weld joints are welded by GTAW [gas tungsten arc welding] or FCAW [flux-cored arc welding], not SMAW. This condition is not applicable.

4. This condition is not applicable.

5. The weld joint is post weld heat-treated according to a heat treatment work instruction ruled by Condition (5) in Code Case N-71-18.

6. The fracture toughness requirements as listed in Code Case N-71-18 apply only to piping supports (ASTM A 500 Grade B) and not to Class 1 component supports.

N-249-14

1. AISI 4340 is used for Class 1 component support.

2. The ultimate tensile strength of AISI 4340 does not exceed 170 ksi.

DCD Tier 2 Table 5.2-4 will be revised with footnotes stating the materials used with each of the applicable ASME Code Cases.

The staff determined that the applicant’s response to RAI 26-7948, Question 05.02.01.02-2, is acceptable because the applicant stated how the materials of Code Cases N-60-5, N-71-18, and N-249-14 are applied in conformance with RG 1.84, Revision 36 and will revise DCD Tier 2 Table 5.2-4 with footnotes stating the materials used with each of the applicable ASME Code Cases.

DCD Tier 2 Table 5.2-4 was updated with footnotes stating the materials used with each of the applicable ASME Code Cases. Therefore, RAI 26-7948, Question 05.02.01.02-2, is resolved and closed.

The staff observed that there is no mention of ASME Code Cases in accordance with RG 1.147 or RG 1.192. Therefore, no finding is necessary with respect to these RGs.

In addition, when reviewing the list of Code Cases in DCD Tier 2 Table 5.2-4, the staff identified multiple locations where this table was initially incomplete or inconsistent with other DCD text.

Several examples of incomplete or inconsistent information in Table 5.2-4 are as follows:

- In DCD Tier 2 Section 3.12.2.2, “ASME Code Cases,” Code Case N-122-2, “Procedure for Evaluation of the Design of Rectangular Cross Section Attachments on Class 1 Piping, Section III, Division 1,” dated February 3, 2003, is mentioned as being used for piping systems and pipe supports; however, this Code Case is not mentioned in DCD Tier 2 Table 5.2-4.

- DCD Tier 2 Section 3.13, “Threaded Fasteners (ASME Section III Class 1, 2, and 3),” also states that ASME Classes 1, 2, and 3 component fasteners are fabricated using materials prescribed in ASME Code Cases allowed by RG 1.84, but does not mention any specific ASME Code Cases.

- DCD Tier 2 Section 3.9.3.4, “Component Supports,” states that ASME Classes 1, 2, and 3 component supports are designed and constructed in accordance with ASME BPV Code, Section III and ASME Code Case(s), but this section does not reference any specific ASME Code Cases. DCD Tier 2 Section 5.4.2.1.1, “Selection, Processing, Testing, and Inspection of Materials,” and Section 6.0, “Engineered Safety Features,” similarly do not have Code Case references.

- DCD Tier 2 Section 6.6.1, “Components Subject to Examination,” and Section 6.6.3, “Examination Techniques and Procedures,” discuss ASME Code Cases used in accordance with RG 1.147. However, no Code Cases in RG 1.147 are mentioned in DCD Tier 2 Section 5.2.1.2.

- In the DCD markups enclosed with the applicant’s letter dated June 1, 2015, ASME Code Cases OMN-1, 3, and 11 are mentioned in proposed revisions to DCD Tier 2 Section 3.9.6.3.1, “Inservice Testing Program for Motor-Operated Valves.” However, this letter did not include associated markups to DCD Tier 2 Section 5.2.1.2.

DCD Table 5.2-4 should be comprehensive in including all ASME Code Cases referenced to support the DC application, and the applicant should clarify in the application which specific ASME Code Cases are used when mentioned, such that the staff can determine the information necessary to make a finding on compliance with GDC 1 and 10 CFR 50.55a, as well as the Code Case RGs incorporated by reference into 10 CFR 50.55a. In addition, combined license (COL) Items 5.2(1), 5.2(2), and 5.2(3) were not initially clear that they address the scope outside the DC application (e.g., developing and executing an operational program such as inservice testing (IST)). The staff issued RAI 26-7948, Question 05.02.01.02-1 (ML15174A374), requesting the applicant to address this issue.

In its response to RAI 26-7948, Question 05.02.01.02-1 (ML15196A670), the applicant stated that DCD Tier 2 Section 5.2.1.2 will be revised to identify the DCD Tier 2 sections related to the application of ASME Code Cases that are not included in DCD Tier 2 Table 5.2-4. DCD Tier 2 Section 3.13.1.1, “Materials Selection,” and Section 5.4.2.1.1, which do not specify Code Cases applicable to the APR1400 design, will be revised to delete the statements regarding ASME BPV Code, Section III Code Cases. The applicant further indicated that the component design for engineered safety features described in Chapter 6, “Engineered Safety Features,” of the DCD may use ASME BPV Code, Section III Code Cases but the specific Code Cases cannot be defined until the actual component supplier is selected. The COL applicant is to address the ASME Code Cases that are approved in NRC RG 1.84 per COL Item 5.2(1). For clarity, the applicant committed to revise COL 5.2(1), 5.2(2) and 5.2(3) to show that they address the scope outside the DC application. COL applicants will address, for example, ASME Code Cases that they intend to implement as part of their ISI program and operations and maintenance program for a specific plant are identified.

The applicant committed to revise DCD Tier 2, Table 1.8-2 and Sections 3.13.1.1, 3.13.4, “References,” 5.2.1.2, 5.2.3.1, “Material Specification,” 5.2.6, “Combined License Information,” 5.4.2.1.1, and 6.0 for consistency with this response.

The staff determined that the applicant’s response to RAI 26-7948, Question 05.02.01.02-1, is acceptable because the applicant committed to revise DCD Tier 2 Section 5.2.1.2 to identify a complete and current set of ASME Code Cases, either through inclusion in DCD Tier 2, Table 5.2-4 or reference to other sections of the DCD where Code Cases are discussed. Specifically, the applicant committed to revise DCD Tier 2, Sections 3.13.1.1, 3.13.4, 5.2.1.2, 5.2.3.1, 5.2.6, 5.4.2.1.1, and 6.0 to clarify that ASME BPV Code, Section III Code Cases used are approved and listed in RG 1.84. This information is sufficient to support the staff’s determination that Code Cases are implemented in accordance with 10 CFR 50.55a, which incorporates by reference RG 1.84.

Based on the review of the DCD the staff has confirmed incorporation of the changes described above; therefore, RAI 26-7948, Question 05.02.01.02-1 is resolved and closed.

5.2.1.2.5 *Combined License Information Items*

DCD Tier 2, Table 1.8-2 and Section 5.2.6 contain the following three COL information items pertaining to Code Cases. The COL items, as modified by the applicant’s RAI response described above, are acceptable.

Table 5.2-1. Combined License Information Items Identified in the DCD

Item No.	Description	Section
COL 5.2(1)	The COL applicant is to address the addition of ASME Code Cases that are approved in NRC RG 1.84 at the time of the application.	5.2.1.2
COL 5.2(2)	The COL applicant is to address the ASME Code Cases approved in NRC RG 1.147 at the time of application and invoked for the ISI program of a specific plant.	5.2.1.2

COL 5.2(3)	The COL applicant is to address the ASME Code Cases approved in NRC RG 1.192 at the time of the application and invoked for operation and maintenance activities of a specific plant.	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 APR1400 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 DCD Tier 2 Table 1.8-2 for Code Cases.

5.2.1.2.6 *Conclusion*

The staff determined that the ASME Code Cases identified in the DCD are acceptable as specified in the applicable NRC RGs, with conformance to conditions in the applicable RGs, as discussed above. The staff concludes that the information provided in the DCD, 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*

For the APR1400, overpressure protection systems include all pressure-relieving devices for the following systems: 1) RCS, 2) primary side of auxiliary or emergency systems connected to the RCS; and 3) secondary side of SGs.

APR1400 includes pilot operated safety relief valve (POSRV) to provide overpressure protection of the RCS. Four POSRVs are connected to the top of the pressurizer by separate inlet lines. The use of shutdown cooling system (SCS) suction line relief valves provide sufficient pressure relief capacity to mitigate the most limiting low temperature overpressure protection (LTOP) events during low temperature conditions. Overpressure protection for the secondary side of the SGs is provided by spring-loaded main steam safety valves (MSSVs).

5.2.2.2 *Summary of Application*

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

DCD Tier 1 Section 2.7.1.2, "Main Steam System," describes the main steam system (MSS) and the features and equipment included for secondary side overpressure protection.

DCD Tier 2: DCD Tier 2 Section 5.2.2, "Overpressure Protection," describes overpressurization of the RCS and SGs is precluded by operation of the pressure relieving devices.

DCD Tier 2 Section 5.4.10, "Pressurizer," contains information on the pressurizer to maintain RCS operating pressure so the minimum pressure during operating transients is above the setpoint for the safety injection actuation signal (SIAS) and low pressure reactor trip and so that the maximum pressure is below the high pressure reactor trip setpoint. In addition, the pressurizer will preclude the opening of POSRVs during normal operational transients.

DCD Tier 2 Section 5.4.14, “Safety and Relief Valves,” contains information on the POSRVs and MSSVs to ensure overpressure protection for the RCPB.

ITAAC: The ITAAC associated with DCD Tier 2 Section 5.2.2, are given in DCD Tier 1 Section 2.4.1 and includes verification of POSRVs’ opening response time, set pressure, and minimum valve capacity.

The ITAAC associated with secondary side MSSV are given in DCD Tier 1 Section 2.7.1.2 for verification of MSSV relief capacity.

The ITAAC associated with SCS suction line relief valves used for LTOP are given in DCD Tier 1 Section 2.4.4, “Shutdown Cooling System,” for verification of SCS relief valve capacity.

TS: APR1400 TS 3.4.10, “Pressurizer Pilot Operated Safety Relief Valves (POSRVs),” requires four POSRVs operable, and each POSRV having sufficient lift capacity and opening time during operation.

APR1400 TS 3.4.11, “Low Temperature Overpressure Protection (LTOP) System,” requires the LTOP to be established.

APR1400 TS 3.7.1, “Main Steam Safety Valves,” requires the MSSVs to be operable for operating modes. In the event that one or more required MSSV is inoperable, TS action requires reduction in maximum power and variable overpower trip setpoint.

Initial Test Program: APR1400 DCD Tier 2 Section 14.2.12.1.3, “Pressurizer Pilot-Operated Safety Relief Valve Test,” includes testing to verify the opening and closing pressure and opening response time of the POSRVs.

APR1400 DCD Tier 2 Section 14.2.12.1.63, “Main Steam Safety Valve Test,” includes testing to verify the popping pressure of the MSSVs.

APR1400 DCD Tier 2 Section 14.2.12.1.20, “Shutdown Cooling System Test,” includes testing to verify the setpoint pressure of the LTOP relief valves.

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 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.

2. 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.

3. Title 10 CFR 52.47(a)(8), provides the requirement for DC reviews to comply with the technically relevant portions of the Three Mile Island (TMI) requirements in 10 CFR 50.34(f).

4. Title 10 CFR 50.34(f)(2)(x) and 10 CFR 50.34(f)(2)(xi) require that RCS safety and relief valves meet TMI Action Plan Items II.D.1 and II.D.3 of NUREG-0737, "Clarification of TMI Action Plan Requirements," issued November 1980.

Acceptance criteria adequate to meet the above requirements include:

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*

The staff reviewed the design for overpressure protection of the APR1400 described in DCD Tier 2, Sections 5.2.2, 5.4.10 and 5.4.14 in accordance with the guidance, review procedures, and regulations outlined and identified in the NRC SRP 5.2.2 of NUREG-0800 and BTP 5-2. The applicant planned to provide overpressure protection for the RCPB by POSRV, MSSVs, and relief valves in the SCS, in combination with the action of the reactor protection system (RPS).

5.2.2.4.1 *Design Bases*

The applicant precluded overpressure of the RCS by operation of the pressurizer, POSRVs and the RPS. Four POSRVs are classified as ASME Section III, Class 1 and are arranged on top of the pressurizer for overpressure protection of the RCPB. The pressurizer is designed to provide the adequate pressurizer size and spray capacity so the POSRVs are not actuated by overpressure events initiated by normal operation transients. The pressurizer POSRVs and RPS are designed to maintain the RCS pressure below 110 percent of design pressure during the worst-case loss-of-load event, as shown in DCD Tier 2 Figure 5.2.2-3, "Primary Pressure Normalized to Design Pressure vs. Time for the Worst-Case Loss-of-Load Event."

The applicant stated that overpressure protection for the secondary side of the SGs and the main steamlines up to the inlet of the turbine stop valve will be provided by direct-acting, spring-loaded, carbon steel MSSVs, designed in accordance with the ASME Section III, Class 2. The main steam system has four main steamlines from the two SGs. The applicant stated that five MSSVs are installed on each of the main steamlines upstream of the main steam isolation valve (MSIV) outside the containment for a total of 20 MSSVs. The MSSVs are sized conservatively to release steam flow equal to the full power level and SG pressure is limited to less than 110 percent of design pressure during the worst-case transient, as shown in DCD Figure 5.2.2-2, "Steam Generator Pressure Normalized to Design Pressure vs. Time for the Worst-Case Loss-of-Load Event."

The SCS contains two suction line relief valves (SI-179 or SI-189) that are designed to provide LTOP. The use of either relief valve provides sufficient pressure relief capacity to mitigate the most limiting LTOP events during low temperature conditions, in accordance with BTP 5-2. A description of the SCS is included in DCD Tier 2 Section 5.4.7, "Shutdown Cooling System."

The applicant performed an analysis to demonstrate the overpressure protection capability provided by the POSRVs and MSSVs during the most limiting AOO. The results of the analysis are provided in DCD Figures 5.2.2-2 through 5.2.2-4. However, the applicant did not provide information regarding the analysis methodology or details regarding input assumptions used in the analyses. The staff issued RAI 233-8244, Question 05.02.02-1 (ML15296A004), requesting the applicant provide additional details regarding the overpressure protection analysis. In its response to RAI 233-8244, Question 05.02.02-1 (ML15348A083), the applicant provided information regarding the analysis methodology, input assumptions, and sensitivity studies conducted to determine the limiting event in terms of overpressure protection. The staff found the applicant's response acceptable because it described a suitably conservative method for identifying and evaluating the limiting AOO in terms of overpressure protection. The applicant's results demonstrated that the overpressure protection system limits peak pressures in the primary and secondary systems to be less than 110 percent of their design values during a limiting AOO. Based on the results of the overpressure analysis, and the additional safety analyses evaluated in Chapter 15 of this SER, the staff finds the overpressure protection provided by the POSRVs and MSSVs satisfies the overpressure protection criteria specified in the SRP.

The applicant also provided additional information on the limiting LTOP transient: the event is assumed to be an inadvertent SIAS actuation with a low initial RCS pressure and one charging pump in operation. Because the RCS is at or very near a water solid condition, the maximum safety injection (SI) flow delivered by the four SI pumps results in a rapid pressure increase until the shutdown cooling suction line relief valves act to mitigate against the pressure increase. The applicant appropriately assumed the LTOP relief valve acts at its design pressure plus measurement uncertainty, conservatively assuming a higher pressure. With regards to energy input assumptions. The staff issued RAI 487-8609, Question 05.02.02-7 (ML16139A580), requesting the applicant to address whether the energy inputs were conservative and how the RCS conditions for this transient were calculated. In its response to RAI 487-8609, Question 05.02.02-7 (ML16161B237), the applicant provided additional information regarding the energy addition transient analysis methodology, assumptions, computer code, and input parameters designed to determine the limiting event in regards to overpressure protection. The applicant identified OVERP as the computer code used to provide the pressure response of the water-solid system of an energy addition transient. OVERP code, assumptions, initial conditions, and input data are described in technical report WCAP-15688, "CE-NSSS LTOP Energy Addition Transient Analysis Methodology." The staff reviewed the response information with respect to the technical report and found it compatible and acceptable. The staff determined that the response is acceptable because the assumptions, input parameters, and initial conditions were adequately conservative to perform the LTOP energy addition transient analysis pressure response.

The applicant did provide information on the secondary-to-primary heat transfer, using a secondary temperature 250 degrees Fahrenheit (°F) (139 degrees Celsius (°C)) greater than the RCS, which the staff concludes is substantially more conservative than the 100 °F (37 °C) difference allowed by technical specifications, and is therefore acceptably conservative.

In DCD Tier 2, Sections 5.2.2 and 5.4.10, the applicant provided the design details of the pressurizer for the APR1400. The pressurizer is a vertically mounted, bottom supported, cylindrical pressure vessel with replaceable direct immersion electric heaters vertically mounted in the bottom head. The pressurizer is furnished with nozzles for the spray, surge, and POSRVs, and with pressure, temperature, and level instrumentation. A manway is provided in the top head for access for inspection of the pressurizer internals. The pressurizer surge line is connected to one of the reactor coolant hot legs and the spray lines are connected to two of the cold legs at the reactor coolant pump discharge. Heaters are supported inside the pressurizer to preclude damage from vibration and seismic loadings.

In DCD Tier 1 Table 2.4.1-1, "Reactor Coolant System Equipment and Piping Location/Characteristics," the applicant stated that the pressurizer is located inside containment and designed in accordance with the ASME Code Section III (Class 1), Seismic Category I. The pressurizer is designed with adequate size and spray capacity to avoid POSRV's actuation during normal operating conditions. Four POSRVs are mounted on top of the pressurizer. In the event of over-pressurization in the pressurizer, the POSRVs will depressurize the pressurizer and discharge steam to the in-containment refueling water storage tank (IRWST). The pressurizer and POSRV flow diagram is given in DCD Tier 1 Figure 2.4.1-2, "Reactor Coolant System (Pressurizer)" and DCD Tier 2 Figure 5.1.2-3, "Pressurizer and POSRV Flow Diagram." DCD Tier 2 Figure 6.8-3, "In-containment Water Storage System Flow Diagram," shows the IRWST.

The pressurizer is designed to maintain RCS operating pressure so that the minimum pressure during operating transients is above the setpoint for the SIAS and low pressure reactor trip, and the maximum pressure is below the high pressure reactor trip setpoint. As indicated in DCD Tier 2 Section 5.4.10.3, "Design Evaluation," it is demonstrated by analysis in accordance with requirements for ASME Section III Class 1 vessels that the pressurizer is adequate for all normal operating and transient conditions expected during the life of the facility. The staff issued RAI 233-8244, Question 05.02.02-1 (ML15296A004), requesting the applicant to provide additional details regarding the analysis of the pressurizer size and spray capacity. In its response to RAI 233-8244, Question 05.02.02-1 (ML15348A083), the applicant contained the results of an analysis of the turbine trip event, which was identified by the applicant as the limiting AOO. The applicant's analysis demonstrated that the pressurizer and associated spray are sized such that the pressurizer POSRVs and RPS are not actuated for the limiting AOO. Based on the turbine trip representing a 100 percent load rejection, and the results of the analysis showing the POSRVs and RPS are not actuated during the turbine trip event, the staff determined that the applicant's response is acceptable for demonstrating the proper sizing of the pressurizer and associated spray capacity; therefore, RAI 233-8244, Question 05.02.02-1, is resolved and closed.

For the APR1400, as described in DCD Tier 2 Section 5.4.11, "Pressurizer Relief Tank," the IRWST is used as the pressurizer relief tank (PRT) to collect and condense the steam discharged from the pressurizer through the POSRVs. Section 5.4.11 of this SER contains the staff's review of the IRWST.

Chapter 15, "Transient and Accident Analysis," of this SER also assesses the pressurizer performance during AOOs. Pressurizer materials are reviewed under SRP Section 5.2.3, "Reactor Coolant Pressure Boundary Materials." Periodic ISI and testing of pressurizers to assess their structural and leaktight integrity are evaluated under SRP Section 5.2.4, "Reactor Coolant Pressure Boundary Inservice Inspection and Testing." Structural integrity of the pressurizer and methods of analysis are evaluated under SRP Sections 3.9.1, "Special Topics for Mechanical Components," 3.9.2, "Dynamic Testing and Analysis of Systems, Structures, and Components," and 3.9.3, "ASME Code Class 1, 2, and 3 Components Supports, and Core Support Structures."

DCD Tier 2, Sections 5.2.2 and 5.4.14 provide the design details of the pressurizer POSRVs to provide overpressure protection for the RCS. As part of the RCPB, the POSRVs are designed to meet the requirements of ASME Section III NB Class 1 components, as outlined in DCD Tier 2 Section 3.2, "Classification of Structures, Systems, and Components," Table 3.2-1, and as shown in DCD Tier 1, Table 2.4.1-2, "Reactor Coolant System Components List." Also, in accordance with RG 1.29, "Seismic Design Classification," Revision 4, issued March 2007, DCD Tier 2 Table 3.2-1 classifies the seismic category for the overpressure protection system related components of the RCS. While the POSRVs and pressurizer were classified as Safety Class 1, Seismic Category I components, the classification applied to the POSRV discharge piping is unclear. The staff issued RAI 233-8244, Question 05.02.02-2 (ML15296A004), requesting the applicant to clarify the classification.

In its response to RAI 233-8244, Question 05.02.02-2 (ML16028A221), the applicant stated that it will revise DCD Tier 1 Table 2.4.1-1, Table 2.4.1-2, and Figure 2.4.2-1, "In-containment Water Storage System" to clarify the seismic classification of the POSRV discharge piping and to be consistent between the DCDs. In its response to part a, the applicant stated that the seismic classification of the POSRV discharging piping is as follows: 1) POSRV discharging piping upstream of 3-way valves: Seismic Category II; and 2) POSRV discharging piping downstream of 3-way valves: Seismic Category I. Because the applicant appropriately clarified the seismic classification of these components, the staff determined that the proposed revision was acceptable. The staff confirmed that the revision was incorporated in DCD Tier 1 Tables 2.4.1-1 and 2, and Figure 2.4.1-2 of DCD; therefore, RAI 233-8244, Question 05.02.02-2 is resolved and closed.

Also, the applicant noted that it revised DCD Tier 2 Table 3.2-1 with its response to RAI 72-8020, Question 03.02.02-5 (ML15269A022). This revision included the addition of line items for the POSRV discharge piping upstream of 3-way valves and for POSRV discharge piping downstream of 3 way valves, indicating the upstream piping is seismic category II and the downstream piping in seismic category I. The staff confirmed that DCD Tier 2 was revised to include these changes, therefore, this item was resolved and closed.

In its response to RAI 233-8244, Question 05.02.02-2, part b (ML16028A221), the applicant stated that the location of the POSRV discharge connection to the IRWST is shown in DCD Tier 1 Figure 2.4.1-2 and DCD Tier 2 Figure 5.1.2-3, but not in DCD Tier 2, Figure 6.8-3 because Figure 2.4.1-2 and Figure 5.1.2-3 are schematic flow diagrams for the pressurizer and POSRV including the portion of the POSRV discharge, respectively. However, Figure 6.8-3 is only a configuration diagram of the in-containment water storage system (IWSS). The staff determined that the applicant's explanation is acceptable because it provided the necessary clarification. In its response to RAI 25-7844, Question 06.02.02-3 (ML15204A705), the applicant stated that it revised DCD Tier 2 Figure 5.1.2-3, to indicate the spargers and associated piping as the POSRV discharge, for additional clarification.

As described in DCD Tier 2 Section 5.4.14, the APR1400 contains four forged-steel type POSRVs connected to the top of the pressurizer by separate inlet lines. The steam from two POSRVs is discharged through one common discharge line. There are two main discharge lines discharging to the IRWST where the steam is released under water to be condensed and cooled.

Each POSRV provides the overpressure protection function with a main valve and two spring-loaded pilot valves in the assembly. A schematics diagram of the typical POSRV is included in Figure 5.4.14-1, "Pilot Operated Safety Relief Valve Schematic Diagram," of the DCD. The pressurizer POSRVs perform the overpressure protection function with two pairs of spring-loaded pilot assemblies and a main valve and perform the rapid depressurization function using two motor-operated pilot valves in series and a main valve. Each spring-loaded pilot valve in the assembly consists of a motor-operated isolation valve, spring-loaded pilot valve, check valve, and manual isolation valve. The spring-loaded pilot valve of each POSRV opens automatically if the system pressure increases to the POSRV set pressure, thus opening the main valve. The motor-operated isolation valves are normally open, but can be manually closed by an operator to prevent discharge when the spring-loaded pilot valves fail to close. The manual isolation valves are normally open, but can be manually closed to block the main valve from opening when repairing a spring-loaded pilot valve or conducting a setpoint test. The POSRVs are also sized to perform feed and bleed to prevent core uncover with the capacity of the safety injection pumps assuming a total loss of feed water accident. Each pressurizer POSRV contains a manual operation option for the rapid depressurization function with a main valve and two motor-operated pilot valves installed in series. The motor-operated pilot valves are normally closed, but an operator can remotely and manually open the valves to open the main valve for the rapid depressurization of the RCS.

DCD Tier 2 Section 5.2.2.9, "System Reliability," states that "the pressurizer POSRVs are pilot-operated mechanisms and cannot fail closed if the setpoint pressure is exceeded. The force of the spring on the main valve disc area is designed to resist the inward pressure force of 2.0 kg/cm^2 D (29 psid)." Since the main valve disc is not spring loaded, the staff is unable to determine which spring is being referenced in this DCD Tier 2 section. The staff issued RAI 233-8244, Question 05.02.02-6 (ML15296A004), requesting the applicant to provide additional details on the main valve spring on the main disc.

In its response to RAI 233-8244, Question 05.02.02-6 (ML15348A083), the applicant clarified the design and operation of the pressurizer POSRVs in reference to DCD Tier 2 Figure 5.2.2-6, "POSRV Schematic Diagram." The main valve, as depicted in the figure, is spring loaded with a designed spring force of 2.0 kg/cm2 D (29 psid). Once opened, the differential pressure at the disc piston of the main valve must be higher or equal to 2.0 kg/cm2 D (29 psid) to maintain the main valve open. Also, to maintain the main in the closed position, the differential pressure at the disc piston pressure required to open the valve from the closed position must be higher or equal to 10.2 kg/cm2 D (145 psid). The staff determined that the applicant's response is acceptable because the applicant provided a POSRV internal design diagram that identified the location of the main valve disc spring; thus, clarifying the proper operation of the POSRV. Therefore, RAI 233-8244, Question 05.02.02-6, is resolved and closed.

The POSRVs are designed to pass sufficient pressurizer steam to limit the RCS pressure to 110 percent of design pressure following a loss of load with delayed reactor trip, which is assumed to be initiated by the safety-grade signal from the RPS. The RCS is protected at elevated pressure by the POSRVs, which have a set pressure of 173.7 kg/cm2A (2,470 psia), as indicated in Table 5.4.14-1. Each POSRV has a minimum valve flow capacity of 244,900 kg/hr (540,000 lb/hr) steam. As shown in Figure 5.2.2-1, the optimized POSRV capacity is determined at the point where an additional increase in the capacity has a negligible effect on reducing the maximum RCS pressure during the loss-of-load transient. The sizing of the POSRVs, as indicated in DCD Tier 2 Section 5.2.2.1.1, is included in a referenced sensitivity study. The DCD describes the assumptions included in this study related to POSRV capacity, indicating the reactor coolant and main steam systems are at maximum rated output plus a 2 percent uncertainty margin at the onset of a loss-of-load transient. The staff noted that no credit is taken for plant control systems such as letdown, charging, pressurizer spray, turbine bypass, reactor power cutback, and feedwater addition (main and auxiliary) after turbine trip in the loss-of-load analysis; and a reactor scram is assumed to be initiated by the second safety grade signal from the RPS. However, the staff is unable to locate the referenced sensitivity study containing assumptions used for the POSRV sizing study. The staff issued RAI 233-8244, Question 05.02.02-1 (ML15296A004), requesting the applicant to provide additional POSRV capacity details, the basis for Figure 5.2.2-1 and to provide access to the analysis referenced in the DCD which contains an assessment describing the basis for POSRV sizing. In its response to RAI 233-8244, Question 05.02.02-1 (ML15348A083), the applicant provided the analysis referenced in the DCD. The analysis included the assumptions with the basis for taking no credit for plant control systems such as letdown, charging, pressurizer spray, turbine bypass, reactor power cutback, and feedwater addition. The staff found the assumptions, initial conditions, sensitivity study, and the detailed node and flow path methodology used in the KISPAC code to model the transient behavior of the fluid systems conservative. The staff finds the analysis acceptable and the staff concerns were adequately addressed.

As indicated in DCD Tier 2 Section 3.9.3.2.1, "Pressure Relief Devices Connected to the Pressurizer," the pressurizer POSRVs and their pilot operators are qualified to operate in saturated steam, water, and steam and water mixtures in hot or cold conditions. Preservice testing of the pressurizer POSRVs and MSSV includes testing as specified in Chapter 14. The testing and inspection requirements are in conformance with ASME OM Code and ASME Section XI including the recommendations of TMI Action Plan Item II.D.1 in 10 CFR 50.34(f)(2)(x). DCD Tier 2 Section 14.2.12.1.3, includes testing to verify the opening/closing pressure and opening time of the POSRVs. In addition, the POSRVs will be subjected to the IST program requirements as provided in DCD 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 SER.

DCD Tier 2 Section 5.2.1, "Compliance with Codes and Code Cases," addresses applicable ASME Section III Code Cases applied in the APR1400, as described in RG 1.84.

In addition, as discussed in Section 5.4.14, "Safety and Relief Valves," of this SER, DCD Tier 2, Figures 5.4.14-1, "Pilot Operated Safety Relief Valve Schematic Diagram," and 5.1.1-1, "Reactor Coolant System Schematic Flow Diagram," illustrate schematic representations of the POSRVs and RCS, respectively. DCD Tier 2 Table 5.4.10-1, "Pressurizer Design Parameters," provides the design parameters of the POSRVs. Open and closed indications of each POSRV are provided in accordance with the recommendations of TMI Action Plan Item II.D.3 in 10 CFR 50.34(f)(2)(xi). Valve leakage is monitored by resistance temperature detectors located on the discharge lines of each pressurizer POSRV and pilot valves. An abnormally high temperature in the discharge lines of the pressurizer POSRV and the pilot valves is an indication of valve leakage and alarmed in the main control room (MCR). Position indication for each pressurizer POSRV is also provided in the MCR.

The MSS presented in DCD Tier 2 Section 10.3, "Main Steam System," describes, in part, the overpressure protection provisions provided for the secondary side of the SG. DCD Tier 2, Sections 5.2.2 and 5.4.14 also provide design details of the MSSVs to provide overpressure protection. The staff reviewed the provisions 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 steamlines from each SG contain five MSSVs as described in DCD Tier 2 Section 10.3.2.2, "Component Description," and design data presented in DCD Tier 2 Table 10.3.2-1, "Main Steam System and Component Design Data," and DCD Tier 2 Table 5.4.14-2, "Main Steam Safety Valve Parameters." The MSSVs are designed in accordance with the ASME Code Section III (Class 2), Seismic Category I as shown in DCD Tier 1, Table 2.7.1-1, "Main Steam System Equipment and Piping Location/Characteristics."

Overpressure protection for the secondary side of the SGs and the main steamlines up to the inlet of the turbine stop valve is provided by the direct-acting, spring-loaded, carbon steel MSSVs. The MSS contains four main steam lines from the two SGs discharging to the main steam common headers. Five MSSVs are installed on each of the main steamlines upstream of the MSIV outside the containment. As the SG pressure rises and pressure setpoints are reached, the MSSVs open and discharge the high-pressure steam to the atmosphere. The

MSSV have staggered set pressures of 1174, 1205, and 1230 psig, in accordance with Article NC-7000 of ASME Section III. These valves are each sized to pass a steam flow of 430,913 kg/hr (950,000 lb/hr) at 92.83 kg/cm2A (1,320 psia), which is 110 percent of SG design pressure. A total of 20 MSSVs are provided for the four main steamlines with a combined relieving capacity greater than 8.62 x 10^6 kg/hr (19 x 10^6 lb/hr) which is sufficient to limit SG pressure to less than 110 percent of SG design pressure during worst-case transients, thereby meeting the SRP Section 5.2.2 acceptance criterion for compliance with the requirements of GDC 15.

Each main steamline also contains an MSIV to maintain tight shutoff against forward and reverse steam flow under its design condition. The applicant stated that the MSIVs are designed in accordance with the ASME Code Section III (Class 2), Seismic Category I and further discussed in Section 10.3 of this SER. Although not used for overpressure protection, the main steamline from each SG also contains one main steam atmospheric dump valve (MSADV) on each main steamline upstream of the MSSVs to allow cooldown of the RCS through a controlled discharge of steam to the atmosphere when the MSIVs are closed or when the main condenser is not available as a heat sink. The applicant stated that these MSADVs are ASME Section III (Class 2), Seismic Category I valves. Each valve is capable of holding the plant at hot standby, dissipating core decay and reactor coolant pump heat, and allowing controlled cooldown from hot standby to SCS initiation conditions in conjunction with auxiliary feedwater system. Section 10.3 of this SER contains the staff's evaluation of the capability of the MSIVs and MSADVs to perform their safety functions. ASME OM Code and ASME Section XI govern ISI and testing of the MSSV.

For the APR1400, the applicant stated that the SCS system is designed to provide sufficient pressure relief capacity to mitigate the most LTOP events during low temperature conditions. The SCS is a safety-related system that will be used to reduce the temperature of the RCS in post shutdown periods from the hot shutdown operating temperature to the refueling temperature, as described in DCD Tier 2 Section 5.4.7. The applicant stated that the LTOP is designed in accordance with BTP 5-2.

The SCS relief valves, located on the SCS suction lines, provide overpressure protection of the RCS during low-temperature conditions. One SCS liquid relief valve is provided in each of the two SCS pump suction lines and discharged to the IRWST. These valves are Seismic Category I and designed in accordance with the ASME Code Section III (Class 2). DCD Tier 2 Section 5.4.7 provides the system description and operation of the SCS. DCD Tier 2, Figure 5.4.7-2, "Shutdown Cooling System; One Train Cooldown," shows the SCS schematic flow diagram, DCD Tier 2 Figure 5.1.2-1 shows the RCS flow diagram, and DCD Tier 2 Figure 6.3.2-1, "Safety Injection/Shutdown Cooling System Flow Diagram," shows the safety injection system (SIS) flow diagram.

Each SCS suction line relief valve (SI-179 or SI-189) is designed to protect the RCS in a failure that initiates a pressure transient. The applicant stated that the use of either SCS suction line relief valve will provide sufficient pressure relief capacity to mitigate the most limiting LTOP events during low temperature conditions. The maximum pressure for low temperature overpressure protection is limited to 43.9 kg/cm2A (625 psia), 20 percent of RCS hydraulic test pressure (219.7 kg/cm2A [3,125 psia]), which is the maximum RCS pressure allowed under the minimum operating temperature required in Appendix G of ASME Section III.

The SCS suction line relief valves are independent of a loss-of-offsite power (LOOP). The applicant stated that each SCS suction line relief valve is a self-actuating, spring-loaded liquid relief valve that opens when the RCS pressure exceeds its setpoint, and does not require control circuitry. As indicated in DCD Tier 2 Table 5.2-3, “SCS Suction Line Relief Valve Valves (SI-179 and SI-189) Design Parameters,” the setpoint of the relief valves is 37.3 kg/cm^{2}G (530 psig) with a flow capacity of 29,337 L/min (7,750 gpm).

As indicated in DCD Tier 2 Section 5.2.2.2.2.2, “Provision for Overpressure Protection,” the LTOP pressure is defined to be the SCS suction line relief valve setpoint pressure adjusted to provide a margin to avoid lifting and to compensate for pressure measuring inaccuracies during plant normal operation. As indicated in DCD Tier 2 Section 5.2.2.2.2.1, “Limiting Transients,” the applicant stated that the most limiting transients are determined by the conservative analyses, which maximize mass and energy additions to the RCS. As described in APR1400-Z-M-NR-14008, Revision 0, “Pressure-Temperature Limits Methodology for RCS Heatup and Cooldown,” submitted December 23, 2014, the maximum pressure for LTOP is defined to be limited to 43.9 kg/cm2A (625 psia), 20 percent of RCS hydraulic test pressure of 219.7 kg/cm2A (3,125 psia), which is the maximum RCS pressure allowed under the minimum operating temperature required in Appendix G of ASME Code, Section XI. To determine the relief valve set-pressure, the following were considered: the maximum pressurizer pressure due to a transient of mass addition by an inadvertent safety injection; the actual elevation difference between the top of the pressurizer and LTOP relief valves; and the bottom pressure of the reactor vessel. The applicant stated that the relief valve set-pressure is determined to be lower than the LTOP limiting pressure of 43.9 kg/cm2A (625 psia). LTOP set-pressure derived from the above considerations is 37.3 kg/cm^{2}G (530 psig), as indicated in DCD Tier 2 Section 5.4.7. While the DCD indicates a conservative analysis was performed, the staff was unable to locate this analysis to determine the basis for the pressure setpoint and flow capacity for the SCS relief valves to demonstrate the limiting case (mass or energy addition) and which transient results in the highest pressure increase. The staff issued RAI 233-8244, Question 05.02.02-1 (ML15296A004), requesting the applicant to provide further information on LTOP. In its response to RAI 233-8244, Question 05.02.02-1 (ML15348A083, the applicant provided a description of the analysis method, discussion of analysis input assumptions, and analysis results. However, in items b and c of Question 05.02.02-1, the staff identified additional questions regarding the methodology for the analysis of the limiting events where LTOP is applied. Therefore, RAI 233-8244, Question 05.02.02-1 was closed as unresolved and the staff then issued a follow-up RAI 487-8609, Question 05.02.02-7 (ML16139A580), to address these issues related to methodology, computer codes, assumptions, and input parameters used in this analysis. In its response to RAI 487-8609, Question 05.02.02-7 (ML16139A580), the applicant provided additional information regarding the energy addition transient analysis methodology, assumptions, computer code, and input parameters designed to determine the limiting event in regards to overpressure protection. The applicant identified OVERP as the computer code used to provide the pressure response of the water-solid system of an energy addition transient. OVERP code, assumptions, initial conditions, and input data are described in WCAP-15688, “CE-NSSS LTOP Energy Addition Transient Analysis Methodology." The staff reviewed the response information with respect to the technical report and found it compatible and acceptable. The staff determined that the response is acceptable because the assumptions, input parameters, and initial conditions were adequately conservative to perform the LTOP energy addition transient analysis pressure response; therefore, RAI 487-8609, Question 05.02.02-7, is resolved and closed.

The SCS suction line relief valves, isolation valves, associated interlocks, and instrumentation are designed as Seismic Category I components and are addressed DCD Tier 2 Section 3.2.1, Table 3.2-1, and further discussed in DCD Tier 2 Section 5.4.7.2.4. The interlocks and instrumentation associated with the SCS suction line isolation valves are described in DCD

Tier 2 Section 5.4.7; Section 7.6.1, “System Description,” Section 7.6.1.1, “Shutdown Cooling System Suction Line Isolation Valve Interlocks,” and Table 7.6-1, “Shutdown Cooling System and Safety Injection Tank Interlock.” DCD Tier 2 Section 5.2.2.1.2.5, “Seismic Design and IEEE Standards 308 and 603 Criteria,” includes commitment to the interlocks satisfying the applicable portions of Institute of Electrical and Electronics Engineers (IEEE) Standard 279, “Criteria for Protection Systems for Nuclear Power Generating Stations,” IEEE Standard 308, “IEEE Standard Criteria for Class 1E Power Systems for Nuclear Power Generating Stations,” and IEEE Standard 603, “IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Stations.”

The applicant discussed the testing and inspections associated with the LTOP system. The applicant stated that the LTOP relief valves, SI-179 and SI-189, located on the SCS suction line are shop-tested, and that the relieving capacity of the valves is certified in accordance with ASME Section III, NC-7000. Additionally, the applicant stated that pre-service and in-service testing of the SCS suction line relief valves (SI-179 and SI-189) setpoint will be performed according to the ASME OM Code. The staff compared the applicant’s description of the testing and inspections associated with the LTOP system against the SRP acceptance criteria. Based on this comparison, the staff determined that the applicant’s description is consistent with SRP acceptance criteria associated with testing and inspections.

5.2.2.4.2 *Technical Report APR1400-Z-M-NR-14008*

The applicant stated that the reactor pressure vessel (RPV) and connected components of the RCPB are designed to withstand the effects of system pressure and temperature variations introduced by controlled heatup and cooldown operations, and operational transients for a specific reactor vessel fluence period or the effective full-power years (EFPYs) in accordance with the DCD TS 5.6.4, “Reactor Coolant System (RCS) Pressure and Temperature Limits Report (PTLR).” However, the RPV is considered the most critical component susceptible to non-ductile failure because of the neutron fluence experienced over the vessel lifetime. NRC Generic Letter (GL) 96-03, “Relocation of the Pressure Temperature Limit Curves and Low Temperature Overpressure Protections System Limits,” Criterion 3, dated January 31, 1996, states that the LTOP system lift setting limits developed using NRC-approved methodologies may be included in the PTLR. The detailed analytical methodology for developing the LTOP system limits is described in the PTLR and APR1400-Z-M-NR-14008.

The APR1400 LTOP mode of operation controls the RCS pressure at low temperatures, thereby ensuring compliance with 10 CFR Part 50, Appendix G, “Fracture Toughness Requirements,” and that the integrity of the RCPB is not compromised. The LTOP mode of operation for overpressure protection of the APR1400 RPV consists of two SCS suction line relief valves (SI-179 and SI-189). The applicant stated that each SCS suction line relief valve provides adequate relief capacity to prevent any pressure transient from exceeding the controlling Pressure-Temperature (P-T) Limit whenever operating below the LTOP disable temperature during heatup, and below the LTOP enable temperature during cooldown. The APR1400, DCD Tier 2 Sections 5.2.2 and 5.4.7 discuss the relief valve design with a summary of the design parameters in DCD Table 5.2-3. In addition, whenever the LTOP relief valves are used during LTOP mode of operation, Technical Specification LCO 3.4.11.a is applicable and requires that the suction relief valves’ lift settings are within the limits specified in the PTLR.

As described in APR1400 DCD Tier 2 Section 5.2.2.1, “Design Bases,” for low temperature operations, the set pressure for the SCS suction relief valves, is established based on the low-temperature pressure limit for the reactor vessel with respect to ASME Code, Section XI,

Appendix G, analyses. The APR1400, DCD Tier 2, Sections 5.2.2 and 5.3.2 discuss the analysis method to determine the heatup and cooldown pressure-temperature curves, as defined in DCD Tier 2 Figure 5.3-7, "Pressure-Temperature Limit Curve (60 years)," in accordance with Appendix G of 10 CFR Part 50 for normal operation of the RCS. However, the COL applicant that references the APR1400 DC will develop plant-specific pressure and temperature limit curves as addressed in the PTLR, consistent with an approved methodology. This action item is identified as COL 5.3(2) in APR1400 DCD Tier 2 Table 1.8-2, "Combined License Information Items." The pressure-temperature limits identified in ASME Code Section XI of Appendix G requires that the applicant's analytical results obtained are from an approved methodology equivalent to methods of analysis described in Appendix G and that the resulting limits are at least conservative as limits obtained by following the methods of analysis and the margins of safety described in Appendix G. Furthermore, whenever the P/T limit curves are revised, the SCS relief valves setpoint must be reevaluated to confirm the validity of the existing pressure setpoint, or a re-analysis to determine a new pressure setpoint of the relief valves based on the revised P/T limit curves. On this basis, the staff determined that the applicant met the provisions of GL 96-03 Criterion 3. The review and evaluation of APR1400-Z-M-NR-14008 is provided in Section 5.3.1, "Reactor Vessel Materials," Section 5.3.2 "Pressure-Temperature Limits, Upper-Shelf Energy, and Pressurized Thermal Shock" and Section 5.3.3 "Reactor Thermal Shock," of this SER report.

5.2.2.4.3 *Overpressure Protection at Power*

For the APR1400, the pressurizer is designed to maintain RCS operating pressure so that the minimum pressure during operating transients is above the setpoint for the SIAS and low pressure reactor trip and so that the maximum pressure is below the high pressure reactor trip setpoint. As described in DCD Tier 2 Section 5.2.2.1.1, the pressurizer POSRVs, MSSVs, and RPS are designed to maintain the RCS pressure below 110 percent of design pressure during the worst-case loss-of-load event with a delayed reactor trip.

During normal power operation, four POSRVs are connected to the top of the pressurizer by separate inlet lines and discharged to the IRWST through one common discharge line. The applicant stated that each pressurizer POSRV provides the overpressure protection function for the primary side of RCS. The portion of the RCPB that is exposed to significant pressure from the secondary side, the SG tubes, is protected from overpressure by the MSSVs. As stated in DCD Tier 2 Section 5.4.11.1, the IRWST is designed as primary heat sink for the POSRV discharge and to accommodate the hydrodynamic loads and the thermal effects in the IRWST due to the steam discharged from the pressurizer. The evaluation of the hydrodynamic loads on the IRWST and the pool temperature of the IRWST are described in DCD Tier 2 Section 6.8, "In-containment Water Storage System."

The primary side overpressure protection is provided by POSRVs designed to pass sufficient pressurizer steam to limit the RCS pressure to 110 percent of design pressure (193.3 kg/cm2A (2,750 psia)) following a loss of load with a delayed reactor trip, which is assumed to be initiated by the second generated safety grade signal from the RPS. The applicant stated that a delayed reactor trip is assumed on a high-pressurizer pressure signal.

The secondary safety valves or MSSVs are sized conservatively to release steam flow equal to the full power level. They are able to pass a total steam flow of 8.62 x 10^6 kg/hr (19 x 10^6 lb/hr) at 8,275 kPa (1,320 psia) to limit SG pressure to less than 110 percent of SG design pressure for design transients.

As indicated in DCD Tier 2 Section 5.2.2.2.1, the functional design evaluation of the overpressure protection system is included in DCD Tier 2 Section 15.2, “Decrease in Heat Removal by the Secondary System.” The evaluation demonstrates the adequacy of the overpressure protection system to maintain primary and secondary operating pressures within their respective pressure limits. The analytical model and assumptions used in the evaluation are discussed in Chapter 15 and were chosen to maximize the required pressure-relieving capacity of the primary and secondary sides. The analyses for the most severe anticipated transient demonstrates that sufficient relieving capacity is provided to prevent the pressure from exceeding 110 percent of the design pressure when acting in conjunction with the reactor protective system.

The staff reviewed DCD Tier 2 Section 15.2 evaluation of the functional design of the overpressure protection system. In DCD Tier 2 Section 15, the capability of the overpressure protection system to maintain secondary and primary operating pressures within 110 percent of design is demonstrated. The events analyzed demonstrate that sufficient relieving capacity has been provided so that when acting in conjunction with the RPS, the safety valves prevent exceeding 110 percent of the design pressure.

Loss of condenser vacuum (LOCV), resulting in turbine trip, is presented as the bounding AOO case for overpressure in DCD Tier 2 Section 15, “Transient and Accident Analyses,” and the results indicate the design is sufficient to remain below the 110 percent pressure requirements. As concluded in DCD Tier 2 Section 15.2 for the LOCV, the maximum RCS pressure remains below 193.34 kg/cm2A (2,750 psia), providing reasonable assurance of primary system integrity. The maximum SG pressure remains below 92.83 kg/cm2A (1,320 psia) providing reasonable assurance of secondary system integrity. The staff’s review of the initial conditions and assumptions established for the beginning of the LOCV event are included in Section 15.2 of this SER. The results of the LOCV analysis presented in DCD Tier 2 Section 15.2.3, “Loss of Condenser Vacuum,” show that the POSRVs can maintain pressure below the ASME limit. Although the pressure margin calculated is small, less than 0.34 kg/cm2A (5 psia), between the RCS design pressure and the ASME 110 percent limit, the peak pressure meets the ASME requirements. This event is discussed further in Section 15.2 of this SER.

The two transients that increase RCS inventory are presented in DCD Tier 2, Sections 15.5.1, “Inadvertent Operation of the Emergency Core Cooling System that Increase Reactor Coolant inventory,” and 15.5.2, “Chemical and Volume Control System Malfunction that Increases Reactor Coolant Inventory.” As shown in Section 15.5, “Increase in Reactor Coolant Inventory,” the transient results show that the maximum pressurizer pressure does reach the POSRV setpoint, but remains below 110 percent of RCS design pressure. Further review of these transients is found in Section 15.5 of this SER.

5.2.2.4.4 *Low Temperature Overpressure Protection*

Guidelines in SRP Section 5.2.2 state that the system for overpressure protection during low-temperature phases of plant operation should be designed in accordance with the requirements of BTP 5-2.

As indicated in DCD Tier 2 Section 5.2.2, the LTOP for the APR1400 is designed in accordance with the guidance provided in BTP 5-2. Two SCS suction relief valves are designed for use during LTOP conditions, and other low pressure systems connected to the RCS are not used. The use of either SCS suction line relief valve provides sufficient pressure relief capacity to mitigate the most limiting LTOP events during low temperature conditions. For temperatures above the LTOP temperature, overpressure protection is provided by the pressurizer POSRVs, which have a set pressure of 173.7 kg/cm2A (2,470 psia).

In the LTOP mode, each SCS suction line relief valve is designed to protect the reactor vessel given a single failure, in addition to the event that initiates the pressure transient. The event initiating the pressure transient is considered to result from either an operator error or equipment malfunction. The SCS suction line relief valve is independent of a LOOP and opens when the RCS pressure exceeds its setpoint.

During heatup, the RCS pressure is maintained below the LTOP pressure until the RCS cold leg temperature exceeds the LTOP disable temperature. During cooldown, the RCS pressure is maintained below the LTOP pressure once the RCS cold leg temperature reaches the LTOP enable temperature. The use of either SCS suction line relief valve provides sufficient pressure relief capacity to mitigate the most limiting LTOP events during low temperature conditions. Alignment of the SCS suction line relief valve to the RCS is specified by plant procedures to provide RCS overpressure protection for all temperatures below the temperature for which LTOP is required. The LTOP pressure is defined as the SCS suction line relief valve set-pressure adjusted to provide a margin to avoid lifting and to compensate for pressure measurement inaccuracies during plant normal operation.

Whenever the SCS suction line relief valves are aligned with the RCS to provide LTOP, an increase in RCS pressure above the maximum SCS alignment pressure of 31.6 kg/cm2A (450 psia) will cause an LTOP transient alarm in the MCR to alert the operator that a pressure transient is occurring. If the SCS is not aligned to the RCS before the cold leg temperature is decreased below the LTOP enable temperature, an alarm will notify the operator to open the SCS suction line isolation valves. Operator actions taken in response to an LTOP transient alarm are described in DCD Tier 2 Section 5.4.7.

APR1400-Z-M-NR-14008, Revision 0, “P-T Limits Methodology for RCS Heatup and Cooldown,” contains additional information regarding development of the LTOP set-pressure, as well as enable and disable temperatures. The applicant developed LTOP enable temperature, for LTOP system initiation, by following the BTP 5-2 guidance. According to the BTP 5-2 guidance, the LTOP system should be operable during startup and shutdown conditions below the enable temperature, defined as the water temperature corresponding to a metal temperature of at least RT (NDT) + 50 ºC (90 ºF) at the beltline location (1/4t or 3/4t) that is controlling in the Appendix G limit calculations. LTOP set-pressure is defined as 37.3 kg/cm^{2}G (530 psig), based on analyzing the most limiting transient which maximize mass and energy additions to the RCS. Conforming with BTP 5-2 guidance, APR1400 has issued APR1400-Z-M-NR-14008 that defines the LTOP enable and disable temperatures as 101.7 °C (215 °F) and 136.1 °C (277 °F), respectively. This technical report is reviewed by the NRC Materials and Chemical Branch (MCB) within Sections 5.3.1, 5.3.2, and 5.3.3. This DCD Tier 2 Section 5.2.2.2.2.1 indicates a conservative analysis was 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 APR1400 over the range of expected conditions. The staff was unable to find the referenced analysis demonstrating the adequacy of the LTOP design. The staff issued RAI 233-8244, Question 05.02.02-1 (ML15296A004), requesting the applicant to provide additional justification of the LTOP design and review of the analysis referenced in DCD.

In its response to RAI 233-8244, Question 05.02.02-1 (ML15348A083), the applicant provided a discussion of the analysis of mass and heat limiting events where LTOP is applied. However, the discussion did not include sufficient information regarding the methodology used in the analysis of the LTOP limiting events. The staff issued RAI 487-8609, Question 05.02.02-7 (ML16139A580), requesting the applicant to address assumptions, evaluation model, analysis methodology, computer codes, and input parameters. In its response to RAI 487-8609, Question 05.02.02-7 (ML16161B237), the applicant provided additional information regarding the energy addition transient analysis methodology, assumptions, computer code, and input parameters designed to determine the limiting event in regards to overpressure protection. The applicant identified OVERP as the computer code used to provide the pressure response of the water-solid system of an energy addition transient. OVERP code, assumptions, initial conditions, and input data are described in WCAP-15688, "CE-NSSS LTOP Energy Addition Transient Analysis Methodology." The staff reviewed the response information with respect to the technical report and found it compatible and acceptable. The staff determined that the response is acceptable because the assumptions, input parameters, and initial conditions were adequately conservative to perform the LTOP energy addition transient analysis pressure response; therefore, RAI 487-8609, Question 05.02.02-7, is resolved and closed.

As stated in DCD Tier 2 Section 5.2.2.2.2.2, the LTOP conditions described above are within the SCS operating range. Section 3.4.11, "Low Temperature Overpressure Protection (LTOP) System," in Chapter 16, "Technical Specifications," TS requires the SCS suction line isolation valves to be open when operating in the LTOP mode. This TS also provides assurance that appropriate action is taken if one or more SCS suction line relief valves are out of service during the LTOP mode of operation.

5.2.2.4.5 *Initial Testing Program*

DCD Tier 2 Section 14.2.12.1.3, describes initial testing of the POSRVs to verify the opening and closing pressure and opening time of the POSRVs. DCD Tier 2 Section 5.2.2.10, "Testing and Inspection," indicates preservice testing of the POSRVs and MSSVs includes testing as specified in Chapter 14, "Verification Programs," and provides a list of verification testing to be performed for preservice and IST of the pressurizer POSRVs. DCD Tier 2 Section 14.2.12.1.3 tests contains POSRV tests to verify the opening and closing pressure (setpoints) and opening response time. Compared to Chapter 14, the Section 5.2.2.10 verification testing seems to contain additional preservice testing (setpoint, response time, leak test, position indicator, closing time, etc.). The staff issued RAI 8244, Question 05.02.02-5 (ML15296A004), requesting the applicant to address the difference between the pre-service testing requirements and Chapter 14 test.

In its response to RAI 233-8244, Question 05.02.02-5 (ML15348A083), the applicant stated that it is in the process of upgrading DCD Tier 2 Section 14.2 in an effort to "focus on adding additional SSCs that are important to safety and risk significant as well as increasing the level of detail described in the DCD for test prerequisites, test methods and acceptance criteria for the various tests including POSRV tests." The staff will evaluate the following tests related to the POSRVs in Chapter 14 of this SER.

- DCD Tier 2 Section 14.2.12.1.2, "Reactor Coolant System Test."
- DCD Tier 2 Section 14.2.12.1.3, "Pressurizer Pilot-Operated Safety Relief Valve Test."
- DCD Tier 2 Section 14.2.12.4.5, "Turbine Trip Test."
- DCD Tier 2 Section 14.2.12.4.6, "Unit Load Rejection Test."

Initial testing of the MSSVs is provided in DCD Tier 2 Section 14.2.12.1.63, “Main Steam Safety Valve Test.” The purpose of this test is to verify the popping pressure of the MSSVs during hot functional testing. In addition, the test will also verify all safety valves have no seat leakage.

Initial testing of the complete SCS is provided in DCD Tier 2 Section 14.2.12.1.20, “Shutdown Cooling System Test.” This testing includes verification of the LTOP relief valves setpoints.

The staff review of these tests is documented in Chapter 14 of this SER.

5.2.2.4.6 *ITAAC*

The ITAAC associated with RCS overpressure protection is given in DCD Tier 1 Section 2.4.1.

The ITAAC associated with MSSV, as it relates to the overpressure protection features for the SG secondary side are given in DCD Tier 1 Section 2.7.1.2, “Inspection, Tests, Analyses, and Acceptance Criteria.”

The staff reviewed both DCD Tier 1, Sections 2.4.1 and 2.7.1.2 and determined 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 operational parameters, among other things. However, ITAAC Item 9.a of Tier 1 Table 2.4.1-4, “Reactor Coolant System ITAAC,” includes an acceptance criteria indicating “maximum opening time (including dead time) of the POSRVs is 0.5/5 seconds (hydraulic/manual). Maximum closing time (including dead time) of the POSRVs is 0.9/9 seconds (hydraulic/manual).” While opening time of 0.5 seconds is consistent with DCD Tier 2 Section 5 and TS for the POSRV design, the staff was unable to locate any details of the manual actuation response time. The staff issued RAI 233-8244, Question 05.02.02-3 (ML15296A004), requesting the applicant to provide bases and assumptions for the five-second and nine-second manual actuation response times.

In its response to RAI 233-8244, Question 05.02.02-3 (ML15348A083), the applicant stated that the POSRV manual actuation response times are based on the motor size related to the required torque and thrust for the manual operation during beyond DBE of a total loss of feedwater. The staff determined that this is acceptable because the response clarifies the bases for the response times. Therefore, RAI 233-8244, Question 05.02.02-3, is resolved and closed.

The SCS ITAAC are given in DCD Tier 1 Section 2.4.4 and include verification of the SCS suction line relief valves provide sufficient relief capacity.

No additional ITAAC items were identified by the staff.

5.2.2.4.7 *Technical Specifications*

The staff reviewed TS 3.4.10 and 3.4.11, “Low Temperature Overpressure Protection (LTOP) System,” for conditions applicable to the overpressure protection and determined that they are adequate to meet the overpressure protection for the primary side of the RCPB, as described in Chapter 16, “Technical Specifications,” of this SER. In addition, the staff also reviewed the TS Bases Sections B 3.4.10, “Pressure Pilot Operated Safety Relief Valves (POSRVs),” and B 3.4.11, “Low Temperature Overpressure Protection (LTOP) System,” and agrees with the bases.

APR1400 TS 3.4.10, "Pressurizer Pilot Operated Safety Relief Valves (POSRVs)," requires that a licensee have four POSRVs operable, and that each POSRV have lift setting of ≥ 171.1 kg/cm^{2}A (2,433 psia) and ≤ 176.3 kg/cm^{2}A (2,507 psia) when in Modes 1, 2, and 3 and in Mode 4 with all RCS cold leg temperature greater than the LTOP enable temperature specified in the PTLR. The opening time of pressurizer POSRV shall also be operable within 0.5 seconds, including dead time.

APR1400 TS 3.4.11, "Low Temperature Overpressure Protection (LTOP) System," requires that LTOP be established when in Mode 4 (when any RCS cold leg temperature is less than or equal to the LTOP enable temperature specified in the PTLR), Mode 5, and Mode 6 (when the reactor vessel head is on). TS Bases B3.4.10 and B3.4.11 provide the bases for TS 3.4.10 and 3.4.11.

TS 3.7.1 requires the MSSVs to be operable for Modes 1, 2, and 3. In the event that one or more required MSSVs are inoperable, TS action requires reduction in maximum power and variable overpower trip setpoint, as indicated in the TS.

5.2.2.5 *Combined License Information Items*

There are no COL information items associated with Section 5.2.2 of the APR1400 DCD.

5.2.2.6 *Conclusion*

The overpressurization protection of the APR1400 was reviewed and evaluated by the staff. The scope of the review included the design bases specification, system description and performance, inspections and tests, and instrumentation. The review included the applicant's referenced technical report. The staff finds that the overpressurization protection system is acceptable and satisfies the intent of the SRP and BTP 5-2 acceptance criteria regarding compliance with GDC 15, and 31, 10 CFR 52.47(a)(8), 10 CFR 50.34(f)(2)(x), and 10 CFR 50.34(f)(2)(xi).

5.2.3 Reactor Coolant Pressure Boundary Materials

5.2.3.1 *Summary of Application*

DCD Tier 2 Section 5.2.3, describes the materials used to fabricate the RCPB. The DCD provides information about material specifications; compatibility with reactor coolant; fabrication and processing of ferritic materials; fabrication and processing of austenitic materials; prevention of primary water stress-corrosion cracking (PWSCC) for nickel-based alloys; and threaded fasteners primarily as these topics pertain to the RCPB. Each of these topics are discussed below.

The APR1400 design follows RG 1.84, "Design, Fabrication and Materials Code Case Acceptability, ASME Section III," Revision 36, issued August 2014; RG 1.44, "Control of the Processing and Use of Stainless Steel," Revision 1, issued March 2011; RG 1.36, "Nonmetallic Thermal Insulation for Austenitic Stainless Steel," issued February 1973; RG 1.50, "Control of Preheat Temperature for Welding of Low-Alloy Steel," Revision 1, issued March 2011; RG 1.43, "Control of Stainless Steel Weld Cladding of Low-Alloy Steel Components," Revision 1, issued March 2011; RG 1.71, "Welder Qualification for Area of Limited Accessibility," Revision 1, issued March 2007; RG 1.28, "Quality Assurance Program Criteria (Design and Construction)," Revision 4, issued June 2010; RG 1.31, "Control of Ferrite Content in Stainless Steel Weld Metal," Revision 4, issued October 2013; and, RG 1.65, "Materials and Inspections for Reactor Vessel Closure Studs," Revision 1, issued April 2010.

Material Specification

DCD Tier 2 Table 5.2-2, lists the pressure-retaining materials and material specifications for the RCPB components. This list includes the RPV, control element drive mechanism (CEDM) component, pressurizer, SG, RCPs, piping, piping nozzles, safe ends, valves, and associated weld materials. All these materials must meet the applicable material requirements of ASME Code, Section III and the applicable ASME Code, Section II material specifications or ASME Code Cases as permitted or approved by the NRC. Any deviations to DCD Tier 2 Table 5.2-2 or use of ASME Code Cases are to be addressed by the COL applicant.

Compatibility with Reactor Coolant

RCS water chemistry is specified to minimize corrosion and is shown in DCD Tier 2, Tables 5.2-5 through 5.2-8. The applicant also provides an extensive description of the RCS chemistry values and controls, the action levels and the diagnostic parameters in accordance to the recommendations of the latest version of the Electric Power Research Institute (EPRI) PWR Primary Water Chemistry Guidelines.

The materials used in the RCPB, including materials that do not act as a pressure boundary, consist of austenitic wrought and cast stainless steel; nickel-based alloys; carbon and low-alloy steels; martensitic stainless steel; and precipitation-hardened stainless steels. The materials of construction used in the RCPB were selected for compatibility with the reactor coolant. All of the construction materials were selected to be resistant to stress corrosion cracking (SCC) in the PWR environment. General corrosion of all materials is expected to be within acceptable limits. The applicant limited the extent of the corrosion of ferritic low-alloy steels and carbon steels in contact with the reactor coolant in the design by cladding all such material with stainless steel or nickel-chromium-iron cladding.

In addition, materials of construction are compatible with reactor coolant through conformance with RG 1.44, "Control of the Processing and Use of Stainless Steel," use of Alloy 690, restriction of cobalt content, and restriction of Inconel X-750 to spiral wound gaskets.

The compatibility of external insulation and environmental atmosphere are established through use of metallic insulation and RG 1.36, "Nonmetallic Thermal Insulation for Austenitic Stainless Steel," compliant non-metallic insulation.

Fabrication and Processing of Ferritic Materials

Fracture toughness requirements for ASME Code Class 1 ferritic materials used for the RCPB components are established in accordance the requirements in ASME Code, Section III and NRC SRP BTP 5-3, "Fracture Toughness Requirements." Welding controls for ferritic materials are to be consistent with RG 1.50. All cladding is to be applied consistent with RG 1.43. Welders are to be qualified consistent with RG 1.71. Nondestructive examination (NDE) requirements are to be consistent with ASME Code, Section III requirements.

Fabrication and Processing of Austenitic Stainless Steel

Austenitic stainless steels are to be fabricated to avoid sensitization to SCC through adherence to RG 1.44. Per RG 1.44, American Society for Testing and Materials (ASTM) A262 Practice A or E is to be used to demonstrate freedom from sensitization in fabricated, unstabilized stainless steel. All raw austenitic stainless steel material to be used in fabrication of components in the RCPB is to be supplied in the annealed condition as specified by pertinent ASME Code requirements. Completed and partially fabricated components are not solution heat treated; rather the extent of chromium carbide precipitation is to be controlled.

Welding procedures are based on testing on stainless steel mockups to establish procedures that do not produce sensitized structures in unstabilized Type 300 stainless steels. Confirmation of this is provided by application of ASTM A262 Practices A or E. The primary parameters from this testing are: weld heat input less than 23.6 kJ/cm, interpass temperature 176.7 °C (350 °F) maximum, and carbon content 0.065 percent maximum. Welds produced under these conditions are considered to be adequate for service coupled with the oxygen content limits on the reactor coolant chemistry.

The unstabilized stainless steel to be used in APR1400 consists of Type 304 and Type 316 material. These materials are not to be exposed to temperatures from 427 °C (800 °F) to 816 °C (1500 °F) to prevent sensitization. In addition the ferrite content and temperature exposure of cast stainless steels and stainless steel weld filler metals are controlled to prevent sensitization and other degradation. The fracture toughness of cast stainless steels is to be ensured by control of ferrite content based on the operating temperature of the component and the specific type of cast steel.

Cleaning and contamination protection are to be established consistent with RG 1.28, "Quality Assurance Program Criteria (Design and Construction)." The cleanness classification of RCPB components is to be based on a graded approach in accord with ASME NQA-1, Part III, Subpart 3.2, Appendix 2.1.

Cold-worked austenitic stainless steel, except for bolting or pin materials, is not to be used for RCPB components. The COL applicant is to submit the actual, as-procured yield strength of austenitic stainless steel materials to the NRC per COL 5.2(7).

NDE of austenitic stainless steel tubular products are to be carried out in accordance with ASME Code, Section III, Subarticle NB-2500 during construction and ASME Code, Section XI during ISI.

Prevention of PWSCC for Nickel-Based Alloys

Prevention of PWSCC of nickel-based alloys is to be ensured through the use of Alloy 690, Alloys 52/52M, and Alloy 152.

Threaded Fasteners

Threaded fasteners used for RCPB components are to be fabricated consistent with ASME Code Section III, Subsection NB requirements. Reactor vessel closure studs are to meet NRC RG 1.65 as well. Material specifications for threaded fasteners for other pressure retaining parts of Class 1 components are listed along with fracture toughness requirements per ASME Code, Section III, Subsubarticle NB-2330 as pertinent. Actual fracture toughness test results are to be provided to the staff at a predetermined time.

5.2.3.2 *Regulatory Basis*

The staff reviewed DCD Tier 2 Section 5.2.3, in accordance with SRP Section 5.2.3, "Reactor Coolant Pressure Boundary Materials," Revision 3. The materials specifications; compatibility of materials with the reactor coolant; fabrication and processing of ferritic materials; and fabrication and processing of austenitic stainless steel within the RCPB are acceptable if they meet the relevant requirements set forth in 10 CFR 50.55a, “Codes and Standards;” GDC 1, “Quality Standards and Records;” GDC 4, “Environmental and dynamic effects design bases;” GDC 14, “Reactor coolant pressure boundary;” GDC 30, “Quality of reactor coolant pressure boundary;” and GDC 31, “Fracture Prevention of Reactor Coolant Pressure Boundary;” Appendix B to 10 CFR Part 50, “Quality Assurance for Nuclear Power Plants and Fuel Processes Plants;” and Appendix G, “Fracture Toughness Requirements,” to 10 CFR Part 50. These requirements are discussed below:

- Compliance with GDC 1 and 10 CFR 50.55a requires that SSCs be designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with the importance of the safety function to be performed.

- Compliance with GDC 4 requires that SSCs important to safety be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operations, maintenance, testing, and postulated accidents, including loss-of-coolant accident (LOCAs).

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

- Compliance with GDC 30 requires that components of the RCPB be designed, fabricated, erected, and tested to the highest quality standards practical.

- Compliance with GDC 31 requires that the RCPB be designed with sufficient margin to ensure that when stressed under operating, maintenance, testing, and postulated accident conditions: (1) the boundary behaves in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized. The design shall reflect consideration of service temperatures and other conditions of the boundary material under operating, maintenance, testing, and postulated accident conditions and the uncertainties in determining (1) material properties, (2) the effects of irradiation on material properties, (3) residual, steady state and transient stresses, and (4) size of flaws.

- Compliance with Appendix B to 10 CFR Part 50 requires, in Criterion XIII, that measures be established to control the cleaning of material and equipment to prevent damage or deterioration.

- Compliance with Appendix G to 10 CFR Part 50 requires that the fracture toughness of RCPB ferritic materials be tested in accordance with the requirements of the ASME Code and that the pressure-retaining components of the RCPB that are made of ferritic materials meet requirements for fracture toughness during system hydrostatic tests and any condition of normal operation, including AOOs.

5.2.3.3 *Technical Evaluation*

As discussed below, the staff divided its evaluation of DCD Tier 2 Section 5.2.3 into six topics mapped to those described in SRP Section 5.2.3 and the DCD. These topics are material specifications; compatibility of the materials with the reactor coolant; fabrication and processing of ferritic materials; processing of austenitic stainless steel; prevention of PWSCC in nickel-based alloys; and threaded fasteners.

5.2.3.3.1 *Material 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 the requirements of GDC 1 and 30 and 10 CFR 50.55a, as they relate to quality standards for design, fabrication, erection, and testing. These requirements are met for material specifications by complying with the appropriate provisions of the ASME Code and by applying the ASME Code Cases identified in RG 1.84.

The staff reviewed DCD Tier 2 Section 5.2.3.1 to determine the suitability of the RCPB materials for this application. The staff determined that the applicant's material specifications listed in DCD Tier 2 Section 5.2.3 and Table 5.2-2 for the APR1400 design conform with the appropriate provisions of the ASME Code, RG 1.84 and other staff guidance.

The staff determined that the applicant's selection of materials for use in the RCPB meets the requirements of the ASME Code or the guidance of RG 1.84 and is therefore, acceptable.

5.2.3.3.2 *Compatibility of Materials with the Reactor Coolant*

The RCPB materials of construction that are in contact with the reactor coolant, contaminants, or radiolytic products must be compatible and must meet the requirements of GDC 4, as they relate to the compatibility of components with environmental conditions. The staff reviewed DCD Tier 2 Section 5.2.3.2, "Compatibility with Reactor Coolant," to determine whether compatibility was adequately considered. The applicant stated that it considered the compatibility of the materials of construction used in the RCPB with the reactor coolant, contaminants, or radiolytic products to which the RCPB is exposed.

The applicant specified that all ferritic low alloy and carbon steel surfaces that come into contact with reactor coolant are to be clad with stainless steel or nickel-chromium-iron cladding. The staff confirmed that underclad cracking should be minimized or absent as the applicant will be using SA-508, Grade 3, Class 1 material, which is resistant to underclad cracking. The staff accepts the cladding approach as an adequate measure ensuring material compatibility with the reactor coolant as the cladding is compatible with the coolant and will isolate the base material from the coolant.

Joints between austenitic safe ends and low alloy or carbon steel nozzles are to be made from Alloy 690, 52, 52M, and/or 152. These materials have demonstrated high resistance to degradation, both in service and in testing, particularly PWSCC. The staff accepts the use of these materials for the above stated use as they have been reviewed in detail for this purpose in prior reviews by the staff and are in routine comparable use within the operating fleet.

The applicant stated that cobalt is restricted to as low a level as practicable in materials that are in contact with reactor coolant and that are in stainless steel or nickel-based alloy components. The usage locations of cobalt are noted as being the CEDM motor assembly pins, link, and latch; and hard-facing for valve components. Inconel X-750 use for pressure retaining components in the RCPB has been restricted to spiral wound gaskets for the primary manways of SGs and the pressurizer. The applicant justified the adequacy of this component based on operating experience in OPR1000 plants. Furthermore, the gasket is replaced each time the manway is opened. The staff considered this a sufficient basis for the adequacy of the subject gaskets. The staff determined that limiting the use of cobalt and X-750 as described, is acceptable as the use of cobalt and X-750 is highly restricted thus limiting the amount of readily activatable material.

DCD Tier 2 Section 5.2.3.2.3, specifies that all metallic insulation used in the plant is stainless steel reflective and that this minimizes insulation contamination in the event of chemical solution spillage. All nonmetallic insulation is required to meet RG 1.36. The staff accepts that conformance with RG 1.36 provides reasonable assurance that non-metallic insulation will not adversely increase the potential for SCC of stainless steel due to the leaching of chloride or fluoride from the non-metallic insulation.

Based on the discussion above, the staff determined that the applicant adequately considered the compatibility of RCPB materials with the reactor coolant and insulation materials with RCPB materials.

5.2.3.3.3 *Fabrication and Processing of Ferritic Materials*

The fracture toughness of ferritic materials in the RCPB must meet the requirements of Appendix G to 10 CFR Part 50. These criteria satisfy the requirements of GDC 14 and GDC 31 regarding prevention of fracture of the RCPB.

Appendix G to 10 CFR Part 50 requires the pressure-retaining components of the RCPB be made of ferritic materials to meet the requirements for fracture toughness during system hydrostatic tests and any condition of normal operation, including AOOs. For piping, pumps, and valves, this requirement is met through compliance with the requirements of ASME Code, Section III, Paragraph NB-2331 or Paragraph NB-2332, and the C_v values specified in Table NB-2332(a)-1, "Required C_V Values for Piping, Pumps, and Valves." Materials for bolting must meet the impact test requirements of ASME Code, Section III, Paragraph NB-2333. Calibration of temperature instruments and C_v impact test machines must meet the requirements of ASME Code, Section III, Subsubarticle NB-2360. The staff reviewed DCD Tier 2 Section 5.2.3.3, "Fabrication and Processing of Ferritic Materials," and verified that the APR1400 design meets the aforementioned requirements regarding fracture toughness of RCPB piping, components, and bolting and equipment calibration. Section 5.3 of this SER presents the staff's evaluation of the fracture toughness requirements of the RPV.

Control of ferritic steel welding by following NRC RGs and adhering to the ASME Code satisfies the quality standards requirements of GDC 1 and GDC 30 and 10 CFR 50.55a. Adherence to the guidance provided in RG 1.50, RG 1.43, RG 1.34, "Control of Electroslag Weld Properties," RG 1.71, and ASME Code non-mandatory Appendix D, "Non-mandatory Preheat Procedures," satisfies the aforementioned quality standard requirements. The staff confirmed that these RGs and ASME Code non-mandatory Appendix D are appropriately required in the DCD consistent with SRP 5.2.3.

DCD Tier 2 Section 5.2.3.3 discusses the use of RG 1.50 and preheat requirements when welding low alloy steel in the APR1400 design. Low alloy steel is used only in the RPV and feedwater piping. RG 1.50 provides guidance that all low alloy steel welds be maintained at the minimum preheat temperature until postweld heat treatment is performed. The staff determined that this portion of the application conforms to RG 1.50 and is therefore acceptable.

DCD Tier 2 Section 5.2.3.3 states that electroslag welding is not used for any RCPB components. Therefore, RG 1.34 does not apply to the APR1400 design.

In DCD Tier 2 Section 5.2.3.3, the applicant stated that the APR1400 design commits to RG 1.71 regarding welder qualification for areas of limited accessibility. The staff determined that this portion of the application is acceptable because the applicant conforms to RG 1.71.

For NDE of ferritic steel and austenitic stainless steel tubular products, compliance with the applicable provisions of the ASME Code meets the requirements of GDC 1 and GDC 30 and 10 CFR 50.55a regarding quality standards. ASME Code, Section III, Subsubarticles NB-2550 through NB-2570 are the applicable provisions of ASME Code, Section III. The applicant stated that NDE of ferritic stainless steel tubular products for components are to be carried out in accordance with ASME Code, Section III. Based on compliance with ASME Code, Section III, the staff determined that this is acceptable.

5.2.3.3.4 *Fabrication and Processing of Austenitic Stainless Steel*

All stages of component manufacturing and reactor construction must include process control techniques, in accordance with the requirements of GDC 1, as it relates to nondestructive testing (i.e., examination) to quality standards; GDC 4; GDC 30 and Criterion XIII, "Handling, Storing, and Shipping," of Appendix B to 10 CFR Part 50. These requirements prevent severe sensitization of the material by minimizing exposure of stainless steel to contaminants that could lead to SCC and reduce the likelihood of component degradation or failure through contaminants.

The staff reviewed DCD Tier 2 Section 5.2.3.4 to confirm that austenitic components of the RCPB are: (1) compatible with environmental conditions to avoid sensitization and SCC, (2) have appropriate controls on welding and material preservation, and (3) receive appropriate NDE.

The DCD indicates that all austenitic stainless steels are supplied in the annealed condition as specified by the pertinent ASME Code and are to be treated consistent with RG 1.44 including use of ASTM A262 Practice A or E tests to confirm and ensure proper heat treatment. The applicant stated that welding procedures were extensively tested on mockups, fabricated using production techniques, to select only the procedures and/or practices demonstrated not to produce a sensitized structure. ASTM A262, Practice A or E is used as the standard for acceptability consistent with RG 1.44. Based on this testing the applicant stated that a weld heat input of less than 23.6 kJ/cm, maximum interpass temperature of 176.7 °C (350 °F), and a maximum carbon content of 0.065 percent would produce adequate non-sensitized results. While the staff cannot determine whether the use of the weld heat input, maximum interpass temperature, and maximum carbon content specified by the applicant will ensure non-sensitization of the material, that staff determined that this is acceptable as conformance with RG 1.44 through the use of ASTM A262 Practice A or E will ensure adequate results.

The DCD also specified that except for bolting or pin materials, no cold-worked austenitic stainless steel is used for components of the RCPB. In addition, COL item, 5.2(7), requires the COL Applicant to submit the actual as-procured yield strength of the austenitic stainless steel used in the RCPB. The staff determined that this is acceptable because the applicant will submit the actual procured yield strength per COL 5.2(7).

The applicant stated that its acceptance criteria for cleaning and cleanliness controls meet the intent of RG 1.28, Revision 4. The staff requested that the applicant clarify how components would be classified with regards to ASME Code NQA-1 cleanliness classifications. The applicant revised the DCD to state that the cleanliness classifications of RCPB components, and hence the NQA-1 cleaning requirements, would be established through use of NQA-1, Part III, Subpart 3.2, Appendix 2.1. The staff concluded that by adhering to RG 1.28 and hence NQA-1; and the non mandatory NQA-1, Part III, Subpart 3.2, Appendix 2.1; that the applicant has adequately defined cleaning and contamination protection requirements.

DCD Tier 2, Sections 5.2.3.4.1.c and 5.2.3.4.5 detail controls on ferrite content of cast austenitic stainless steel or welds. The applicant noted that ferrite content is to be calculated using Hull's equivalent factor. The limits on ferrite content are consistent with the staff approved requirements found in NRC License Renewal Issue No. 98-0030, "Thermal Aging Embrittlement of Cast Austenitic Stainless Steel Components," dated May 19, 2000.

DCD Tier 2 Section 5.2.3.4.4 details the controls on welding. The applicant listed adherence to RGs 1.31 with ferrite limits, 1.34, and 1.71. The ferrite limits are consistent with the staff position stated in License Renewal Issue 98-0030. The staff determined that the application is consistent with these RGs and, with the addition of the ferrite limits discussed above, is acceptable with regards to welding austenitic stainless steel.

For NDE of austenitic stainless steel tubular products, the applicant complied with the requirements of GDC 1 and GDC 30 and 10 CFR 50.55a regarding quality standards by specifying the appropriate provisions of the ASME Code, which are in Section III, Subsection NB-2500. The staff determined that the NDE will be acceptable based on compliance with ASME Code, Section III.

Based on the above the staff have determined that the applicant has adequately addressed the requirements of GDC 1, 4, 30, and Criterion XIII Appendix B to 10 CFR Part 50 as they pertain to fabrication and processing of austenitic stainless steel for RCPB materials.

5.2.3.3.5 *Prevention of PWSCC for Nickel-Based Alloys*

DCD Tier 2 Section 5.2.3.5, states that thermally treated Alloy 690 and Alloys 52/52M and 152 weld metals are to be used for the APR1400 design. The applicant cited EPRI Report MRP-111, "Materials Reliability Program: Resistance to Primary Water Stress Corrosion Cracking of Alloys 690, 52, and 152 in Pressurized Water Reactors," issued March 2004, and operating experience as supporting the selection of these materials as highly resistant to PWSCC. The applicant further stated that a boric acid corrosion prevention program or ISI program is applied to provide reasonable assurance of the integrity of Alloy 690 base and weld metals. The staff concurred with the applicant's assessment and determined that adequate measures have been taken, as described above, to protect the nickel-based alloy RCPB components from PWSCC.

5.2.3.3.6 *Threaded Fasteners*

DCD Tier 2 Section 5.2.3.6, expands on the discussion in DCD Tier 2 Section 3.13 regarding threaded fasteners in the context of RCPB applications. The staff confirmed that the information provided in DCD Tier 2 Section 5.2.3.6 is consistent with DCD Tier 2 Section 3.13. The staff evaluation of this information is found in Section 3.13 of this SER.

5.2.3.3.7 *Reactor Coolant Chemistry*

It is essential that coolant purity be maintained carefully, since many dissolved species can enhance material corrosion or otherwise damage internal components. One way of meeting this is by having RCS chemical parameters that are consistent with the limiting values presented in the EPRI PWR Primary Water Chemistry Guidelines (here after "EPRI Guidelines"). Although the staff does not formally review or issue a safety evaluation for the various EPRI water chemistry guidelines, these guidelines are recognized as representing the industries' best practices in water chemistry control. Extensive experience in operating reactors has demonstrated that following the EPRI Guidelines minimizes the occurrence of corrosion-related failures. Further, the EPRI Guidelines are periodically revised to reflect evolving knowledge with respect to best practices in chemistry control. Therefore the staff accepts the use of the latest version of the EPRI guidelines as an acceptable basis for reactor coolant chemistry considerations (for example as stated in SRP Section 9.4.3).

The applicant addressed the EPRI guidelines through two COL items. These COL items ensure that the COL applicant references the latest edition of the EPRI guidelines (COL 5.2(5)) and provides threshold values and operator actions for primary water chemistry that are in compliance with the latest version of the EPRI Guidelines (COL 9.3(7)). These COL Items ensure the APR1400 primary water chemistry will conform to the latest EPRI Guidelines and thus will follow industry best practices, and the staff therefore determined that the APR1400 RCS water chemistry values, are acceptable.

EPRI Guidelines also specify certain water chemistry concentrations as control parameters, which require strict adherence to limits in order to achieve material protection. In APR1400 DCD Tier 2 Section 5.2.3.2.1, "Reactor Coolant Chemistry," the applicant provided a detailed discussion of the control parameters for dissolved oxygen, ammonia, lithium, dissolved hydrogen, fluoride and sulfate. For each of these chemical species, except Li, the applicant provided limiting values equal to, or more stringent than, the Action Level 2 values from the EPRI Guidelines. (For chloride and fluoride ion concentrations, these values are also mentioned in RG 1.44 as strict limits.) For Li concentration, no exact limits are provided in the EPRI Guidelines, as this component is determined by the pH control. The EPRI Guidelines only state Action Level 1 limits for H_2 and O_2, and DCD Tier 2 Table 5.2-8, "Reactor Coolant Detailed Power Operations Specifications," standard values are consistent with, or more stringent than, these limits. The EPRI Guidelines stipulate sampling three times/week for all but sulfate (once/week) and dissolved O (as stipulated in plant TS). However, they note that sampling frequencies may vary and that they will be determined in the plant TS. The staff concluded that the applicant has provided appropriate limits for the RCS water chemistry control parameters since the limits are the same, or more stringent that the limits recommended by the EPRI Guidelines.

EPRI Guidelines specify certain water chemistry parameters as "diagnostic", which do not have mandated limits, but which should nevertheless be monitored as they provide an additional level of protection from corrosion, radiation protection and other failures. These are listed as conductivity, silica, and suspended solids. The applicant stated the following concerning these parameters:

- For conductivity, the measurement will only be used as an auxiliary measurement to assess general ionic activity. The staff determined that this is acceptable as there is recommended limit in the EPRI Guidelines and is mostly site specific.

- For suspended solids the applicant will use a standard value of 350 ppb. The EPRI Guidelines state that normal operational values are typically < 10 ppb, but recognize that this value varies widely and cannot be mandated. However, in Section 4.2.3, "Parameters with Negligible Impact on Structural Integrity," of the EPRI Guidelines, suspended solids are classified as a parameter having negligible effect on RCPB or fuel cladding integrity. Further, Table 3.8, "Reactor Coolant System Startup Chemistry Diagnostic Parameters (Following Fill-and-Vent to Reactor Critical)," of the EPRI Guidelines recommends suspended solids be less than 350 ppb prior to reactor criticality. Based on the above, the staff determined that the proposed limit for suspended solids, is acceptable.

- For silica the applicant recommended a value of 1 ppm, which is consistent with the EPRI Guidelines. The EPRI Guidelines mention that no deposits have been observed if Si is below 1 ppm; they suggest a plant-specific target of 3 ppm. It should be observed that this value is not mandated, but a good practice for better operation. Based on the above, the staff determined that the proposed limit for Si, is acceptable.

DCD Tier 2 Table 5.2-7, "Reactor Coolant Detailed Plant Startup Operation Specifications," states that the standard value for pH at 25 °C (77 °F) is between 4.6 and 7.3. The applicant provided a description of its pH control program that complies with the EPRI Guidelines. The applicant also stated that during prior to initial criticality, the boron will range from near zero to 4,400 ppm, and lithium from 0 to 3.5 ppm. This will give a pH range of 4.2 to 10.7. During operation boron will range from near zero at end of cycle to 4,400 ppm at refueling. Lithium, the alkalizing additive employed, will vary from 0.2-3.5 ppm. In addition the applicant agreed to rewrite Table 5.2-79.3.4-1B. The staff finds this acceptable as it conforms to the EPRI Guidelines.

The staff considers consistency with the EPRI Guidelines an acceptable method of ensuring GDC 14 will be met for the RCPB, since the EPRI Guidelines are recognized as representing industry best practice in water chemistry control. Based on the applicant's commitments to the EPRI Guidelines, the staff concludes that the methods for controlling water chemistry in the DCD are acceptable.

5.2.3.4 *Combined License Information Items*

Table 5.2.3-1 lists the item numbers and descriptions from Table 1.8-2 of the DCD.

Table 5.2.3-1 Combined License Information Items Identified in the DCD

Item No.	Description	Section
5.2(4)	The COL applicant is to address the material specifications, which are not shown in Table 5.2-2, as necessary.	5.2.3.1
5.2(5)	The COL applicant will review and confirm at the time of COL submittal based on the latest revision of EPRI PWR Primary Side Water Chemistry Guidelines.	5.2.3.2.1
5.2(6)	The COL applicant is to address the actual, as-procured, fracture toughness data of the RCPB materials to the staff at a predetermined time by an appropriate method.	5.2.3.3
5.2(7)	The COL applicant is to submit the actual, as-procured yield strength of the austenitic stainless steel materials used in RCPB to the staff at a predetermined time agreed-upon by the regulatory body.	5.2.3.4.3

The staff deems COL 5.2(4), 5.2(5), 5.2(6), and 5.2(7) appropriate.

5.2.3.5 *Conclusions*

The staff concludes that the RCPB materials are acceptable in accordance with the requirements of GDC 1, 4, 14, 30, and 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. Title 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 ASME Section III and ASME Section XI, as well as the Operations and Maintenance Code (ASME OM Code). ASME Section XI defines, for each component Code Class, the specific ISI and testing requirements (e.g., methodology, periodicity, acceptance criteria). ISI includes a preservice inspection (PSI) prior to initial plant startup. The term “testing”, as it is commonly used in this SER Section, refers only to system pressure testing required by ASME Section XI. IST of APR1400 components such as pumps, valves, and dynamic restraints, as required by the ASME OM Code, is addressed in Section 3.9 of this SER.

5.2.4.2 *Summary of Application*

DCD Tier 1: There are no DCD Tier 1 entries for this area of review. The system-based descriptions of DCD Tier 1 Chapter 2, “Design Descriptions and ITAAC,” address ASME design-related Code requirements for system components.

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

The application stated that accessibility to equipment for maintenance, testing, and inspection is a basic element of the APR1400 design process. Systems and components are designed such that design, materials, and geometry do not restrict the performance of inspection required by ASME Section XI. Provisions are made in the plant design for the removal and storage of all vessel internals during ISI. General provisions are made for removable insulation, removable shielding, installation of handling machinery, adequate personnel, and equipment access space, and laydown space for all temporarily removed or serviced components. Storage space for the removable insulation panels is also provided. Working room for personnel is provided adjacent to each weld in order to examine all piping system welds manually. With the internals removed, the entire inner surface of the reactor vessel is accessible for surface and volumetric inspection. With the internals in place, the reactor vessel nozzle-to-shell welds and the inner radii of the reactor vessel outlet nozzles are accessible using remote automated equipment. An access tunnel is provided to allow examination of the bottom head. Primary coolant piping is designed to provide access to both sides of the welds. Manways are provided to allow access to the SG, pressurizer, and other components as needed.

The application addresses: (1) general description of the system boundary subject to inspection; (2) arrangement of systems and components to provide accessibility; (3) examination categories and methods (e.g., visual, liquid penetrant, magnetic particle, eddy current, ultrasonic, radiography); (4) inspection intervals; (5) evaluation of examination results; and (6) system pressure tests; (7) Code exemptions; (8) Code cases; (9) other inspection programs; and the (10) PSI and testing program. In each of these areas, the application references the applicable ASME Code requirements. The application also provided six combined operating license (COL) information items related to the PSI and ISI program.

ITAAC: The ITAAC associated with DCD Tier 2 Section 5.2.4 are given several sections of DCD Tier 1. These ITAAC indicate that inspections will be performed on as-built components and piping, and that reports exist that conclude the following:

- As-built ASME Code components, piping, and supports are designed and constructed in accordance with ASME Section III requirements.
- The ASME Section III requirements are met for non-destructive examination of the pressure boundary welds in as-built ASME Code components and piping.
- The results of hydrostatic testing of the as-built ASME Code components and piping conform to ASME Section III requirements.

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

5.2.4.3 *Regulatory Basis*

The relevant requirements of the NRC regulations for this area of review and the associated acceptance criteria are given in SRP Section 5.2.4, "Reactor Coolant Pressure Boundary Inservice Inspection and Testing," and are summarized below. SRP Section 5.2.4 includes review interfaces with other SRP sections.

- GDC 32, "Inspection of Reactor Coolant Pressure Boundary," as it relates to periodic inspection and testing of the RCPB.
- Title 10 CFR 50.55a, as it relates to the requirements for inspecting and testing ASME BPV Code Class 1 components of the RCPB as specified in ASME Section XI.
- ASME Code Case N-729-1, as required by 10 CFR 50.55a(g)(6)(ii)(D) for reactor vessel head inspection.

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

- RG 1.26, as it relates to the quality group classification of components.
- RG 1.147, as it relates to ASME Section XI Code Cases acceptable for use.
- GL 88-05, "Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components in PWR Plants," dated March 17, 1988, as it relates to the establishment of a program to detect and correct potential RCPB corrosion caused by boric acid leaks.

NRC Bulletin 2003-02, "Leakage from Reactor Pressure Vessel Lower Head Penetrations and Reactor Coolant Pressure Boundary Integrity," dated August 21, 2003, as it relates to providing additional measures to inspect PWR RPV lower heads that contain penetrations.

5.2.4.4 *Technical Evaluation*

The staff reviewed APR1400 DCD, DCD Tier 2 Section 5.2.4, using SRP Section 5.2.4. The ASME BPV Code of record for the APR1400 is the 2007 edition with the 2008 Addenda (see

DCD Tier 2 Section 5.2.1.1.). Subject to the conditions of 10 CFR 50.55a, ASME Section III, Subsection NB presents the construction requirements for Class 1 components, and ASME Section XI, Subsection IWB, presents the PSI and ISI requirements.

5.2.4.4.1 *Inservice Inspection and Testing Program*

DCD Tier 2 Section 5.2.4.1 states that the ISI and testing program for Quality Group A components of the RCPB conforms to the regulatory requirements of 10 CFR 50.55a and GDC 32.

The applicant defined Quality Group A components of the RCPB as ASME BPV Code Class 1 components. This is consistent with DCD Tier 2 Section 5.2.1.1, which stated that RCPB components are designed and fabricated as ASME Code Class 1, except for the components that meet the exclusion requirements of 10 CFR 50.55a(c). The staff's review to verify that systems and components are appropriately classified in accordance with regulatory requirements and NRC quality group classification guidance, including verification that Quality Group A components meet the requirements for ASME BPV Code Class 1 components, is documented in SER Section 3.2.2, "System Quality Group Classification," and SER Section 5.2.1.1, "Conformance with 10 CFR 50.55a."

DCD Tier 2 Section 5.2.4.1 also states that the ISI and IST programs for the APR1400 consist of the following there subprograms:

a. The component inspection program, which includes nondestructive inspection of major components, piping systems, and support systems.

b. The pump and valve IST program.

c. The hydrostatic testing program.

The following sections of the SER document the staff's review of the ISI and testing program. As stated in the introduction, the staff's review of the pump and valve IST program is documented in Section 3.9, "Mechanical Systems and Components," of this SER.

<u>System Boundary Subject to Inspection</u>. The definition of the system boundary subject to inspection is acceptable if it is in agreement with the definition of the RCPB provided in 10 CFR 50.2. Per 10 CFR 50.2, the RCPB includes all pressure-retaining components of boiling and PWRs, such as pressure vessels, piping, pumps, and valves, which are part of the RCS, or connected to the RCS, up to and including any and all of the following: (a) the outermost containment isolation valve in system piping that penetrates the primary reactor containment; (b) the second of two valves normally closed during normal reactor operation in system piping that does not penetrate primary reactor containment; (c) the RCS safety and relief valves. The examination requirements of ASME Section XI, Subsection IWB, apply to all Class 1 pressure retaining components and their welded attachments.

DCD Tier 2 Section 5.2.4.1.1 states that the RPV, pressurizer, primary side of the SG, and associated piping, pumps, valves, bolting, and component supports are subject to inspection. The applicant further stated that all ASME Code Class 1 pressure-retaining components are subject to inspection. The staff confirmed that the applicant appropriately enumerated all components subject to inspection compliant with 10 CFR 50.55a and ASME Section XI.

Arrangement of Systems and Components to Provide Accessibility. The design and arrangement of system components are acceptable if an adequate clearance is provided in accordance with ASME 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 III or XI of the ASME Code of record.

DCD Section 5.2.4.1.2 states that the layout and arrangement of the APR1400 plant provides adequate working space and access for inspection, maintenance, and repair of the Class 1 components of the RCPB in accordance with ASME Section XI, Subarticle IWA-1500. The applicant also stated that all Class 1 components shall be designed for and provided with access to enable the performance of ASME Section XI inspections in the installed condition. The applicant also described the provisions provided in the APR1400 design to allow access for to perform the required examinations of the RPV, reactor coolant piping, RCPs, the pressurizer, the SGs, and other RCPB components. In addition, the application states that provisions are made for removable insulation, removable shielding, the installation of handling machinery, adequate personnel and equipment access space, and laydown space for all temporarily removed or serviced components. Storage space for the removable insulation panels is also provided as well as working room adjacent to each piping system weld to allow for manual examination. Relevant to this, DCD Tier 2 Section 5.2.4.1, states that dissimilar metal welds and austenitic welds in piping will be examined from both sides. It is further stated that when ultrasonic examination from both sides is not possible, then single-sided ultrasonic examination will be performed in accordance with ASME Section XI, Appendix VIII. The staff determined that the applicant’s approach is acceptable because it is in accordance with ASME Section XI.

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

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

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

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

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

In DCD Tier 2 Section 5.2.4.1.3, “Examination Categories and Methods,” the applicant stated that the visual, surface, and volumetric examination techniques and procedures agree with the requirements of Section XI, Subarticle IWA-2200, Article IWB-2000, and Table IWB-2500-1. Surface examinations techniques used for the APR1400 include the liquid penetrant, magnetic particle, or eddy current methods, while volumetric examinations are performed using either ultrasonic or radiographic testing. The applicant also stated that the categories and requirements appropriate for each examination area are consistent with those categories and requirements specified in ASME Section XI, Table IWB-2500-1. In addition, PSI and subsequent ISI are conducted with equivalent equipment and techniques. The staff determined that the information described above is acceptable because it meets the requirements of ASME Section XI and 10 CFR 50.55a.

DCD Tier 2 Section 5.2.1.1 indicated that the baseline ASME BPV Code used for the APR1400 DC is the 2007 edition with the 2008 addenda of ASME, Section XI. This edition and addenda of ASME Section XI require 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, RPV welds, and RPV closure studs. In DCD Tier 2 Section 5.2.4.1.3, the applicant stated that the methods, procedures and requirements for qualification of personnel performing ultrasonic examination are in accordance with the requirements of ASME Section XI, Appendix VII. The applicant also stated that the performance demonstration for ultrasonic examination procedures, equipment, and personnel used to detect and size flaws is in accordance with the requirements of ASME Section XI, Appendix VIII. The staff determined that the information described above is acceptable because it meets the requirements of ASME Section XI and 10 CFR 50.55a.

In DCD Tier 2 Section 5.2.4.1.3, “Examination Categories and Methods,” personnel performing visual, liquid penetrant, magnetic particle, eddy current, or radiographic examinations as part of the PSI or ISI program are to be qualified in accordance with ASME Section XI, Subarticle IWA-2300. The staff determined that these personnel qualifications are acceptable because they are in compliance with the requirements of ASME Section XI.

<u>Inspection Intervals</u>. The required examinations and pressure tests must be completed during each 10-year interval of service, hereinafter designated as the inspection interval. In addition, the scheduling of the program must comply with the provisions of ASME Section XI, Article IWA-2000, “Examination and Inspection.” As stated in DCD Tier 2 Section 5.2.4.1.4, “Inspection Intervals,” the inspection interval for the APR1400 is defined as 10 years. The applicant also stated that the inspection period may be reduced or extended by as much as one year to enable an inspection to coincide with a plant outage. This is acceptable because the inspection interval for the APR1400 is in compliance with ASME Section XI and 10 CFR 50.55a, and ASME Section XI allows an inspection period to be reduced or extended by as much as one year to enable an inspection to coincide with a plant outage.

<u>Evaluation of Examination Results</u>. The standards for evaluation of examination results are acceptable if they are in accordance with the requirements of ASME Section XI, Article IWB-3000. DCD Tier 2 Section 5.2.4.1.5, “Evaluation of Examination Results,” states that the evaluation of examination results for Class 1 components is conducted in accordance with ASME Section XI, Articles IWA-3000 and IWB-3000. This is acceptable because it meets the applicable requirements of ASME Section XI.

The proposed program regarding repair or replacement of components containing defects is acceptable if the program is in accordance with the requirements of ASME Section XI, Article

IWA-4000. The criteria that establish the need for repair or replacement are described in ASME Section XI, Article IWB-3000. DCD Tier 2 Section 5.2.4.1.5 states that unacceptable indications are repaired in accordance with ASME Section XI, Article IWA-4000 and that the criteria for establishing the need for repair or replacement are in accordance with ASME Section XI, Article IWB-3000. This is acceptable because it meets the applicable requirements of ASME Section XI.

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

DCD Tier 2 Section 5.2.4.1.6, “System Pressure Tests,” states that the leakage and hydrostatic pressure tests of the RCPB Code Class 1 components are conducted in accordance with the requirements of ASME Section XI, Article IWB-5000 and the TS requirements for operating limits during heat-up, cooldown, and system hydrostatic testing. Since the applicant’s methodology for performing pressure testing of the Class 1 boundary and components meets the requirements of the ASME Code, the methodology for performing system pressure testing is, therefore, acceptable.

Code Exemptions. Exemptions from Code examinations should be permitted if the criteria in ASME Section XI, Subsubarticle IWB-1220, “Components Exempt from Examination,” are met. In DCD Tier 2 Section 5.2.4.1.7, “Code Exemptions,” the applicant stated that there are no ASME Code Class 1 components (or portions of components) to be exempted from ASME Section XI, Subarticle IWB-2500 examination requirements, except for the items allowed by ASME Section XI, Subsubarticle IWB-1220. The staff determined that the noted exemption class is acceptable because it is in compliance with the requirements of ASME Section XI.

Code Cases. The applicable ASME Code Cases pertinent to DCD Tier 2 Section 5.2.4, are listed in DCD Subsection 5.2.4.18 as conditioned in RG 1.147, Revision 17, “Inservice Inspection Code Case Acceptability ASME Section XI, Division 1.” The staff finds the use of these Code Cases acceptable as they have been incorporated by reference in 10 CFR 50.55a.

Augmented ISI to Protect Against Postulated Piping Failures. DCD Tier 2 Section 5.2.4.1.1 stated that high energy system piping between containment isolation valves receives an augmented ISI as described in DCD Tier 2 Section 6.6.8, “Augmented ISI Protect against Postulated Piping Failure.” Therefore, the staff’s evaluation of the augmented ISI to protect against postulated piping failures is documented in Section 6.6, “Inservice Inspection and Testing of Class 2 and 3 Components,” of this SER.

Other Inspection Programs. For PWR plants, the applicant must establish an inspection program to detect and correct potential RCPB corrosion caused by boric acid leaks as described in NRC Generic Letter (GL) 88-05. DCD Tier 2 Section 5.2.4.1.9, “Other Inspection Program,” states that the boric acid corrosion (BAC) program includes the selection of locations of degradation caused by small leakage, identification of small leakage locations, implementation methods of inspection and evaluation, and corrective action procedures for preventing recurrences of leakage. The DCD further describes the design features of the APR1400 that enable effective boric acid leak detections through insulation design. The staff determined that the applicant’s response is acceptable because it provides a reasonable assurance that the boric acid corrosion program will be effectively implemented on the APR1400 plant.

Operating experience has shown that bottom-mounted instrumentation (BMI) nozzles on PWRs are susceptible to cracking and leakage. In 2003, the NRC issued Bulletin 2003-02 to advise PWR licensees that current methods of inspecting the RPV lower heads may need to be supplemented with additional measures (e.g., bare metal visual inspections) to detect RCPB leakage. In DCD Tier 2 Section 5.1, "Summary Description" and Section 5.3.3, "Reactor Vessel Integrity," the applicant stated that the lower head contains 61 in-core instrumentation nozzle penetrations. However, the materials used for the APR1400 reactor bottom head penetrations and their welds are Alloy 690 and Alloy 52/152 which are very resistant to SCC in a PWR environment. The staff determined that the use of Alloy 690 and Alloy 52/152 is acceptable because the use of improved materials of construction precludes the need to perform inspections in addition to those required by ASME Section XI.

5.2.4.4.2 *Preservice Inspection Program*

DCD Tier 2 Section 5.2.4.2, "Preservice Inspection and Testing Program," states that the preservice examination program is in accordance with the requirements of ASME Section III, Subsubarticle NB-5280 and that the PSI program conforms with the edition and addenda of ASME Section XI, as required by 10 CFR 50.55(b) and 10 CFR 50.55a(g)(3). Title 10 CFR 50.55a(b)(2) provides conditions on the use of the 1970 edition through the 1976 Winter Addenda and the 1977 edition through the 2007edition with the 2008 Addenda of ASME Code, Section XI, Division 1. Title 10 CFR 50.55a(g)(3) places a limit on the ASME Section XI editions and Addenda that may be used for the PSI program. Specifically, 10 CFR 50.55a(g)(3) requires ASME Code Classes 1, 2, and 3 components to meet the PSI requirements applied to the construction of the particular component or a later edition incorporated by reference into 10 CFR 50.55a. The staff determined that the applicant appropriately cited the necessary requirements for the preservice examination program in compliance with 10 CFR 50.55a, as it pertains to which edition and addenda of ASME Section XI applies.

5.2.4.5 *Combined License Information Items*

Table 5.2.4-1 provides a list of ISI and testing of the RCPB related COL information item numbers and descriptions from DCD Tier 2 Table 1.8-2:

Table 5.2.4-1 APR1400 Combined License Information Items

Item No.	Description	Section
5.2(8)	The COL applicant is to provide and develop the implementation milestones for the ISI and testing program for the RCPB, in accordance with ASME Code Section XI and 10 CFR 50.55a.	5.2.4.1
5.2(9)	The COL applicant is to address the provisions to accessibility of Class 1 components for ISI if the design of the APR1400 Class 1 component is changed from the DCD design.	5.2.4.1.2
5.2(10)	The COL applicant is to provide the list of Code exemptions in the ISI program of the specific plants, if it exists.	5.2.4.1.7
5.2(11)	The COL applicant is to prepare and implement a BAC prevention program compliant with Generic Letter 88-05.	5.2.4.1.9
5.2(12)	The COL applicant is to prepare the PSI and testing program.	5.2.4.2

5.3(5)	The COL applicant is to provide and develop the ISI and testing program for the RCPB in accordance with ASME Section XI and 10 CFR 50.55a (COL 5.3(4))	5.3.3.7

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

5.2.4.6 *Conclusion*

The design of the RCPB incorporates provisions for access to enable the performance of ISI examinations in accordance with 10 CFR 50.55a(g)(3) and the 2007 edition with the 2008 Addenda of ASME Section XI. The final ISI program is required to meet the latest ASME 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 PSI and ISI plan. The periodic inspections and pressure testing of pressure-retaining components of the RCPB 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 PSI and ISI program required by the ASME Code constitutes an acceptable basis for satisfying, in part, the requirements of GDC 32.

The staff concluded the description of the PSI and 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 Section XI, Division 1, "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 Reactor Coolant Pressure Boundary (RCPB) Leakage Detection

5.2.5.1 *Introduction*

The APR1400 design includes the RCPB leakage detection systems that provide a means for detecting and, to the extent practical, identifying the source of reactor coolant leakage and monitoring leaks from the reactor coolant and associated systems. Leakage detection systems, including monitoring of containment sump level and flow monitor, airborne particulate radioactivity monitor, and atmosphere humidity monitoring system, are included in the plant TS. Additional methods used to diversely indicate the leakage inside containment include, but are not limited to, RCS inventory methods, containment pressure and temperature monitoring, and acoustic leak monitoring system.

5.2.5.2 *Summary of Application*

The DCD describes RCPB leakage detection as follows:

DCD Tier 2: DCD Tier 2 Section 5.2.5, "Reactor Coolant Pressure Boundary Leakage Detection," provides information on the RCPB leakage detection system including detection methods, readout instrumentation in the MCR, maximum allowable total leakage, intersystem leakage, sensitivity and response time, operability testing and calibration, and applicable limits for reactor coolant leakage rates. DCD Tier 2 Section 5.2.5.7 lists COL information item COL 5.2(15) to address the procedures for operator responses to prolonged low-level leakage. DCD

Tier 2, Section 3.6.3.2.3.2, “Leakage Detection Systems (LDS),” Section 3.6.3.3.3, “Stability Analysis Acceptance Criteria,” and Section 3.6.3.5.2.1, “Leak Detection System,” describe the leak detection system to support leak-before-break (LBB) analysis procedures.

DCD Tier 1 and ITAAC: DCD Tier 1 Section 2.4.7, “Leakage Detection System,” indicates that Table 2.4.7-1, “Leakage Detection System ITAAC,” describes the ITAAC for the RCPB leakage detection system.

TS: DCD Tier 2 Chapter 16, “Technical Specifications” requires in LCO 3.4.12 to verify RCS operational leakage within specified limits, and in LCO 3.4.14 to address RCS leakage detection instrument requirement.

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 NUREG-0800, Section 5.2.5, “Reactor Coolant Pressure Boundary Leakage Detection,” and are summarized below.

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

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

Acceptance criteria 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, “Guidance on Monitoring to Reactor Coolant System Leakage.”

5.2.5.4 *Technical Evaluation*

The staff reviewed the RCPB leakage detection systems described in the DCD, in accordance with NUREG-0800, Section 5.2.5.

The staff’s evaluation included the following areas: leakage detection capability, sensitivity, and response time; leakage detection systems; leakage instrumentation in the control room; separation of identified and unidentified leakage; intersystem leakage; plant TSs; initial testing program; ITAAC; seismic qualification; and COL information items.

GDC 30 requires that the components, which are part of the RCPB, be 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. SRP Section 5.2.5 states that, for GDC 30, the review of RCPB leakage detection is based on meeting the guidance in RG 1.45. Therefore, staff reviewed the following items in accordance with RG 1.45.

5.2.5.4.1 *Leakage Detection Capability, Sensitivity, and Response Time*

RG 1.45 states that the capability of the leakage monitoring system includes overall response time, detector sensitivity and accuracy. The instrument should be able to detect leakage of one gallon per minute (gpm) within an hour. In addition, if an LBB approach is used, the overall capability of the leakage monitoring system should be sufficient to support the LBB analysis procedures. SRP Section 3.6.3, "Leak-Before-Break Evaluation Procedures," recommends that a safety margin of 10 on the predicted leakage rate be required for determining the leakage size flaw for LBB analysis.

DCD Tier 2 Section 5.2.5.1.1 states that the sensitivity and response time of leakage detection equipment for unidentified leakage is such that a leakage rate, or its equivalent, of 0.5 gpm can be detected in less than one hour. In DCD Tier 2 Section 3.6.3.2.3.2, "Leakage Detection Systems (LDS)," Section 3.6.3.3.3, "Stability Analysis Acceptance Criteria," and Section 3.6.3.5.2.1, "Leak Detection System," the applicant described the leak detection system to support LBB. The LBB evaluations are based on a leak detection capability of 0.5 gpm within one hour, which is more stringent than the "one gpm within one hour" criterion established in RG 1.45, because a safety margin of 10 as specified in SRP Section 3.6.3 for the LBB analyses is applied. The staff's review of the LBB analyses is documented in Section 3.6.3 of this SER.

Based on the above DCD information, the staff concluded that APR1400 RCS leakage detection capability of 0.5 gpm in less than one hour satisfies RG 1.45 and supports LBB evaluations. Moreover, the COL applicant is to verify the sensitivity and response times by performing tests in the "Initial Testing Program," and the "Inspections, Tests, Analyses and Acceptance Criteria (ITAAC)", which will be discussed in Sections 5.2.5.4.9 and 5.2.5.4.10 of the following evaluation.

5.2.5.4.2 *Leakage Detection System*

RG 1.45 states that plant TSs should identify at least two independent and diverse instruments and/or methods that have the quantitative detection and monitoring capabilities to detect and monitor RCS leakage. In addition, the plant should have additional diverse systems to qualitatively detect and monitor RCS leakage.

In DCD Tier 2 Section 5.2.5.1.1, the applicant identified the following leakage detection methods: containment sump level, containment airborne particulate radioactivity, and containment atmosphere humidity. DCD Tier 2 Section 5.2.5.1.1.2, "Sump Level and Flow Method," and Section 5.2.5.1.1.3, "Containment Air Particulate Radioactivity Monitoring," state that the containment sump level and flow method and the containment air particulate monitoring method both can detect 0.5 gpm within one hour. Further, TS LCO 3.4.12, "RCS Operational Leakage," specifies the leakage limits of 0.5 gpm for unidentified leakage. TS LCO 3.4.14, "RCS Leakage Detection Instrumentation," identified containment sump level monitor, containment atmosphere radioactivity (particulate) monitor, and containment atmosphere humidity monitor to detect and monitor RCS leakage. In addition, DCD Tier 2

Section 5.2.5.1.1.4, “Other Methods,” states that consistent with RG 1.45, containment pressure, temperature, humidity monitoring, and an acoustic leak monitoring system are used as diverse qualitative indications for RCS leakage detection.

The staff reviewed the above information against the guidance in RG 1.45 where, in a section relating to “Detector Response Time,” it stated that the functional requirements for leakage monitoring systems should include the detector response time. Further it stated that plants should use multiple instrument locations to ensure that the transport delay time of the leakage effluent from its source to the detector will yield an acceptable overall system response time. The staff issued RAI 80-8040, Question 05.02.05-1 (ML15295A317), requesting the applicant to provide the following information to verify consistency with the above guidance.

a) Clarify whether containment atmosphere humidity has the capability to detect RCS leakage of 0.5 gpm within one hour and provide the correlation between the RCS leakage and containment atmosphere humidity with respect to time to demonstrate the capability. How many humidity detectors are there in the containment and where are they located in the containment?

b) DCD Tier 2 Section 5.2.2.1.1.3, states that containment air particulate monitoring can detect RCS leakage of 0.5 gpm within one hour. How many containment air particulate monitoring detectors are there in the containment and where are there located? Provide the correlation between the RCS leakage and containment air particulate radiation level with respect to time to demonstrate the capability of 0.5 gpm within one hour. Explain the assumption being used for the primary coolant radioactivity concentration to derive the correlation. It should be noted that RG 1.45 states that the analysis of the capabilities of leakage monitoring systems that measure radioactivity should use a realistic primary coolant radioactivity concentration assumption consistent with plant normal operations (as opposed to the maximum concentration permitted by TSs or used in accident analysis).

In its response to RAI 80-8040, Question 05.02.05-1 (ML15243A074), the applicant responded to Part (a) of the RAI, the applicant clarified the humidity detectors in a markup of DCD stating that “the humidity sensor is designed to indicate a sudden and significant increase of humidity level by annunciating an alarm in the MCR due to unidentified leakage.” The accuracy and sensitivity of the humidity sensor used in the APR1400 design is not sufficient enough to establish a quantitative correlation between the unidentified RCS leakage and the humidity sensor to detect the level of detail of 0.5 gpm within one hour. There are four humidity sensors being located in different locations of the containment.

The staff determined that the above response to Part (a) of RAI 80-8040, Question 05.02.05-1, is acceptable because it adequately clarified the capability of the humidity detectors and that the detectors are not relied upon to quantitatively detect 0.5 gpm leakage within an hour. Instead, it is designed to promptly indicate a sudden and significant increase of humidity level by setting off an alarm in the MCR due to unidentified leakage. RG 1.45 states that the RCPB leakage detection system should have at least two independent and diverse instruments and/or methods that have the quantitative detection and monitoring capabilities detailed above. In addition, the plant should have other diverse systems to qualitatively detect and monitor RCS leakage. DCD Tier 2, Sections 5.2.5.1.1.2 and 5.2.5.1.1.3 state that containment sump level method and containment air particulate monitoring can both detect 0.5 gpm within one hour. Therefore, the guidance on “two independent and diverse instruments and/or methods that have the quantitative detection monitoring capabilities” is satisfied by the containment sump level method and containment air particulate monitoring. The humidity detector satisfies the guidance on the additional diverse systems to qualitatively detect and monitor RCS leakage.

In its response to Part (b) of RAI 80-8040, Question 05.02.05-1 (ML15243A074), the applicant stated that there are two radiation monitors sensors being located in two different locations of the containment. The applicant provided the correlation between the RCS leakage and containment air particulate radiation level and demonstrated the detection capability of 0.5 gpm within one hour. The applicant assumed the reactor coolant "expected" radioactivity concentration based on ANSI/ANS-18.1-1999, entitled "Radioactive Source Terms for Normal Operation of Light Water Reactors." The staff determined that the above response to Part (b) of RAI 80-8040, Question 05.02.05-1, is acceptable because it adequately addressed the containment air particulate detectors with respect to the number and location of the detectors, the correlation and capability, and the assumption on the primary coolant radioactivity concentration.

Based on the above, the staff determined that the APR1400 design satisfies RG 1.45 because at least two independent RCS leakage detection methods identified in the plant TSs can detect leakage of 0.5 gpm within one hour. In addition, there are sufficient diverse methods to monitor RCS leakage qualitatively. The staff confirmed that the DCD was revised as committed in the response to RAI 80-8040, Question 05.02.05-1. Therefore, RAI 80-8040, Question 05.02.05-1, is resolved and closed.

5.2.5.4.3 *Leakage Instrumentation in the Main Control Room*

Guidance (i.e., SRP Section 5.2.5 and RG 1.45) states that the plant should provide in the MCR the output and alarms from RCS leakage monitoring systems. Procedures for converting the instrument output to a leakage rate should be readily available to the operators. Periodic calibration and testing of leakage monitoring systems should take place.

DCD Tier 2 Section 5.2.5.2, "Leakage Instrumentation in the Main Control Room," provides the relevant information. DCD Tier 2 Section 5.2.5.2.1 indicates that leakage indication of pilot-operated safety relief valve is continuously monitored and provides an alarm in the control room. DCD Tier 2 Section 5.2.5.2.2, "Primary Indicators of Reactor Coolant Unidentified Leakage," indicates that the following primary RCS leakage detection instruments have indications and alarms in the MCR: containment air radiation (particulate), pumped flow from containment drain sump and reactor cavity sump, levels of containment drain sump and reactor cavity sump and acoustic leakage monitoring. DCD Tier 2 Section 5.2.5.2.3, "Other Indicators of Reactor Coolant Leakage," lists control room leakage instrumentation that may indicate significant reactor coolant leakage. DCD Tier 2 Section 5.2.5.2.4, "Leakage Conversion to Equivalent," states that procedures for converting the instrument output to a leakage rate are available to the operators for containment radioactive air particulate monitoring and leakage to containment sumps. DCD Tier 2 Section 5.2.5.6, "Operability Testing and Calibration," describes periodic calibration and testing of the RCS leakage detection instrumentation. In addition, periodic inspection of the floor drainage system to the containment sump is conducted to check for blockage and provide reasonable assurance of unobstructed pathways.

Based on the above DCD description, the staff determined that the design, procedures, periodic testing of APR1400 control room leakage instrumentation are in accordance with the relevant RG 1.45 positions and guidance in SRP Section 5.2.5.

5.2.5.4.4 *Prolonged Low-Level RCS Leakage*

The operating experiences at Davis Besse (NRC Bulletin 2002 01) indicated that prolonged low level unidentified reactor coolant leakage inside containment could cause corrosion and material degradation such that it could compromise the integrity of a system leading to the gross rupture of the RCPB. RG 1.45, Regulatory Position on "Operations-Related Positions," provides guidance to address the issue. For example, the plant should establish procedures for responding to prolonged low-level RCS leakage. In addition, the procedures should specify operator actions in response to prolonged low-level unidentified reactor coolant leakage conditions that exist above normal leakage rates and below the TS limits to provide operators sufficient time to take action before the TS limit is reached. The procedures would include identifying, monitoring, trending, and managing prolonged low-level leakage.

In DCD Tier 2 Section 5.2.5.7, "Limits for Reactor Coolant Leakage Rates within RCPB," it states in COL 5.2(13), that the COL applicant is to address and develop the milestones for the preparation and implementation of the procedure for operator responses to prolonged low-level leakage according to the guidance in RG 1.45 Revision 1.

Based on the above, the staff determined that the proposed approach is consistent with the guidance in RG 1.45, Revision 1, pertaining to managing the prolonged low-level RCS leakage. The COL information item will require a COL applicant to address the prolonged low-level RCS leakage in more details according to RG 1.45. Therefore, the staff determined that the applicant's approach is acceptable.

5.2.5.4.5 *Leakage Separation (Identified and Unidentified Leakage)*

The regulatory positions in RG 1.45 indicate that leakage from identified sources should be monitored and collected separately from the unidentified sources. SRP Section 5.2.5 indicates that provisions for collecting, detecting, and monitoring unidentified leakage are separated from those for identified leakage.

In DCD Tier 2 Section 5.2.5.1.1, "Unidentified Leakage," the applicant described unidentified leakage and that the indication of unidentified coolant leakage into containment is provided by a containment sump level and flow monitor, an airborne particulate radioactivity monitor and an atmosphere humidity monitoring system. Unidentified leakages are routed to the containment drain sump, or incore instrumentation (ICI) cavity sump. In DCD Tier 2 Section 5.2.5.1.2, the applicant described the identified leakage. Identified leakage is defined in accordance with the guidance of NRC RG 1.45 as follows: (1) leakage (such as pump seal or valve packing leakage) that is captured, flow-metered, and conducted to a sump, collecting tank, or collection system and (2) leakage into the containment atmosphere from a known source, which does not interfere with the operation of unidentified leakage monitoring systems and is not attributable to leakage in the RCPB.

Based on the above DCD information, the staff determined that APR1400 has adequately demonstrated, in accordance with RG 1.45, that the RCPB leakage detection system can separately monitor and collect leakage from both identified and unidentified leakage without masking between the two types.

5.2.5.4.6 *Intersystem Leakage*

The regulatory positions in RG 1.45 states that the plant should monitor intersystem leakage for systems connected to the RCPB. SRP Section 5.2.5, "Reactor Coolant Pressure Boundary

Leakage Detection," indicates that the applicant should identify all potential intersystem leakage paths and the instrumentation to monitor the intersystem leakage.

In DCD Tier 2 Section 5.2.5.4, "Intersystem Leakage," the applicant described the leakage detection of the intersystem leakages that include safety injection system (SIS), steam generator (SG) leakage, shutdown cooling system (SCS), and component cooling water system (CCWS). Leakage from the RCS to the SIS under normal operation is detected by SIS pressure and level increases. Leakage into the safety injection tanks is detected by an increase in pressure between the check valves isolating the tanks from the RCS. This pressure is indicated and alarmed in the MCR. The leakage rate is computed from the rate of change of the level in the tank. The detection of leakage across the SG boundary between the primary to secondary side is addressed in DCD Tier 2 Section 5.2.5.1.2.5. Leakage across this boundary would be quantified, after the indication of radioactivity in the N-16 radiation monitors and the condenser vacuum vent effluent radiation monitor, by performing an RCS inventory balance. If the amount of leakage is small, chemical and radioisotope analyses of both the primary and secondary sides may be necessary to determine the leakage rate. DCD Tier 2, Appendix 11B describes the methods that are used to detect primary-to-secondary leakage. The primary-to-secondary leakage TS limit of 150 gallons per day (gpd) is specified in LCO 3.4.12 (d). The evaluation of the primary-to-secondary leakage is in Section 5.4.2.2, "Steam Generator Program," of this report. Leakage from the RCS to SCS under normal operation, when the system is isolated from the RCS, would be detected by relief valve discharges. The CCWS cools the RCPs, the SCS heat exchanger (HX), the letdown HX, and the containment spray pump and SCS pump miniflow HXs. Leakage from the RCS to the CCWS is detected by the component cooling water (CCW) radiation monitors and/or the CCW surge tank level. The change in surge tank level is utilized to quantify any leakage.

The staff issued 552-9083, Question 05.02.05-4 (ML17228A993), requesting the applicant revise DCD Tier 2 Section 5.2.4 to identify (1) the CVCS as a potential intersystem leakage path and (2) the instrumentation provided to monitor intersystem leakage between the RCS and the CVCS. In its response to RAI 552-9083, Question 05.02.05-4, (ML17248A376), the applicant revised DCD Tier 2 Section 5.2.5.4 to address these issues. The staff determined the response acceptable because it adequately described the intersystem leakage between the RCS and the CVCS. Based on the review of the DCD, the has confirmed incorporation of the changes describe above; therefore, RAI 552-9083, Question 05.02.05-4, is resolved and closed.

Based on the above, the staff determined that the design of APR1400 intersystem leakage detection is in accordance with RG 1.45 and SRP Section 5.2.5, because the design provides for identifying the paths and monitor intersystem leakage for systems connected to the RCPB, and is therefore acceptable.

5.2.5.4.7 *Technical Specifications*

RG 1.45 provides guidance on the TS requirements for RCS leakage detection systems.

In DCD Tier 2 Chapter 16, "Technical Specifications," the applicant provided 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.

LCO 3.4.12, RCS operational leakage shall be limited to the following:

a) No pressure boundary leakage

b) 1.89 L/min (0.5 gpm) unidentified leakage

c) 37.8 L/min (10 gpm) identified leakage

d) 0.39 L/min (150 gpd) primary-to-secondary leakage through any one SG

LCO 3.4.14, the following RCS leakage detection instrumentation shall be OPERABLE:

a) One containment sump level monitor

b) One containment atmosphere radioactivity (particulate) monitor

c) One containment atmosphere humidity monitor

The adequacy of the specified TS limit of 0.5 gpm for unidentified leakage and leakage detection systems is evaluated above in SER Section 5.2.5.4.2. The limit of 10 gpm for identified leakage is consistent with Combustion Engineering Standard TS. The limit for the primary-to-secondary leakage in TS LCO 3.4.12 (d) is reviewed in SER Subsection 5.4.2.2, "Steam Generator Program." The adequacy of three diverse leakage detection monitors in TS LCO 3.4.14, which include sump level monitor, radioactivity (particulate) monitor, and humidity monitor, is evaluated above in SER Section 5.2.5.4.2. Additional review on TS LCO 3.4.14 is in SER Section 16.

Based on the above DCD information and previous staff's evaluation, the staff determined that the proposed TSs are in accordance with the guidelines in RG 1.45 because the plant TS include the limiting conditions for identified, unidentified, and RCPB leakage; the availability of various types of instrumentations to ensure adequate coverage during all phases of plant operation; and at least two (containment sump level and containment airborne radioactivity) independent RCS leakage detection methods being able to detect the leakage of 0.5 gpm within one hour.

5.2.5.4.8 *GDC 2 and Seismic Qualification*

GDC 2 requires that safety-related SSCs be designed to withstand the effects of natural phenomena, including earthquakes, without loss of capability to perform intended safety functions. RG 1.29 describes an acceptable method of identifying and classifying system portions that should be designed to withstand the effects of a safe shutdown earthquake (SSE). In SRP Section 5.2.5, it states that the RCPB leakage detection system detects leakage after an early indication of degradation so that corrective action can be taken before such degradation becomes severe enough to result in a leak rate greater than the capability of the makeup system to replenish the coolant loss. In RG 1.45, it states that at least one of the leakage monitoring systems specified in the plant TSs should be capable of performing its functions following any seismic event that does not require plant shutdown.

In DCD Tier 2 Section 5.2.5.1.1.2, "Sump Level and Flow Method," it states that the sump level monitoring system is qualified for a SSE. In DCD Tier 2 Section 5.2.5.1.1.3, "Containment Air Particulate Radioactivity Monitoring," it states that the containment air radiation monitor is capable of functioning when subject to SSE. Based on the above, the staff determined that the guidance in RG 1.45 is satisfied with respect to the seismic qualification, and therefore, GDC 2 is satisfied.

5.2.5.4.9 *Initial Testing Program*

RG 1.68, Appendix A, "Initial Test Program," June 2013, Revision 4 provides guidance on the initial testing program for RCS leakage detection, which includes the relevant leak tests such as:

- The RCS leak detection systems;
- The primary to secondary leakage detection system through SGs;
- The verification of the capability of RCS leakage detection instrumentations;
- Verification of the RCS leak rates being within specified limits;
- Calibration of the instrumentation being used for reactor coolant leak detection systems;
- The operation of computer programs used to calculate RCS leakage rates.

DCD Tier 2 Section 14.2.12.1.134, "Leakage Detection System Test," demonstrates the operation of the various leakage detection systems. The test prescribed in DCD Tier 2 Section 14.2.12.1.134 will test the sump level switches and flow monitors, airborne radioactivity monitor, and/or atmosphere humidity monitors using simulated signals. In Subsection 5.0/5.1, "Acceptance Criteria," of this test, it states that "the leakage detection system operates as described in Subsection 5.2.6.1."

The staff reviewed the above information but could not find the referenced Subsection 5.2.6.1 in the DCD to review the acceptance criteria. The staff issued RAI 80-8040, Question 05.02.05-2 (ML15295A317), requesting the applicant to provide Subsection 5.2.6.1 in the DCD and demonstrate that the proposed initial test program (ITP) has adequately addressed the tests identified in RG 1.68, Appendix A.

In its response to RAI 80-8040, Question 05.02.05-2 (ML15243A074), and updated on September 15, 2017 (ML17258A210), the applicant clarified an editorial error on the referenced Subsection. Subsection 5.0/5.1 is revised as follows: "The leakage detection system operates as described in Subsection 5.2.5.1."

Subsection 5.2.5.1, "Leakage Detection Method," provides a description of the detection methods and operation of the unidentified leakage and identified leakage detection system. The unidentified leakage detection system will be tested in accordance with Section 14.2.12.1.134, "Leakage Detection System Test," including calibration and verification of the capability of leakage detection instrumentation.

The RCS leak rate measurements for unidentified leakage and identified leakage will be tested in accordance with Section 14.2.12.1.57, "Pre-Core Reactor Coolant System Leak Rate Measurement," and Section 14.2.12.2.7, "Post-Core Reactor Coolant System Leak Rate Measurement." These tests are performed for the verification of RCS leak rates being within the limits described in Subsection 5.2.5, and include the operation of computer programs used to calculate RCS leakage rates, if applicable.

The primary to secondary leakage detection system through the SGs will be tested in accordance with Section 14.2.12.1.106, "Process and Effluent Radiological Monitoring System (PERMS) Test," including the test described in of Appendix 11B, "Primary to Secondary

Leakage Detection." The N-16 radiation monitors for detection of primary to secondary leakage are described in Subsection 11.5.2.2.m, "Gaseous PERMS."

Based on the analysis above, the staff determined that the response is acceptable, because DCD Tier 2 Section 14.2.12.1.134, "Leakage Detection System Test," adequately demonstrated the operation of the various leakage detection systems. The staff confirmed that the DCD was revised as committed in the response to RAI 80-8040, Question 05.02.05-2. Therefore, RAI 80-8040, Question 05.02.05-2, is resolved and closed.

5.2.5.4.10 *Inspections, Tests, Analyses and Acceptance Criteria*

DCD Tier 1 Section 2.4.7, "Leakage Detection System," indicates that Table 2.4.7-1, "Leakage Detection System ITAAC," describes the ITAAC for RCPB leakage detection system. This ITAAC will perform the as-built inspection to verify the design of the RCS leakage detection systems. The inspection includes verifying the capability of 0.5 gpm within one hour for containment sump level instruments, and containment airborne particulate radioactivity monitor, the associated display and alarms in the MCR.

In its review of DCD Tier 1 Section 2.4.7, the staff noticed that the acceptance criteria for item 1.e, the acceptance criteria for the containment airborne particulate radioactivity monitor, states that it has the capability for detecting a "change in" leakage rate of 1.89 L/min (0.5 gpm) or greater within an hour. The staff noticed that containment sump (1.c) has the criterion of being able to detect a "change in" leakage rate of 0.5 gpm or greater within an hour. The staff issued RAI 369-8486, Question 05.02.05-3 (ML16019A275), requesting the applicant to clarify if the criteria should be able to detect a leakage rate of 0.5 gpm or greater within an hour, instead of a "change in" leakage rate of 0.5 gpm or greater within an hour.

In its response to RAI 369-8486, Question 05.02.05-3 (ML16029A437), the applicant clarified that the acceptance criteria for containment sump level monitor and containment airborne particulate radioactivity monitor will be revised to delete "change in" from items No's 1.c and 1.e of Table 2.4.7-1, "Leakage Detection System ITAAC." In addition, the containment atmosphere humidity monitor will be deleted from Subsection 2.4.7.1, "Design Description," and the Table 2.4.7-1 as a result of the applicant's response to RAI 80-8040, Question 05.02.05-1. The humidity sensor is designed to indicate a sudden and significant increase of humidity level by annunciating an alarm in the MCR due to unidentified leakage." The accuracy and sensitivity of the humidity sensor used in the APR1400 design is not sufficient enough to establish a quantitative correlation between the unidentified RCS leakage and the humidity sensor to detect the level of detail of 0.5 gpm within one hour. The staff determined that the response to RAI 369-8486, Question 05.02.05-3 is acceptable because the criteria should be a leakage rate instead of a "change in" leakage rate. In addition, a quantitative ITAAC criterion for the humidity monitor can be removed, because the quantitative acceptance criterion of the response time for the leakage detection is not applicable as discussed in previously in Section 5.2.5.4.2 of this SER.

Based on the above, the staff determined that the ITAAC for the APR1400 RCS leakage detection is adequate because it has addressed the capability of all the RCS leakage detection methods listed in the plant TS and the associated display and alarms in the MCR. The staff confirmed that the DCD was revised as committed in the response to RAI 369-8486, Question 05.02.05-3. Therefore, RAI 369-8486, Question 05.02.05-3, is resolved and closed.

5.2.5.5 *Combined License Information Items*

DCD Tier 2 Section 5.2.5.7 lists COL information item COL 5.2(13) and states:

> *The COL applicant is to address and develop the milestones for the preparation and implementation of the procedure for operator responses to prolonged low-level leakage per guidance in RG 1.45, Revision 1.*

The staff reviewed this COL item and determined that it is acceptable because it is consistent with the guidance in RG 1.45, Revision 1, regarding prolonged low-level RCS leakage.

5.2.5.6 *Conclusion*

Based on the above, the staff concluded that the design of the RCPB leakage detection system follows the guidelines of SRP Section 5.2.5 and RG 1.45 and, therefore, meets the requirements of GDC 2 and GDC 30.

5.3 Reactor Vessel

The reactor vessel (RV) is a vertically mounted cylindrical shell with a hemispherical lower head welded to the cylindrical shell and a removable hemispherical upper closure head. The RV contains the reactor fuel and the vessel internals, which direct the flow of reactor coolant. The RV has four inlet and two outlet nozzles located in a horizontal plane just below the RV flange, but above the top of the fuel. The reactor coolant enters the RV through the inlet nozzles, is guided downward into the annulus between the RV shell and the core barrel, and then upward through the core, acquiring thermal energy. The reactor coolant leaves the RV through the outlet nozzles. The RV closure head contains penetrations for CED mechanism adapters, in-core instrumentation adapters, and a high point vent. The bottom contains 61 in-core instrumentation penetration nozzles and four external shear key supports that mate with the keyway in the RV support column base plate.

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 NDE, special controls and special processes used for ferritic steels and austenitic stainless steels, fracture toughness, material surveillance, and RV fasteners. This section of the DCD should contain pertinent data in sufficient detail to provide assurance that the materials (including weld materials), fabrication methods, and inspection techniques used for the RV and applicable attachments and appurtenances conform to all applicable regulations. Other RCS materials are addressed in Section 5.2.3 of this SER.

5.3.1.2 *Summary of Application*

DCD Tier 1: The DCD Tier 1 information associated with this section is found in DCD, Tier 1, Section 2.4.1, “Reactor Coolant System,” which describes the RCS.

DCD Tier 2: The applicant provided a Tier 2 description of the materials used in the RV in DCD Tier 2 Section 5.3.1, “Reactor Vessel Materials,” summarized here in part as follows:

The RV is fabricated in accordance with ASME Section III requirements. It consists of forged rings, forged hemispherical heads, forged flanges on the closure head, and forged nozzles. The forgings are supplied in a quenched and tempered condition. Vacuum degassing is used as part of the fabrication process to improve the quality of the steel by lowering the hydrogen levels. To reduce the effects of embrittlement, limits are placed on the weight percent of residual elements, such as copper, nickel, phosphorus, and sulfur, in the forgings and welds that compose the beltline region. Cladding is used on the internal surfaces of the RV that are in contact with the reactor coolant. No special manufacturing methods that could compromise the integrity of the RV are used.

Welding materials and processes are in accordance with the ASME Code. Welding processes used to fabricate the vessel include submerged arc welding, flux core arc welding, gas tungsten arc welding, and shielded metal arc welding. Electroslag welding is not used in the RV.

NDE is performed on all welds and forgings, as required, prior to, during, and after fabrication of the RV. All full penetration, pressure-retaining welds are 100 percent radiographed. All forgings are inspected by ultrasonic testing. All bolting material is examined using ultrasonic and magnetic-particle techniques. Upon completion of all post-weld heat treatments, the RV is hydrostatically tested. Cladding is ultrasonically inspected.

Special controls for welding of ferritic and austenitic stainless steel are described in DCD Tier 2 Section 5.2.3. The DCD also references the use several NRC RGs related to the RV. Special controls are also provided for tools used in abrasive work operations to ensure that austenitic steels are not contaminated with ferritic steels or other materials that could contribute to SCC.

The RV beltline materials are expected to meet all of the requirements of 10 CFR Part 50, Appendix G, "Fracture Toughness Requirements." To assess the effects of neutron fluence during the plant's operating life, a RV materials surveillance program is used. The DCD describes the various aspects of the RV materials surveillance program, including material selection, the type and quantity of test specimen, the design of the surveillance capsules, capsule locations, and the withdrawal schedule.

The bolting for the RV closure head is made from SA-540, Grade B24, Class 3 material. The DCD describes how the bolting will be fabricated, mechanically tested, and inspected in accordance with the ASME Code, the regulations, and NRC RG 1.65, "Materials and Inspections for Reactor Vessel Closure Studs."

ITAAC: The ITAAC associated with DCD Tier 2 Section 5.3.1 are given in DCD, Tier 1, Table 2.4.1-4, "Reactor Coolant System ITAAC." Table 2.4.1-4, Items 2.a, 3.a, and 4.a, indicate that inspections will be performed on as-built components and that reports exist that conclude the following:

(1) As-built ASME Code components are designed and constructed in accordance with the requirements of the 2007 Edition with the 2008 Addenda of ASME Section III (hereafter referred to as ASME Section III).

(2) The ASME Section III requirements are met for NDE of the pressure boundary welds in as-built ASME Code components.

(3) The results of hydrostatic testing of the as-built ASME Code components conform with the ASME Section III requirements.

- Table 2.4.1-4, Item 10 indicates that an inspection of the RV is performed to ensure that at least six surveillance capsules are in the RV. Table 2.4.1-4, Item 11 indicates that an inspection of the RV surveillance capsule material specimens is performed to ensure that the surveillance capsule specimens are made from material used to fabricate the RV and that they include the appropriate specimen types (charpy V-notch, tensile, etc.).

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

5.3.1.3 *Regulatory Basis*

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

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

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

3. GDC 14, as it relates to prevention of rapidly propagating failures 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. Title 10 CFR 50.55a, as it relates to quality standards for design, and determination and monitoring of fracture toughness. The staff notes that 10 CFR 50.55a endorses ASME Section III and ASME Section XI.

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

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

10. Title 10 CFR Part 50, Appendix H, as it relates to the determination and monitoring of fracture toughness. The staff notes that 10 CFR Part 50, Appendix H requires that surveillance programs be developed in accordance with ASTM E185-82.

5.3.1.4 *Technical Evaluation*

The staff reviewed APR1400 DCD, Tier 2 Section 5.3.1, “Reactor Vessel Materials,” using NUREG-0800, Section 5.3.1, “Reactor Vessel Materials.” The ASME Code of record for the APR1400 is the 2007 Edition with the 2008 Addenda). Subject to the conditions of 10 CFR 50.55a, ASME Section III, Subsection NB presents the construction requirements for the RV.

5.3.1.4.1 *Materials Specifications*

The materials specifications for the RV are acceptable if they are in accordance with ASME Section III, NB-2000. ASME Section III, NB-2121 states that pressure-retaining material shall conform to the requirements of one of the specifications for material given in ASME Section II, Part D, Subpart 1, Tables 2A, “Section III, Classes 1, TC, and SC Design Stress Intensity Values S_m for Ferrous Materials,” and 2B, “Section III, Classes 1, TC, and SC Design Stress Intensity Values for S_m for Nonferrous Materials.” ASME Section III, NB-2128 states that materials for bolts and studs shall conform to the requirements of one of the specifications listed in listed in Section II, Part D, Subpart 1, Table 4, “Section III, Classes 1, TC, and SC, and Section VIII, Division 2 Design Stress Intensity Values S_m for Bolting Materials.” The Code also states that welding and brazing material shall comply with an SFA specification in ASME Section II, Part C, except as otherwise permitted in ASME Section IX.

DCD Tier 2 Section 5.3.1.1, “Material Specifications,” states that the principal ferritic materials used in the RV are listed in Table 5.2-2, “Reactor Coolant System Materials and Weld Materials.” Ferrous materials used to fabricate the RV include SA-508 Grade 3, Class 1 (forgings), and SA-182 Grade F316LN (direct vessel injection nozzle safe ends). NiCrFe Alloy 690 is used for the RV head control element drive mechanism (CEDM) nozzles, the RV flow skirt, and the instrument nozzles on the bottom head. The RV closure head studs are SA-540 Grade B24, Class 3. Based on the review of the information described above, the staff determined that the material specifications are acceptable because they meet the requirements of ASME Section III, and therefore complies with 10 CFR 50.55a.

DCD Table 5.2-2, “Reactor Coolant System Materials and Weld Materials,” also provides the specifications for the weld materials used in the RV and DCD Tier 2 Section 5.3.1.2 states that welding materials for the RV conform to ASME Section II and ASME Section III, or satisfy the requirements for other welding materials as permitted in ASME Section IX. The weld materials used to fabricate the RV include SFA 5.1, SFA 5.5, SFA 5.18, SFA 5.23, MIL-E-18193 B-4, and NiCrFe filler metal. The staff noted that the MIL-E-18193B weld electrode specification was cancelled in 1989, and was superseded by a different military specification. This issue was discussed with the applicant in a public meeting on June 30, 2015, and the applicant responded that MIL-E-18193 B-4 is not used for the APR1400 and that DCD Table 5.2-2 will be revised to delete MIL-E-18193 B-4 from the weld material specifications. Removal of the MIL-E-18193 B-4 weld electrode from APR1400 DCD Table 5.2-2 is acceptable because all other weld electrodes identified for P-1 to P-3 or P-3 to P-3 welding are in accordance with the ASME Code. The staff confirmed that the applicant appropriately deleted the subject material from the DCD Table 5.2-2 in the DCD.

DCD Table 5.2-2 states that the NiCrFe filler metal will be used for buttering of j-groove welds in the RV closure head and in the RV CEDM nozzles, however the filler metal material specification was not provided. This issue was discussed with the applicant in a public meeting on June 30, 2015, and the applicant responded that DCD Table 5.2-2 will be revised to add the specifications for the NiCrFe filler metals, which are SFA 5.11 ENiCrFe-7 and SFA 5.14 ERNiCrFe-7(A). The applicant’s response is acceptable because the filler metal specifications comply with the ASME Code and therefore complies with 10 CFR 50.55a. The applicant revised DCD Table 5.2-2 in the DCD. The staff determined that the applicant has appropriately addressed the issue in the DCD.

5.3.1.4.2 *Special Processes Used for Manufacture and Fabrication of Components*

The special processes used for the manufacture and fabrication of the RV are acceptable if they are in accordance with ASME Section III. Special processes that do not have Code requirements are reviewed on a case-by-case basis.

DCD Tier 2 Section 5.3.1.2, "Special Process Used for Manufacturing and Fabrication," states that the RV is fabricated in accordance with ASME Section III, NB-4000 and its materials satisfy the requirements of ASME Section III, NB-2000. Application of the appropriate Code Symbol and completion of a data report are in accordance with ASME Section III, NCA-8000. The applicant also states that no special manufacturing methods are used that could compromise the integrity of the RV. The staff determined that this is acceptable because it meets the requirements of ASME Section III. Compliance with ASME Section III is an acceptable approach to satisfy the requirements of 10 CFR Part 50, Appendix A, GDC 1 and GDC 30.

5.3.1.4.3 *Special Methods for Nondestructive Examination*

DCD Tier 2 Section 5.3.1.3, "Special Methods for Nondestructive Examinations," states that prior to, during, and after fabrication of the RV, NDE based on ASME Section III are performed on all welds and forgings as required. The applicant also described, in the DCD, the NDE to be performed on the RV and its appurtenances. The DCD states that all full penetration, pressure-containing welds are 100 percent radiographed to the standards of ASME Section III. Weld preparation areas, back chip areas, and final weld surfaces are magnetic particle or liquid penetrant examined. Other pressure-retaining welds are inspected by liquid penetrant testing. All forgings are inspected by ultrasonic testing. All carbon-steel and low alloy forgings and ferritic welds are also inspected by magnetic particle testing after stress relief. All RV bolting is examined using ultrasonic and magnetic particle testing. DCD, Tier 2, Tables 5.3-8, "Inspection Plan for Reactor Vessel Materials," and 5.3-9, "Inspection Plan for Reactor Vessel Welds," summarize the NDE methods that are used for the RV base materials and welds. Based on the review of the information described above, the staff determined that the applicant's plan for NDE of the RV is acceptable because it is in accordance with ASME Section III, NB-5000, and therefore complies with 10 CFR 50.55a.

5.3.1.4.4 *Special Controls for Ferritic and Austenitic Stainless Steels*

Special controls and special welding processes used for welding the RV and its appurtenances are acceptable if they are in accordance with the requirements of the ASME Code. DCD Tier 2 Section 5.3.1.2, "Special Process Used for Manufacturing and Fabrication," describes the controls on welding for the RV. The DCD states that the welding processes used to fabricate the APR1400 RV include submerged arc welding, flux core arc welding, gas tungsten arc welding, and shielded metal arc welding. Electroslag welding is not used in the RV. The applicant also stated that welding of pressure boundary parts of the RV is performed in accordance with welding procedure specifications that satisfy the requirements of ASME Section III and Section IX. Preheat temperatures utilized for low alloy steel is in accordance with ASME Section III, Appendix D. Also, post-weld heat treatment temperature and time for welds of low alloy steels are in accordance with ASME Section III, NB-4620. The information described above is acceptable because it meets the requirements of ASME Section III and ASME Section IX respectively, and therefore complies with 10 CFR 50.55a.

DCD Tier 2 Section 5.3.1.4, "Special Controls for Ferritic and Austenitic Stainless Steels," the applicant stated that the tools used in abrasive work operations on austenitic steel do not contain and are not contaminated with ferritic carbon steel or other materials that could contribute to intergranular SCC. This is acceptable because it is in accordance with ASME NQA-1.

In DCD Tier 2 Section 5.3.1.4, the applicant identified the NRC RGs that are applicable to the RV. The applicability of the following RGs to the APR1400 is addressed in DCD Tier 2 Section 5.2.3; “Reactor Coolant Pressure Boundary Materials,” RG 1.31, RG 1.34, RG 1.43, RG 1.44, RG 1.50, and RG 1.71. As such, the staff’s evaluation of the applicant’s use of these RGs is documented in Section 5.2.3 of this SER. The staff’s evaluation of the applicant’s use of RG 1.99, “Radiation Embrittlement of Reactor Vessel Materials,” is documented in Sections 5.3.1.4.5, “Fracture Toughness,” and 5.3.2, “Pressure-Temperature Limits, Upper-Shelf Energy, and Pressurized Thermal Shock,” of this SER, and the staff’s evaluation of the applicant’s use of RG 1.190, “Calculation and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence,” is documented in Section 4.3, “Nuclear Design,” of this SER.

5.3.1.4.5 *Fracture Toughness*

DCD Tier 2 Section 5.3.1.5, “Fracture Toughness,” describes how the fracture toughness requirements of 10 CFR Part 50, Appendix G are met for RV beltline materials in the APR1400. The DCD states that RV beltline materials have a minimum upper-shelf energy (USE) of 102 Joules (75 ft-lbs) as determined by charpy V-notch tests on unirradiated specimens. DCD Tier 2 Section 5.3.2.4, “Upper-Shelf Energy,” states that the end-of-life charpy USE is estimated to be 69.4 Joules (51 ft-lbs). The applicant also stated that charpy V-notch test coupons, test specimens, testing procedures, testing requirement, and acceptance criteria for nil ductility reference temperature (RT_{NDT}) determination are in accordance with ASME Section III, NB-2300. DCD Tier 2 Section 5.3.1.5, “Fracture Toughness,” also states that the effect of neutron irradiation is taken into account in accordance with RG 1.99, Revision 2. The staff determined that the information provided in the DCD is acceptable because it meets the requirements of 10 CFR Part 50, Appendix G, Section III and Section IV.A.1.

The staff’s evaluation of the applicant’s predicted charpy USE for the beltline materials is documented in Section 5.3.2, “Pressure-Temperature Limits, Upper-Shelf Energy, and Pressurized Thermal Shock,” of this SER.

5.3.1.4.6 *Material Surveillance*

GDC 32 requires that the RCPB components shall be designed to permit an appropriate material surveillance program for the RV. Title 10 CFR Part 50, Appendix H states that RVs that are projected to have a peak neutron fluence exceeding 10^{17} n/cm^2 (E > 1.0 MeV) must have their beltline materials monitored by a surveillance program complying with ASTM International (formerly the American Society for Testing and Materials) (ASTM) E-185. The latest edition of ASTM E-185 incorporated by reference into 10 CFR Part 50, Appendix H, “Reactor Vessel Material Surveillance Program Requirements,” is the 1982 Edition (ASTM E-185-82).

To meet the requirements of GDC 32, the APR1400 design includes provisions for a material surveillance program to monitor changes in the fracture toughness caused by exposure of the RV beltline materials to neutron radiation. In DCD Tier 2 Section 5.3.1.6, “Material Surveillance,” the applicant described various aspects of the RV materials surveillance program, including material selection, the type and quantity of test specimen, the design of the surveillance capsules, capsule locations, and the withdrawal schedule. Six identical surveillance capsule assemblies are provided. Four of the assemblies are for retrieval and two are for standby. The type and quantity of specimens contained in each capsule assembly are presented in DCD Table 5.3-4. DCD Table 5.3-4, "Type and Quantity of Specimens Contained in Each Irradiation Capsule Assembly."

DCD Tier 2 Section 5.3.1.6.1, "Test Material Selection," states that materials selected for the surveillance capsule program are those judged to be controlling with regard to radiation embrittlement. Test materials are prepared from the actual material used in fabricating the beltline region of the RV and include the base metal, weld metal, and heat affected zone (HAZ) material. This is acceptable because it meets the requirements of ASTM E185-82, and therefore complies with 10 CFR Part 50, Appendix H.

DCD Tier 2 Section 5.3.1.6.2, "Test Specimens," describes the type and quantity of test specimen provided for the surveillance capsule program. Standard charpy impact, tensile, and compact tension (CT) and precracked charpy fracture toughness specimens are provided for unirradiated baseline and irradiated testing. For baseline testing, twelve drop weight test specimens are provided for each base metal, weld metal, and HAZ material for establishing the nil-ductility transition temperature. Twenty-four unirradiated standard charpy test specimens (transverse) are provided for each base metal, weld metal, and HAZ material. In addition,12 unirradiated precracked charpy test specimens (transverse) are provided for each base metal and weld metal with 12 longitudinal precracked charpy specimens for base metal. Also, 12 tensile test specimen (transverse), 8 1T CT specimen (transverse), and 4 1/2 T CT specimen (transverse) are provided for each base metal and weld metal. For irradiated testing, a total of 360 standard charpy V-notch impact, 162 precracked charpy fracture toughness, 72 1/2T CT fracture toughness, and 54 tensile test specimen are provided to account for 60 years of operation. The type and quantity of test specimen provided for the surveillance capsule program exceeds the requirements of ASTM E-185-82, and is therefore acceptable because it complies with 10 CFR Part 50, Appendix H.

DCD Tier 2 Section 5.3.1.6.3, "Surveillance Capsules," describes the design and layout of the surveillance capsule assemblies and DCD Tier 2 Section 5.3.1.6.5, "Irradiation Locations," describes their location within the RV and associated lead factors. A diagram of the capsule assemblies is provided in DCD Figure 5.3-1, "Typical Surveillance Capsule Assembly." The applicant stated that the capsule assemblies are corrosion resistant and that all compartments are sealed with helium. The surveillance capsules are equipped with lock assemblies that fix the locations of the capsules within the holders and prevent relative motion. The lock assemblies also serve as a point of attachment for the tooling used to remove the capsules from the reactor. DCD Tier 2 Section 5.3.1.6.5 states that the design of the capsule assemblies and holders also permits the remote installation of replacement capsule assemblies. The capsule holders are welded to the RV cladding on the inside surface, and the welds are subject to inspection in accordance with ASME Section III and Section XI. The surveillance capsules are located to produce a lead factor (ratio of the neutron flux at the location of the capsule to that at the RV inner surface at the peak neutron fluence location) of approximately 1.4. The axial and vertical position of the capsule assemblies within the RV are illustrated on DCD Figures 5.3-5, "Location of Surveillance Capsule Assemblies (Plan View)," and 5.3-6, "Location of Surveillance Capsule Assemblies (Elevation View)." The azimuthal locations of the capsule assemblies are also provided in DCD Table 5.3-7, "Capsule Assembly Removal Schedule." The design, location, and associated lead factors of the surveillance capsule assemblies and holders are acceptable because they meet the requirements of ASTM E-185-82, and therefore comply with 10 CFR Part 50, Appendix H.

DCD Tier 2 Section 5.3.1.6.4, "Neutron Irradiation and Temperature Exposure," describes the neutron dosimeters and temperature monitors provided in the surveillance capsules. Three sets of flux spectrum monitors, capable of monitoring thermal and fast neutron spectra, and one set of temperature monitors are included in each capsule assembly. The materials to be used for the neutron threshold detectors are listed in DCD Table 5.3-5, “Material for Neutron Threshold Detectors.” The composition and melting points of the candidate materials for temperature monitors are listed in DCD Table 5.3-6, “Composition and Melting Points of Candidate Material for Temperature Monitors.” The information provided in the DCD is acceptable because it meets the requirements of ASTM E-185-82, and therefore complies with 10 CFR Part 50, Appendix H.

DCD Tier 2 Section 5.3.1.6.6, “Withdrawal Schedule,” as modified by a letter dated July 17, 2015, describes the surveillance capsule withdrawal schedule. For a predicted transition temperature shift of less than 100 ^{o}F (56 ^{o}C), ASTM E 185-82 requires a minimum of three surveillance capsules. However, four primary surveillance capsules are provided for the APR1400 surveillance program because the design life of the APR1400 RV (60 years) is longer that the design life indicated in ASTM E 185-82 (40 years or 32 EFPY). A comparison of the withdrawal schedule required by ASTM E185-82 to the APR1400 withdrawal schedule is shown below:

Table 5.3-1: Comparison of APR1400 Withdrawal Schedule to the requirements of ASTM E185-82

	ASTM E 185-82, Table 1 (Left column – Predicted shift less than or equal to 100 ^{o}F)	**APR1400 DCD**
1st Capsule	No later than 6EFPY	6 EFPY
2nd Capsule	No later than 15EFPY	15 EFPY
3rd Capsule	EOL (Withdraw at a neutron fluence not less than once or greater than twice the peak end of design life vessel neutron fluence (this capsule may be held without testing following withdrawal)).	32 EFPY
4th Capsule	Not Required	EOL

Based on the above comparison, the staff determined that the APR1400 withdrawal schedule meets the minimum requirements of ASTM E185-82 and that the addition of a fourth surveillance capsule is appropriate because it provides a reasonable assurance that the mechanical properties of the reactor vessel will be monitored throughout its design life. On this basis, the staff determined that the withdrawal schedule for the APR1400 surveillance capsule program was acceptable.

In DCD Tier 2 Section 5.3.1.6.7, "Irradiation Effects Prediction Basis," the applicant stated that when data from the surveillance capsules becomes available, it will be used to adjust the pressure and temperature limit curves. This is acceptable because it is in compliance with 10 CFR Part 50, Appendix H, and it provides a reasonable assurance that the pressure and temperature limits will be determined in accordance with 10 CFR Part 50, Appendix G throughout the operating life of the APR1400.

Based on the review described above, the staff determined that the description of the RV materials surveillance program for the APR1400 is acceptable because it is in accordance with ASTM E-185-82 and meets the requirements of 10 CFR Part 50, Appendix H, "Reactor Vessel Material Surveillance Program Requirements."

5.3.1.4.7 *Reactor Vessel Fasteners*

DCD Tier 2 Section 5.3.1.7, "Reactor Vessel Fasteners," states that the bolting material for the RV closure head is fabricated from SA-540, Grade B24, Class 3. The applicant indicated that the bolting material meets the fracture toughness requirements of 10 CFR Part 50, Appendix G and the intent of RG 1.65. The DCD also states that the bolting materials are tested in accordance with ASME Section III, NB-2220 and NB-2300. NDE is performed in accordance with ASME Section III, NB-2580 during the manufacturing process. Manganese phosphate coating is used on the studs, nuts and washers to improve anti-galling properties and corrosion resistance. Also, nickel-based anti-seize lubricant is added to the threads and bearing surfaces to prevent galling. The information described above is acceptable because it is in accordance with RG 1.65. The integrity of the APR1400 RV 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. The staff's evaluation of the conformance of the threaded fasteners used in the APR1400 design with the recommendations of RG 1.65 is further discussed in Section 3.13, "Threaded Fasteners – ASME Code Class 1, 2, and 3," of this SER.

5.3.1.5 *Combined License Information Items*

The following is a list of COL information item numbers and descriptions from DCD Tier 2, Table 1.8-2, "Combined License Information Items:"

Table 5.3.1-1 APR1400 Combined License Information Items Identified in the DCD

Item No.	Description	Section
COL 5.3(1)	A COL applicant is to provide a reactor vessel material surveillance program for a specific plant.	5.3.1.6

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

5.3.1.6 *Conclusion*

The staff concludes that the APR1400 RV materials and associated manufacturing and fabrication processes, NDE methods, fracture toughness testing, and material surveillance meet the applicable requirements of the ASME Code, ASTM E185-82, 10 CFR 50.55a, and 10 CFR Part 50 Appendices G and H, which provide an acceptable basis for satisfying the requirements of GDC 1, 4, 14, 30, 31, and 32.

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

5.3.2.1 *Introduction*

Neutron radiation is known to cause embrittlement, or a reduction in ductility, in the RV. This degradation is most severe in the beltline region due to its exposure to the highest average neutron flux at power. This reduction in ductility is typically measured in terms of a change in RT_{NDT} or a change in charpy USE. To limit the effects of radiation embrittlement, controls are placed on the weight percentage of residual elements, such as copper, nickel and phosphorus in the materials used to fabricate the RV. An additional requirement placed on beltline materials is that they must also maintain charpy USE at an acceptable level throughout the life of the RV. In addition, pressure-temperature (P-T) limits are imposed on the RCS to provide adequate safety margins against non-ductile fracture during normal operation, heat-up, cooldown, AOOs, and system hydrostatic, pre-service and inservice leakage tests.

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

5.3.2.2 *Summary of Application*

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

DCD Tier 2: The applicant provided a DCD Tier 2 description of how it addresses P-T limits, PTS, and charpy USE 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 P-T limits for heatup, cooldown, low temperature operation, criticality, and hydrostatic testing, as well as heatup and cooldown rates shall be established and documented per the TS provided in DCD Tier 2, Chapter 16, Section 3.4.3, “RCS Pressure and Temperature (P-T) Limits,” and Section 3.4.11, “Low Temperature Overpressure Protection.” The detailed methodology for developing the P-T limit curves is provided in APR1400-Z-M-NR-14008, “Pressure-Temperature Limits Methodology for RCS Heatup and Cooldown.” Generic P-T limit curves based on bounding RV material properties are provided in DCD Figure 5.3-7, “Pressure-

Temperature Limit Curves (60 years)." A COL applicant that references the APR1400 DC will develop P-T limit curves based on plant-specific data.

Bounding PTS and charpy USE values are determined based on neutron fluence projections over a 60 year design life. A COL applicant that references the APR1400 design is to verify the PTS reference temperature (RT_{PTS}) and USE values based on plant-specific material properties and neutron fluence.

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

TS: The TS associated with Section 5.3.2 of this report are given in DCD Tier 2, Chapter 16, Sections 3.4.3, "RCS Pressure and Temperature (P/T) Limits," B 3.4.3, "RCS Pressure and Temperature (P/T) Limits," 3.4.11, "Low Temperature Overpressure Protection (LTOP) System," and B 3.4.11, "Low Temperature Overpressure Protection (LTOP) System." In addition, TS 5.6.4, "Reactor Coolant System (RCS) Pressure and Temperature Limits Report (PTLR)," specifies the content of the RCS PTLR.

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 NUREG-0800, Section 5.3.2 and are summarized below. Review interfaces with other SRP sections can be found in NUREG-0800, Section 5.3.2.

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

2. Title 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.

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

4. Title 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 prevention of rapidly propagating failures of the RCPB.

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

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

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

Acceptance criteria adequate to meet the above requirements include:

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

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

3. GL 96-03 as it relates to the submittal of a pressure and temperature limits report (PTLR).

5.3.2.4 *Technical Evaluation*

5.3.2.4.1 *Pressure-Temperature Limits*

To address the requirements of 10 CFR Part 50, Appendix G relating to P-T limits, the applicant submitted APR1400-Z-M-NR-14008, “Pressure-Temperature Limits Methodology for RCS Heatup and Cooldown,” Revision 0 for the NRC’s review and approval in a letter dated December 23, 2014. This PTLR follows the guidelines of GL 96-03, “Relocation of Pressure Temperature Limit Curves and Low Temperature Overpressure Protection System Limits,” and provides the bounding P-T limits, LTOP system limits, and the complete methodology for their development. The staff’s review of review of APR1400-Z-M-NR-14008, hereafter called the APR1400 PTLR, is complete (ML18087A364).

5.3.2.4.2 *Pressurized Thermal Shock*

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

The PTS Rule, 10 CFR 50.61, “Fracture toughness requirements for protection against pressurized thermal shock events,” established screening criteria to serve as a limiting level of RV material embrittlement beyond which operation cannot continue without further plant-specific evaluation. The screening criteria are given in terms of reference temperature, RT_{PTS}. The screening criteria are 270 °F (132.2 °C) for plates and axial welds, and 300 °F (148.9 °C) 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, neutron fluence and calculational procedures

In DCD Tier 2 Section 5.3.2.3, the applicant provided RT_{PTS} values for the limiting beltline material, which is a weld conservatively assumed to be in the center of the RV beltline and subjected to a neutron fluence of 9.5×10^{19} n/cm^2 (E > 1.0 MeV). However, 10 CFR 50.61 requires projected values of RT_{PTS} be provided for each RV beltline material. For the APR1400 design, the beltline material to be considered should include, at a minimum, the lower shell section of the RV (adjacent to the active height of the core) and the two circumferential welds that connect the lower shell to the bottom head and to the upper shell section (nozzles section). This issue was discussed with the applicant in a public meeting on June 30, 2015, and the applicant responded that the DCD will be revised to provide all values used to calculate the RT_{PTS} for all RV beltline materials. The applicant also provided a markup copy of the proposed changes to the DCD in a letter dated July 17, 2015. Using the projected neutron fluence values at 60 years, the staff verified that the limiting RT_{PTS} values for all RV forgings and welds satisfied the PTS screening criteria of 10 CFR 50.61. On this basis, the staff determined that the applicant's response is acceptable because the RT_{PTS} values for all RV beltline materials meet the requirements of 10 CFR 50.61. Therefore, the staff determined that the applicant has appropriately addressed the issue and confirmed that the proposed changes were included in the DCD. In addition, DCD Tier 2 Section 5.3.2.3, "Pressurized Thermal Shock," indicates that a COL applicant that references the APR1400 DC will provide plant-specific RT_{PTS} values in accordance with 10 CFR 50.61 for RV beltline materials.

5.3.2.4.3 *Charpy Upper-Shelf Energy*

DCD Section 5.3.2.4, "Upper-Shelf Energy," states that initial USE is determined from charpy V-notch tests on unirradiated specimens in accordance with ASME Section III, NB-2320. The applicant also stated that the change in USE due to radiation embrittlement can be predicted in accordance with RG 1.99. With an initial USE of 75 ft-lbs (102 Joules), the applicant determined that the end-of-life (EOL) upper-shelf energy (USE) for the RV materials would be greater than 50 ft-lbs (68 Joules). Using the applicant's projected neutron fluence and the copper contents for the base metal and welds, the staff verified that the reduction in USE resulted in EOL USE values that were greater than 50 ft-lbs (68 Joules). On this basis, the staff determined that the applicant's USE values meet the requirements of 10 CFR Part 50, Appendix G and are therefore acceptable because the USE values exceed the minimum USE values provided in the regulation.

5.3.2.5 *Combined License Information Items*

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

Table 5.3-1 APR1400 Combined License Information Items Identified in the DCD

Item No.	Description	Section
COL 5.3(2)	The COL applicant is to develop P-T limit curves based on plant-specific data.	5.3.2.1
COL 5.3(3)	The COL applicant is to verify the RT_{PTS} value and the USE at EOL based on plant-specific material property and neutron fluences.	5.3.2.3, 5.3.2.4

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

5.3.2.6 *Conclusion*

Pressure-Temperature Limits

The staff concluded 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 RV beltline materials during operation will be verified through a material surveillance program developed in compliance with 10 CFR Part 50, Appendix H by a COL applicant. 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, 10 CFR Part 50, Appendix G, and GDC 1, GDC 14, GDC 31, and GDC 32 with respect to P-T limits.

Pressurized Thermal Shock

The staff concluded that the APR1400 RV meets the relevant requirements of 10 CFR 50.61, because calculations show that the RV beltline materials will be substantially below the PTS screening criteria 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 RV 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 RV materials in accordance with the acceptance criterion specified in 10 CFR Part 50, Appendix G, paragraph IV.A.1.a. Therefore, the staff concluded that the APR1400 RV materials meet the relevant requirements of 10 CFR Part 50, Appendix G.

5.3.3 Reactor Vessel Integrity

5.3.3.1 *Introduction*

While most of the features and topics addressed in this section are being reviewed separately in other sections of this report, the integrity of the RV 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 RV integrity.

5.3.3.2 *Summary of Application*

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

DCD Tier 2: The applicant has provided a DCD Tier 2 description of RV integrity in Section 5.3.3, “Reactor Vessel Integrity,” summarized here, in part, as follows:

The RV is designed to operate for 60 years. The RV consists of three cylindrical shell sections (upper, intermediate, and lower) and a bottom head. These sections are welded together to form the complete RV cylindrical shell assembly. The closure head is fabricated separately and joined to the RV using 54 closure stud bolts. The RV contains four primary inlet and two primary outlet nozzles. The closure head contains 101 CEDM nozzles and two heated junction thermocouple nozzles, and the bottom head contains 61 ICI penetration nozzles and four external shear key supports that mate with the keyways in the RV support column base plate. The RV is supported by vertical columns under each of the four RV inlet nozzles.

The RV closure head, shell sections, bottom head, and major nozzles are fabricated from SA-508, Grade 3, Class 1. The design of the RV permits all required inspections to be performed and does not preclude access to areas requiring ISI in conformance to ASME Section XI.

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

TS: There are no TS 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 NUREG-0800, Section 5.3.3 and are summarized below. Review interfaces with other SRP sections can be found in NUREG-0800, Section 5.3.3.

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

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

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

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

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

5.3.3.4 *Technical Evaluation*

Although the staff reviewed most areas separately in accordance with other SRP sections, the importance of the RV integrity warranted a special summary review of all factors relating to RV integrity. SRP Section 5.3.3 provides the acceptance criteria and references that form the bases for this evaluation. The staff reviewed DCD Tier 2 Section 5.3.3 and the following information that is discussed in other sections of the DCD.

- ASME Code classification and Code Edition and Addenda of Record for the RV (Section 5.2.1.1).

- Materials, fabrication methods, NDE, and threaded fasteners used for the RCPB (Section 5.2.3).

- Preservice and ISI, system pressure testing, and provisions for accessibility to inspect the RCPB (Section 5.2.4).

- Materials specifications, special fabrication methods, special NDE methods, fracture toughness testing, and surveillance of the RV (Section 5.3.1).

- P-T limits, PTS and USE (Section 5.3.2).

Based on the review of the information provided in the DCD, the staff determined that the RV will be designed, fabricated, inspected, and tested to the high standards of quality required by the ASME Code, and meets the requirements of GDC 1, GDC 30, GDC 31, GDC 32, 10 CFR 50.55a, 10 CFR Part 50, Appendix G, and 10 CFR Part 50, Appendix H. Meeting the aforementioned requirements provides reasonable assurance that the integrity of the RV will be maintained. A detailed discussion of the staff's findings is documented in the SERs associated with the DCD Tier 2 Sections listed above. Other areas of special interest to the staff in determining that the integrity of the APR1400 RV is ensured are discussed below.

5.3.3.4.1 *Accessibility to Inspect the Reactor Vessel*

To provide assurance that the integrity of the RV is maintained throughout the life of the APR1400 plant, the RV must be designed and provided with access to perform the required preservice and ISIs. DCD Tier 2 Section 5.2.4, "Inservice Inspection and Testing of the Reactor Coolant Pressure Boundary," describes the provisions in the APR1400 design that enable the performance of the required preservice and ISIs. With the internals removed, the entire inner surface of the RV is accessible for the required surface and volumetric examinations. With the internals in place, the nozzle-to-shell welds and inner radii of the outlet nozzles are accessible from the inside of the RV using remote automated equipment. An access tunnel is provided to allow personnel into the area below the bottom head. Insulation over the bottom head weld seams is removable. Access to the RV closure flange upper shell-to intermediate shell weld, and closure studs, nuts, stud holes, and ligaments is available when the closure head is removed. Access to the underside of the head is provided in the head laydown area. The staff's evaluation to ensure that ASME Class 1 components, including the RV, are designed and provided with access to perform the required examinations is documented in Section 5.2.4 of this SER. In SER 5.2.4, the staff determined that the APR1400 Class 1 components were designed and inspection access was provided in accordance with ASME Section XI and 10 CFR 50.55a. On this basis, the staff determined that the access provisions incorporated into the design of the APR1400 RV provides assurance that it's structural and leaktight integrity will be maintained.

5.3.3.4.2 *Special Considerations Related to Fracture Toughness*

In SER Section 5.3.1, "Reactor Vessel Materials," the staff determined that the materials of construction for the APR1400 RV were selected in accordance with the ASME Code. Although many materials are acceptable for RVs according to ASME Section III, the special considerations relating to fracture toughness and radiation effects effectively limit the basic

Code-approved materials that are currently acceptable for most parts of RVs to SA 533 Grade B, Class 1; SA 508 Grade 2, Class 1; and SA 508 Grade 3, Class 1. The APR1400 design utilizes SA 508 Grade 3, Class 1 for the RV, and is therefore, acceptable.

5.3.3.4.3 *Shipping and Installation*

With regard to shipment and installation, the integrity of the RV is maintained by ensuring that the as-built characteristics of the RV are not degraded by improper handling. DCD Tier 2 Section 5.3.3.5, "Shipment and Installation," states that the requirements of ASME NQA-1 are followed for the packaging and shipment of the RV. The RV is prepared to be shipped by barge or rail to a plant site while mounted on the shipping skid used for installation. All surfaces and covers are sprayed with a strippable coating or wrapped with shrink-wrap to protect against corrosion during shipping and installation. All openings are sealed to provide protection against dust, moisture, and/or detrimental materials during shipment. A desiccant is also applied to the interior of the closure head in the area surrounding the nozzles. The applicant also described how the strippable coatings will be removed after arrival on site to facilitate future inspections. The information provided in the DCD 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 to satisfy the requirements of 10 CFR Part 50, Appendix B, and Criterion XIII.

5.3.3.5 *Combined License Information Items*

No additional information is required to be provided by a COL applicant in connection with Section 5.3.3, "Reactor Vessel Integrity," of the APR1400 DCD. In DCD Tier 2 Section 5.3.3.7, "Inservice Surveillance," the applicant has provided COL Information Item 5.3(4), stating that the COL applicant is to develop and provide the ISI and testing program for the RCPB in accordance with ASME Section XI and 10 CFR 50.55a. This COL item is discussed in SER Section 5.2.4, which describes the staff's evaluation of the ISI program for the RCPB.

5.3.3.6 *Conclusion*

The staff concluded that the structural integrity of the APR1400 RV is acceptable because it meets the applicable requirements of 10 CFR Part 50, Appendix A, GDC 1, 4, 14, 30, 31, and 32; 10 CFR Part 50, Appendices G and H; 10 CFR 50.61, and 10 CFR 50.55a. The basis for this conclusion is that the design, materials, fabrication, inspection, and quality assurance requirements of the APR1400 plant conforms to the applicable NRC regulations and the ASME Code. The APR1400 design meets the fracture toughness requirements of the regulations and ASME Section III, including requirements for surveillance of RV material properties throughout its service life. In addition, COL applicants will establish operating limitations on temperature and pressure in accordance with the regulations and the ASME Code.

5.4 Component and Subsystem Design

The review of reactor thermal-hydraulic systems includes the review of the various components and subsystems associated with the RCS. RCS design bases, descriptions of design features and the associated operation, and necessary tests and inspections for these components and subsystems (including radiological considerations from the viewpoint of how radiation affects operation, and the viewpoint of how radiation levels affect the operators and their capabilities of operation and maintenance) are to be evaluated for the following subsystems and components: RCPs, SGs, RCS piping, SCS, pressurizer, PRT, RCS high-point vents, main steamline flow restriction, pressurizer pilot operated safety relief valves, and RCS component supports. In its DCD Tier 2 description in Section 5.4, the applicant provided information regarding the performance requirements and design features of these subsystems and components. The descriptions of the design bases, fabrication and inspection, and various operational conditions are provided for the RCPs in Section 5.4.1, SGs in Section 5.4.2, “Steam Generators,” reactor coolant piping in Section 5.4.3, “Reactor Coolant Piping,” SCS in Section 5.4.7, “Shutdown Cooling System,” pressurizer in Section 5.4.10, “Pressurizer,” PRT in Section 5.4.11, “Pressurizer Relief Tank,” RCS high-point vents in Section 5.4.12, “Reactor Coolant System High Point Vents,” pressurizer pilot operated safety relief valves in Section 5.4.13, “Main Steamline Flow Restrictor,” and RCS supports in Section 5.4.14,, “Safety and Relief Valves.” According to RG 1.206, the NRC reserved DCD Tier 2 Sections 5.4.4, 5.4.5, and 5.4.9, and the applicant noted that DCD Tier 2 Sections 5.4.6, “Reactor Core Isolation System (Boiling Water Reactors Only,” and 5.4.8, “Reactor Water Cleanup Systems (Boiling Water Reactors Only),” are not applicable to the APR1400. Note that RG 1.206, “Combined License Applications for Nuclear Power Plants,” and NUREG-0800 do not have a Section 5.4.10 and have no other section for the pressurizer. Therefore, the applicant used Section 5.4.10 of the APR1400 DCD to provide the information on the pressurizer.

5.4.1 Reactor Coolant Pumps

5.4.1.1 *RCP Flywheel*

5.4.1.1.1 *Introduction*

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

The staff’s review of the APR1400 DCD, related to this section, ensures that the potential for failures of the flywheels of RCPs are minimized and the materials are adequate to assure a quality product commensurate with the importance to safety.

5.4.1.1.2 *Summary of Application*

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

DCD Tier 2: The applicant has provided a DCD Tier 2 description in Section 5.4.1.1, summarized here in part, as follows:

RCP flywheels are designed, manufactured, and inspected to minimize the possibility of generating high-energy fragments (missiles) under normal conditions, anticipated transients, LOCA with the largest mechanical pipe break remaining after application of leak before break as described in Subsection 3.6.3, “Leak-Before-Break Evaluation Procedure,” and a SSE, consistent with the intent of the guidelines set forth in Section 5.4.1.1, “Pump Flywheel Integrity,” of NUREG-0800 and RG 1.14,, “Reactor Coolant Pump Flywheel Integrity.” The flywheel uses a shrink fit design to couple it to the shaft. Flywheels are tested at 125 percent of the normal speed of the motor.

The flywheel integrity analysis is summarized in the APR1400-A-M-NR-14001, “KHNP APR1400 Flywheel Integrity Report,” issued November 2014, which performed a stress analysis at standstill, normal and overspeed conditions, and a fracture mechanics analysis to predict the critical speed for fracture of the flywheel.

The flywheel material is a quenched and tempered forging with the German material designation 26NiCrMoV14-5, which is a high strength ductile forged material with mechanical properties equal to or exceeding SA-508 Class 2, which is a typical U.S. forged flywheel material. This material is its resistance to non-ductile-type failures and has adequate fracture toughness properties under operating conditions. A surface examination of all exposed surfaces and a 100 percent volumetric examination by ultrasonic methods are conducted at

approximately 10-year intervals during the plant shutdown coinciding with the ISI schedule as required by ASME Section XI. The flywheel is accessible for 100 percent in-place volumetric ultrasonic inspections.

ITAAC: APR1400 DCD, Revision 0, DCD Tier 1 Section 2.4, “Reactor Systems,” Table 2.4.1-4, “Reactor Coolant System ITAAC,” provides an ITAAC commitment No. 9.b, which requires that each reactor coolant pump flywheel be tested at a minimum of 125 percent of operating speed.

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

Technical Reports: KHNP APR1400-A-M-NR-14001, (WCAP-17942), Revision 3, issued July 2017.

5.4.1.1.3 *Regulatory Basis*

The relevant requirements of the Commission’s regulations for this area of review, and the associated acceptance criteria, are given in Section 5.4, “Reactor Coolant System Component and Subsystem Design,” Section 5.4.1.1, “Pump Flywheel Integrity (PWR),” of NUREG-0800 and are summarized below. Review interfaces with other SRP sections can be found in Section 5.4 and Section 5.4.1.1 of NUREG-0800.

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

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

Acceptance criteria adequate to meet the above requirements include:

RG 1.14, as it relates to the RCP flywheel design, materials selection and fabrication, PSI program, ISI program, and overspeed test of each pump flywheel assembly.

5.4.1.1.4 *Technical Evaluation*

The staff reviewed APR1400 DCD, Revision 0, Tier 2, Section 5.4.1.1, describing the materials used in the fabrication of the RCP flywheel, so that the structural integrity of the RCP 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. The following evaluation addresses the acceptance criteria outlined in SRP Section 5.4.1.1.

<u>Material Selection, Fabrication and Fracture Toughness</u>

APR1400 DCD Tier 2 Section 5.4.1.1.2, “Fracture Toughness,” states that the RCP flywheel material is a quenched and tempered forging with the German material designation 26NiCrMoV14-5, which is a high strength ductile forged material with mechanical properties equal to or exceeding SA-508 Class 2, which is a typical U.S. forged flywheel material. This material is resistant to non-ductile-type failures and has adequate fracture toughness properties under operating conditions. The flywheel material is produced by the vacuum melting and degassing process that minimizes flaws and therefore improves its fracture toughness. In addition, the APR1400 DCD incorporates the cut surfaces guidance of RG 1.14 in that it specifies that all cut surfaces are to be removed by machining to a depth of 1/2 inch (13 mm) minimum below the cut surface to minimize any loss of fracture toughness during fabrication. DCD Tier 2 Section 5.4.1.1.1, “Material Selection and Fabrication,” also discusses the guidance in RG 1.14 on the prohibition of welding on the flywheel which will preserve the material properties, especially toughness. The staff determined that the design material and manufacturing methods are acceptable because the material is produced by vacuum melting and degassing, which is an acceptable method of producing material of improved purity and toughness. Also, the staff determined 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.

DCD Tier 2 Section 5.4.1.1.2 provides information about the fracture toughness of the flywheel material, which states that:

> *K_{IC} of the flywheel material at the normal operating temperature of the flywheel is greater than 165 MPa√m (150 ksi√in). Conformance is demonstrated by an indirect test. Justification is provided to establish the equivalence of fracture toughness in the proposed flywheel material and certain steels (ASME SA-533-B Class 1, ASME SA-508 Class 2, ASME SA-508 Class 3, and ASME SA-516 Grade 65). The RT_{NDT} (nil-ductility reference temperature) of the flywheel materials is determined in accordance with NB-2320 and NB-2330 of the ASME Section III.*

In order for the staff to determine whether the APR1400 design meets the criteria in SRP 5.4.1.1 and RG 1.14 with regard to RCP flywheel integrity, as it relates to fracture toughness. The staff issued RAI 341-8410, Question 05.04.01.01-1 (ML15352A308), requesting that the applicant provide the following information:

a. Provide the method used to determine the fracture toughness.

b. Provide the justification that the flywheel material, 26NiCrMoV14-5, is equivalent to the steels specified above so that an indirect method of determining fracture toughness can be used. Otherwise, a direct method should be used, since this is the preferred method as stated in SRP 5.4.1.1, paragraph II.2, “SRP Acceptance Criteria.”

c. Provide the RT_{NDT} of the flywheel material.

d. Provide operating experience of the flywheel material, 26NiCrMoV14-5.

In its response to RAI 341-8410, Question 05.04.01.01-1 (ML16120A475), the applicant stated that the indirect method will be used to determine fracture toughness of the flywheel material. However, the applicant did not provide adequate justification that the flywheel material, 26NiCrMoV14-5, is equivalent to the steels specified in APR1400 DCD Tier 2 Section 5.4.1.1.2, and NUREG 0800, Section 5.4.1.1, in order to use an indirect method of determining the fracture toughness. RG 1.14 states,

> *…past evaluations have shown that ASME SA-533-B Class1 and SA-508 Classes 2 and 3 materials generally have suitable toughness for typical flywheel applications provided stress concentrations are kept within reasonable limits and the reference temperature RT_{NDT}, determined in accordance with Article NB-2331(a) of Section III of the ASME Code, is at least 50°C (90°F) below the lowest temperature at which operating speed is achieved. For other materials that may be considered for flywheels, the strength and toughness properties should be evaluated and justified for this application.*

The staff issued follow-up RAI 503-8641, Question 05.04.01.01-7 (ML16188A055), requesting the applicant to revise the information in the APR1400 DCD to document an acceptable approach for determining the fracture toughness of the RCP flywheel materials used in the APR1400 design. An example of an acceptable approach discussed in a June 29, 2016, public meeting is to specify the direct method of determining fracture toughness, and provide either the fracture toughness value using the direct method or include fracture toughness as an ITAAC in Table 2.4.1-4, of APR1400 DCD Tier 1. Therefore, RAI 503-8641, Question 05.04.01.01-7, was being tracked as an open item.

In its response to RAI 503-8641, Question 05.04.01.01-7 (ML16256A805), the applicant stated that the direct method will be used to determine that fracture toughness of the flywheel material is greater than 165 MPa√m (150 ksi√in). The staff finds that the direct method is consistent with NUREG-0800, Section 5.4.1.1 and RG 1.14, and therefore is an acceptable approach for this material. Therefore, this open item is considered resolved and closed. The staff confirmed that DCD Tier 2, was revised as committed in the applicant's response to RAI 503-8641, Question 05.04.01.01-7. Therefore, RAI 503-8641, Question 05.04.01.01-7, is resolved and closed.

Also, in response to RAI 341-8410, Question 05.04.01.01-1 (ML16120A475), the applicant provided operating experience of the flywheel material. However, the operating experience cited for the material includes spare flywheels that have not seen service or plants that have not been operated. The staff issued follow-up RAI 503-8641, Question 05.04.01.01-8 (ML16188A055), requesting the applicant to clarify how this information is relevant to demonstrating the performance of the proposed flywheel material under the APR1400 operating conditions. The operating experience that is cited to justify the use of the specific material, 26NiCrMoV14-5, having a yield strength of 640 N/mm^2 (92,825 psi) should be the same material (including yield strength) cited in the APR1400 design. Therefore, RAI 503-8641, Question 05.04.01.01-8, was being tracked as an open item.

In its response to RAI 503-8641, Question 05.04.01.01-8 (ML16256A805), the applicant provided operating experience that demonstrates the performance of the proposed flywheel material under the APR1400 operating conditions. The staff finds the operating experience acceptable since it provides information that the proposed material with a yield strength of 640 N/mm^2 (92,825 psi) has shown no known integrity issues for the past forty years in operation. Therefore, RAI 503-8641, Question 05.04.01.01-8, is resolved and closed.

<u>Flywheel Design</u>

The flywheel is required to be designed to withstand normal conditions, anticipated transients, the design basis LOCA, and the SSE without loss of structural integrity, and the potential of generating a missile. APR1400 DCD, Section 5.4.1.1.3, "Design," states that the flywheel

analysis is summarized in APR1400-A-M-NR-14001 and includes a stress analysis at standstill, normal and overspeed conditions and a fracture mechanics analysis to predict the critical speed for fracture of the flywheel.

APR1400-A-M-NR-14001, states that the flywheel is shrink-fit onto a hub, and this hub is shrink-fit onto the shaft. However, APR1400 DCD Tier 2 Section 5.4.1.1 states that the "flywheel uses a shrink fit design to couple it to a shaft." The staff issued RAI 341-8410, Question 05.04.01.01-2 (ML15352A308), requesting the applicant to provide additional information regarding the consistency of the APR1400 DCD Tier 2 Section 5.4.1.1, "Pump Flywheel Integrity," with the description of the flywheel design in the technical report.

In its response to RAI 341-8410, Question 05.04.01.01-2 (ML16120A475), the applicant provided a proposed change to the APR1400 DCD Tier 2 Section 5.4.1.1 which states that the outer flywheel is shrink-fit to the hub, and the hub is shrink-fit to the shaft. The staff determined that this is acceptable since this is consistent with the flywheel design in the technical report. The staff confirmed that DCD Tier 2, was revised as committed in the response to RAI 341-8410, Question 05.04.01.01-2. Therefore, RAI 341-8410, Question 05.04.01.01-2, is resolved and closed.

For the excessive deformation (including shrink-fit stresses), fatigue evaluation and the total stresses in the flywheel at normal and design overspeed conditions, the staff could not complete its review, since additional information is required. The staff issued RAI 341-8410, Question 05.04.01.01-3 (ML15352A308), requesting that the applicant provide the following information concerning APR1400-A-M-NR-14001:

a. Paragraph 2.4 of the technical report specifies that: "the total stress in the flywheel at standstill and normal operating speed does not exceed one-third of the ultimate tensile strength." In addition, paragraph 5.11 specifies that "the total stresses in the flywheel at standstill and normal operating speed shall not exceed one-third of the minimum specified yield strength" and that the "total stresses is 266.647 N/mm^2 (38,674 psi)." The staff identified inconsistencies in Paragraphs 2.4 and 5.11 of the technical report. The total stresses should not exceed one-third of the minimum specified yield strength as stated in SRP 5.4.1.1 paragraph II.4.a.

   Section 5.11, "Shrink Fit Requirements," of the technical report states that "one-third of the ultimate strength of 800/3 (N/mm^2) = 38,677 psi." The staff noted that one-third of the ultimate strength is actually 266.599 N/mm^2 (38,667 psi), and therefore, the calculated total stresses of the flywheel at normal operating speed of 266.599 N/mm^2 (38,667 psi) exceeds 266.599 N/mm^2 (38,667 psi). Therefore, the calculated stresses do not meet the one-third of yield strength or one-third of ultimate tensile strength acceptance criteria. Therefore, the staff requested additional information as to how the flywheel design meets the guidance and acceptance criteria of SRP 5.4.1.1.Paragraph 5.15 of the technical report performed the fatigue crack growth rate for the flywheel, but not for the hub. In addition, paragraph 5.15 of the technical report specifies that "the fatigue crack growth due to 6,000 cycles from standstill to normal operation can be predicted by the fatigue crack growth rates available in reference 5 [ASME Code, Section III, Division 1, Appendix A]." The staff requested that the applicant explain how the technical report provides acceptable fatigue crack growth rates for the flywheel material. In addition, this analysis should also be done on the hub since it is a critical part that attaches the flywheel to the shaft, and a potential hub failure could release the flywheel as a missile.

b. Submit Reference 6 (Siemens Document, 4D5.0170.83-575711F, Revision F, “Flywheel Calculation,” dated May 30, 2011) from the technical report to support the staff’s review of the APR1400 design.

In its response to RAI 341-8410, Questions 05.04.01.01-3a and 05.04.01.01-3b (ML16120A475), the applicant stated that in order to maintain a shrink-fit stress up to 150 percent of normal operation, the criterion on combined stresses at normal operation was chosen to limit the maximum stress to one-third of the minimum ultimate strength, instead of one-third of the minimum yield strength. However, there was no basis provided for using the design acceptance criteria of one-third ultimate strength in lieu of one-third yield strength for the flywheel design stress limit. The use of one-third of the yield strength as a design acceptance criteria has been documented by the staff in SRP Section 5.4.1.1 and RG 1.14, as providing an acceptable level of safety for this component. The use of one-third of the ultimate strength of the material as the basis for the flywheel design stress limit is unacceptable absent a technical basis demonstrating why the use of such a criteria will provide for an acceptable level of safety against flywheel failure. The staff issued follow-up RAI 503-8641, Question 05.04.01.01-9 (ML16188A055), requesting the applicant to revise the APR1400 DC to apply a RCP flywheel stress limit of one-third of the yield strength of the material, or provide a technical justification regarding why the use of one-third of the ultimate strength as the design stress limit will provide an acceptable level of safety against potential failure of the flywheels. Examples of an acceptable approach discussed in the June 29, 2016, public meeting included specifying a yield strength of 800 N/mm^2 (116 ksi) that was in original flywheel analysis (dated August 13, 2014) in lieu of the current specified yield strength of 640 N/mm^2 (92.825 ksi) or providing sufficient analytical basis to demonstrate that the change in the probability of creating a missile by using ultimate strength as the design stress limit, in lieu of the yield strength, is small. The staff issued follow-up RAI 503-8641, Question 05.04.01.01-9 (ML16188A055), requesting the applicant to address this issue. If the use of the one-third ultimate strength criteria is to be justified, as noted above, the acceptance criteria in the APR1400-A-M-NR-14001, “KHNP APR1400 Flywheel Integrity Report,” Revision 0, dated November 24, 2014, needs to be revised. Therefore, RAI 503-8641, Question 05.04.01.01-9, was being tracked as an open item.

In its responses to RAI 503-8641, Question 05.04.01.01-9(ML17233A366 and ML17240A437), the applicant stated thatAPR1400-A-M-NR-14001, was revised to apply a flywheel stress limit of one-third of the material’s yield strength, which is 640 N/mm^2 (92.825 ksi), which supersedes the previous RAI response (ML16256A805). In addition, in its revised response to RAI 503-8641, Question 05.04.01.01-9 (ML17240A437), the applicant revised APR1400 DCD, Section 5.4.18 to include Revision 3 of APR1400-A-M-NR-14001, issued July 2017, which supersedes Revision 0 of APR1400-A-M-NR-14001, dated November 24, 2014. Upon reviewing Revision 3 of APR1400-A-M-NR-14001, issued July 2017, the staff notes that the design specification for the flywheel was modified by optimizing the shrink-fit stresses to meet the normal operating stress limit of one-third of yield strength. Based on this redesign, APR1400-A-M-NR-14001 was modified extensively to demonstrate that the design stress of the flywheel at design speed does not exceed one-third of the yield strength of the material, consistent with Section 5.4.1.1 of NUREG-0800. The staff found APR1400-A-M-NR-14001, Revision 3, issued July 2017, which was submitted by the applicant in its letter dated August 28, 2017, acceptable and is discussed below. Therefore, RAI 503-8641, Question 05.04.01.01-9, is resolved and closed.

Revision 3 of APR1400-A-M-NR-14001, issued July 2017, specified that the design specification was revised to optimize the shrink-fit stresses, deleted the requirement of total stresses at joint release since it is not applicable, and used yield strength of the material as the design stress limit in lieu of the ultimate strength of the flywheel material. The staff verified that the calculated stresses of the flywheel during normal operating conditions are below one-third of the yield strength, and that the stresses during design overspeed conditions are less than the two-thirds of the yield strength of the flywheel material.

For the ductile fracture analysis, APR1400-A-M-NR-14001, Revision 3, uses the elastic stress analysis method of the ASME Code, Section III, Article F-1330 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 should be equal to 0.7 of the minimum specified ultimate tensile strength of the flywheel material. The staff verified that the minimum calculated limiting speed (2748 rpm) assuming a 13 mm (0.50-inch) crack is at least twice the normal operating speed (1200 rpm). The staff determined that the critical speed for ductile fracture meets the criterion in RG 1.14. In addition, the staff confirmed that the critical speed for ductile fracture (2748 rpm) is greater than the LOCA overspeed of 1500 rpm. Therefore, the staff determined that the ductile fracture analysis in APR1400-A-M-NR-14001, Revision 3 is acceptable.

For the non-ductile fracture analysis, the report uses the linear elastic stress analysis method of the ASME Code, Section III to predict the critical speed for non-ductile fracture of the flywheel. The analysis in the report uses the minimum fracture toughness of 165 MPa√m (150 ksi√in) for the flywheel material and a crack depth of 13 mm (0.50 inch), to predict that the calculated critical speed is 3203 rpm. Half of this speed, approximately 1601 rpm, is higher than the operating speed of 1200 rpm. Thus, the calculated critical speed meets the pertinent criterion of RG 1.14 and is therefore, acceptable.

In addition, APR1400-A-M-NR-14001, Revision 3 demonstrates that the normal speed of the flywheel (1200 rpm) is less than one-half of the critical speeds for the ductile fracture (2748 rpm), non-ductile fracture (3203 rpm) and excessive deformation (2938 rpm) failure modes. This report also confirms that the predicted LOCA overspeed (1500 rpm) is less than the critical speeds for the ductile facture, non-ductile fracture and excessive deformation failure modes. The staff reviewed the evaluation in APR1400-A-M-NR-14001, Revision 3 to the regulatory position of RG 1.14 for the flywheel design based on these critical speeds.

APR1400-A-M-NR-14001, Revision 3 also provided a fatigue crack growth that was determined from the crack growth rate in Appendix A of Section XI to the ASME Code. An initial crack length of 13 mm (0.50 inch) was assumed, with an assumed loading cycle of 6000 starts and stops for the life of the pump. A crack growth of 0.08128 mm (0.0032 inch) was calculated. Since the fatigue crack growth of 0.08128 mm (0.0302 inch) for a 60-year period is minimal, fatigue is not a major contributor for crack growth. Also, APR1400-A-M-NR-14001, Revision 3 demonstrated that at design overspeed, the critical crack length is over 152.4 mm (6 inches). Since, the inspection technique is capable of detecting flaws of 13 mm (0.50 inch), the staff concludes that the structural integrity of the flywheel is ensured because the ISI performed at regular intervals, and will detect the flaw before it will reach the critical crack size.

In its response to RAI 341-8410, Question 05.04.01.01-3c (ML16120A475), the applicant did not provide an analysis of the hub nor an acceptable justification for the fatigue crack growth rates used for the flywheel. The use of fatigue crack growth rates from ASME Code, Section XI, Appendix A, Paragraph A-4300 for the proposed flywheel material is unacceptable, as those fatigue crack growth rates are for SA-533 Grade B, Class 1 and SA-508, Class 3 steels, and no justification has been provided for using them for high alloy 26NiCrMoV14-5 material. The staff issued follow-up RAI 503-8641, Question 05.04.01.01-10 (ML16188A055), requesting the applicant to revise the technical report, or explain why no revision is needed, to include an appropriate analysis of the hub, including an appropriate fatigue evaluation for the applicable hub material, and revise the technical report to use appropriate fatigue crack growth rates for the proposed flywheel material. Therefore, RAI 503-8641, Question 05.04.01.01-10, was being tracked as an open item.

In its response to RAI 503-8641, Question 05.04.01.01-10, (ML16256A805), the applicant stated that Revision 3 of the APR1400-A-M-NR-14001, issued July 2017, will include an analysis of the hub that demonstrates the hub is in compression, and therefore, fatigue is not a concern. The staff reviewed the analysis of the hub and verified that the hub has compressive stresses during design speed due to the shrink fit of the outer flywheel. Therefore, the staff finds that the fatigue evaluation for the hub demonstrates that a flaw will not propagate due to fatigue, thereby minimizing the potential of generating missiles from the RCP flywheel. In addition, the applicant stated that only the generic crack growth rates of ASME Code, Section XI, Appendix A, Figure A-4300, “Reference Fatigue Crack Growth Curves for Carbon and Low Alloy Ferritic Steels Exposed to Air Environments (Subsurface Flaws),” was used in the analysis Since the applicant did not use specific stress intensity factors for Alloys SA-533 and SA-508 steels, which are not applicable to the 26NiCrMoV14-5 hub material, the staff finds that the use of reference crack growth rates in ASME Code to be acceptable because the crack growth rates are conservative for the hub material. Therefore, RAI 503-8641, Question 05.04.01.01-10, is resolved and closed.

<u>Pre-service Inspection</u>

DCD Tier 2 Section 5.4.1.1.4 states that each flywheel, prior to final assembly, is inspected by a 100 percent ultrasonic inspection in accordance with ASME Code, Section III, and a surface inspection using liquid penetrant or magnetic particle examination of areas of high stress concentrations. In addition, each flywheel receives a preservice baseline inspection and that the examination procedures and acceptance criteria are determined in accordance with ASME Code, Section III.

RG 1.14, Section C.4.a states that, following the spin test, each finished flywheel receives a check of critical dimensions and a non-destructive examination. This section of RG 1.14 also states that the non-destructive examination includes surface examination of areas of high stress concentrations using procedures in accordance with NB-2540, and acceptance criteria in NB-2545 or NB-2546 of Section III to the ASME Code, and a 100 percent volumetric examination using procedures and acceptance criteria specified in accordance with NB-2530 or NB-2540 of Section III to the ASME Code. However, APR1400 DCD, Section 5.4.1.4 does not provide this information. The staff issued RAI 341-8410, Question 05.04.01.01-4 (ML15352A308), requesting the applicant to provide additional information in APR1400 DCD Tier 2 Section 5.4.1.1.4.f, regarding the acceptance criteria that will be used for the inspection. In addition, the applicant was asked to specify in the DCD whether the maximum flaw size used as the acceptance criteria for this inspection is bounded by the flaw size used in determining the critical flaw size in APR14001-A-M-NR-14001.

In its response to RAI 341-8410, Question 05.04.01.01-4 (ML16120A475), the applicant did not revise the APR1400 DCD to specify the maximum flaw size used as the acceptance criteria for the PSI and that it is bounded by the flaw size used in determining the critical flaw size in

APR1400-A-M-NR-14001. The staff issued follow-up RAI 503-8641, Question 05.04.01.01-8 (ML16188A055), requesting the applicant to provide additional information on this subject. Therefore, RAI 503-8641, Question 05.04.01.01-11, was being tracked as an open item.

In its response to RAI 503-8641, Question 05.04.01.01-11, (ML16256A805), the applicant revised APR1400 DCD Tier 2 Section 5.4.1.1, to include the maximum flaw size used as the acceptance criteria for the PSI that is bounded by the flaw size used in determining the critical flaw size. Therefore, the staff finds that the PSI flaw size of less than 12.7mm (0.5 inch) to be acceptable since this bounds the flaw size used in APR1400-A-M-NR-14001, Revision 3. The staff confirmed that DCD Tier 2, was revised as committed in the response to RAI 503-8641, Question 05.04.01.01-11. Therefore, RAI 503-8641, Question 05.04.01.01-11, is resolved and closed.

In addition to the PSI discussed above, APR1400 DCD, Section 5.4.1.1.4 states that the flywheel is tested at design speed (125 percent of normal operating speed). However, APR14001-A-M-NR-14001 details that the flywheel is shrink-fit onto a hub, and this hub is shrink-fit onto the shaft. Therefore, since the hub can affect the integrity of the flywheel if it fails and release the flywheel as a potential missile, the test and inspections proposed in APR1400 DCD Tier 2 Section 5.4.1.1.4 for the flywheel should also apply to the hub.

The staff issued RAI 341-8410, Question 05.04.01.01-5 (ML15352A308), requesting the applicant to provide additional information regarding the tests and inspections proposed for the flywheel will also apply to the hub in APR1400 DCD Tier 2 Section 5.4.1.1.4. The applicant was also asked to address whether the hub can be inspected without the removal of the flywheel from the pump.

In its response to RAI 8410, Question 05.04.01.01-5 (ML16120A475), the applicant did not revise the APR1400 DCD to include that the hub will be inspected for both PSI and ISI in the same manner as the flywheel. In addition, the applicant's response also stated that the hub has oil channels that would make it difficult to perform UT inspection. The staff issued follow-up RAI 503-8641, Question 05.04.01.01-12 (ML16188A055), requesting the applicant to discuss this issue and to request additional information in APR1400 DCD Tier 2 Section 5.4.1.1.4, to address whether the hub will be inspected for both PSI and ISI in the same manner. The staff issued follow-up RAI 503-8641, Question 05.04.01.01-12 (ML16188A055) (and as discussed in the June 29, 2016, meeting), requesting the applicant to address the extent and acceptance criteria of UT inspections that could be performed or other alternatives of performing ISIs given these geometric interferences (oil channels).

In its response to RAI 503-8641, Question 05.04.01.01-12 (ML16256A805), the applicant revised APR1400 DCD Tier 2 Section 5.4.1.1.4, to include the extent and acceptance criteria for the PSI and ISI of the hub. The applicant stated that ultrasonic inspection of the hub will be performed prior to final machining for PSI. Surface examinations (liquid dye penetrant or magnetic particle testing) will be performed for the hub after final machining and for ISI due to the oil channels machined into the hub that could interfere with an ultrasonic examination. The staff finds that 100 percent ultrasonic inspection of the hub prior to final machining, and surface examinations (liquid dye penetrant or magnetic particle testing) after final machining for the hub to be acceptable since volumetric examination will be able to detect flaws below the crack size specified in the flywheel analysis, and a surface examination after final machining is adequate to detect surface breaking flaws.

The staff also finds the inspection performed for PSI on the flywheel, which includes 100 percent ultrasonic inspection in accordance with ASME Code, Section III, and a surface inspection using liquid penetrant or magnetic particle examination of areas of high stress concentrations for the flywheel acceptable since it has the capability to detect flaws that are bounded by the flywheel analysis in APR1400-A-M-NR-14001, Revision 3. The initial flywheel condition, along with the flywheel analysis, provides a baseline for future ISI to ensure that no flaws will propagate resulting in the fracture of the flywheel and generation of potential missiles. The staff confirmed that DCD Tier 2, was revised as committed in the response to RAI 503-8641, Question 05.04.01.01-12. Therefore, RAI 503-8641, Question 05.04.01.01-12, is resolved and closed.

Overspeed Testing

DCD, Section 5.4.1.1.4 states that the flywheel is tested at design overspeed. The design overspeed of the flywheel is determined to be 1500 rpm based on 125 percent of normal operating speed of 1200 rpm. The flywheel will be tested at 1500 rpm, and is therefore acceptable since it will be tested at the design overspeed prior to service to ensure the flywheel will maintain its structural integrity during an overspeed event, in accordance with the guidelines in Section C.3, "Testing," of RG 1.14. In addition, APR1400 DCD, Tier 1, Section 2.4, Table 2.4.1-4 provides ITAAC Commitment No. 9.b, which prescribes that the as-built RCP flywheel assembly be tested at 125 percent of operating speed. This ITAAC is considered acceptable since it provides the necessary information for the flywheel spin test to be performed, including the acceptance criteria.

DCD, Section 5.4.1.1.4.c states that the PSI is performed on the flywheel after final machining and the overspeed test. This is consistent with SRP Section 5.4.1.1, Paragraph II.3, which states that the preservice inspection results are performed after the spin test, and are used as baseline information for future ISIs. Therefore, the staff determined that this is acceptable.

Inservice Inspection

DCD, Section 5.4.1.1.4.d states that the ISI program of the flywheel includes ultrasonic examinations of the areas of high stress concentration at the bore and keyway at about three and one-third year intervals, during the refueling or maintenance shutdown coinciding with the ISI schedule as required by ASME Code, Section XI.

Although APR1400 DCD Tier 2 Section 5.4.1.1.4.d specifies an ISI of the keyway in the flywheel, APR1400-A-M-NR-14001 shows that the flywheel does not have a keyway.

The staff issued RAI 341-8410, Question 05.04.01.01-6 (ML15352A308), requesting the applicant to provide a drawing of the flywheel design, so that staff can confirm whether the flywheel has a keyway. The applicant was also asked to revise APR1400-A-M-NR-14001 and APR1400 DCD Tier 2 Section 5.4.1.1.4, as necessary, to be consistent with the actual flywheel design.

In its response to RAI 341-8410, Question 05.04.01.01-06 (ML16120A475), the applicant stated that the actual flywheel design has no keyway, and therefore the technical report includes the correct flywheel design. The response also provided a proposed change to the APR1400 DCD Tier 2 Section 5.4.1.1.4.d that removed inspection of the keyway, which the staff determined is acceptable since there is no keyway in the flywheel design. The staff confirmed that DCD Tier 2, was revised as committed in the response to RAI 241-8410, Question 05.04.01.01-6. Therefore, RAI 341-8410, Question 05.04.01.01-6, is resolved and closed.

In addition, APR1400 DCD, Section 5.4.1.1.4.e specifies that a surface examination of all exposed surfaces of the outer flywheel and a 100 percent volumetric examination by ultrasonic methods are conducted at approximately 10-year intervals during the plant shutdown coinciding with the ISI schedule as required by ASME Code, Section XI. The staff determined that this is acceptable since the ISI will consist of an ultrasonic inspection of areas of high stress concentrations and all exposed surfaces to detect flaws thereby ensuring the basis for safe operation of the RCP. The staff also determined that the ISI of the outer 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. However, the flywheel design consists of an outer flywheel and a hub. The hub is an integral and critical part of the flywheel that attaches the outer flywheel to the shaft, and a potential hub failure could release the flywheel as a missile The staff issued follow-up RAI 503-8641, Question 05.04.01.01-12 (ML16188A055), requesting the applicant to provide additional information on whether APR1400 DCD Tier 2 Section 5.4.1.1.4 provides that the hub will be inspected for both PSI and ISI in the same manner and that the applicant provide additional information on the extent and acceptance criteria of UT inspections that could be performed or other alternatives of performing ISIs given these geometric interferences (oil channels).

In its response to RAI 503-8641, Question 05.04.01.01-12 (ML16256A805), the applicant revised APR1400 DCD Tier 2 Section 5.4.1.1.4, to include the extent and acceptance criteria for ISI of the hub. The applicant stated that surface examinations (liquid dye penetrant or magnetic particle testing) will be performed on the hub for ISI, in lieu of ultrasonic examinations, due to the presence of oil channels machined into the hub that could interfere with the ultrasonic examination. In addition, the hub is in compression based on Revision 3 of the APR1400-A-M-NR-14001, issued July 2017. The staff finds that a surface examination (liquid dye penetrant or magnetic particle testing) for the hub to be acceptable since a surface examination is adequate to detect surface breaking flaws, and since there are compressive stresses in the hub which will inhibit flaw growth by fatigue or SCC. Therefore, the staff also determined that the ISI of the hub is acceptable, since it will 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. The staff confirmed that DCD Tier 2, was revised as committed in the response to RAI 503-8641, Question 05.04.01.01-12. Therefore, RAI 503-8641, Question 05.04.01.01-12, is resolved and closed.

5.4.1.1.5 *Combined License Information Items*

There are no COL information items applicable to this issue.

5.4.1.1.6 *Conclusion*

The staff concludes based on the above evaluation, that the material selection; fabrication practices; and PSI provide reasonable assurance that the materials used for the RCP flywheel structures will preclude inservice deterioration and maintain its structural integrity. In addition, the flywheel analysis results, along with the corresponding ISI, will ensure that flaws do not propagate beyond those predicted for the service life of the flywheel. Therefore, the staff concludes that Section 5.4.1.1, of the APR1400 FSAR is acceptable because it meets the requirements of 10 CFR 50.55a, GDC 1 and 4 of Appendix A to 10 CFR Part 50, and Sections III and XI of the ASME Code, in that the probability of a flywheel failure is sufficiently small, thereby minimizing the potential of generating missiles from the RCP flywheel.

5.4.1.2 *RCP Design*

5.4.1.2.1 *Introduction*

The RCPs provide forced circulation flow of the reactor coolant to transfer heat from the reactor core to the SGs. The RCPs circulate the water through the reactor vessel and SGs at a sufficient rate to ensure proper heat transfer and to 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.

5.4.1.2.2 *Summary of Application*

DCD Tier 1: There is no Tier 1 information regarding any specific design features of the RCPs beside those for the flywheel integrity described in above Section 5.4.1.1.

DCD Tier 2: The applicant provided a DCD Tier 2 description of the RCPs in Section 5.4.1.2, summarized here, in part, as follows:

There are four identical RCPs in the APR1400 design, two in each reactor coolant loop. The RCPs are vertical, single-stage, centrifugal pumps with mechanical shaft seals driven by synchronous squirrel-cage induction motors.

The motors are open and cooled by two air-to-water heat exchanges.

Each RCP assembly has one common vertical shaft line for the pump and motor with a water-lubricated radial bearing within the pump housing, radial and thrust bearings located in the motor stand, journal and thrust bearings in the motor housing and a flywheel located at the lower portion of the motor shaft.

The flywheel consists of an outer wheel shrunk-fit to an inner hub which, in turn, is shrunk fit to the motor shaft. The flywheel, in combination with the RCP rotating assembly, the motor rotor, and other rotating parts, provides sufficient rotational inertia for the RCPs to maintain a departure from nucleate boiling (DNB) margin during the gradual loss of forced RCS flow that occurs during RCP coastdown following a LOOP event.

Each motor has an anti-reverse rotation device to prevent impeller rotation in the reverse direction.

The applicant stated that internal oil systems lubricate the pump and motor bearings. External oil pumps are not needed during normal pump operation and coastdown. Bearing lubrication is accomplished by the internal pumping devices. Lubricating oil is water-cooled by cooling coils submerged in the oil sump.

The applicant provided a separate oil lift system for startup of the pump assembly. Interlocking devices prevent pump startup until oil lift flow is established.

The shaft seal assembly consists of two face-type, mechanical seals in series, with a controlled leakage bypass network to provide differential pressure equally across each seal, and also a third low-pressure vapor seal designed to withstand system operating pressure only when the pumps are not operating. The applicant designed each of the mechanical seals for the full pressure differential of 175.8 kg/cm^2 (2500 psi) and in combination with the controlled leakage bleed-off design feature, to reduce the pressure from the RCS normal operating pressure to the volume control tank pressure.

ITAACs: There is no specific ITAAC related to design and operation of the RCPs.

TS: The TS related to RCPs are given in the following DCD Tier 2 Chapter 16:

- Section 3.4.1, “RCS Pressure, Temperature, and Flow DNB Limits.”
- Section 3.4.4, “RCS Loops – MODES 1 and 2.”
- Section 3.4.5, “RCS Loops – MODE 3.”
- Section 3.4.6, “RCS Loops – MODE 4.”
- Section 3.4.7, ““RCS Loops – MODE 5, Loops Filled.”

5.4.1.2.3 *Regulatory Basis*

The relevant requirements of NRC regulations for this area of review and the associated acceptance criteria are given in NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition” (SRP), Section 5.4, “Reactor Coolant System Component and Subsystem Design,” and are summarized below.

GDC 1, as it relates to the quality standards and records applicable to the design, fabrication, erection, and testing of the RCP.

5.4.1.2.4 *Technical Evaluation*

The staff’s evaluation of RCP design features, as described in DCD Tier 2 Sections 5.4.1.2 through 5.4.1.4, are as follows:

RG 1.206 designates Section 5.4.1 for RCPs, but NUREG-0800 does not have a Section 5.4.1. However, the RCPs are identified in SRP Section 5.4, “Reactor Coolant System Component and Subsystem Design,” Section I, “Area of Review,” Part 1, “Reactor Coolant Pumps or Circulation Pumps [BWR],” and are classified within the various components and subsystems associated with the RCS. The specific areas of review of other SRP sections related to RCPs are provided below. Detailed information about the RCPs and the staff’s evaluation and conclusion regarding various APR1400’s common design characteristics and performance requirements, such as material selection and fabrication of RCS pressure boundary components, are discussed in the respective sections of this report related to the following SRP sections:

- Section 3.9.1, “Special Topics for Mechanical Components.”
- Section 3.9.2, “Dynamic Testing and Analysis of Systems, Components, and Equipment.”
- Section 3.9.3, “ASME Code Class 1, 2, and 3 Components, Component Supports, and Core Support Structures.”

- Section 3.9.6, “Functional Design, Qualification, and Inservice Testing Programs for Pumps, Valves, and Dynamic Restraints.”
- Section 3.10, “Seismic and Dynamic Qualification of Mechanical and Electrical Equipment.”
- Section 5.2.3, “Reactor Coolant Pressure Boundary Materials.”
- Section 5.2.4, “Reactor Coolant Pressure Boundary Inservice Inspection and Testing.”
- Section 7.2, “Reactor Trip System.”
- Section 7.3, “Engineered Safety Features.”
- Section 7.4, “Safe Shutdown.”
- Section 7.5, “Information Systems Important to Safety.”
- Section 9.2.2, “Component Cooling Water System.”
- Section 9.3.4, “Chemical and Volume Control System.”
- Section 15.3.1-15.3.2, “Loss of Forced Reactor Coolant Flow Including Trip of Pump Motor and Flow Controller Malfunctions.”
- Section 15.3.3-15.3.4, “Reactor Coolant Pump Rotor Seizure and Reactor Coolant Pump Shaft Break.”

The staff’s evaluation of the design features and performance requirements specific to the RCPs not included in the above listed sections of this SER is presented below.

Pump Seal Cooling

The RCP seal section consists of three seals. The applicant stated that the first and second primary seals are the RCS pressure boundary during pump normal operation, but when the pump completely stops a third secondary seal is the RCS pressure boundary. The applicant stated that it configures the first and second face type, mechanical seals, with a controlled leakage bypass flow network that equally divides the total pressure drop across each seal during normal pump operation. The third seal is a face type, vapor seal which can withstand system operating pressure when the pumps are not operating.

The applicant stated that it cools the RCP seals by: (1) seal injection water from the chemical and volume control system (CVCS) and (2) the component cooling water (CCW) through a high-pressure seal cooler.

The CVCS continuously supplies seal injection water to the reactor coolant pump seals, as required by the reactor coolant pump design. In addition to the centrifugal charging pump found in a typical PWR design, the APR1400 design includes a positive displacement auxiliary charging pump to provide a diverse means of seal injection to the RCPs if the normal means of seal cooling are lost.

The applicant stated that it supplies CCW to the pump bearing oil cooler, the motor air coolers, the motor bearing oil coolers, and the high-pressure seal cooler. The applicant stated that it will monitor reactor coolant leakage into the component cooling water system (CCWS) in the high-pressure seal cooler to alarm control room operators of a tube rupture condition so that they can take action to prevent further contaminating the CCWS.

In DCD Tier 2 Section 5.4.1.2, the applicant stated that if either seal injection water or the CCW is available the pump seal operation may continue indefinitely. Additionally, the staff noted that the APR1400 design includes a positive displacement pump to provide back-up injection to the RCPs in the event that the normal seal cooling method is lost. The staff could not find a detailed description of the seal injection flow paths into and within each of the seal stages in the DCD Tier 2 Sections 5.4.1, "RCPs," 9.2.2, "CCWS," or 9.3.4, "CVCS." Without such details, the staff could not validate that pump seal operation may continue indefinitely as stated above. At the staff's request in the non-Chapter 15 audit plan, the applicant presented a non-proprietary document which shows the seal injection flow paths including the existence of an auxiliary impeller, integral to the pump shaft, which provides the driving force for recirculation of seal water through the external high-pressure seal cooler. The staff issued RAI 307-7835, Question 05.04-1, Part 1, (ML15320A152), requesting the applicant to add those details to DCD Tier 2 Section 5.4.1, as part of the pump design description. In its response to RAI 307-7835, Question 05.04-1, Part 1, (ML16007A682), the applicant proposed to revise the DCD to add new Figure 5.4.1-2, "Flow Diagram for Reactor Coolant Pump Shaft Seal System," as well as clarifying details of the seal water flow path in DCD Tier 2 Section 5.4.1.2. The staff determined that the proposed changes are acceptable because the added details clearly explain how pump seal operation may continue indefinitely with either loss of seal water injection from the CVCS or loss of CCW to the high pressure cooler. The staff confirmed that DCD Tier 2, was revised as committed in the response to RAI 307-7835, Question 05.04-1. Therefore, RAI 307-7835, Question 05.04-1 is resolved and closed.

Also, in DCD Tier 2 Section 5.4.1.2, the applicant stated that it performed a series of tests and analyses and determined that a minimum margin of 12.2 °C (22 °F) exists between the seal cooling water outlet temperature and the seal cooling water temperature limit specified by the pump manufacturer. However, the staff could not find a summary of these performance tests and analyses within DCD Tier 2 Section 5.4.1. Without such information, the staff could not validate the stated temperature margin. The staff issued RAI 307-7835, Question 05.04-1, Part 3, (ML15320A152), requesting that the applicant provide additional information regarding a summary of the test results in DCD Tier 2 Section 5.4.1, and a reference to the document where these tests and analyses are described. In its response to RAI 307-7835, Question 05.04-1, Part 3, the applicant stated:

> *A summary report of the test results for the 22 °F temperature margin has been uploaded to [electronic reading room] ERR, titled "Justification that the APR1400 RCP seals will not exceed the exit temperature specification limit" which is proprietary and not intended to be included in DCD. This report provides 1) documentation of the basis for the 22 °F margin and 2) a justification that this margin is applicable to the APR1400 RCP seals. APR1400 RCP specification identifies the same 180 °F limit for the RCP outlet temperature as the System 80 RCP specification and similar loss of component cooling water and/or loss of seal injection tests have been performed on both pumps to sufficiently justify the 22°F margin for the APR1400 RCPs.*

The staff reviewed the proprietary documents (LTR-APR-15-09, Revision 0, "Justification that the APR1400 RCP Seals will not Exceed the Exit Temperature Specification Limit," issued November 2015, and 11A60-FS-DS480, Revision 03, "Design Specification for Reactor Coolant Pumps," issued March 2015) that were made available to the staff during an audit and agrees with the applicant's position that the provided test results based on RCP seal packages used in CE 80 design plants (Palo Verde Nuclear Generating Station) and in APR1400 design plants (Shin Kori 3) sufficiently justify the 22 °F (12.2 °C) margin for the APR1400 RCPs. Therefore, the staff determined that the applicant's response is acceptable. Therefore, RAI 307-7835, Question 05.04-1, Part 3, is resolved and closed.

Loss of Seal Injection

With regard to loss of RCP seal injection flow, in DCD Figure 5.1.2-2, "Reactor Coolant Pump Flow Diagram," the staff noted the use of a jet pump in the seal injection line from CVCS. However, DCD Tier 2 Section 5.4.1 does not include a description of this jet pump and its operational requirements. Without a detailed description of operational requirements for the jet pump, the staff could not clearly determine the effect of the seal injection flow loss, even when CCW is still available to the high-pressure seal cooler. The staff issued RAI 307-7835, Question 05.04-1, Part 4 (ML15320A152), requesting the applicant to provide the needed details. In its response to RAI 307-7835, Question 05.04-1, Part 4, (ML16007A682), the applicant proposed to add, in DCD Tier 2 Section 5.4.1.2, clarifying details of a cyclone filter/jet pump assembly which functions only as filter for particulates in the RCS fluid before it is allowed to enter the seal chamber. The staff determined that this response is acceptable because the added details provide a complete description of a component shown in DCD Figure 5.1.2-2. The staff confirmed that DCD Tier 2, was revised as committed in the response to RAI 307-7835, Question 05.04-1. Therefore, RAI 307-7835, Question 05.04-1, is resolved and closed

Loss of Component Cooling Water

With regard to loss of CCW to the high-pressure seal coolers, seal injection flow from the CVCS would provide sufficient cooling to the seals to allow safe operation of the RCP seals indefinitely.

The loss of CCW to the pump/motor bearing oil coolers would result in an increase in oil temperature and a corresponding rise in bearing metal temperature. In DCD Tier 2 Section 5.4.1.2, the applicant stated that the RCP can operate for at least 30 minutes without bearing seizure affecting normal RCP coastdown in the event of the simultaneous loss of CCW to all RCP and motor bearing assemblies but seal injection water is available to the seals. In the design, the applicant provided several transmitters to monitor CCW flow to motor bearing oil coolers. The applicant designed these transmitters to provide flow indication and actuate low-flow alarms in the control room. However, the staff considered the applicant's description of this event to be incomplete because CCW is common to both the CVCS and the pump/motor bearing oil coolers of the RCP. The staff issued RAI 307-7835, Question 05.04-1, Part 5 (ML15320A152), requesting the applicant to provide more detail on the following:

a) Since the CCWS provides cooling water to both the CVCS (mini flow HXs) and oil coolers for the pump and motor bearings, explain what is meant by loss of CCW, and

b) If "loss of CCW" includes cooling flow to CVCS, explain how seal injection flow to the RCPs will not be affected for "up to 30 minutes" to prevent seal damage.

In its response to RAI 307-7835, Question 05.04-1, Part 5 (ML16007A682), the applicant stated that it provided seal injection flow by one of two charging pumps or an auxiliary charging pump when the normal charging pumps are not available. The applicant further clarified that, unlike the charging pumps, the loss of CCW does not affect the auxiliary charging pump. Therefore, the applicant concluded that the auxiliary charging pump can supply seal injection flow to the RCP to prevent seal damage. However, the staff noted that the description of the CVCS in DCD Tier 2 Section 9.3.4.2, "System Description," states that the auxiliary charging pump requires a manual start by the plant operator. Therefore, the staff concluded that the auxiliary charging pump would not provide the seal injection water to the RCPs within 30 minutes after a loss of CCW to all RCP coolers. The staff issued follow-up RAI 443-8555, Question 05.04-2, Part 1 (ML16075A370), requesting the applicant to provide further clarifications regarding RCP seal water injection during this 30-minute period.

In its response to RAI 443-8555, Question 05.04-2 (ML16120A439), the applicant clarified that the charging pump operation does not rely on CCW for cooling because it is air cooled; therefore, it can remain in operation. In addition, the applicant stated the "loss of letdown may occur as a result of loss of CCW to the letdown heat exchanger and source for suction for charging pump is automatically changed to the boric acid storage tank (BAST) of which normal operating temperature is 60 ~ 120 °F." The staff determined the response is acceptable because the charging pump can continue to provide seal injection during the loss of CCW. Therefore, RAI 443-8555, Question 05.04-2, Part 1, is resolved and closed.

RCP Seal Protection

In the event that the CCW to all RCPs is isolated on a low-low CCWS surge tank level signal and the seal water injection from the CVCS is also lost, the applicant stated that it is possible to override this isolation signal by manual operation from the MCR to restore RCP seal cooling.

In the event where the Containment pressure reaches a high-high limit, a containment spray actuation signal (CSAS) isolates the RCP controlled bleedoff line to the CVCS. However, a diversion of the seal controlled bleedoff flow to the reactor drain tank (RDT) via CV-507 in the CVCS on receipt of a CSAS supports continuous RCP seal cooling operation.

Internal Motor Parts Cooling

The applicant stated that internal air flow cools the RCP motor internal parts. Fans located on each end of the rotor draws air in through cooling slots in the motor frame before the air is routed to the external water/air coolers supplied by the CCWS. Each motor has two such coolers, mounted diametrically opposed to each other. The applicant sized the coolers to maintain the outlet air temperature below the maximum containment ambient temperature.

Shaft and Frame Vibration Monitoring

The applicant stated that vibration sensors mounted on the RCP shaft and the motor stand continuously monitor the vibration levels. The applicant also described plans to mount probes on the rigid coupling to measure shaft displacement, and on the motor stand to measure the frame displacement. In addition, the applicant stated that it will install a seismoprobe on top of the motor. These installed probes will then transmit the measured signals of shaft vibration and frame vibration to the MCR.

Pump and Motor Bearings

A water lubricated radial bearing within the pump and radial and thrust bearings located in the motor stand support the pump shaft. The applicant stated that it will mount these bearings, together with an oil reservoir and associated oil cooler, as a single, easily removed assembly between the flexible coupling and the rigid coupling, to allow access to the seal assembly below without displacing the motor. However, the applicant stated that the removal of the motor and the motor stand is necessary to access the pump internals. Low oil level in the oil reservoirs triggers an alarm in the control room. Further, each bearing has built-in temperature detectors with a high-bearing-temperature setpoint that triggers an alarm in the control room.

Two journal bearings with their own separate oil reservoir and associated oil cooler support the motor shaft. Low oil level in the oil reservoirs triggers an alarm in the control room. Each bearing has built-in temperature detectors with a high-bearing-temperature setpoint that triggers an alarm in the control room.

In DCD Tier 2 Section 5.4.1.2, the applicant provided limited information as to the types of thrust bearings it used to support the pump shaft and/or motor shaft. At the staff's request in the non-Chapter 15 audit plan, the applicant presented a non-proprietary document which shows the needed details. The staff issued RAI 307-7835, Question 05.04-1, Part 8, (ML15320A152), requesting the applicant to add those details in DCD Tier 2 Section 5.4.1 as parts of the pump design description. In its response to RAI 307-7835, Question 05.04-1, Part 8, (ML16007A682), the applicant proposed to add these clarifying details to DCD Tier 2 Section 5.4.1.2. The staff determined that this response is acceptable for the above discussed reason. The staff confirmed that DCD Tier 2, was revised as committed in the response to RAI 307-7835, Question 05.04-1, Part 8. Therefore, RAI 307-7835, Question 05.04-1, Part 8, is resolved and closed.

Oil Lift System

The oil lift system furnishes high pressure oil to the pump assembly thrust bearings, thereby lifting the rotor and reducing bearing friction during pump startup. The applicant stated that it will furnish interlocking devices, which prevent pump startup until oil lift flow is established. The oil lift system shuts down automatically when the pump reaches full speed. Because an oil lift is not necessary during normal operation, an oil leak in this system does not cause a bearing failure.

Oil Collection System

The RCP assembly is equipped with an oil collection system for protection against oil spillage. The applicant designed this system to meet Seismic I Category requirements.

Anti-Reverse Rotation Device

The applicant stated that it provided each motor with an anti-reverse rotation device. Although the applicant provided the basis for including the anti-reverse rotation device as a part of the pump design, the staff could not find a description of this device in DCD Tier 2 Section 5.4.1. At the staff's request in the non-Chapter 15 audit plan, the applicant presented a non-proprietary document which shows details of the anti-reverse rotation device. The staff issued RAI 307-7835, Question 05.04-1, Part 6 (ML15320A152), requesting the applicant to add those details to DCD Tier 2 Section 5.4.1 as part of the pump design description. In its response to RAI 307-7835, Question 05.04-1, Part 6, (ML16120A439), the applicant stated:

The only functional requirement of the anti-reverse rotation device for APR1400 RCP Motor is to prevent rotation in the reverse direction without damage when reverse torques are induced as described in Section 5.4.1.2 of DCD. Safety and function of APR1400 are not affected by the type of an anti-reverse rotation device. The document that was reviewed by staff in the non-Chapter 15 audit is actually proprietary as opposed to nonproprietary. Thus the details on the design of the device are vendor specific data; therefore, APR1400 DCD will not incorporate the details on the device.

The staff agreed with the applicant's position that the safety and the function of the APR1400design are not affected by the type of an anti-reverse rotation device used as long as its functional requirements have been met; that is, for example, the motor will not rotate in the reverse direction if motor power leads are incorrectly wired, and there is no need to have this vendor specific information in the DCD. The staff determined that this response is acceptable. Therefore, RAI 307-7835, Question 05.04-1, Part 6, is resolved and closed.

Motor/Pump Sensors

The applicant provided the following RCP sensors for monitoring of the motor/pump performance:

- Stator winding temperature
- Lower and upper oil reservoir level (for motor bearings)
- Upper and lower journal bearing temperature (for motor bearing metal)
- Stator winding and motor bearing temperature recorder
- Motor vibration transmitters
- CCW flow and temperatures (outlets of motor oil coolers and high-pressure coolers)
- Upper and lower journal bearing temperature (for pump bearing metal)
- Upper thrust bearing temperature (for pump thrust bearing plate)
- Pump shaft vibration transmitter (on the rigid coupling)
- Frame vibration transmitter
- Pump speed
- Seal water inlet, 1st Stage seal, and 2nd Stage seal pressures
- Oil lift pump pressure and oil lift flow

Pump Performance

The applicant designed the RCP to maintain the required flow rate. Initial plant testing will confirm total flow delivery capability of the RCP. The applicant stated that it will check the RCP hydraulic performance and test the RCP motor mechanical performance at over-speed condition up to and including 125 percent of normal speed.

The applicant stated that it designed the shaft seal system with three seals; two primary face type, mechanical seals in series to support the pump normal operation and a third secondary face type, vapor seal to serve as a pressure boundary barrier only when the pump is not operating. The applicant routed the seal leak-off past the primary seals to the volume control tank (VCT) in the CVCS, and routed any seal leak-off passed the secondary seal to the reactor drain tank (RDT). Although each primary seal is picking up one-half of the pressure drop between the RCS pressure and the VCT pressure, the applicant designed the primary seals to withstand the full system pressure in case a failure of one seal occurs during normal plant power operation. The applicant also designed the secondary seal to withstand the full system pressure, but only when the pump is not running. The staff determined that the three-seal design is adequate because the features of the three-tier seal provide an effective barrier to minimize RCS leakage into the containment atmosphere.

RCP Seal Integrity during Station Black-Out (SBO)

The need for improved SBO performance led to selected improvements in the design of the seals used in the APR1400 design. The seal uses an enhanced manufacturing process and improved composition of elastomer materials to provide high temperature seal performance at 300 °C (572 °F) and 163.1 kg/cm²G (2,320 psig). Key materials used in the seals are silicon carbide/graphite compound material for the glide rings (primary seals) and specially treated (high-temperature-resistant) ethylene propylene diene monomer (EPDM) elastomers (secondary seals). Both materials are capable of withstanding long term exposure to a high temperature and pressure environment. The applicant included plans for a robust test program to verify satisfactory performance of the improved seal material under adverse SBO condition. At the staff's request, in the non-Chapter 15 audit plan, the applicant presented two proprietary documents which shows the basis for the test program and the associated test procedure. The staff issued RAI 307-7835, Question 05.04-1, Part 7 (ML15320A152), requesting the applicant to provide a summary of the test results to be included in DCD Tier 2 Section 5.4.1, and a reference to the document describing this program. In its response to RAI 307-7835, Question 05.04-1, Part 7 (ML16007A682), the applicant stated that the planned seal tests have not been completed at the time of the response, and that these test results are considered proprietary for the seal supplier and inappropriate to be included or referred to in the DCD. However, after the tests have been completed, the information will be available to staff audit. Also, the applicant committed to incorporate a general summary of tests into the DCD. The staff issued follow-up RAI 443-8555, Question 05.04-2, Part 2 (ML16075A370), to track the applicant's commitment as stated above regarding the seal test results. In its response to RAI 443-8555, Question 05.04-2 (ML16120A439), the applicant included a summary of the seal test results in the DCD and added the seal results document in the reference section. Based on the review of DCD Tier 2, the staff confirmed incorporation of the changes described above; therefore, RAI 443-8555, Question 05.04-2, Part 2, is resolved and closed.

Also, during a SBO event, the applicant stated that it maintained RCP seal integrity via seal water injection by the auxiliary charging pump of the CVCS powered from an alternate alternating current (ac) power source.

Coastdown Capability

In the event of a LOOP, the RCP coastdown characteristics are necessary to provide adequate flow to protect the reactor core. The RCP is designed with a flywheel (See Section 5.4.1.1 of this SER), which, along with the other rotating parts, provides high rotational inertia, thereby providing reactor core cooling during the coastdown period. Section 15.3, “Decrease in Reactor Coolant System Flow Rate,” of this SER describes the coastdown flow and core flow transients.

Rotor Seizure/Shaft Break Events

The seizure of either the upper or the lower thrust-journal bearings can cause an instantaneous seizure of the pump rotor. A manufacturing defect in the shaft can cause a pump shaft break. In both events, the flow reduction in the affected loop results in a reactor trip. Section 15.3 of this SER describes flow transients related to pump rotor seizure/shaft break events.

Initial Test Program

The staff issued RAI 279-8175, Question 14.02-41 and Question 14.02-42, (ML15303A547), requesting the applicant to address incomplete information in DCD Tier 2Sections 14.2.12.1.2, “Reactor Coolant System Test,” and 14.2.12.1.1, “Reactor Coolant Pump Motor Initial Operation Test,” related to preoperational test methods for the coupled pumps and the associated un-coupled motors, respectively, in DCD Tier 2 Section 14.2, “Initial Test Program.” Section 14.2 of this SER contains the staff’s evaluation of the responses to these two RAIs.

Technical Specifications

Chapter 16 of this SER contains the staff’s evaluation of TS requirements identified in Section 5.4.1.2(B) above regarding operation of the RCPs.

5.4.1.2.5 *Combined License Information Items*

There are no COL information items associated with Section 5.4.1.2, “Description,” of the APR1400 DCD.

5.4.1.2.6 *Conclusion*

The staff reviewed DCD Tier 2 Sections 5.4.1.2 through 5.4.1.4 in accordance with SRP Section 5.4, “Reactor Coolant System Component and Subsystem Design,” Revision 2. The staff determined that GDC 1 is satisfied because Sections 5.4.1.2 through 5.4.1.4 comply with the quality standards and records applicable to the design, fabrication, erection, and testing of the RCP.

5.4.2 Steam Generators

5.4.2.1 *Steam Generator Materials*

5.4.2.1.1 *Introduction*

The SGs transfer heat from the reactor core to the secondary system to produce the steam required for turbine operation. The reactor coolant is inside the SG tubes; therefore, the SG tubing is a large part of the RCPB.

5.4.2.1.2 *Summary of Application*

DCD Tier 1: In DCD Tier 1 Subsection 2.4.1, “Reactor Coolant System,” the applicant discussed the components of the RCS, including the two SGs. The Tier 1 information includes design information and ITAAC related to verification of that the SGs meet requirements for ASME Boiler and Pressure Vessel Code design, seismic design, instrumentation, and controls.

DCD Tier 2: In DCD Tier 2 Subsection 5.4.2.1, "Steam Generator Materials," the applicant described the SGs, including materials selection, materials fabrication and processing, cleanliness and compatibility with the primary and secondary coolants, flow-induced vibration, and design features for accessing the secondary side. The APR1400 plant design has two recirculating SGs with a combination of inverted U-bend tubes and square-bend tubes. The tube material is thermally treated nickel-base Alloy 690 (Alloy 690 TT, ASME SB-163), and each SG contains 13,102 tubes.

On the primary side, reactor coolant enters through the channel head inlet nozzle, flows through the tubes, and exits through the channel head outlet nozzle. Inside the tubes, the heat is transferred to the secondary water on the outside of the tubes to produce steam. Feedwater can be introduced through both a sparger ring near the top of the tube bundle and an economizer (preheater) section near the exit ("cold") end of the tubes. A recirculation nozzle above the tube bundle enables circulation during wet lay-up and the addition of chemical cleaning agents. The steam generated on the outside of the tubes and the steam-water mixture from the tube bundle flow into centrifugal moisture separators and corrugated plate steam dryers. The dry steam exits the SG through two steam outlet nozzles.

DCD Tier 2 Subsection 5.4.2.1 includes Subsection 5.4.2.1.2.1.1, "Flow-Induced Vibration of the Tube Bundle." The staff reviewed this information under NUREG-0800, SRP Section 3.9.2, "Dynamic Testing and Analysis of Systems, Components, and Equipment," and the staff's evaluation is discussed in that section of the SER.

ITAAC: As noted above, the ITAAC related to the SGs are listed in DCD Tier 1 Subsection 2.4.1.

TS: The APR1400 TS related to the SGs are located in DCD Tier 2, Chapter 16, Section 3.4, "Reactor Coolant System (RCS)"; Section," 5.5.9, "Steam Generator (SG) Program"; Section, 5.6.7, "Steam Generator Tube Inspection Report"; and Bases Section 3.4 (B 3.4), "Reactor Coolant Systems (RCS)." There are no TS specifically related to SG materials and design. TS related to the SG Program are discussed in the next section of this report, Section 5.4.2.2. The purpose of these TS is to maintain tube structural and leakage integrity.

5.4.2.1.3 *Regulatory Basis*

The relevant requirements of the Commission's regulations for this area of review, and the associated acceptance criteria, are given in SRP Section 5.4.2.1, "Steam Generator Materials," and are summarized below. Review interfaces with other SRP sections are listed in SRP Section 5.4.2.1.

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 operations, 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 RCS 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 AOOs.

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. Title 10 CFR 50.35, “Technical Specifications,” applies to the SG program in the TS.

8. Title 10 CFR 50.55a(c), 10 CFR 50.55a(d), and 10 CFR 50.55a(e), generally require certain groupings of components, including those comprising the pressure boundaries, to meet the requirements of Section III of the ASME Code.

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

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

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

Acceptance criteria adequate to meet the above requirements include:

1. ASME Boiler and Pressure Vessel Code, Section II, Section III, and Section IX.

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

3. RG 1.36, as it relates to nonmetallic thermal insulation for austenitic stainless steel.

4. RG 1.28, as it relates to onsite cleaning and cleanliness controls.

5. RG 1.44, as it relates to the use of sensitized stainless steel.

6. RG 1.84, as it relates to ASME Code Case acceptability.

5.4.2.1.4 *Technical Evaluation*

5.4.2.1.4.1 Steam Generator Design and Materials

The staff reviewed DCD Tier 2 Section 5.4.2.1, "Steam Generator Materials," in accordance with SRP Section 5.4.2.1, "Steam Generator Materials," to ensure that the integrity of the SG materials is maintained, and that the SG materials meet the relevant requirements of GDC 1, GDC 4, GDC 14, GDC 15, GDC 30, GDC 31, and Appendix B to 10 CFR Part 50. These requirements are met through compliance with appropriate requirements of the ASME Code and conformance to guidance in RGs by specifying design features shown to preserve SG tube integrity, and by specifying water chemistry practices that limit degradation of SG materials. The staff also reviewed supplemental information that the applicant provided in letters dated August 4, and December 10, 2015.

In the letter dated August 4, 2015 (ML15216A454), and December 10, 2015 (ML15344A185), the applicant stated that wear is the only form of degradation observed in nearly identical SGs in Korea. This is consistent with similarly designed and operated SGs in the U.S. Of particular note are the replacement SGs (RSGs) at Palo Verde Units 1, 2, and 3, which, like the APR1400 SGs, closely resemble those in the System 80+ certified design (10 CFR Part 52, Appendix B). The Palo Verde RSGs were installed between 2003 and 2007, and the only active degradation mechanism thus far is wear from support structures and foreign objects. Relatively few tubes have been plugged, and the wear is managed and tube integrity maintained through the SG Program. The applicant also stated that wear is also the only active degradation mechanism in the Korean OPR1000 plants, which have SGs nearly identical to the APR1400 SGs in terms of design and materials. In its letter dated August 4, 2015, the applicant proposed modifying DCD Subsection 5.4.2," Steam Generators," to describe the similarity between the OPR1000 and APR1400 SGs rather than stating they are identical. The staff confirmed that DCD Tier 2, was revised to include this change. The staff's review of the SG Program is discussed in Subsection 5.4.2.2 of this report, and the staff's review of flow-induced vibration is discussed in Subsection 3.9.2, "Dynamic Testing and Analysis of Systems, Components, and Equipment."

5.4.2.1.4.2 Selection, Processing, Testing, and Inspection of Materials

The SG materials proposed for the APR1400 are described in DCD Tier 2, Section 5.4.2.1.1, "Selection, Processing, Testing, and Inspection of Materials," Section 5.2.3.1, "Material Specification," and Table 5.2-2, "Reactor Coolant System Materials and Weld Materials." The materials proposed are ferritic carbon and low-alloy steels; austenitic, ferritic, and martensitic stainless steels; and nickel-base alloys. The staff reviewed the materials selected 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 SGs are acceptable if they are selected, fabricated, tested, and inspected (during fabrication and manufacturing) in accordance with the ASME Code. As revised by the applicant's letter dated August 4, 2015, DCD Tier 2 Subsection 5.4.2.1.1 will state that the selection, fabrication, testing, and inspection of all SG pressure boundary materials meet the applicable ASME Code requirements, and it references DCD Tier 2 Subsection 5.2.3 for the description of how these requirements are met. The staff confirmed that DCD Tier 2 was revised to incorporate this change.

The primary side of each SG is designed and fabricated to comply with ASME Code Class 1 requirements. The secondary side of each SG is designed and fabricated to comply with ASME

Code Class 2 requirements. These design criteria are identified in DCD Tier 2 Table 5.2-1. The applicant proposed to use Alloy 690 TT for the tubing material. This material is listed in Section II of the ASME Code and is, therefore, permitted by 10 CFR 50.55a. In addition, this material is appropriate based on operating experience. Alloy 690 TT tubes were first used in U.S. operating plant SGs in 1989, and have thus far resisted degradation by corrosion mechanisms. The tubes have a nominal outside diameter of 0.75 inch and a nominal wall thickness of 0.042 inch, which are typical for Alloy 690 TT tubes at operating plants.

Carbon and alloy steels with ASME Code specifications are used for the head, shell, tubesheet stay, nozzles, safe ends, and tubesheet. Welding consumables matched to Alloy 690 (Alloys 52/52M/152) are used for welds in contact with the primary coolant, including the tubesheet weld-deposited cladding. The tubesheet cladding thickness is 4.8 mm (0.19 inch) and the overall thickness including the cladding is 653 mm (25.5 inch). The addition of these tubesheet details to the DCD (alloy specification in Table 5.2-2, cladding thickness, and tubesheet thickness) is based on Enclosure 8 to the applicant's August 4, 2015, letter. The staff confirmed that DCD Tier 2 was revised to incorporate this change.

The only use of austenitic stainless steel is for the weld-deposited cladding on the primary side of the bottom head, stay, and nozzles. DCD Tier 2 Subsection 5.4.2.1.1 states that there is no use of austenitic stainless steel for the primary or secondary pressure boundary. As revised by the applicant's letter dated August 4, 2015, DCD Tier 2 Subsection 5.4.2.1.1 will state that austenitic stainless steel is not used for the primary or secondary pressure boundary. The staff confirmed that DCD Tier 2 was revised to incorporate this change. Type 409 and Type 405 ferritic stainless steels are used for the tube support structures. Type 410S martensitic stainless steel is used for the channel head divider plate. These stainless steel grades are listed in Section II of the ASME Code and are, thus, permitted by 10 CFR 50.55a and are typical of modern SGs. In one case, for the Type 409 tube support material, DCD Table 5.2-2 lists the ASTM International specification (ASTM A240) rather than the ASME specification (SA-240). The staff determined that this material is still permitted by 10 CFR 50.55a, since the applicable ASME SA-240/SA-240M specification states that it is identical to ASTM A240.

The proposed bolting materials are precipitation-hardened nickel-base Alloy 718 (SB-637) on the primary side and SA-106 or SA-193 alloy steel on the secondary side. These materials are listed in Section II of the ASME Code and are, thus, permitted by 10 CFR 50.55a. The staff's review of threaded fasteners is discussed in Section 3.13, "Threaded Fasteners – ASME Code Class 1, 2, and 3," of this report.

The staff determined that the SG materials meet the requirements of GDC 1, 4, 14, 15, 30, and 31, Appendix B to Part 50, and Appendix G to Part 50, and 50.55a, as they relate to material selection. The conclusion is based on determining that the materials selection, fabrication, testing, and inspection meet the ASME Code requirements.

5.4.2.1.4.3 Steam Generator Design

The staff reviewed the adequacy of the design and fabrication process proposed for the APR1400 SGs to determine whether crevice areas are limited, residual stresses are limited in the tube bends and tubesheet crevice region, corrosion-resistant materials are used, corrosion allowances are specified, and suitable bolting materials are used. As discussed above, the SGs 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.

The joints between the tube ends and the tubesheet form part of the RCPB and must satisfy the design requirements of the ASME Code Class to comply with GDC 1, GDC 14, GDC 15, GDC 30, GDC 31, and 10 CFR 50.55a. For the APR1400, the tube ends are welded to the nickel-base cladding on the primary side of the tubesheet using structural welds that meet the design requirements of ASME Code Sections III and IX, including the design stress limits in Subarticle NB-3200 of Section III. This is acceptable to meet the requirements of GDC 1, GDC 14, GDC 15, GDC 30, GDC 31, and 10 CFR 50.55a, as they relate to the primary-to-secondary pressure boundary formed by the tube-end welds. The statement that the welds satisfy the limits of NB-3200 is based on Enclosure 8 to the applicant's August 4, 2015, letter. The staff confirmed that DCD Tier 2 was revised to incorporate this statement.

Crevices around SG tubes in the tubesheet region have caused corrosion in earlier SG designs. In order to minimize or eliminate crevice areas, the APR1400 design includes hydraulic expansion of the tubes into the tubesheet for the entire tubesheet thickness. This meets the staff's acceptance criteria in SRP Section 5.4.2.1, which states that full-depth expansion of the tube through the tubesheet region limits the crevice between the tube and tubesheet.

The DCD states that tubes are formed into both square bends and U-bends at the top of the bundle. The U-bends are in the central portion of the bundle, Rows 1-17, and have a bend radius up to 280 mm (11 inches). The bend radius of all square bends is 254 mm (10 inches). All tubes with U-bends receive a thermal treatment to relieve residual stresses. This conforms to the staff guidance in SRP Section 5.4.2.1, which states that short-radius U-bends should be thermally treated to reduce residual stresses. It also meets the specific industry guidance in EPRI TR-016743, "Guidelines for Procurement of Alloy 690 Steam Generator Tubing," (Reference 1) which specifies thermal treatment for bends with a radius less than 191 mm (7.5 inches) for this tube diameter. The addition of bend radius details to Section 5.4.2.1 of the DCD is based on Enclosure 8 to the applicant's August 4, 2015, letter. The staff confirmed that DCD Tier 2 incorporated the bend radius details.

The DCD describes two patterns for the tube layout, depending on the location. The straight portion of the tubes is a triangular array with a 25.4 mm (1.00 inch) center-to-center spacing (pitch). The bend portion is a rotated square array with a 31.3 mm (1.23 inch) pitch. The addition of tube layout to Section 5.4.2.1 of the DCD is based on Enclosure 8 to the applicant's August 4, 2015, letter. The staff confirmed that DCD Tier 2 was revised to incorporate the tube layout information. The tubes are supported by three types of structures fabricated from Type 409 ferritic stainless steel: eggcrate, vertical, and diagonal supports. In addition, three different configurations are used for the eggcrate supports: a full circular structure, a half circular structure, and a partial structure bounded by the circumference and a chord. The type of eggcrate support depends on its location in the SG. The vertical and diagonal supports provide vibration support in the bend region of the tubes. In its letter dated August 4, 2015, the applicant proposed adding a statement to DCD Subsection 5.4.2.1.2.2, "Material Design," that operating experience at Korean nuclear power plants shows that the tube support design has been effective in limiting tube wear. The staff confirmed that DCD Tier 2 included this statement.

All of the supports are constructed of flat strips except for the flow distribution plate (FDP) just above the entrance to the economizer section. The flat-strip design prevents creviced or stagnant areas where impurities or deposits could accumulate and cause corrosion or stress on the tubes. The vertical and diagonal supports in the bend region are made of perforated strips to support flow in the upper bundle. The FDP in the economizer is Type 405 stainless steel with circular drilled holes. Both tubes and secondary coolant pass through the holes. The hole diameter is 0.64 mm (0.025 inch) larger than the tube outside diameter to provide a flow area. In its letter dated August 4, 2015, the applicant proposed modifying DCD Subsection 5.4.2.1.2.2 to describe the geometry of the economizer FDP, including the diameter of the holes. The staff confirmed that the economizer FDP information was incorporated into DCD Tier 2. The support system includes another FDP at the uppermost full eggcrate support. This plate does not support the tubes. It is made from Type 405 stainless steel and has drilled holes for fluid flow. No tubes pass through the upper FDP. In its letter dated August 4, 2015, the applicant proposed modifying DCD Subsection 5.4.2.1.2.2 to describe the geometry of the eggcrate FDP and state that no tubes pass through the holes. The staff confirmed that DCD Tier 2 was revised to incorporate this change. In summary, the materials, sizes, and configuration of the support structures are designed to resist corrosion, support flow along the tubes, and provide support against the postulated loads.

The acceptance criteria for SG design in SRP Section 5.4.2.1 state that the design is acceptable with respect to the tube support structure if the support structures are fabricated from a corrosion-resistant material and the design provides for flow along the tubes. A lattice grid (eggcrate) support system is identified as an example. The material proposed for the support structures is Type 409 ferritic stainless steel, which has shown negligible corrosion in replacement SGs (e.g., NUREG-1841, "U.S. Operating Experience with Thermally Treated Alloy 690 Steam Generator Tubes," Reference 2), including the Palo Verde Units RSGs, which are designed and operated nearly the same as the proposed APR1000 SGs. Since the support design uses corrosion-resistant material and a design that promotes flow along the tube surface, it meets the staff's guidance.

The staff reviewed the design of the feedwater inlet rings and economizer section with respect to materials and the potential for loose parts. The spray nozzles in the downcomer and recirculation feedwater inlet rings have 5.8 mm (0.23 inch) diameter holes to limit the entry of foreign objects into the SG, and the nozzles in the downcomer feedwater ring can be detached to remove foreign objects from the ring. The feedwater box bottom plate, which is near the bottom of the economizer section, has 5.8 mm (0.23 inch) diameter holes to limit foreign object entry, and 127 mm (5 inch) inspection holes for foreign object removal. These features are described in Enclosure 8 to the applicant's August 4, 2015, letter. In DCD Subsections 5.4.2.1.2.1.2 and 5.4.2.1.2.1.3, respectively, the applicant described design and operating features to prevent water hammer in the feedwater rings and thermal stratification at the feedwater nozzle. These design features include an elevated feedwater ring, goose-neck bends in the piping inside the SG, and top discharge spray nozzles. The proposed design features to prevent water hammer and thermal stratification are consistent with staff guidance in BTP 10-2, "Design Guidelines for Avoiding Water Hammers in Steam Generators," and industry guidance in the "EPRI Steam Generator Reference Book" (Reference 3).

The staff issued RAI 292-8306, Question 05.04.02.01-1 (ML15314A017), requesting the applicant to provide the materials for the feedwater rings and nozzles, since material selection is a key factor in resistance to flow-accelerated corrosion (FAC). In its response to RAI 292-8306, Question 05.04.02.01-1 (ML15344A185), the applicant stated that the feedwater ring and feedwater nozzle thermal sleeves are fabricated from low alloy steel containing nominally 1.25 weight percent chromium (P11 material), and the spray nozzles are nickel-base Alloy 690 (30 weight percent chromium). The economizer box is Grade SA-516 Grade 70 low-alloy steel specified with 0.2 weight percent chromium with nickel-base weld cladding (30 weight percent chromium). The NRC does not have detailed requirements for the materials of construction of these internal components, but the response indicates that these components are designed with materials that will resist FAC because they contain sufficient chromium as defined by industry guidelines the staff finds acceptable. Those guidelines describe 0.1 weight percent chromium as the minimum required to achieve significant FAC resistance. Because the response identified the materials and how they are selected for compatibility with the environment. Therefore, RAI 292-8306, Question 05.04.02.01-1, is resolved and closed.

The staff noted that the SG design includes provisions for detecting and removing loose parts, as discussed below in Section 5.4.2.1.4.8 (access to the secondary side). The design also includes thickness allowances for corrosion, as discussed below under the topic of compatibility with the coolant.

In DCD Subsection 5.4.2.1.2.1.1, the applicant described how the SG design addresses flow-induced vibration (FIV) of the tube bundle. The staff evaluated this as part of its FIV review in Section 3.9.2, "Dynamic Testing and Analysis of Systems, Structures, and Components," of this report.

The staff determined that the SG design meets the acceptance criteria in SRP Section 5.4.2.1 as they relate to limiting the potential for degradation of the tubes and other secondary-side components. These criteria, in conjunction with the acceptance criteria for interacting reviews and appropriately performed ISIs, provide assurance that: (1) the probability of abnormal leakage, rapidly propagating failure, and gross rupture will be extremely low, (2) the design conditions of the RCPB are not exceeded during operation, and (3) sufficient margin is available to prevent rapidly propagating failure, consistent with the requirements of GDC 14, 15, and 31.

5.4.2.1.4.4 Fabrication and Processing of Ferritic Materials

To comply with GDC 14, GDC 15, and GDC 31, the fracture toughness of the ferritic materials forming the primary and secondary pressure boundaries of the SGs must resist rapidly propagating failure and ensure that the design conditions will not be exceeded during operation. The pressure-retaining ferritic materials selected for use in SGs are acceptable with respect to fracture toughness if they: (1) comply with Appendix G to 10 CFR Part 50, 10 CFR 50.55a(c), (d), and (e), and (2) follow the provisions of Appendix G to Section III of the ASME Code. For Class 1 and Class 2 SG components, the regulations cited above require the use of Section III of the ASME Code. Subarticle NB-2300, Subarticle NC-2300, and Appendix G of Section III of the ASME Code address fracture toughness requirements for Class 1 and Class 2 components.

The APR1400 design addresses these Code requirements in DCD Tier 2, Subsections 5.2.3.3 and 5.4.2.1.3. Although the secondary side of the SGs is classified as Class 2, the applicant designs both the Class 1 and Class 2 boundary components to the Class 1 Section III, Subsection NB requirements. DCD Subsection 5.4.2.1.3 states that the primary and secondary pressure boundary components meet the fracture toughness requirements of Section III, Subsection NB. As revised by the applicant's letter dated August 4, 2015, DCD Subsection 5.4.2.1.3 will state that the primary and secondary pressure boundary components meet the fracture toughness requirements of Section III, Subsection NB. The staff confirmed that these statements were incorporated into DCD Tier 2. Therefore, the SG ferritic material 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 the ferritic steel for the primary and secondary pressure boundary of the SGs must meet the requirements of 10 CFR 50.55a(c), (d), and (e). Ferritic steel pressure-boundary welding must also meet the requirements of Subsubarticle D-1210 of Appendix D to ASME Code Section III, as well as adhere to the guidance in RG 1.50, RG 1.71, and RG 1.43. The APR1400 design follows these requirements and RGs for both the primary and secondary SG pressure boundary components, as stated in DCD Tier 2 Subsection 5.4.2.1.3. As revised by the applicant's letter dated August 4, 2015, DCD Subsection 5.4.2.1.3 will state that the primary and secondary pressure boundary components of the SG meet the welding requirements described in Subsection 5.2.3.3. DCD Subsection 5.2.3.3 describes how the APR1400 design meets the welding requirements for RCPB components. Based on its review of the DCD, the staff confirmed incorporation of the changes discussed above and determined that the design satisfies the requirements of 10 CFR Part 50 related to welding for the SG ferritic materials. The staff's review of welding of RCPB materials is discussed in more detail in Section 5.2.3 of this report.

5.4.2.1.4.5 Fabrication and Processing of Austenitic Stainless Steel

To comply with GDC 1, 14, 15, 30, and 31, the use of austenitic stainless steel in SG pressure boundary applications must include limiting the susceptibility to SCC and performing welding according to quality standards. The requirements of GDC 4 and 10 CFR Part 50, Appendix B, Criterion XIII, are met through compliance with the applicable provisions of the ASME Code and with the regulatory positions of RG 1.31, RG 1.34, RG 1.36, RG 1.44, and RG 1.71.

According to DCD Tier 2 Subsection 5.4.2.1.4, the APR1400 design uses no austenitic stainless steel for the primary or secondary pressure boundary. Revision of the DCD to replace the original information with this statement is based on Enclosure 8 to the applicant's August 4, 2015, letter. The staff confirmed that DCD Tier 2 was revised to incorporate this statement. Therefore, the NRC requirements listed above for austenitic stainless steel are not applicable to the SG design. The staff's review of fabrication and processing requirements for austenitic stainless steel used in other RCPB applications is discussed in Subsection 5.2.3, of this report.

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

The SG 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. According to DCD Subsections 5.2.3.2, 5.4.2.1, 5.4.2.2, and 9.3.4, the APR1400 bases primary water chemistry control on the EPRI "PWR Primary Water Chemistry Guidelines" (Reference 4). The reference to the EPRI primary water guidelines and Subsection 5.2.3.2 is based on Enclosure 8 to the applicant's August 4, 2015, letter. The staff confirmed that DCD Tier 2 included the addition of these references. The staff issued RAI 292-8306, Question 05.04.02.01-2 (ML15314A017), requesting the applicant to clarify whether the APR1400 primary water chemistry will be controlled according to the EPRI guidelines or "at a level comparable" to the EPRI guidelines, as stated in the August 4, 2015, letter. In its response to RAI 292-8306, Question 05.04.02.01-2 (ML15344A185), the applicant proposed changing DCD Subsection 5.4.2.1.5, to state that primary water chemistry is controlled "in accordance" with EPRI guidelines. The staff has determined that the EPRI guidelines are acceptable for primary water chemistry, as discussed in the staff's review in Subsections 5.2.3.2 and 9.3.4 of this report. Therefore, the staff determined that the applicant's response is acceptable. The staff confirmed that DCD Tier 2 was revised as committed in the response to RAI 292-8306, Question 05.04.02.01-2. Therefore, RAI 292-8306, Question 05.04.02.01-2, is resolved and closed.

According to DCD Subsections 5.4.2.2 and 10.3.5, secondary water chemistry is based on the EPRI “PWR Secondary Water Chemistry Guidelines” (Reference 5), which conforms to SRP BTP-1, “Monitoring of Secondary Side Water Chemistry in PWR Steam Generators.” In addition, the APR1400 TSDCD Tier 2 Chapter 16, Section 5.5.10) incorporate the secondary water chemistry requirements in the Standard Technical Specifications (STS). Therefore, the applicant’s design for secondary water chemistry meets 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 SG materials include nickel-base Alloy 690 TT, stainless steels, and carbon and low-alloy steels. DCD Subsection 5.4.2.1.2 states that a corrosion allowance of 1.6 mm (0.063 inch) will be applied to low-alloy steel that comprises the secondary-side pressure boundary, and a corrosion allowance of 0.053 mm (0.002 inch) will be applied to austenitic stainless steel or nickel-base cladding on the primary-side pressure boundary. As revised by the applicant’s letter dated August 4, 2015, DCD Subsection 5.4.2.1.2 will add a statement that the a 0.053 mm (0.002 inch) corrosion allowance is used for austenitic stainless steel or nickel-base cladding. The staff confirmed that the amount of corrosion allowance was identified in DCD Tier 2. Also in its letter dated August 4, 2015, the applicant identified the references for these corrosion allowances. For secondary environments, the 0.06 in (1.6 mm) allowance is based on tests described in EPRI Report NP-2791 (Reference 6) for candidate tube support materials. The tests were conducted on carbon and low-alloy steels and stainless steels (including Types 405 and 409) in “faulted” secondary environments with acid and chloride contamination. For primary water, the 0.0021 in (0.053 mm) allowance is based on tests described in a research symposium paper on the corrosion effects of zinc added to primary coolant (Reference 7). That research included measurements of the corrosion rate of austenitic stainless steels and nickel-base alloys (including Alloy 690 TT) in a simulated PWR environment. The primary water corrosion rates for Alloy 690 TT and stainless steel reported in Reference 7 were about 0.001 mils per year (mpy) and lower, which is effectively zero.

The staff verified the applicant’s references and checked additional references. The staff issued RAI 292-8306, Question 05.04.02.01-3 (ML15314A017), requesting the applicant to address an apparent error in a conversion factor the applicant used in calculating corrosion rates to evaluate corrosion allowance for SG materials. In its response to RAI 292-8306, Question 05.04.02.01-3 (ML15344A185), the applicant confirmed the error and proposed correcting the August 4, 2015, letter. The staff determined that this is acceptable because it clarifies how the applicant evaluated corrosion allowance. Based on its review of the DCD, the staff confirmed incorporation of the changes discussed above; therefore, RAI 292-8306, Question 05.04.02.01-3, is resolved and closed. The laboratory and operating experience for the SG materials proposed for the APR1400 (low-alloy steel, nickel-base alloy, austenitic stainless steel, and ferritic stainless steel) indicates no general corrosion is expected for Alloy 690 TT or stainless steels in the primary or secondary coolant (Reference 8, for example). In another EPRI study of the tube support materials in a simulated faulted secondary environment of mixed water and steam, the measured corrosion rates were 0.17 mpy for A570 low-alloy steel, 0.04 mpy for Type 409 stainless steel, and 0.03 mpy for Type 405 stainless steel (Reference 9). In NP-2791, referenced by the applicant (Reference 6), the corrosion rate was 0.12 – 0.38 mpy for A508 low-alloy steel, and 0.03-0.06 mpy for Type 405 and Type 409 stainless steel in the simulated faulted secondary environment.

ISI performed on Alloy 690 TT tubes in replacement SGs since approximately 1990, have revealed no instances of corrosion-related degradation of the tubes or stainless steel support structures. The inspection results through 2004 are documented in NUREG-1841 (Reference 2) and on an ongoing basis in the staff's review of inspection reports submitted by licensees in accordance with its TS. This experience has revealed negligible thinning of nickel-base alloys and stainless steels.

The staff determined that the proposed corrosion allowances are acceptable because: (1) they are reasonable with respect to the corrosion rates measured under aggressive chemistry conditions for the secondary side and representative conditions on the primary side, (2) general thinning of the proposed steam general materials is not being observed under normal operating conditions, and, (3) secondary water chemistry control has improved since the support material corrosion tests were performed, decreasing the exposure of SGs to faulted conditions.

According to DCD Subsection 5.4.2.1.5, onsite cleaning and cleanliness controls for the SGs will meet the regulatory provisions of RG 1.28, Revision 4 and associated ASME NQA-1 requirements. The acceptance criterion for this topic in the current revision of SRP Section 5.4.2.1 is RG 1.37, which was withdrawn in 2014. However, RG 1.37 was withdrawn based in part on the availability of RG 1.28, Revision 4, as discussed in the Bases for withdrawal (Reference 10). Because RG 1.28, Revision 4 contains the staff's updated guidance on cleaning and cleanliness controls, the design meets the requirements of Criterion XIII of Appendix B to 10 CFR Part 50, with respect to the cleaning and cleanliness controls for the SG design.

The controls placed on the primary and secondary coolant chemistry limit the susceptibility of the SGs to corrosion in the operating environment so that the ISI program can manage any degradation that may occur. In addition, the proposed secondary water chemistry program conforms to the latest revision of the STS. Pending confirmation of the proposed DCD changes described in this subsection, the staff determined that these water chemistry controls meet, in part, the requirements of GDC 4 to ensure that the materials are compatible with the environment.

5.4.2.1.4.7 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 DCD Subsection 5.4.2.1.6. The design provides openings for access to the tube bundle, moisture separators, feedwater ring, downcomer annulus, tubesheet, and feedwater box. The openings are designed and located to facilitate cleaning, inspection, repairs, and foreign object search and retrieval.

The design includes two, 533 mm (21 inch) diameter manways to the moisture separator and dryer area, which contains a hatch to access the top of the tube bundle and feedwater ring. There are two, 203 mm (8 inch) handholes in the tubesheet region that provide access for tubesheet sludge lancing, downcomer annulus inspection, and inspection for loose parts. Two 127 mm (5 inch) inspection holes provide access to the feedwater box to remove foreign objects.

The staff determined that this level of secondary-side access is acceptable for inspection, cleaning, and foreign object removal because it has been determined adequate in similarly designed operating SGs. In addition, the staff determined that this level of access is 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 SG meets, in part, the requirements of GDC 14 and GDC 15.

5.4.2.1.5 *Combined License Information Items*

There are no COL information items related to this section of the application.

5.4.2.1.6 *Conclusion*

Pending confirmation of the proposed DCD changes described in this review section, on the basis of its review of DCD Tier 2 Subsection 5.4.2.1, the staff concluded that the APR1400 SG materials satisfy the acceptance criteria for materials selection, design, fabrication, compatibility with the service environments, and secondary-side accessibility. The staff further concluded that these materials as specified are acceptable and meet the requirements of GDC 1, 4, 14, 15, 30, and 31, as well as the requirements of 10 CFR Part 50, Appendices B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," "and G, "Fracture Toughness Requirements."

5.4.2.2 *Steam Generator Program*

5.4.2.2.1 *Introduction*

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

5.4.2.2.2 *Summary of Application*

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

DCD Tier 2: In DCD Subsection 5.4.2.2, "Steam Generator Program," the applicant described the SG Program to monitor and manage tube degradation and to provide prompt preventive and corrective actions to maintain the structural and leak-tight integrity of the SG tubes. The program is based on Nuclear Energy Institute (NEI) 97-06 and the STS. The major program elements are degradation assessment, tube inspection, tube integrity assessment, tube plugging, primary-to-secondary leak monitoring, foreign material exclusion (including loose parts management), maintenance of SG secondary-side integrity, contractor oversight, self-assessment, reporting, and maintaining compatibility between the SG tubes and coolant.

ITAAC: There are no ITAAC for this review topic.

TS: The APR1400 TS related to the SG Program are located in DCD Chapter 16, Sections 3.4.12, 3.4.17, 5.5.9, 5.6.7, and Bases Sections B 3.4.12 and B 3.4.17. The purpose of these TS is to maintain tube structural and leakage integrity.

5.4.2.2.3 *Regulatory Basis*

1. GDC 32 requires, in part, that 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.

2. Title 10 CFR 50.55a(g) requires that ISI programs meet the applicable inspection requirements in Section XI of the ASME Code.

3. Title 10 CFR 50.36 applies to the SG program in the TS.

4. Title 10 CFR 50.65 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.

5. Appendix B to 10 CFR Part 50 applies to quality assurance in the implementation of the SG Program.

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

Acceptance criteria adequate to meet the above requirements include:

1. NEI 97-06, "Steam Generator Program Guidelines."

2. NUREG-1430, NUREG-1431, and NUREG-1432, "Standard Technical Specifications."

3. TS Task Force 510, "Revision to Steam Generator Program Inspection Frequencies and Tube Sample Selection."

4. RG 1.121, as it relates to determining the plugging criteria for degraded SG tubes.

5. BTP 5-1, as it relates to monitoring secondary-side water chemistry.

5.4.2.2.4 *Technical Evaluation*

The staff reviewed DCD Subsection 5.4.2.2, "Steam Generator Program," in accordance with SRP Section 5.4.2.2, "Steam Generator Program," to ensure that the SG tube bundle, as part of the RCPB, is designed to permit periodic inspection and testing of the tubes and critical areas, and includes features to assess the structural and leakage integrity of the tubes, as required by GDC 32. The staff also reviewed supplemental information the applicant provided in a letter dated August 4, 2015; December 16, 2015; March 2, 2016; May 3, 2016; July 26, 2016; and August 21, 2017.

The proposed SG Program incorporates prevention of degradation, inspection, evaluation, corrective actions, leakage monitoring, and maintaining performance criteria that define SG tube integrity. According to DCD Tier 2 Section 1.9.2, "Conformance with Standard Review Plan," the design conforms to the guidance in SRP Section 5.4.2.2 with no exceptions. This statement is based on a correction to a typographical error in Table 1.9-2, "APR1400 Conformance with the Standard Review Plan," in Enclosure 9 to the applicant's August 4, 2015, letter. The staff confirmed that the correction to Table 1.9-2 was made in DCD Tier 2. DCD Subsection 5.4.2.2.1 states that all of the tubes can be accessed from the primary side of the SG for full-length inspection, which conforms to SRP 5.4.2.2. This statement is based on Enclosure 9 to the applicant's August 4, 2015, letter. The staff confirmed that this statement of access to the tubes was incorporated in DCD Tier 2.

DCD Subsection 5.4.2.2 states that the program will be established and maintained based on NEI 97-06 and its referenced EPRI guidelines, and that it will comply with the applicable sections of Section XI of the ASME Code as required by 10 CFR 50.55a. As described in DCD Subsection 5.4.2.2.2.3, the SG Program is governed by the TS. These bases for the SG Program conform to SRP 5.4.2.2. This statement about the TS (which includes deleting two unnecessary paragraphs) is based on Enclosure 9 to the applicant's August 4, 2015, letter. However, the applicant's description of deleting the unnecessary paragraphs was different than the proposed DCD revisions. The staff issued RAI 299-8310, Question 05.04.02.02-1, (ML15314A024), requesting consistency in the response of the applicant's August 4, 2015, letter and the proposed DCD revision. Specifically, the response incorrectly stated that only one paragraph would be deleted.

In its response to RAI 299-8310, Question 05.04.02.02-1 (ML17233A369), the applicant confirmed the intent to delete the two unnecessary paragraphs from DCD Subsection 5.4.2.2.2.12 and move a third paragraph to DCD Subsection 5.4.2.2.2.3. The response included the proposed DCD revisions submitted in the initial response. The staff determined that the proposed DCD revisions are acceptable because they delete unnecessary information and relocate information about the bases for the SG Program to a more appropriate location. The staff confirmed that DCD Tier 2 was revised as committed in the response to RAI 299-8310, Question 05.04.02.02-1. Subsequently, in its revised response to RAI 299-8310, Question 05.04.02.02-1 (ML17233A369), the applicant replaced the term "repair criteria" with "plugging criteria" in two places in DCD Subsection 5.4.2.2.2.3. The staff determined this is acceptable because "plugging criteria" is consistent with the STS as modified by TSTF-510. Based on the review of the DCD, the staff has confirmed incorporation of the changes described above; therefore, RAI 299-8310, Question 05.04.02.02-1 is resolved and closed.

DCD Subsection 5.4.2.2 includes several references to repairs and sleeves. This is not applicable to the APR1400 application, and the staff requested that the references be deleted. The applicant agreed and proposed revisions in Enclosure 9 to the applicant's August 4, 2015, letter. The staff found that some references to repairs and sleeves remained in Revision 1 of the DCD and TS. Based on its review of the DCD, the staff confirmed the deletion of these references in Subsection 5.4.2.2 and the TS. The DCD markup for these changes included another wording change inconsistent with the STS as modified by TS Task Force (TSTF)-510, Revision 2, "Revision to Steam Generator Program Inspection Frequencies and Tube Sample Selection." The staff issued RAI 299-8310, Question 05.04.02.02-2 (ML15314A024), requesting that the applicant change "degradation" to "indications" as originally proposed. In its response to RAI 299-8310, Question 05.04.02.02-2 (ML15352A301), the applicant proposed changing the last word in DCD Item 5.4.2.2.2.12.d from "degradation" back to "indications" as originally proposed. The staff determined that this is acceptable because the proposed wording is consistent with the STS. Based on the review of the DCD, the staff has confirmed incorporation of the changes described above; Therefore, RAI 299-8310, Question 05.04.02.02-2 is resolved and closed. The staff also requested information as to whether a statement should be deleted from DCD Subsection 5.4.2.2.2.1, which suggested the SG degradation assessment would focus only on cracking. The applicant agreed and proposed a revision in Enclosure 9 to the applicant's letter dated August 4, 2015. The staff confirmed that the statement was deleted in DCD Tier 2. With respect to water chemistry programs, the DCD stated they are based on the EPRI programs (primary and secondary) but did not describe them or indicate where in the DCD they are described. The applicant proposed revisions to DCD Subsections 5.4.2.2.2.7, "Secondary-Side Water Chemistry," and 5.4.2.2.2.8, "Primary-Side Water Chemistry," in Enclosure 9 to the applicant's letter dated August 4, 2015. The staff confirmed that these revisions were incorporated into DCD Tier 2.

The staff reviewed the associated TS with respect to the latest revision of the STS/TSTF-510. The staff used NUREG-1432, Revision 4 of the STS as the basis for the review. This is the version for Combustion Engineering (CE) Plants, which are most similar to the APR1400, but the staff noted that with respect to the SG Program the three versions of the STS are the same. The staff noted that the applicant did not state that the TS would govern if there are discrepancies between the TS and Article IWB-2000 of Section XI of the ASME Code. Such a statement is identified as an acceptance criterion in SRP 5.4.2.2; however, it was based on a requirement in 10 CFR 50.55a that has since been deleted. Therefore, the applicant can meet the guidance in SRP 5.4.2.2 without this statement in the DCD. In Enclosure 9 to its letter dated August 4, 2015, the applicant proposed revisions to DCD Subsections 5.4.2.2 and 5.4.2.2.2.12, to delete the statement about differences between the TS and Section XI of the ASME Code. The staff confirmed that these revisions were incorporated into DCD Tier 2. The TS sections directly related to the SG Program are the following:

- TS 3.4.12 and B 3.4.12, “RCS Operational LEAKAGE” (RCS).
- TS 3.4.17 and B 3.4.17, “Steam Generator (SG) Tube Integrity.”
- TS 5.5.9, “Steam Generator Program.”
- TS 5.6.7, “Steam Generator Tube Inspection Report.”

The staff determined that the proposed APR1400 TS (including the Bases), and the description of them in DCD Subsection 5.4.2.2, were generally consistent with the STS and TSTF-510. However, the staff identified several discrepancies and identified them to the applicant. The applicant addressed most of these satisfactorily in Enclosure 9 to the applicant’s letter dated August 4, 2015. Based on its review of the DCD, the staff confirmed the incorporation of the changes discussed above. However, several issues remained or were created in the applicant’s proposed DCD revision. The staff issued RAI 299-8310, Question 05.04.02.02-3 (ML15314A024), requesting the applicant to address these issues. These nine issues, designated (a) through (i), were all related to wording or the applicability of STS and TSTF-510 provisions to the APR1400. In its response to RAI 299-8310, Question 05.04.02.02-3 (ML16062A276), the applicant proposed DCD changes to address these issues. The staff determined that the responses to items (a) through (g) are acceptable because they are consistent with the STS and TSTF-510. Based on the review of the DCD, the staff has confirmed incorporation of the changes described above; therefore, RAI 299-8310, Question 05.04.02.02-3 is resolved and closed.

For items (h) and (i) of RAI 299-8310, Question 05.04.02.02-3, it was not to clear to the staff that proposed changes to the primary-to-secondary leakage values were consistent with the accident analyses performed for the APR1400. The staff issued RAI 494-8620, Question 05.04.02.02-6 (ML16160A379), requesting the applicant to provide clarification of the proposed leakage values and consistency across the TS and DCD. In its response to RAI 494-8620, Question 05.04.02.02-6 (ML16208A488), the applicant clarified its accident analysis and leakage assumptions, and proposed revisions to TS Bases Sections 3.4.12 and 3.4.17. The applicant’s analysis assumes primary-to-secondary leakage of 2.27 L/min (0.6 gpm). The staff determined that the applicant’s response is acceptable because the leakage value of 2.27 L/min (0.6 gpm) is consistent with TS 5.5.9.b.2 and the value was clarified in the proposed revision to the TS Bases. The staff confirmed that DCD Tier 2 was revised as committed in the response to RAI 494-8620, Question 05.04.02.02-6. Therefore, RAI 494-8620, Question 05.04.02.02-6, is resolved and closed.

DCD Subsection 5.4.2.2.2.2, states that PSI of the tubes will be conducted by examining the full length of each tube after fabrication and prior to placing the SGs in service (i.e., after the field hydrostatic test as suggested in Section 3.2.1 of the EPRI SG Examination Guidelines referenced in NEI 97-06). This statement is based on Enclosure 9 to the applicant's letter dated August 4, 2015. The staff confirmed that this description of the PSI requirement was incorporated in DCD Tier 2. The DCD also states that the PSI will be performed using techniques capable of detecting degradation and fabrication abnormalities along the length of the tube from the tube-to-tubesheet weld at the tube inlet to the tube-to-tubesheet weld at the tube outlet. This is consistent with industry practice and is acceptable because the timing and method of the PSI supports the objective of detecting flaws and discriminating between service-related degradation and manufacturing imperfections during ISI.

The tube plugging criteria establish a minimum acceptable SG tube wall thickness that accounts for flaw growth and uncertainty in measuring the size of a flaw. The plugging criteria are based on maintaining tube structural and leakage integrity. Tubes with flaws exceeding the plugging criteria will be removed from service by plugging. For the APR1400, the tube plugging criteria in the TS for plugging tubes were determined using the methodology specified in RG 1.121 and were based on EPRI guidelines specified in NEI 97-06. This is acceptable because it satisfies the staff's recommendations in SRP Section 5.4.2.2 related to tube plugging criteria.

The staff issued RAI 299-8310, Question 05.04.02.02-4 (ML15314A024), requesting the applicant to address the recent operating experience described in Information Notice (IN) 2013-20, "Steam Generator Channel Head and Tubesheet Degradation." IN 2013-20 describes cases of exposure of the carbon steel base material in the SG channel head and primary side of the tubesheet. The base material in those cases was originally protected by corrosion-resistant weld cladding (stainless steel or nickel-base alloy). In its response to RAI 299-8310, Question 05.04.02.02-4 (ML16062A276), the applicant described cleanliness and repair requirements for cladding and proposed adding COL 5.4(7) and COL 13.4(4), with modifications to DCD Subsections 5.4.16, 13.4.1, and Table 1.8-2. COL 5.4(7) requires a COL applicant to prepare an inspection and monitoring program and implementation program for SG channel head cladding integrity. COL 13.4(4) requires a COL applicant to develop an implementation plan for the cladding integrity program. The staff determined that these changes are acceptable because a required inspection and monitoring program provides a way for all APR1400 plants to address SG channel head cladding integrity. Based on the review of the DCD, the staff has confirmed incorporation of the changes described above; therefore, RAI 299-8310, Question 05.04.02.02-4 is resolved and closed.

As discussed in Section 5.4.2.1 above, the APR1400 SGs are designed to be accessible for inspection. All tubes can be inspected from the primary side using currently available NDE techniques. The SGs 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 determined that the design of the APR1400 SGs is acceptable as it relates to providing access to allow ISI.

5.4.2.2.5 *Combined License Information Items*

In Enclosure 9 to the applicant's letter dated August 4, 2015, the applicant proposed COL item in DCD Chapter 5.4(6), requiring the COL applicant to prepare PSI and ISI programs for the SG tubes. The applicant also added the COL item to DCD Table 1.8-2. The applicant also proposed a related COL item in Chapter 13, "Conduct of Operations." The staff considers Chapter 5 COL 5.4(6) adequate, and a second, similar COL item potentially confusing. The staff issued RAI 299-8310, Question 05.04.02.02-5 (ML15314A024), requesting the applicant update the DCD to include a single COL item in Section 5.4 to require COL applicants to provide implementation milestones for the SG PSI and ISI programs. In its response to

RAI 299-8310, Question 05.04.02.02-5 (ML15352A301), the applicant proposed revisions to DCD Subsections 5.4.2.2.2.2, “Inspection,” and 5.4.16, “Combined License Information,” and Table 1.8-2, but it still included two COL items. In its revised response to RAI 299-8310, Question 05.04.02.02-5 (ML16124B186), the applicant proposed to include one COL item and corresponding revisions to DCD Subsections 5.4.2.2.2.2 and 5.4.16, and Table 1.8-2. Based on the revised response, COL 5.) would require a COL applicant to prepare the PSI and ISI programs. The staff determined that this is acceptable because it meets the requirements in 10 CFR 50.55a and the guidance in NEI 97-06 for individual licensees to develop these programs. The COL item was incorporated into the DCD as COL 5.4(8). Based on the review of the DCD, the staff has confirmed incorporation of the changes described above; therefore, RAI 299-8310, Question 05.04.02.02-5 is resolved and closed.

5.4.2.2.6 *Conclusion*

On the basis of its review of DCD Tier 2 Subsection 5.4.2.2, the staff concludes that the APR1400 SG Program as described in the DCD, satisfies the acceptance criteria for accessibility for periodic inspection and testing of critical areas for structural and leakage integrity. The staff also concludes that the proposed TS are consistent with the STS, and the tube plugging criteria were determined using the methodology specified in RG 1.121. Therefore, the staff concludes that the SG Program is acceptable and meets the requirements of GDC 32, 10 CFR 50.55a, 10 CFR 50.36,10 CFR 50.65, 10 CFR 50.47(b)(1), and Appendix B to 10 CFR 50.

5.4.3 Reactor Coolant Piping

The RCS piping includes the main coolant lines for the two loops, the pressurizer surge line and the pressurizer spray lines. Portions of other systems such as the SIS piping (refer to DCD Tier 2 Section 6.3,, “Safety Injection System”), SCS piping (refer to DCD Tier 2 Section 5.4.7, “Shutdown Cooling System”), and chemical and volume control system piping (refer to DCD Tier 2 Section 9.3.4, “Chemical and Volume Control System”) constitute a part of the RCPB, but are not part of the RCS piping.

Detailed information about the reactor coolant piping and the staff’s evaluation and conclusion regarding APR1400’s reactor coolant piping design features and performance requirements are discussed in the following sections of this report:

- Section 3.9.1, “Special Topics for Mechanical Components.”
- Section 3.9.2, “Dynamic Testing and Analysis of Systems, Components, and Equipment.”
- Section 3.9.3, “ASME Code Class 1, 2, and 3 Components, Component Supports, and Core Support Structures.”
- Section 3.9.6, “Functional Design, Qualification, and Inservice Testing Programs for Pumps, Valves, and Dynamic Restraints.
- Section 3.10, “Seismic and Dynamic Qualification of Mechanical and Electrical Equipment.”
- Section 3.12, “ASME Code Class 1, 2, and 3 Piping Systems, Piping Components and Their Associated Supports.”

- Section 5.2.3, “Reactor Coolant Pressure Boundary Materials.”
- Section 5.2.4, “Reactor Coolant Pressure Boundary Inservice Inspection and Testing.”
- Section 5.2.5, “Reactor Coolant Pressure Boundary Leakage Detection.”
- Section 5.4.7, “Residual Heat Removal System.”
- Section 6.3, “Emergency Core Cooling System.”
- Section 6.6, “Inservice Inspection of Class 2 and 3 Components.”

5.4.7 Shutdown Cooling System

5.4.7.1 *Introduction*

The SCS is a safety-related system designed to remove decay heat from the reactor cooling system during different modes of reactor operations when the reactor conditions are below approximately 350 °F (176.7 °C) and 450 psia (31.6 kg/cm2A). Also, the SCS is operated while the plant is shutdown during reduced inventory, such as mid-loop operation. Additionally, the SCS is used to cooldown the RCS following a small break LOCA (SBLOCA) (refer to Section 6.3). The SCS is designed with two independent trains that can operate in both single and two loop operation. Each train has its own suction and discharge connections to the RCS.

This section of the DCD describes the design basis, system operation, instrumentation and testing requirements of the SCS. DCD Tier 2 Section 6.3, “Emergency Core Cooling System,” provides additional information of the SCS as it relates to the interface with the SIS.

5.4.7.2 *Summary of Application*

After the reactor coolant temperature and pressure have been reduced to below their designed values for SCS operation, as shown in the table below, the SCS is placed into shutdown cooling service to maintain or reduce the RCS temperature to the temperature defined by the SCS mode of operation: (1) Normal Shutdown, (2) Safety Shutdown, and (3) Refueling Shutdown. In the safety shutdown mode, the SCS will be used for accidents such as SBLOCA, steam and feedwater line breaks, and SG tube ruptures. In addition, the SCS is used for plant heatup operations that bring the RCS from cold shutdown to hot standby. The SCS operational modes are summarized in the table below with approximate initial and final reactor conditions.

Mode Of Operation	**Initial Conditions**			**Final Conditions**		
	Pressure (kg/cm2A) / (psia)	**Temperature (°C) / (°F)**	**Flow Rate (L/min) & (gpm)**	**Time After Shutdown (Hours)**	**Temperature (°C) / (°F)**	**Trains In Service**
Normal S/D	31.6 / 450	~176.7 / ~350	18,927 / (5,000)	24 40	60 / 140 54.4 / 130	2 2

Safety S/D	28.1 / 400	~193.3 / ~380	18,927 / (5,000)	24	93.3 / 200	1
Refueling			18,927 / (5,000)	96	~48.9 / ~120	2
Startup	Variable	Variable	18,927 / (5,000)	Pressure: 31.6 kg/cm^{2}A / 450 psia	176.7 / 350	2

Prior to fuel loading, the SCS configuration and component operational performance is verified through a series of inspections, tests, and analyses whereupon the results are compared against the acceptance criteria as identified in DCD Tier 1 Section 2.4.4, "Shutdown Cooling System," Table 2.4.4-4, "Shutdown Cooling System ITAAC." During the initial startup following fuel loading, operability tests will be performed to determine and confirm the SCS safety related functional capabilities are within the system designed parameters. Also, other data is obtained to determine system operational limitations, such as the SCS flow rate, to avoid vortexing during mid-loop operation.

The SCS is considered in the intersystem loss-of-coolant accident (ISLOCA)) event analysis because the SCS is directly connected to the RCS, which makes it a primary interface through which an ISLOCA event can occur. The event is postulated by over-pressurization of the SCS originating from the hot leg and out of containment through the containment isolation valves to the low-pressure sections of the SCS. Once the SCS is over-pressurized, other low-pressure systems are susceptible, such as the CVCS, sampling system (SS), containment spray system (CSS), and SIS.

Since the SCS is a safety related system, the two trains are powered from both the normal plant electrical system and the emergency electrical system with each SCS train having their independent power source from the other train to allow SCS operations with any single electrical failure.

DCD Tier 1: The DCD Tier 1 information associated with this section is found in DCD Tier 1 Section 2.4.4, "Shutdown Cooling System."

DCD Tier 2: The applicant provided a description of the SCS in APR1400 DCD Tier 2 Section 5.4.7, summarized here in part, as follows:

The SCS is comprised of two physically separate and independently powered trains of safety-related equipment. Each train utilizes a low-head pump to draw RCS water from a RCS hot leg and return to the RCS through SIS direct vessel injection nozzles. The RCS water passes through the tube side of a shutdown cooling HX where it is cooled by CCWS water and returned to the reactor vessel.

The controls required to operate the SCS are provided in the MCR, with control room indications for parameters such as system flow, pressure, and temperature. Controls and displays are also available at the remote shutdown room (RSR).

ITAAC: The ITAAC associated with DCD Tier 2 Section 5.4.7 is given in DCD Tier 1 Section 2.4.4, Table 2.4.4-4, “Shutdown Cooling System ITAAC.”

TS: The TS associated directly or indirectly with DCD Tier 2 Section 5.4.7 are given in DCD Tier 2 Chapter 16:

- Section 3.4.6, “RCS Loops – MODE 4.”
- Section 3.4.7, “RCS Loops – MODE 5 (Loops Filled).”
- Section 3.4.8, “RCS Loops – MODE 5 (Loops Not Filled).”
- Section 3.4.10, “Pressurizer Pilot Operated Safety Relief Valves (POSRVs).”
- Section 3.4.11, “Low Temperature Overpressure Protection (LTOP) System.”
- Section 3.4.13, “RCS Pressure Isolation Valve (PIV) Leakage.”
- Section 3.7.7, “Component Cooling Water System (CCWS).”
- Section 3.7.8, “Essential Service Water System (ESWS).”
- Section 3.9.4, “Shutdown Cooling System (SCS) and Coolant Circulation – High Water Level.”
- Section 3.9.5, “Shutdown Cooling System (SCS) and Coolant Circulation – Low Water Level.”

Initial Plant Testing:

- Section 14.2.12.1.20, “Shutdown Cooling System Test.”
- Section 14.2.12.1.46, “Pre-Core Hot Functional Test Controlling Document.”
- Section 14.2.12.1.76, “Component Cooling Water System Test.”
- Section 14.2.12.1.126, “Mid-Loop Operations Verification Test.”
- Section 14.2.12.4.22, “Natural Circulation Test.”

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 SCS.

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, “Control Room,” as it relates to including necessary instrumentation and controls for the SCS 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 the SCS.

6. Title 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 SCS 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 which 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 SCS 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 SCS 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 SCS system.

5.4.7.4 *Technical Evaluation*

The staff reviewed and evaluated the functional design bases with the system design to determine that the SCS system complies with the relevant regulatory requirements of GDC 2, 4, 5, 19, and 34 and BTP 5-4.

The applicant stated that the SCS is designed to remove decay heat, RCS sensible heat, and heat generated by the shutdown cooling pumps (SCPs) during normal plant cooldown, safe cold shutdown and refueling conditions after partial cooldown has been accomplished. In addition, the SCS will provide heatup from cold shutdown conditions during startup until the RCS pumps are placed in service.

The applicant also stated that the SCS is designed with two independent SCS trains to accomplish heat removal and heat injection; therefore, no single failure prevents operation of one complete SCS train with system control from the MCR during normal plant cooldown, refueling, and a DBE.

The two independent SCS trains consist of their own primary physical components, electrical power sources, and instrumentation and controls. The staff reviewed numerous figures and tables provided by the applicant in the DCD Tier 2, as well as appropriate startup testing procedures and the Tier 1 ITAAC sections to provide a preliminary confirmation that the SCS trains are indeed independent.

The applicant stated that the isolation valves, relief valves, and appropriate piping between high pressure sources, such as the RCS, and lower pressure designed SCS components will provide overpressurization protection of the SCS components. The applicant also stated that SCS components are designed for a maximum pressure of 900 psig (63.3 kg/cm^2G) but operate in a range well below this value as established by the LTOP analysis and P-T limits based on the limit curves developed for protecting the integrity of the RCPB to withstand the effects of system pressure and temperature changes during normal heatup and cooldown of the RCS and AOOs. Isolation valves are designed to protect SCS components through a series of interlocks actuated from pressure sensors set at 450 psia (31.6 kg/cm2A). In addition, pressure relief valves will be installed on the SCS suction line with a rated capacity to adequately provide over-pressure protection of the reactor cooling system during shutdown cooling from accidental events related to the operation of the SIPs, pressurizer heaters, the charging pump, and the reactor coolant pump (RCP). The applicant also stated that the thermal relief valves will be installed in isolated sections of common piping shared with SIS to inhibit overpressurization due to thermal transients. The staff reviewed the Tier 2 DCD RCS, SCS, and SIS PI&Ds and evaluated the relief valves component characteristics. Based on this review and evaluation, the staff concludes that the SCS components are adequately protected from overpressurization events.

The applicant stated that a relief valve on each of the SCS suction lines (SI-179, SI-189) is sized to have sufficient capacity to provide LTOP for the RCS due to accidental operation of the safety injection pumps (SIPs), pressurizer heaters, the charging pump, and the RCP during shutdown cooling. The staff reviewed and determined that the relief valve design parameters are reasonable as summarized in Table 5.2-3, “SCS Suction Line Relief Valve Valves (SI-179 and SI-189) Design Parameters.” The staff concluded that the SCS relief valve design is acceptable to mitigate a pressure transient during plant operation where the SCS is exposed to RCS pressure while providing heat removal.

In addition, the applicant stated that no single failure allows the SCS to be over-pressurized by the RCS. The staff reviewed the failure modes and effects analysis (FMEA) provided in DCD Table 5.4.7-2, “SCS Failure Modes and Effects Analysis,” which included all SCS component failures that have the potential to disable the SCS. The staff concluded that the applicant successfully demonstrated that a single failure cannot lead to failure of the SCS because of redundancy and separation of the SCS trains. The staff agreed that there is no credible single failure that would disable the SCS or allow the SCS to be over pressurized by the RCS.

To improve the availability of the CSS when one or two containment spray pumps (CSPs) are out of service, the applicant designed these systems with interchangeable pumps. The staff reviewed the piping and instrumentation diagrams (P&IDs) and confirmed that the Train 1 SCS pump is paired with the Train 1 CSS pump, and the same for the Train 2 SCS to the Train 2 CSS. The staff determined that the pump interchange configuration is acceptable. In addition, the staff confirmed the SCS pump characteristics and the CSS pump characteristics were comparable; however, there were no CSS net positive suction head (NPSH) available/required data to compare with SCS NPSH data. Also, there was no CSS pump characteristic curve to compare with the SCS pump curve. The staff could not find CSS and SCS elevation diagrams to evaluate the NPSH. The staff issued RAI 221-8248, Question 05.04.07-2 (ML15295A513), requesting the applicant to provide additional information regarding SCP and CSP NPSH during shutdown and long term cooling, since these pumps are interchangeable.

In its response to RAI 221-8248, Question 05.04.07-2 (ML16093A008), the applicant provided the requested NPSH data for both SCPs and CSPs. The applicant provided NPSH data required to determine the NPSH available for both the SCPs and CSPs. The staff reviewed the data that indicated sufficient NPSH available margin existed to allow the use of the SCPs with the CSS for long term cooling. The staff determined that this is acceptable because in the long term cooling SCPs/CSS configuration, the SCPs will provide adequate flow. In addition, the applicant will revise DCD Tier 2Table 5.4.7-1, “Shutdown Cooling System Design Parameters,” to clarify the parameter as “Minimum NPSH Available.” The staff confirmed that DCD Tier 2 was revised as committed in the response to RAI 221-8248, Question 05.04.07-2; therefore, RAI 299-8310, Question 05.04.02.02-5 is resolved and closed**.**

RG 1.29 provides guidance used to establish the seismic design classification to meet the requirements of GDC 2. Compliance with GDC 2 requires that nuclear power plant SSCs important to safety be designed to withstand the effects of natural phenomena, including earthquakes, without loss of capability to perform their safety functions. The staff confirmed that SCS seismic design classification is seismic Category I which satisfies GDC 2.

During the Chapter 5 ACRS subcommittee presentation, it was noted during the CSP/SCP interchange discussion that the suction line of the CSP is not isolated when an operable SCP is interchanged with an inoperable CSP. As stated in the “Interim Letter: Chapters 2, 5, 8, 10, and 11 of the NRC Staff’s Safety Evaluation Report with Open Items Related to the Certification of the APR1400 Design,” issued by the ACRS (ML17052A307), the concern is that inspections or maintenance on a containment spray pump may require isolation of the pump suction flow path which cannot be achieved while the SCP is interchanged with a CSP. The staff confirmed that a shutdown cooling pump cannot provide automatic containment spray flow during conditions when the suction paths for the associated containment spray pump is isolated because the valve that provides isolation to the CSP suction line must also be in the open position to link the SCP to the containment spray system and provide permissive for signals to automatically start the shutdown cooling pump in the spray mode.

In response to the ACRS conclusion and recommendations, the applicant stated (ML17284A105):

> *When a containment spray (CS) pump requires maintenance or failure during plant MODES 1, 2, 3 and the maintenance or failure require a breach of pressure boundary of the CS pump, the shutdown cooling pumps cannot perform its interchangeable function. Therefore, the affected train will be isolated and the plant will operate under LCO (TS 3.6.6). However, if the CS pump maintenance*

> *or failure during plant MODES 1, 2, 3 and the maintenance or failure does not require a pressure boundary breach, the shutdown cooling pumps can perform its interchangeable function. In this case, the plant would not need to operate under LCO.*

Therefore, the staff determined that the applicant's response is acceptable because the CSP/SCP interchange operation is controlled by TS 3.6.6.

Natural Circulation Cooldown

In DCD Tier 2 Sections 5.4.7.3.1.1 and 5.4.7.3.1.2, the applicant discussed a LOOP to the RCPs that results in a reactor trip, and the sequence of operator actions to establish a natural circulation cooldown to the shutdown cooling range where SCS is initiated. The applicant described the steps taken by the operator to manually control atmospheric dump valves (ADVs) to restore and maintain the secondary pressure to no-load hot standby conditions. Then, the SG level is restored by the operator manually controlling the auxiliary feedwater flow rate before manually manipulating the pressurizer vent, SIS, and reactor vessel upper head to lower the RCS pressure to allow the operation of the SCS.

This AOO event is also discussed in DCD Tier 2 Section 15.3.1, "Loss of Forced Reactor Coolant Flow," where the applicant described the event with respect to thermal hydraulics and the core protection calculator that generates a reactor trip provides assurance the minimum departure from nucleate boiling ratio (DNBR) value will remain above the specified acceptable fuel design limit (SAFDL) for DNBR.

However, in regard to startup testing, the staff could not identify any information in DCD Tier 2 Sections 5.4.7 and 15.3, related to the natural circulation power-to-flow ratio of less than 1.0 being an adequate test acceptance criterion for DCD Tier 2 Section 14.2.12.4.22. The staff issued RAI 384-8100, Question 05.04.07-3 (ML16032A106), requesting the applicant to provide: (1) additional information in DCD Tier 2 Sections 5.4.7 and 14.2.12.4.22 to address and justify meeting NRC SRP BTP 5-4, Section B, BTP, Section 5, "Test Requirements" and (2) documents related to the natural circulation analysis and test including a discussion of the relationship of the above conditions with respect to the reactor power conditions at the initiation of the test.

In its response to RAI 384-8100, Question 05.04.07-3 (ML16077A291), the applicant included a definition of power to flow ratio and referencing of RG 1.68, Appendix A, Subsection 4.t requirements as the reason to perform this test. In addition, the applicant stated that it will revise DCD Tier 2 Section 5.4.7.3.1 and Section 14.2.12.4.22, to clarify the natural circulation analysis and testing.

The applicant defined the power to flow ratio as the enthalpy rise across the reactor core during natural circulation to the enthalpy rise across the reactor core at full power design conditions. The staff determined that this is reasonable because it indicates that natural circulation is adequate to remove a sufficient amount heat to maintain the enthalpy below the design value; delta enthalpy is equated to the ratio of (power or decay heat) to mass flow rate. This also implies that the fuel temperature is maintained within the design limits. The staff determined that the applicant's response is acceptable because it adequately discussed the relationship of the power to flow ratio to the natural circulation test and, since the ratio was not defined in the current DCD, the applicant committed to revise DCD Tier 2 Section 5.4.7.3.1, to include a clarification of the power to flow ratio. The staff confirmed that DCD Tier 2 was revised as committed in the response to RAI 384-8100, Question 05.04.07-3. Therefore, RAI 384-8100, Question 05.04.07-3, is resolved and closed.

Gas Accumulation

In DCD Tier 2 Section 5.4.7, staff determined that additional information or references were needed regarding the following issues described in DC/COL-ISG-019, "Review of Evaluation to Address Gas Accumulation Issues in Safety Related Systems and Systems Important to Safety," related to potential gas accumulation in the SCS, as described in GL 2008-01:
(1) identification of potential gas accumulation locations and intrusion mechanisms,
(2) confirmation of the P&ID and isometric drawing against the as-built configuration, and
(3) implementation of surveillance and venting procedures. In addition, there was no reference to the NRC endorsed guidance documented in NEI 09-10, Revision 1a-A, "Guidelines for Effective Prevention and Management of System Gas Accumulation."

In DCD Tier 1 Section 2.4.4.1, the applicant stated "the ASME Code piping including supports, and design features described in the design basis to limit potential gas accumulation, identified in Table 2.4.4-1 is designed and constructed in accordance with ASME Section III requirements." The staff determined that the applicant's discussion and Table 2.4.4-1, were of high level general information that did not provide sufficient technical information to identify potential gas intrusion and accumulation locations and address measures to avoid or minimize gas accumulation. The applicant included an ITAAC, however the staff determined that additional information was needed regarding minimizing gas accumulation and gas intrusion to ensure that SCPs would function properly during SCS operation.

SCS gas accumulation and vortexing are also addressed in DCD Tier 2 Section 19.2, "Severe Accident Evaluation," related to mid-loop operation. The SCS mid-loop evaluation is discussed in Chapter 19 of this SER.

The staff issued RAI 42-7945, Question 19-2 (ML16203A282), requesting the applicant to provide additional information regarding gas accumulation and vortexing, during mid-loop operation. The staff asked whether ITAAC consistent with DC/COL ISG -019 would be included to verify that decay heat removal will not be impaired by gas entrainment during mid-loop operation while the system is operating at its maximum allowable flow rate, and the reactor coolant hot leg level is at the lowest level allowable and to address the SCS pipe slope.

In its response to RAI 42-7945, Question 19-2 (ML15212A687), the applicant provided the staff's evaluation of the response discussed in a public meeting with the applicant in May 2016. The staff is awaiting a revised response from the applicant. This issue is being evaluated under SER Section 19.2.

The staff issued RAI 492-8614, Question 05.04.07-4 (ML16147A594), requesting the applicant to clarify gas accumulation and gas entrainment requirements with respect to ITAAC. In its response to RAI 492-8614, Question 05.04.07-4 (ML16190A320), the applicant included revisions to DCD Tier 1 Sections 2.4.1.1, 2.4.3.1, 2.4.4.1, and 2.11.2.1 and ITAAC Tables 2.4.1-4, 2.4.3-4, 2.4.4-4, and 2.11.2-4 to address recommendations in GL-2008-01 and NEI 09-10, Revision 1a-A regarding gas accumulation and gas entrainment during power and mid-loop operations. The staff finds the ITAAC revision acceptable because it is consistent with the recommendations of GL-2008-01 and NEI 09-10, Revision 1a-A. The staff confirmed that DCD Tier 2 was revised as committed in the response to RAI 492-8614, Question 05.04.07-4; therefore, RAI 492-8614, Question 05.04.07-4, is resolved and closed.

Preoperational Testing and Post-Core Hot Functional Testing

In DCD Tier 2 Section 5.4.7.4.1, "Preoperational Testing," the applicant summarized the SCS tests performed during the ITP to confirm that the as-built SCS will satisfy system operability and provides a level of performance that satisfies design analyses for a safe cold shutdown.

The SCS is functionally tested in Phases I and II that includes functional tests: (1) 14.2.12.1.20, "Shutdown Cooling System Test," (2) 14.2.12.1.46, "Pre-Core Hot Functional Test Controlling Document," 14.2.12.1.126, "Mid-Loop Operations Verification Test," and 14.2.12.2.1 "Post-Core Hot Functional Test Controlling Document." The staff reviewed and determined that additional information is needed to complete the SCS startup testing evaluation because the appendices referenced in the test could not be found. The staff issued RAI 221-8248, Question 05.04.07-1 (ML15295A513**)**, requesting the applicant to provide or indicate the location of the appendices related to the execution of the pre-core hot functional test.

In its response to RAI 221-8248, Question 05.04.07-1 (ML16034A613), the applicant stated that DCD Tier 2 Sections 14.2.12.1.46 and 14.2.12.2.1, will be revised to replace the appendices with Tables 14.2-1 and 14.2-2. The staff finds the response acceptable because the applicant included tables that identify the plant conditions for the test. The staff confirmed that DCD Tier 2 was revised as committed in the response to RAI 221-8248, Question 05.04.07-1. Therefore RAI 221-8248, Question 05.04.07-1, is resolved and closed.

5.4.7.5 *Combined License Information Items*

There are no COL information items associated with Section 5.4.7 of the APR1400 DCD.

5.4.7.6 *Conclusion*

The staff reviewed and evaluated Section 5.4.7 of the APR1400 DCD. The scope of the review included the design bases, system description and performance, FMEA, inspections and tests, and instrumentation requirements.

The staff has determined that DCD Tier 2 Section 5.4.7 is acceptable based on meeting the regulatory requirements listed below.

- GDC 2 is satisfied because SCS complies with RG 1.29 seismic design classification for a safety related system.

- GDC 4 is satisfied because the SCS is designed to self-venting due to an upward slope of the piping which reduces the potential of voiding, thus water hammer events.

- GDC 5 is met because the APR1400 is a single unit design.

- GDC 19 requirements are satisfied because SCS has adequate instrumentation and controls in the MCR and at the RSR.

- GDC 34 is satisfied because the CSC design complies with the functional, isolation, pressure relief, pump protection, and test-related criteria specified BTP 5-4.

- Title 10 CFR 50.34(f)(2)(xxvi) requirements are met because SCS is designed with a leakage detection and control program to minimize the leakage from of the SCS outside of the containment.

5.4.10 Pressurizer

5.4.10.1 *Introduction*

RCS pressure is controlled by the pressurizer, where steam and water are maintained in thermal equilibrium. Steam is formed by energizing immersion heaters in the pressurizer, or is condensed by the pressurizer spray to limit pressure variations caused by contraction or expansion of the reactor coolant.

5.4.10.2 *Summary of Application*

The pressurizer is a vertically mounted, bottom supported, cylindrical pressure vessel. Replaceable direct immersion electric heaters are vertically mounted in the bottom head. The pressurizer is furnished with nozzles for the spray, surge, and pilot-operated safety relief valves, and with pressure, temperature, and level instrumentation. The pressurizer surge line is connected to one of the reactor coolant hot legs and the spray lines are connected to two of the cold legs at the reactor coolant pump discharge. The pressurizer is described in Section 5.4.10, “Pressurizer.”

5.4.10.3 *Regulatory Basis*

Although NUREG-0800 does not contain a Section 5.4.10, the pressurizer is included in the review scope of NUREG-0800, Section 5.4, “Reactor Coolant System Component and Subsystem Design.” Detailed information about the pressurizer (PZR) and the staff’s evaluation and conclusion regarding pressurizer design features and performance requirements is discussed in the following sections of this report:

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

5.4.11 Pressurizer Relief Tank

The typical PWR primary systems include a PRT to condense and cool the discharge from the pressurizer safety and relief valves. The applicant stated that the APR1400 does not include a PRT, but routes the discharge from the pressurizer safety and relief valves to the safety related IRWST. The applicant designed the piping between the pressurizer safety and relief valves and the IRWST to Seismic Category II standards to protect safety related equipment from a pipe failure. Section 6.8, “In-containment Water Storage System,” of this SER presents the staff’s evaluation of the IRWST.

5.4.12 Reactor Coolant System High Point Vents

5.4.12.1 *Introduction*

The staff reviewed DCD Tier 2 Section 5.4.12, “Reactor Coolant System High Point Vents,” to ensure that RCS high point vents are provided and adequate to remove noncondensible gases from points in the RCS where gas accumulation could inhibit natural circulation core cooling. This high point venting capability may be necessary to maintain long-term core cooling following an accident. The APR1400 high point vents are connected to the reactor vessel closure head (RVCH) and pressurizer steam space and are collectively called the reactor coolant gas vent system (RCGVS). The staff’s review focused on the following aspects of the RCGVS: system design, operability from the control room, venting capability and performance, the probability of inadvertent or spurious actuation, and assessment of potential vent line breaks with respect to a loss-of-coolant accident (LOCA).

5.4.12.2 *Summary of Application*

DCD Tier 1: The DCD Tier 1 information associated with this section is found in DCD Tier 1 Section 2.4.5, “Reactor Coolant Gas Vent System.” The high point vents arrangement is shown in DCD Tier 1 Figure 2.4.5-1, “Reactor Coolant Gas Vent System.” Equipment and piping locations and characteristics for the RCGVS are provided in DCD Tier 1 Table 2.4.5-1, “Reactor Coolant Gas Vent System Equipment and Piping Location/Characteristics.” Components and instruments are listed in Tables 2.4.5-2, “Reactor Coolant Gas Vent System Component List,” and 2.4.5-3, “Reactor Coolant Gas Vent System Instrument List.”

DCD Tier 2: The applicant provided a DCD Tier 2 system description in DCD Tier 2 Section 5.4.12 that is summarized as follows:

The RCGVS has two major safety-related functions for post-accident conditions: to remotely discharge noncondensible gases and/or steam from the high points of the RCS (the RVCH and pressurizer steam space) to the IRWST; and to control RCS pressure should pressurizer main and auxiliary sprays be unavailable. The RCGVS also provides a non-safety-related path to vent noncondensible gases to the IRWST or RDT when filling the RCS during plant startup.

DCD Tier 2, Figure 5.4.12-1, “Reactor Coolant Gas Vent System Flow Diagram,” presents a flow diagram of the RCGVS. The RCGVS connects to the RCS at the RVCH and at the steam sample/vent line on the pressurizer upper head. Illustrations of the reactor vessel and the pressurizer in DCD Tier 2, Figures 5.3-8, “Reactor Vessel Assembly,” and 5.4.10-1, “Pressurizer,” show these connections. The vent paths coming from both the RVCH and pressurizer divide into parallel branches that contain two normally closed solenoid-operated valves in series. Independent divisions of the 125 V direct current (DC) power system power the valves. The 125 V DC power system battery chargers receive power from either the normal or emergency ac power source.

The MCR and RSR are equipped to remotely operate the RCGVS valves and view their position indications. The MCR and RSR also contain temperature and pressure readouts for instrumentation downstream of the RCGVS valves.

The applicant designed the RCGVS equipment and piping up to and including the second vent valve to seismic Category I, Class 1E and ASME Code Section III, Class 1 requirements. The solenoid-operated valves in the vent path parallel branches are qualified using ANSI/IEEE Std. 344. The applicant described ISI and testing for the ASME Code Class 1 components that are part of the RCPB in DCD Tier 2 Section 5.2.4, “In-service Inspection and Testing of the Reactor Coolant Pressure Boundary,” and described ISI and testing for valves in DCD Tier 2 Section 3.9.6, “Functional Design, Qualification, and In-service Testing Programs for Pumps, Valves, and Dynamic Restraints.”

In addition, the applicant designed the RCGVS to meet the requirements of 10 CFR 20.1406, “Minimization of Contamination.”

ITAAC: The ITAAC associated with DCD Tier 2 Section 5.4.12 are given in DCD Tier 1 Section 2.4.5, Table 2.4.5-4, “Reactor Coolant Gas Vent System ITAAC.”

TS: The TS associated with DCD Tier 2 Section 5.4.12 are provided in DCD Tier 2, Chapter 16, Section 3.4.16, “Reactor Coolant Gas Vent (RCGV) Function.”

5.4.12.3 *Regulatory Basis*

The relevant requirements of NRC regulations for RCS high point vents, and the associated acceptance criteria, are identified in NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition” (SRP), Section 5.4.12, “Reactor Coolant System High Point Vents,” and are summarized below. Review interfaces with other SRP sections can be found in NUREG-0800, Section 5.4.12. The requirements governing ITAAC are provided in NUREG-0800, Section 14.3, “Inspections, Tests, Analyses, and Acceptance Criteria.”

1. GDC 1, “Quality Standards and Records,” GDC 30, “Quality of Reactor Coolant Pressure Boundary,” and 10 CFR 50.55a, “Codes and Standards,” as they relate to the vent system components that are part of the RCPB being designed, fabricated, erected, tested, and maintained to high quality standards.

2. GDC 14, “Reactor Coolant Pressure Boundary,” 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, “Electric Power Systems,” and GDC 34, “Residual Heat Removal,” as they relate to the provision of normal and emergency power for the vent system components to assure that specified acceptable fuel design limits and RCPB limits are not exceeded.

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

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

6. Title 10 CFR 50.34(f)(2)(vi), as it relates to the capability of high point venting of noncondensible gases from the RCS, the venting systems being operable from the MCR, and venting operation not leading to an unacceptable increase in LOCA probability or an unacceptable challenge to containment integrity.

7. Title 10 CFR 50.44, “Combustible Gas Control for Nuclear Power Reactors,” as it relates to combustible gas control in the containment.

8. Title 10 CFR 50.46(b)(5), as it relates to decay heat removal for an extended period of time following any calculated successful initial operation of the ECCS.

9. Title 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 noncondensible gases would cause the loss of function of these systems.

10. Title 10 CFR 50.49, “Environmental Qualification of Electric 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.

The related acceptance criteria are as follows:

1. The reactor coolant vent design must ensure that use of these vents during and following an accident does not aggravate the challenge to containment or the course of the accident.

2. Vent capability should be provided on high points of the RCS (including the pressurizer on PWRs) to vent gases, which may inhibit core cooling. For reactors with U-tube SGs, 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. In general, vent paths are not required for local high points at locations where gas accumulation would not be expected to jeopardize core cooling, such as a reactor coolant pump valve body.

3. A single failure of a vent valve, power supply, or control system should not prevent isolation of the vent path.

4. The design should incorporate sufficient redundancy to minimize the probability of inadvertent actuation. Other methods to reduce the chances of inadvertent actuation, such as removing power or administrative controls, may be considered.

5. Since the RCS vent will be part of the RCPB, all requirements for the RCPB must be met.

6. The size of the vent should be smaller than the size corresponding to the definition of a LOCA (Appendix A to 10 CFR Part 50) to avoid unnecessary challenges to the ECCS, unless the applicant provides justification for a larger size.

7. Vent paths to the containment should discharge into areas that provide good mixing with containment air and are able to withstand steam, water, noncondensibles, and mixtures of the above.

8. The vent system should be operable from the control room and provide positive valve position indication. Power should be supplied from emergency buses.

9. It is important that the control room displays and controls for the RCS vents do not increase the potential for operator error. A human-factor analysis should be performed that considers the following:

    A. The use of this information by an operator during both normal and abnormal plant conditions.

    B. Integration into emergency procedures.

    C. Integration into operator training.

    D. Other alarms during an emergency and need for prioritization of alarms.

10. The design should have provisions for testing the operability of the reactor coolant vent system. Testing should be performed in accordance with Subsection IWV of Section XI of the ASME Code for Category B valves.

11. The reactor coolant vent system (i.e., vent valves, block valves, position indication devices, cable terminations, and piping) should be seismically and environmentally qualified in accordance with IEEE Std. 344, "IEEE Recommended Practice for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations," as supplemented by RG 1.100, "Seismic Qualification of Electrical and Active Mechanical Equipment and Functional Qualification of Active Mechanical Equipment for Nuclear Power Plants," and RG 1.92, "Combining Modal Responses and Spatial Components in Seismic Response Analysis." Environmental qualifications must be in accordance with 10 CFR 50.49.

12. The reactor coolant vent system should be designed to withstand the dynamic loads that will be encountered during operation from high RCS pressure to the approximate atmospheric pressure at the vent system exhaust.

13. Procedures to effectively operate the vent system must consider when venting is needed and when it is not needed. A variety of initial conditions for which venting may be required should be considered. Operator actions and the necessary instrumentation should be identified.

14. The reactor coolant vent system should meet the quality assurance acceptance criteria provided in SRP Chapter 17, "Quality Assurance."

5.4.12.4 *Technical Evaluation*

The staff reviewed DCD Tier 2 Section 5.4.12 in accordance with NUREG-0800, Section 5.4.12 to ensure the adequacy of the RCGVS to remove from high points in the RCS noncondensible gases that could hinder natural circulation core cooling after a DBE. The review covered the RCGVS to ensure conformance with the requirements of GDC 1, GDC 14, GDC 17, GDC 19, GDC 30, GDC 34, GDC 36, 10 CFR 50.34(f)(2)(vi), 10 CFR 50.44, 10 CFR 50.46(b)(5), 10 CFR 50.46a, 10 CFR 50.49, and 10 CFR 50.55a.

In addition to RAIs addressing specific technical issues related to RCGVS, discussed in detail below, the staff prepared several editorial or administrative RAIs to clarify certain typographical errors and inconsistent wording or information. This technical evaluation does not discuss these editorial and administrative RAIs because they do not alter the substantive technical information provided by the applicant.

System Design

The safety-related functions of the RCGVS are to vent noncondensible gases from the high points of the RCS to the IRWST and to depressurize the RCS following a non-LOCA DBE. The RCGVS also has the non-safety-related function of venting noncondensible gases during plant startup and shutdown. The staff issued RAI 175-8034, Question 05.04.12-2 (ML15295A499), requesting the applicant to ensure that the non-safety-related function would have no impacts on the safety venting function.

In its response to RAI 175-8034, Question 05.04.12-2 (ML16026A577), the applicant emphasized that the safety- and non-safety-related venting functions would not be necessary at the same time, so there would be no adverse impacts to the safety venting function. The staff agrees and notes that TS 3.4.16 requires RCGVS operability when the RCS is heating up or at power. Therefore, RAI 175-8034, Question 05.04.12-2, is resolved and closed.

DCD Tier 2, Figure 5.1.3-2, “Reactor Coolant System Arrangement Elevation,” shows that the high points of the RCS where gases could possibly accumulate are the RVCH, the pressurizer, and the U-tubes of the SGs. The RCGVS includes vents for the RVCH and the pressurizer. According to 10 CFR 50.46a, high point vents are not required for the tubes in U-tube SGs. Therefore, in accordance with 10 CFR 50.34(f)(2)(vi) and 10 CFR 50.46a, vents are provided for the RCS high points for the APR1400.

Although high point vents are not required for SG U-tubes, SRP Section 5.4.12 states that an applicant should develop procedures to remove gas from SG U-tubes to ensure continued core cooling. The applicant described a procedure to operate one or more reactor coolant pumps per loop for short periods to purge gas from the U-tubes. The staff notes that this procedure would be used prior to reactor startup, consistent with current plant operating practices. The procedure would not be used during a transient or accident, which may result in a natural circulation event.

As shown in DCD Tier 2, Figure 5.4.12-1, “Reactor Coolant Gas Vent System Flow Diagram,” both portions (RVCH and pressurizer) of the RCGVS contain two parallel paths that branch from a piping tee section connected to the respective vent line. The figure shows that these parallel paths contain two solenoid-operated valves in series that are normally closed and fail closed on loss of motive power. Each of the valves is powered from a separate division of the Class 1E 125 V power supply, as listed in DCD Tier 2, Table 5.4.12-1, “Reactor Coolant Gas Vent System – Active Valve List.” The alternate ac power source supplies power to the 125 V DC power system during a SBO.

The staff noted that the series arrangement of the valves minimizes the probability of spurious actuation of the RCGVS because the vent paths from the RVCH and pressurizer can be isolated given a single failure of a normal or emergency power division, and the fail-closed valves provide extra assurance of ability to isolate the vent paths. In accordance with 10 CFR 50.34(f)(2)(vi), this ensures that the vents do not lead to an unacceptable increase in the probability of a LOCA. The staff also notes that the parallel paths and their power scheme ensure that vent paths can be established given a single failure of a power train or division. The staff therefore concludes that the RCGVS satisfies GDC 17 and 34 with respect to the provision of normal and emergency power. Chapter 8, “Electric Power,” of this SER provides a detailed review of electrical separation, redundancy, independence, and power sources.

In accordance with 10 CFR 50.34(f)(2)(vi), 10 CFR 50.46a, and GDC 19, the isolation valves for the RCGVS are operable from the MCR and RSR. The applicant noted that both the MCR and RSR provide open or closed valve position indication. MCR and RSR displays and controls for the RCGVS should not increase the potential for operator error, a topic addressed in Chapter 18 of this SER.

The RVCH and pressurizer vent path portions merge downstream of the vent valves and discharge either to the RDT or through a sparger to the IRWST, a seismic Category I structure. As described in DCD Tier 1 Section 2.4.5, the non-safety function of the RCGVS during plant startup and shutdown can use either the RDT or the IRWST, while the RCGVS discharges only to the IRWST for the safety function. However, the Tier 2 information did not provide a clear description of when the RCGVS uses the paths to the IRWST and RDT. The staff issued RAI 175-8034, Question 05.04.12-1 (ML15295A499), requesting the applicant to provide clarification in Tier 2, including delineation between safety-related and non-safety-related functions.

In its response to RAI 175-8034, Question 05.04.12-1 (ML16107A070), the applicant proposed revising DCD Tier 2 to clarify that the RCGVS discharges to the IRWST through the RCGVS sparger for the safety vent function and to the RDT or IRWST for the non-safety function. The staff determined that the response is acceptable because the applicant clearly explained the destinations for RCGVS discharge for the safety versus non-safety gas vent operation. Based on the review of DCD Tier 2, the staff has confirmed incorporation of the changes described above; therefore, RAI 175-8034, Question 05.04.12-1, is resolved and closed.

In accordance with 10 CFR 50.34(f)(2)(vi), discharges from the RCGVS do not challenge containment design limits. Section 6.2.2 of this SER, “Containment Heat Removal Systems,” evaluates the adequacy of the IRWST to withstand discharge under postulated conditions. The top of the IRWST is equipped with swing panels to allow any combustible gases to mix with the containment atmosphere. The staff concludes that the RCGVS therefore satisfies 10 CFR 50.44 as it relates to adequate mixing in the containment air; however, Section 6.2.5 of this SER, “Combustible Gas Control in Containment,” provides a more detailed evaluation of combustible gas control in containment. In addition, the IRWST is part of the APR1400 engineered safety features (ESFs). Section 6.1 of this SER, “Engineered Safety Features Materials,” evaluates material selection for the ESFs.

Final Interim Staff Guidance DC/COL-ISG-019, “Review of Evaluation to Address Gas Accumulation Issues in Safety Related Systems and Systems Important to Safety,” identifies issues related to gas accumulation in nuclear systems. The staff noted that the RCGVS alleviates gas accumulation in the RVCH and pressurizer, but the potential exists for gas accumulation in the RCGVS piping; this could lead to water hammer, challenging piping integrity. The staff issued RAI 175-8034, Question 05.04.12-7 (ML15295A499), requesting the applicant to provide what measures it is taking to prevent gas accumulation in the RCGVS piping from the vent entrance points to the discharge locations.

In its response to RAI 175-8034, Question 05.04.12-7 (ML16055A434), the applicant discussed provisions to ensure that water cannot accumulate in the system and contribute to gas accumulation. These provisions include sloping of the RCGVS piping and inclusion of a vacuum relief valve to ensure the system can be drained even after system operation and closure of the vent isolation valves. The applicant committed to add this information to the DCD per the attached markups in its response. The staff determined that the response is acceptable because the measures facilitate gravity drainage of fluids that could cause gases to accumulate in the RCGVS.

Based on the review of DCD Tier 2, the staff has confirmed incorporation of the changes described above; therefore, RAI 175-8034, Question 05.04.12-7, is resolved and closed.

DCD Tier 2 Section 5.4.12 states that the RCGVS satisfies 10 CFR 50.49 and environmental qualifications. Although DCD Tier 1 Table 2.4.5-2 identifies the safety-related high point vent valves as qualified for a harsh environment, DCD Tier 2 Section 3.11, "Environmental Qualification of Mechanical and Electrical Equipment," did not discuss the RCGVS, nor did DCD Table 3.11-2, "Equipment Qualification Equipment List," include the RCGVS. The staff issued RAI 175-8034, Question 05.04.12-5 (ML15295A499), requesting the applicant to address the environmental qualification of the RCGVS.

In its response to RAI 175-8034, Question 05.04.12-5 (ML16103A461), the applicant clarified that, as a safety-grade system, the RCGVS valves are designed as Class 1E and to withstand harsh environments and that the environmental qualification of the RCGVS is performed in accordance with 10 CFR 50.49, RG 1.89, "Environmental Qualification of Certain Electric Equipment Important to Safety for Nuclear Power Plants," IEEE Std. 323, "IEEE Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations," and available equipment qualification standards. The applicant committed in the attached DCD markups to add the RCGVS valves to the equipment qualification list in DCD Tier 2 Table 3.11-2 and to correct DCD Tier 1 Table 2.4.5-2 to show that all safety-related RCGVS valves are qualified for a harsh environment.

The staff determines that the applicant's response to RAI 175-8034, Question 05.04.12-5, is acceptable because the applicant appropriately identified the environmental qualification of the RCGVS in accordance with 10 CFR 50.49. Based on the review of DCD Tier 1 and DCD Tier 2, the staff has confirmed incorporation of the changes described above: therefore, RAI 175-8034, Question 05.04.12-5, is resolved and closed. Section 3.11 of this SER, "Environmental Qualification of Mechanical and Electrical Equipment," contains the staff's evaluation of the applicant's environmental qualification program.

The RCGVS is a safety-related system. SER Section 3.9.1, "Special Topics for Mechanical Components," Section 3.9.2, "Dynamic Testing and Analysis of Systems, Components, and Equipment," Section 3.9.3, "ASME Code Class 1, 2, and 3 Components, and Component Supports, and Core Support Structures," and Section 3.12, "ASME Code Class 1, 2, and 3 Piping Systems, Piping Components and their Associated Supports," discuss the structural integrity of safety-related components and piping. In addition, the seismic qualifications of the high point vent components and piping are presented in DCD Tier 2 Table 3.2-1, "Classification of Structures, Systems, and Components," DCD Tier 2 Table 3.9-4, "Seismic Category I Active Valves," and DCD Tier 1 Table 2.4.5-1.

The applicant stated that it designed the RCGVS in accordance with DCD Tier 2 Chapter 17, "Quality Assurance and Reliability Assurance," and APR1400-K-Q-TR-11005, Revision 4, "KHNP Quality Assurance Program Description (QAPD) for the APR1400 Design Certification." The quality assurance program (QAP), together with the structural reviews and seismic qualifications of safety-related components and piping, provide reasonable assurance that the RCGVS meets GDC 1, 14, and 30 and 10 CFR 50.55a. Chapter 17 of this SER evaluates the QAP in detail.

According to DCD Tier 2 Section 5.4.12.2.3, “Design Features for Minimization of Contamination,” the RCGVS incorporates design features to meet the requirements of 10 CFR 20.1406, “Minimization of Contamination.” The adequacy of the design with regards to 10 CFR 20.1406 is reviewed in Section 12.4, “Dose Assessment and Minimization of Contamination,” of this SER.

The staff determined that the RCGVS meets the requirements of GDC 1, 14, 17, 19, 30, and 34; 10 CFR 50.34(f)(2)(vi); 10 CFR 50.44; 10 CFR 50.46a; 10 CFR 50.49; and 10 CFR 50.55a as they relate to high point vent design. The conclusion is based on equipment qualification, compliance with the QAP, meeting the requirements of the RCPB, the provision of normal and emergency power to the RCGVS vent valves, operability of the RCGVS from the control room, provisions for mixing of gases with the containment atmosphere, and a system arrangement designed to ensure availability of the RCGVS safety function without inadvertent or irreversible actuation.

RCGVS Operation

The applicant designed the RVCH vent portion of the RCGVS to have sufficient capacity to vent a steam bubble formed during natural circulation cooldown to allow refill of the reactor pressure vessel, both cooling the RVCH and depressurizing the RCS. The pressurizer vent portion is designed to reduce RCS pressure consistent with plant cooldown requirements should pressurizer main and auxiliary spray be unavailable, and its flow rates can be controlled using valves on top of the pressurizer and RVCH. The vent capacities of these portions are 4,797 kg/hr (10,576 lb/hr) and 14,023 kg/hr (30,915 lb/hr), respectively, at the design pressure of 175.8 kg/cm^2 A (2,500 psia). The staff audited RCGVS and natural circulation cooldown analysis flow rate calculations and ensured that the maximum RCGVS flow rate is sufficient to fulfill the RCGVS safety function.

DCD Tier 2 Section 5.4.12.3, “Performance Evaluation,” stated that a break in the vent line on the RVCH is categorized as a SBLOCA and that RVCH vent break phenomena are similar to those in DCD Tier 2 Section 15.6.5, “Loss-of-Coolant Accidents Resulting from the Spectrum of Postulated Piping Breaks within the Reactor Coolant Pressure Boundary.” The applicant therefore concluded that the results of DCD Tier 2 Section 15.6.5 conservatively bound the RVCH vent line break.

However, the SBLOCA analyses performed in DCD Tier 2 Section 15.6.5 are for breaks in the cold leg, direct vessel injection (DVI) line, or top of the pressurizer. A break in the RVCH vent line may behave differently from the analyzed locations. In addition, a break in the RVCH line would have an area of 3.37 cm^2 (0.00362 ft^2), which is considerably smaller than even the smallest break analyzed, an 18.6 cm^2 (0.02 ft^2) break in the DVI line. It was therefore unclear to the staff how the breaks analyzed could bound a potential break in the RVCH vent line. In addition, DCD Tier 2 Section 5.4.12.3 did not discuss a break in the pressurizer vent line or present supporting LOCA analyses. The staff issued RAI 175-8034, Question 05.04.12-3 (ML15295A499), requesting the applicant to address a break in the pressurizer vent line, providing supporting LOCA analyses if necessary, and to provide justification for why the results in DCD Tier 2 Section 15.6.5 conservatively envelop the RVCH vent line break scenario.

In its response to RAI 175-8034, Question 05.04.12-3 (ML16134A410), the applicant stated that the SBLOCA analyses in DCD Tier 2 Section 15.6.5 performed for nine different break sizes and an in-core instrument tube bound both the RVCH and pressurizer vent line breaks and that all of the SBLOCA analyses conclude the SIS satisfies SRP acceptance criteria for SBLOCAs. The staff noted that the analysis of a break in the top of the pressurizer presented in DCD Tier 2

Section 15.6.5 assumes a break size about twice the size of the RCGVS pressurizer vent line; therefore, the staff agrees that the pressurizer SBLOCA analysis bounds a RCGVS pressurizer vent line break. The staff notes that the in-core instrument tube break discussed in DCD Tier 2 Section 15.6.5 is close in size to a postulated RVCH vent line break, but the in-core instrument tube break would occur at the bottom of the reactor pressure vessel instead of the top. Because of potentially different thermal-hydraulic behaviors at the two vessel locations, the staff performed hand calculations to estimate the mass flow rates for both break scenarios. The staff determined that the mass flow rate for the in-core instrument tube break would be about two to three times as large as that for an RVCH vent line break. Therefore, the staff concludes that the existing SBLOCA analyses bound the RVCH vent line break. The staff's evaluation of the SBLOCA analyses is provided in Section 15.6.5 of this SER.

The staff determines that the applicant's response to RAI 175-8034, Question 05.04.12-3, is acceptable because it clarified how the SBLOCA analyses in DCD Tier 2 Section 15.6.5 bound potential breaks in the RCGVS. Based on the review of DCD Tier 2, the staff has confirmed incorporation of the changes discussed above; therefore, RAI 175-8034, Question 05.04.12-3, is resolved and closed. The staff's evaluation of the SBLOCA analyses is provided in Section 15.6.5 of this SER.

DCD Tier 2 Section 15.6.5 indicates the core remains covered for SBLOCAs. Because of this and the applicant's response to RAI 175-8034, Question 05.04.12-3, the staff finds that the high point vents comply with 10 CFR 50.46.

Required instrumentation for vent operation includes reactor vessel water level indication for the RVCH vents and pressurizer pressure and cold leg temperature indication for the pressurizer vents. Though not required for post-accident operation, the APR1400 design includes temperature and pressure instrumentation for the RCGVS valves after the RVCH and pressurizer vent portions merge to detect leakage and pressure buildup.

However, the description of system operation in DCD Tier 2 Section 5.4.12.3, "Performance Evaluation," including how the above instrumentation is used, was incomplete. The applicant did not provide details as to the initial conditions, such as the accident scenarios leading up to the need for RCGVS operation or how the operator would know when to initiate and terminate operation of the RVCH and pressurizer vent portions of the RCGVS. The staff issued RAI 175-8034, Question 05.04.12-6 (ML15295A499), requesting the applicant to provide procedures for the pressurizer portion of the RCGVS.

In its response to RAI 175-8034, Question 05.04.12-6 (ML16117A581), the applicant clarified in the attached markup of DCD Tier 2 Section 5.4.12.3 that the evaluation applies to both the RVCH and pressurizer vents. The applicant's markup also added a reference to DCD Tier 2 Section 5.4.7.3.1, "Performance Evaluation Assuming the Most Limiting Single Failure and Only Onsite Power Available," for information on the need for, and how an operator would know when to start and stop, RCGVS operation. DCD Tier 2 Section 5.4.7.3.1 provides a performance evaluation of a natural circulation cooldown given a LOOP and the most limiting single failure. The description therein clarifies that the operator initiates RCGVS pressurizer venting to depressurize the RCS when subcooling is high and terminates pressurizer venting when the subcooling margin reaches its minimum value or when the maximum-sized steam void forms in the RVCH. The operator initiates RVCH venting to cool the RVCH and reduce the steam void volume and terminates RVCH venting when the steam void collapses. The RAI response also clarified that a COL applicant will develop the RCGVS operating procedure as part of COL Information Items 13.5(4) and 13.5(5). These COL information items are evaluated in

Section 13.5 of this SER. In addition, the staff noted that the Bases for TS 3.4.16 specify that the RCGVS provides depressurization capability during natural circulation cooldown and SG tube rupture events.

The staff determines that the applicant's response to RAI 175-8034, Question 05.04.12-6, is acceptable because it clarified that the performance evaluation in DCD Tier 2 Section 5.4.12.3 is applicable to the entire RCGVS and pointed to other parts of the DCD that explain when an operator would initiate and terminate system operation. The staff also notes that a COL applicant's operating procedures will provide instructions on RCGVS operation to an operator.

Based on the review of DCD Tier 2, the staff has confirmed incorporation of the changes discussed above; therefore, RAI 175-8034, Question 05.04.12-6, is resolved and closed.

The staff determines that the RCGVS meets the requirements of 10 CFR 50.34(f)(2)(vi) and 10 CFR 50.46(b)(5) as they relate to high point vent operation. The conclusion is based on the applicant's justification that RCGVS operation will not lead to an unacceptable increase in LOCA probability or a challenge to the ECCS or containment and the applicant's handling of operating procedures.

RCGVS Tests and Requirements

The preoperational test procedures for the RCGVS described in DCD Tier 2 Section 14.2.12.1.37, titled "Safety Depressurization and Vent System Test" in Revision 0 of the DCD, intended to verify proper operation of the RCGVS. However, the staff noted that the test acceptance criteria were insufficient to verify RCGVS operation and incorrectly referenced other sections of the DCD. The staff issued RAI 175-8034, Question 05.04.12-9 (ML15295A499), requesting the applicant to address inconsistencies in DCD Tier 2 Subsection 14.2.12.1.37, and to either provide the testing requirements and associated acceptance criteria in DCD Tier 2 Sections 5.4.12 and 5.4.14, or to provide the specific test acceptance criteria for each test method in Subsection 14.2.12.1.37.

In its response to RAI 175-8034, Question 05.04.12-9 (ML16125A479), the applicant clarified that DCD Tier 2 Section 14.2.12.1.37 describes RCGVS testing, whereas DCD Tier 2 Section 14.2.12.1.3 describes testing of the safety depressurization function of the pressurizer POSRVs. Therefore, the applicant committed to revise the title of Section 14.2.12.1.37 to "Reactor Coolant Gas Vent System Test" and remove references to the POSRVs. The applicant provided a markup of Section 14.2.12.1.37 that removed incorrect references to other parts of the DCD and whose acceptance criteria clearly relate to the test objectives and methods. The staff concludes that the revised preoperational test procedures and their acceptance criteria are adequate to verify RCGVS operation; therefore, the staff determines that the response to RAI 175-8034, Question 05.04.12-9, is acceptable. Based on the review of DCD Tier 2, the staff has confirmed incorporation of the changes described above; therefore, RAI 175-8034, Question 05.04.12-9, is resolved and closed.

To satisfy GDC 36, the design and arrangement of the RCGVS should permit periodic ISI and testing. The in-service testing and inspection of valves is described in DCD Tier 2 Section 3.9.6, "Functional Design, Qualification, and In-service Testing Programs for Pumps, Valves, and Dynamic Restraints," and the requirements for the RCGVS valves are given in Table 3.9-13, "Inservice Testing of Safety-Related Pumps and Valves." The ISI and testing of ASME Code, Class 1 components that are part of the RCPB is described in DCD Tier 2 Section 5.2.4, "In-service Inspection and Testing of the Reactor Coolant Pressure Boundary." The respective sections of this SER evaluate the adequacy of ISI and testing.

The TS associated with the RCGVS are in DCD Tier 2 Chapter 16, Section 3.4.16, “Reactor Coolant Gas Vent (RCGV) Function.” TS 3.4.16 and the associated LCO and surveillance requirements provide assurance of RCGVS operability. Chapter 16 of this SER contains the detailed evaluation of the TS, including TS 3.4.16.

The staff determined that the RCGVS meets the requirements of GDC 36, as it relates to high point vent testing. The conclusion is based on the design of the RCGVS permitting periodic ISIs and tests.

Tier 1 Information

DCD Tier 1 Section 2.4.5 provides a design description for the RCGVS; tables listing system equipment, piping, components, and instruments; proposed ITAAC; and a diagram of the RCGVS. During the review of the RCGVS DCD Tier 1 material, the staff noted that the location of the RCGVS was not included in its associated design description. Following the guidance provided in SRP Section 14.3.4, the NRC staff issued RAI 83-7962, Question 14.03.04-2 (ML15197A267), to address this discrepancy. In its response to RAI 83-7962, Question 14.03.04-2 (ML15259A765), the applicant stated that the RCGVS location will be provided in DCD Tier 1 Section 2.4.5.1, “Design Description,” and provided proposed revisions to the DCD. The staff finds that the applicant’s revised DCD Tier 1 RCGVS design description supports the associated RCGVS ITAAC. Based on the review of DCD Tier 1, the staff has confirmed incorporation of the changes described above; therefore, RAI 83-7962, Question 14.03.04-2, is resolved and closed.

The staff also noted that DCD Tier 1 Figure 2.4.5-1, “Reactor Coolant Gas Vent System,” did not indicate which tank the RCGVS empties into, even though the associated design description indicates that RCGVS effluent from the pressurizer or reactor vessel closure head is transported to the IRWST through the RCGVS sparger. In accordance with the guidance provided in SRP Section 14.3.4, the staff issued RAI 83-7962, Question 14.03.04-3 (ML15197A267), to address the inadequacy of the figure. In its response to RAI 83-7962, Question 14.03.04-3 (ML15259A765), the applicant stated that DCD Tier 1 Figure 2.4.5-1 will be revised to show that the RCGVS discharges into the IRWST and provided a markup of the figure. The staff finds that the applicant’s revised DCD Tier 1 Figure 2.4.5-1 is consistent with its associated design description. Based on the review of DCD Tier 1, the staff has confirmed incorporation of the changes described above; therefore, RAI 83-7962, Question 14.03.04-3, is resolved and closed.

The staff concludes that the Tier 1 information is consistent with DCD Tier 2. In addition, the staff determines that the proposed ITAAC in Table 2.4.5-4 of DCD Tier 1 are necessary and sufficient to provide reasonable assurance that, if met, the RCGVS will be in conformity with the certified design, the applicable regulations, and the Atomic Energy Act, as amended. This conclusion is based on the inclusion of comprehensive inspections, tests, and analyses that verify the acceptability of RCGVS components. In addition, the RCGVS ITAAC are consistent with corresponding ITAAC for similar reactor designs.

5.4.12.5 *Combined License Information Items*

There are no COL information items associated with Section 5.4.12 of the APR1400 DCD. However, as noted in Section 5.4.12.4 of this SER, a COL applicant will develop the RCGVS operating procedure as part of COL 13.5(4) and 13.5(5).

5.4.12.6 *Conclusion*

The staff reviewed the RCS high point vent system to confirm its capability to remove noncondensible gases from the RCS that could inhibit natural circulation core cooling. Based on the evaluation presented above, the staff concludes the following:

- The applicant met the relevant requirements of GDC 1, GDC 14, GDC 30, and 10 CFR 50.55a with respect to the RCGVS because the RCGVS components that are part of the RCPB are designed, fabricated, erected, tested, and maintained to high quality standards; meet the RCPB requirements; are seismically qualified; and are designed to withstand the dynamic loads of operation.

- The applicant met the relevant requirements of GDC 17 and 34 by ensuring normal and emergency power supply to the RCGVS valves as well as a vent path when system operation is needed and isolation when venting is not needed.

- The applicant met the relevant requirements of GDC 19, 10 CFR 50.46a, and 10 CFR 50.49 because the RCGVS is operable from the control room, and the environmental qualification and conformance of the RCGVS valves to 10 CFR Part 50, Appendices A and B provides reasonable assurance that their safety function is performed without inadvertent or irreversible actuation.

- The applicant met the relevant requirements of GDC 36 since the design of the RCGVS permits periodic ISIs and tests.

- The applicant met the relevant requirements of 10 CFR 50.34(f)(2)(vi) and 10 CFR 50.46(b)(5) by providing high point vents that are operable from the control room, a description of operating procedures, and justification that RCGVS operation will not lead to an unacceptable increase in LOCA probability or a challenge to the ECCS or containment.

- The applicant met the relevant requirements of 10 CFR 50.44 because the swing panels on the IRWST allow mixing of any combustible gases with the containment atmosphere.

5.4.13 Main Steam Flow Restrictor

The applicant designed the steam nozzle of the SG with a flow restrictor to limit steam flow in the unlikely event of a break in the main steamline. The small flow area decreases the flow rate which is limited to sonic velocity. This provides several protective advantages including: preventing a rapid rise in containment pressure; keeping the rate of heat removal from the reactor coolant within acceptable limits; reducing thrust forces on the main steamline piping; and limiting stresses on internal SG components, particularly the tubesheet and tubes.

Detailed information about the main steam flow restrictor and the staff's evaluation and conclusion regarding APR1400 main steam flow restrictor design features and performance requirements are discussed in the following sections of this SER:

- Section 3.9.1, "Special Topics for Mechanical Components."

- Section 3.9.2, "Dynamic Testing and Analysis of Systems, Components, and Equipment."

- Section 3.9.3, “ASME Code Class 1, 2, and 3 Components, Component Supports, and Core Support Structures.”

- Section 15.1.5, “Steam System Piping Failure Inside and Outside the Containment.”

5.4.14 Safety and Relief Valves

The use of safety and relief valves and the reactor protection system ensures overpressure protection for the RCPB. Overpressurization of the RCS and SGs is precluded by operation of the pressurizer POSRVs and MSSVs.

Overpressure protection for the RCPB is provided by four POSRVs installed on top of the pressurizer. These valves discharge to the IRWST where the steam is released under water to be condensed and cooled. Overpressure protection for the RCS is addressed in Section 5.2.2, “Overpressure Protection,” of this SER.

Overpressure protection for the secondary side of the SGs is provided by spring-loaded MSSVs installed on each of the main steam line upstream of the MSIV outside the containment. As the SG pressure rises and pressure setpoints are reached, the MSSVs open and discharge the high-pressure steam to the atmosphere. MSSVs are further described in DCD Tier 2 Section 10.3.2.2.3, “Main Steam Safety Valves.”

Although NUREG-0800 does contain a Section 5.4.14, the RCS pressure relief devices are included in the review scope of NUREG-0800, Section 5.4, “Reactor Coolant System Component and Subsystem Design.” Detailed information about the safety and relief valves and the staff’s evaluation and conclusion regarding SRV design features and performance requirements are discussed in the following sections of this SER:

- Section 3.9.1, “Special Topics for Mechanical Components.”

- Section 3.9.2, “Dynamic Testing and Analysis of Systems, Components, and Equipment.”

- Section 3.9.3, “ASME Code Class 1, 2, and 3 Components, Component Supports, and Class CS Core Support Structures.”

- Section 3.9.6, “Functional Design, Qualification, and Inservice Testing Programs for Pumps, Valves, and Dynamic Restraints.”

- Section 3.10, “Seismic and Dynamic Qualification of Mechanical and Electrical Equipment.”

- Section 5.2.2, “Overpressure Protection.”

- Section 5.2.3, “Reactor Coolant Pressure Boundary Materials.”

- Section 5.2.4, “Inservice Inspection and Testing of the Reactor Coolant Pressure Boundary.”

- Section 6.6, “In-service Inspection of Class 2 and 3 Components.”

- Section 6.8, “In-containment Water Storage System.”

- Section 10.3, “Main Steam System.”
- Section 15.6.1, “Inadvertent Opening of a PWR Pressurizer Pressure Relief Valve.”

5.4.15 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 major RCS components support and restrain the RCS piping.

Detailed information about the component supports and the staff’s evaluation and conclusion regarding APR1400 component supports design features and performance requirements are discussed in the following sections of this SER:

- Section 3.9.1, “Special Topics for Mechanical Components.”
- Section 3.9.2, “Dynamic Testing and Analysis of Systems, Components, and Equipment.”
- Section 3.9.3, “ASME Code Class 1, 2, and 3 Components, Component Supports, and Core Support Structures.”

Section 3.9.6, “Functional Design, Qualification, and Inservice Testing Programs for Pumps, Valves, and Dynamic Restraints.”

5.A EVALUATION OF THE APR1400 DESIGN AND INTERSYSTEM LOSS-OF-COOLANT ACCIDENT CHALLENGES

This section documents the staff's review of DCD Tier 2, Appendix 5.A, "Evaluation of the APR1400 Design and Intersystem Loss-of-Coolant Accident Challenges," which discusses intersystem loss-of-coolant accident (ISLOCA) challenges of systems directly and indirectly connected to the RCS. The staff's review was based on the guidance in Information Notice (IN) 92-36, "Intersystem LOCA Outside Containment," NUREG/CR-5102, "Interfacing Systems LOCA, Pressurized Water Reactors," and NUREG-0800.

5.A(A) Introduction

IN 92-36 defines ISLOCAs as a class of accidents in which a break occurs in a system connected to the RCS, causing a loss of the primary system inventory. This type of accident can occur when a low pressure system, such as the SCS, is inadvertently exposed to high RCS pressures beyond its capacity which results in a system breach that leads to a loss-of-coolant. Of most concern are the ISLOCAs result in a break discharge flow outside the reactor containment building; this system breach can result in high offsite radiological consequences but also because the RCS inventory lost cannot be retrieved for long-term core cooling during the long term cooling phase. An ISLOCA-like event is an event that results from the failure, degradation, or inadvertent opening of the system pressure isolation valves between the RCS and lower pressure systems.

5.A(B) Summary of Application

DCD Tier 1: There is no Tier 1 information in the DCD directly related to this section. However, information associated with the primary systems connected directly to the RCSRCS, as evaluated, in this section is found in DCD Tier 1. These primary systems, which are further discussed in the appropriate sections of this SER are: the SIS; the SCS; the CVCS; and the sampling system.

DCD Tier 2: The applicant provided a discussion of the ISLOCA in APR1400 DCD Tier 2 Chapter 5, "Reactor Coolant System and Connecting Systems," Appendix 5A, "Evaluation of the APR1400 Design to Interfacing System LOCA Challenges." This appendix identifies the systems directly connected to RCS and subsystems indirectly connected to the RCS through pressurization pathways. The primary systems in one or more pressurization pathways include: the SIS; the SCS; the CVCS; and the sampling system.

The pressurization pathways include: shutdown cooling line; safety injection delivery line; letdown line; charging line; RCP seal injection; RCP controlled bleedoff line; sampling – hot leg; sampling – pressurizer surge line; and sampling – pressurizer steam space. Table 5A-1, "APR1400 ISLOCA Design," in Section 5.A(D) of this SER contains additional information.

ITAAC: There are no ITAAC associated with this section. However, ITAAC are associated with DCD Tier 2 systems identified above that are related to the staff evaluation and are discussed in the appropriate sections of this SER.

TS: The TS associated with DCD Tier 2 Appendix 5.A is given in DCD Tier 2 Chapter 16:

- Section 3.4.13, "RCS Pressure Isolation Valve (PIV) Leakage."

This TS is based on the WASH-1400, “Reactor Safety Study,” which identified potential IS LOCAs as a significant contributor to the risk of core melt.

5.A(C) Regulatory Basis

The relevant requirements of NRC regulations for this area of review, and the associated acceptance criteria, are consistent with the guidance in NRC IN 92-36.

The systems that are susceptible to an ISLOCA are to be designed so that the following requirements are satisfied:

1. GDC 2, as it relates to the seismic design of SSCs such that the ISLOCA systems retain their structural integrity throughout the event. Structural integrity, and therefore the safety function, is preserved if the system maintains its pressure boundary.

2. GDC 19, as it relates to including necessary instrumentation and controls for the ISLOCA systems identified in this SER section in the control room.

3. Title 10 CFR 52.47(a)(2)(iv), as it relates to offsite dose limitations.

4. Title 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 ISLOCA systems outside of the containment that contain or may contain radioactive material following an accident.

The ISLOCA systems design responses to conditions described in DCD Tier 2 Appendix 5A are evaluated against acceptance criteria consistent with the guidance in NRC IN 92-36. The systems that are susceptible to an ISLOCA are to be designed so that the following conditions are satisfied:

- The system maintains its pressure boundary,

- System maintains its structural integrity,

- Any leakage caused by the event is limited to the makeup system capabilities, and

- Offsite doses are limited to a small fraction of the limit specified in 10 CFR 52.47(a)(2)(iv).

The applicant designed the systems with direct and indirect interfacing pathways to the reactor coolant system with acceptance criteria enhancements that alert the operator to an ISLOCA challenge or terminate and limit the scope of an ISLOCA event. The acceptance criteria enhancements include the following:

- High pressure alarms to warn the operator when rising pressure approaches the design pressure of low-pressure systems

- Check valves that isolate the RCS from the low-pressure systems

- Isolation valves with position indication in the MCR
- Pressure relief valves
- Increased the low pressure systems design pressure and temperature
- Leakage detection alarms in the MCR
- Pressure reducing devices between the RCS and high pressure systems to the low pressure systems

There is no SRP for the ISLOCA and the acceptance criteria enhancements come from SECY-93-087, SECY-90-016, NUREG/CR-5603, IN 92-36, and engineering judgement. Therefore, whenever the acceptance criteria enhancements are mentioned in the following technical evaluation, it refers to enhancements and documents identified above.

5.A(D) Technical Evaluation

SECY-90-016, "Evolutionary Light Water Reactor (LWR) Certification Issues and Their Relationship to Current Regulatory Requirements," specifies the staff's position on protection against the possibility of a LOCA occurring outside the containment for those systems linked to the RCS. The NRC's position is that advanced light-water reactor designs should reduce the possibility of a LOCA outside the containment by designing the systems and subsystems connected to the RCS to the extent practicable or alternatively, provide for leak testing and valve position indication of isolation valves as well as high-pressure alarms for the low-pressure systems in the control room.

The staff based its review on the design requirements to minimize the potential for ISLOCAs as described in IN 92-36, NRC Letter, "Preliminary Evaluation of the Resolution of the Intersystem Loss-of-Coolant-Accident (ISLOCA) Issue for the Advanced Boiling Water Reactor (ABWR) – Design Pressure for Low Pressure Systems," SECY-90-016, "Evolutionary Light Water Reactor (LWR) Certification Issues and their Relationship to Current Regulatory Requirements," and NUREG/CR-5102, "Interfacing Systems LOCA: Pressurized Water Reactors."

IN 92-36 defines an ISLOCA as "a class of accidents in which a break occurs in a system connected to the RCS, causing a loss of the primary system inventory." An ISLOCA is further defined as a beyond DBE and is also discussed in DCD Chapter 19. For evaluation convenience, the staff developed Table 5A-1 in this SER section to summarize the systems and subsystems directly or indirectly connected to the RCS as well as the pathway design pressure and the containment isolation valves. All containment isolation valves have position indication in the MCR. In addition, ISLOCA systems have pressure indication alarms in the MCR to alert the operator to a potential over-pressurization event and allow sufficient time to isolate the system/subsystem.

The applicant defined a pressurization pathway as an abnormal pressurization of a system/subsystem directly from the RCS which is due to: (1) an inadvertent opening of a valve or valves (operator error), (2) a failure of motor operated containment isolation valves, or (3) the assumption that valves failed fully open, such as check valves. With this interpretation, the applicant identified and addressed all causes that may adversely affect low-pressure systems.

Based on the DCD Tier 2 Chapter 5 RCS flow diagrams, the staff agrees with the applicant's selection of systems/subsystems. The staff reviewed the available P&IDs and DCD Tier 2 section figures to confirm the pathway and components such as isolation valves and check valves and instrumentation such as valve position and pressure indications in the MCR in reaching its conclusion.

Based on the above information, the staff evaluated the systems and subsystems directly connected to the RCS including the once or twice removed systems associated with the primary system under review.

Shutdown Cooling System

The SCS supply line and return line is directly connected to the RCS and is a primary interface through which an ISLOCA event can occur. The pressurization pathway is from the hot leg and out of containment through the containment isolation valves to the low-pressure sections of the SCS. Then the SCS return line is connected directly to the RCS through the containment isolation valves and DVI nozzles through which an ISLOCA event can be initiated. The SCS supply and return lines are designed to withstand pressures equal to or greater than 63.3 kg/cm2G (900 psig), which corresponds to 40 percent of normal RCS pressure as discussed in NUREG/CR-5603, "Pressure-Dependent Fragilities for Piping Components: Pilot Study on Davis-Besse Nuclear Power Station." The staff concluded that this is consistent with past applicant practices that the staff found acceptable, including for the ABWR and System 80+ designs. It remains acceptable for the APR1400 because there is reasonable assurance that, at this design pressure, the integrity of the SCS will be maintained. This is consistent with the guidance in SECY-90-016. In addition, the other controls prohibit operation of the SCS above an enable temperature and the low pressure portions of the SCS have relief valves designed to prevent over-pressurization during cool-down conditions. The staff concluded that the SCS design pressure, temperature/pressure detection, system isolation, and over pressurization protection mitigates a potential ISLOCA transient. Therefore, the staff determined that the SCS design is acceptable because it meets the acceptance criteria in Section 5.A(C) of this SER.

Once pressurized, the other low-pressure systems indirectly in the pathway are vulnerable to over-pressurization. Table 5A-1 of this SER identifies the various pathways an ISLOCA in this configuration can take and the interfacing systems it would affect, assuming all downstream interfacing valves are open. For this case, other low-pressure systems indirectly in the pathway include the CVCS, SS, and SIS. Since these systems also have a direct pathway to the RCS in other configurations, they meet RCS requirements, and their design is adequate in terms of interfacing systems. The staff concluded that the SCS pathway temperature or pressure detection, motor-operated isolation valves, and pressure relief valves provides isolation and overpressurization protection that terminates a potential SCS ISLOCA transient interaction with other indirectly interfacing low pressure systems. The staff determined that is the design of the system is acceptable because it meets the acceptance criteria described in Section 5.A(C) of this SER.

Safety Injection System

The applicant stated that the SIS's injection line is directly connected to the reactor vessel, which makes it a primary interface through which an ISLOCA can start. The pressurization pathway is introduced from the DVI nozzle and out of containment through the containment isolation valves to the low-pressure sections of the SIS where the system is vulnerable over-pressurization transients.

The design pressure for most of the SIS is 144.1 kg/cm^{2}G (2,050 psig), which exceeds 40 percent of normal RCS pressure. The design pressures of the SIS sections outside of containment are not designed to 144.1 kg/cm^{2}G (2,050 psig) but are designed to 63.3 kg/cm^{2}G (900 psig). This low pressure design includes SIS sections from the containment to the SIS pump suctions. As previously discussed with regards to the SCS, the staff has reasonable assurance that, at this design pressure, the integrity of the SIS will be maintained. Therefore, the SIS lines and all interfacing systems from an ISLOCA event are protected without adversely affecting performance or operations of these systems. In addition, these systems include check valves, isolation valves, and MCR instrumentation that provide the operator with systems configurations and thermal hydraulic conditions to alert the operator to the start of an adverse pressure transient. The staff concluded that the SIS pathway has pressure detection, system isolation, and pressure relief that terminates a potential ISLOCA transient. Therefore, the staff finds the applicant's SIS pressurization pathway analysis and conclusion acceptable because the applicant met the acceptance criteria described in Section 5.A(C) of this SER.

Letdown Line

The letdown line is directly connected to the RCS, which makes it a primary interface through which an ISLOCA event can initiate. The applicant identified the pressurization pathway from the letdown nozzle, through the regenerative HX, letdown orifices, letdown HX, containment isolation valves, and letdown control valves to the low-pressure sections of the system located outside of the containment. With the downstream interfacing valves opened, the low pressure interfacing systems are susceptible to over-pressurization transients. A high-pressure alarm has been added to the letdown line located downstream of the letdown control valves to alert the MCR operator to an impeding over-pressurization transient that the operating pressure is exceeding the design pressure; thus, the operator can isolate the letdown line to prevent additional pressurization downstream of the low pressure system. The staff concludes that the letdown line pathway pressure detection and manual isolation terminates a potential ISLOCA transient. Therefore, the staff finds the ISLOCA acceptance criteria in Section 5.A(C) of this SER are satisfied based on the following:

- System integrity is preserved since the portion of the letdown line pressurized by the ISLOCA is designed to full RCS design pressure.

- The letdown line downstream of the letdown control valves and all other downstream interfacing systems are designed to be protected from an ISLOCA by manual action of the isolation valves. Therefore, the low pressure systems integrity is ensured.

- The RCS pressurization pathway is isolated, which prevents primary coolant loss and no increase in offsite dose.

Charging Line

The charging line is directly connected to the RCS, creating a primary pressurization pathway interface in which an ISLOCA transient can occur. The pressurization pathway originates from the charging nozzle, through the shell side of the regenerative HX, charging control valve, and charging pump to the low pressure sections of the system. Once pressurized, other low-pressure interfacing systems, as noted in Table 5A-1 of this SER, are vulnerable. A pressure indicator alarms in the MCR to alert the operator that the operating pressure is exceeding the design pressure; thus, the operator can isolate the charging line to prevent additional pressurization downstream of the low pressure system. The staff concludes that the charging line pathway pressure detection and manual isolation terminates a potential ISLOCA transient. .

Therefore, the staff finds the design response to this pathway satisfies the ISLOCA acceptance criteria in Section 5.A(C) of this SER based on the following:

- System integrity is preserved since the portion of the charging line pressurized by the ISLOCA is designed to full RCS design pressure.

- The Charging line upstream of the charging pumps and all other downstream interfacing systems are designed to be protected from an ISLOCA by manual action of the isolation valves. Therefore, the low pressure systems integrity is ensured.

- The RCS pressurization pathway is isolated, which prevents primary coolant loss and no increase in offsite dose.

The staff reviewed the impact of the difference between the letdown line and the charging line. In the letdown line, the pressurization pathway is in the direction of normal flow. Therefore, the "letdown line has been designed to reduce and control the fluid pressure under normal and ISLOCA operating conditions." Besides isolation, a backup relief valve is provided which is sized to provide reasonable assurance that the design pressure, in the low pressure sections, is not exceeded. However, for the charging line, the pressurization pathway direction in the charging line is opposite to the flow. To remedy this condition, the charging line has three check valves in series to prevent overpressurization of the CVCS. This check valve configuration has a high probability of not failing when the charging pump and auxiliary charging pump (ACP) are not in operation. The applicant noted that a pressure indicator upstream of the charging pumps provides MCR indication upon sensing high pressure to initiate an operator manual isolation of the charging line by the containment isolation valves. To further improve over-pressurization protection, the applicant stated that there will be installed relief valves "in the volume control tank [VCT] discharge line, charging pump mini-flow line, charging pump discharge line, and ACP discharge line." The staff determined that this pathway configuration is acceptable because it is consistent with the guidance in SECY-90-016, meets the ISLOCA acceptance criteria in Section 5.A(C) of this SER, and additional relief valves provide additional assurance that the design pressure, in the low pressure sections, is not exceeded.

Reactor Coolant Pump Seal Injection Line

The seal injection line is a primary pathway system connected directly to the RCS through which an ISLOCA event can start. Pressurization is composed of two separate pathways: (1) from the charging nozzle through the shell side of the regenerative HX to the seal injection line and (2) from the RCP seals through the tube side of the high pressure seal cooler to the discharge side of the charging pump. As noted in Table 5A-1 of this SER, the applicant identified the boron recovery system and atmospheric system as low-pressure interfacing systems with RCP seal injection line which presents a potential over-pressurization transient event.

The design pressure of the seal injection line is 212.7 kg/cm^{2}G (3,025 psig), which is greater than normal RCS pressure. Therefore, the staff finds that the seal injection line meets the acceptance criteria in Section 5.A(C) of this SER

Reactor Coolant Pump Controlled Bleed-Off Line

The applicant stated that the controlled bleed-off line is directly connected to the RCS and the controlled bleed-off line is a primary pathway interface through which an ISLOCA event can occur. The pressurization pathway is identified from the RCP seals through a flow-limiting orifice and a flow meter out of containment to the charging pump mini-flow HX (MFHX) or volume control tanks (VCT). In DCD Tier 2 Section 5A.4.7.2, "Design Evaluation Change," the applicant stated the following:

> The design pressure for the controlled bleed-off system is divided into high and low pressure sections. The high pressure, 174.7 kg/cm^{2}G (2,485 psig), exceeds the minimum design pressure requirement for an ISLOCA, as provided in SECY-90-016. The low pressure section outside containment is designed to 5.3 kg/cm^{2}G (75 psig) and 14.1 kg/cm^{2}G (200 psig) to be compatible with the design pressure of the VCT and charging pump MFHX, respectively.

The staff concluded that there is no credible event that would pressurize the low-pressure section of the RCP controlled bleed-off line above its design pressure because a fixed-resistance flow control orifice limits the upstream pressure. The applicant stated that relief valves protect the low-pressure section of the VCT and charging pump MFHX inlet from over-pressurization. The relief valves discharge to the equipment drain tank where indication and alarm for liquid level, temperature, and pressure alert the operators in the MCR. Furthermore, the applicant stated that "the flow through the controlled bleed-off line during normal operation is within the makeup capability and can be collected in the reactor drain tank." In addition, the VCT provides the MCR with indication and alarm for liquid level, temperature, and pressure. For these reasons, the RCP controlled bleed-off line low-pressure section meets the ISLOCA acceptance criteria in Section 5.A(C) of this SER.

Sampling System

The sampling system related to ISLOCA includes sampling for the following components, as described below: hot leg, pressurizer surge line, and pressurizer steam space.

<u>Hot Leg Sampling</u>

The sampling system has a direct pathway to the RCS hot leg in which the interface has the potential of ISLOCA event. The applicant identified the pressurization pathway "from the sampling nozzle on the hot leg through the containment isolation valves and out of containment to the low-pressure sections of the system." As noted in Table 5A-1 of this SER, the applicant identified various pathways.

The applicant noted that the hot leg sampling line is designed with a fixed-resistance orifice and a "small line size that limits the flow and pressure during all modes of plant operation." The VCT receives the sampling line discharge. As noted above, a relief valve that discharges into the equipment drain tank protects the VCT from overpressurization. The staff used the Tier 2 figures listed below to aid in evaluating the intersystem configuration interface. The staff concluded that the sampling system is protected from an ISLOCA because the hot leg sampling line has a fixed-resistance orifice that limits the pressure and pressure relief valve which terminates a potential ISLOCA transient protects the VCT. Therefore, the staff determined that the hot leg portion of the sampling system meets the acceptance criteria in Section 5.A(C) of this SER.

<u>Pressurizer Surge Line Sampling</u>

The sampling system includes another direct pathway to the pressurizer surge line which may result in a primary interface ISLOCA event. The applicant identified the pressurization pathway "from the sampling nozzle on the surge line through the containment isolation valves and out of containment to the low-pressure sections of the system." The pressurizer surge line also incorporates a "fixed resistance orifice and a small line size that limits the flow and pressure during all modes of plant operation." This line also discharges to the volume control tank and, as discussed above, a relief valve protects the line from overpressurization. The staff used the Tier 2 figures listed below to aid in evaluating the intersystem configuration interface. The staff concluded that the sampling system is protected from an ISLOCA because the pressurizer surge line sampling line also has a fixed-resistance orifice that limits the pressure and, as discussed above, a pressure relief valve protects the VCT and terminates a potential ISLOCA transient. Therefore, the staff determined that the pressurizer steam surge line portion of the sampling system meets the acceptance criteria in Section 5.A(C) of this SER.

<u>Pressurizer Steam Space Sampling</u>

The sampling system includes another direct pathway to the pressurizer steam space which may result in a primary interface with a potential of an ISLOCA event. Pressurization pathway is from the sampling nozzle on the pressurizer steam space through the containment isolation valves and out of containment to the low-pressure sections of the system.

The design is identical to the two sampling lines discussed above. The staff used the Tier 2 figures listed below to aid in evaluating the intersystem configuration interface. The staff concluded that the sampling system is protected from an ISLOCA because the pressurizer steam space sampling line also has a fixed-resistance orifice that limits the pressure which terminates a potential ISLOCA transient. Therefore, the staff determined that the pressurizer steam space portion of the sampling system meets the acceptance criteria in Section 5.A(C) of this SER.

Containment Spray System

The applicant stated that the CSS is not directly connected to the RCS during the modes of operation in which an ISLOCA challenge can occur. However, CSPs provide backup to the SCS for decay heat removal and cooling of the IRWST during feed and bleed operations utilizing the SIS and the POSRV. When connected in this SCS configuration, the staff determined that the SCS relief valves protect the CSPs, as shown on DCD Tier 2 Figure 6.3.2-1, "Safety Injection/Shutdown Cooling System Flow Diagram."

As part of the Appendix 5A evaluation, the staff reviewed the following DCD Tier 2 figures to confirm the applicant's analysis and the correctness of the simplified drawings of this appendix.

- Figure 5.1.2-1, "Reactor Coolant System Flow Diagram."
- Figure 5.1.2-2, "Reactor Coolant Pump Flow Diagram."
- Figure 5.1.2-3, "Pressurizer and POSRV Flow Diagram."
- Figure 6.2.2-1, "Containment Spray System Flow Diagram."
- Figure 6.3.2-1, "Safety Injection / Shutdown Cooling System Flow Diagram."

- Figure 9.2.2-1, “Component Cooling Water System Flow Diagram.”
- Figure 9.3.4-1, “Chemical and Volume Control System Flow Diagram.”

Table 5A-1: Potential Pressurization Pathways during ISLOCA Events

System Directly Interfacing With RCS	Pressurization Pathway Design For System Directly Interfacing With RCS	Containment Isolation Valves	Systems Once Removed	Systems Twice Removed
SCS	Design pressure for all sections outside containment: 63.3 kg/cm2G (900 psig).	Supply Line SI-653/654 SI-655/656 Return Line SI-600/601 MCR IND	CVCS	SS, CSS, GWMS, HSS, Air Supply, NSS, PCPS LWMS, ATM, SWMS, MBRS
			ATM	
			SS	CVCS, SIS, CSS, ATM
			CSS	SIS, ATM, CVCS, SS
			SIS	ATM, CSS, SS
SIS	SIS Delivery Line Design pressure for all sections outside containment: 63.3 kg/cm2G (900 psig).	SI-616/626 636/646 SI-322/332 SI-602/603 SI-321/331 MCR IND		
			SCS	CVCS, CSS, SS, ATM
			CSS	SCS, ATM, CVCS, SS
			SS	CVCS, SCS, CSS, ATM
			ATM	
Letdown Line	High-pressure alarm installed downstream of the letdown control valves to initiate a manual isolation of the letdown line by the containment isolation valve when high pressure is sensed.	CV-515/516 CV-522/523 MCR IND	MBRS	SFP, ATM, SS, LWMS, SWMS, SCS
			ATM	
			SS	SCS, CSS, SIS
			SCS	CSS, SIS, SS, ATM
			H2, N2, GWMS, SWMS	
Charging Line	Pressure sensor installed upstream of the charging pumps to initiate a manual isolation of the charging line by the containment isolation valve when high pressure is sensed.	CV-240 CV-524 MCR IND	MBRS, Seal Injection, Makeup	SFP, ATM, SS, LWMS, SWMS, SCS
			ATM	
			SS	SCS, CSS, SIS
			H2, N2, GWMS, SWMS	
Seal Injection		CV-241/242 243/244/255 MCR IND	MBRS	Purification, Makeup, CCWS, SWMS, LWMS, GWMS, SS
			ATM	
RCP Controlled Bleedoff		CV-505/506 MCR IND	MBRS	Purification, Makeup, CCWS, SWMS, LWMS, GWMS, SS
			Purification	MBRS, ATM, SS, SCS, GWMS, SWMS, H2, N2
			ATM	
Hot leg Sampling	Pressure relief valve installed upstream of discharge isolation valves to VCT and recycle drain header.	SS-001/002 MCR IND	SCS	CSS, SIS, ATM
			ATM	
			CVCS	ATM, SCS, GWMS, SWMS, H2, N2, MBRS
			LWMS	
			SIS	ATM, CSS
			CSS	SIS, CVCS, ATM
		SS-003/004	SCS	CSS, SIS, ATM

System Directly Interfacing With RCS	Pressurization Pathway Design For System Directly Interfacing With RCS	Containment Isolation Valves	Systems Once Removed	Systems Twice Removed
Pressurizer Surge Line Sampling	Pressure relief valve installed upstream of discharge isolation valves to VCT and recycle drain header.	MCR IND	ATM	
			CVCS	ATM, SCS, GWMS, H2, N2, MBRS
			MLWMS	
			SIS	ATM, CSS
			CSS	SIS, CVCS, ATM
Pressurizer Steam Space Sampling	Pressure relief valve installed upstream of discharge isolation valves to VCT and recycle drain header.	SS-005/006 MCR IND	SCS	CSS, SIS, ATM
			ATM	
			CVCS	ATM, SCS, GWMS, SWMS, H2, N2, MBRS
			MLWMS	
			SIS	ATM, CSS
			CSS	SIS, CVCS, ATM

5.A(E) Combined License Information Items

There are no COL information items associated with Appendix 5.A of the APR1400 DCD.

5.A(F) Conclusion

The staff determined that the applicant's ISLOCA analysis is acceptable because it demonstrated that the systems' direct and indirect pathways will ensure adequate protection with respect to potential ISLOCA transients by providing pressure detection, system isolation, pressure relief, and fixed-resistance orifices that terminates potential ISLOCA transients. The staff determined that the APR1400 intersystem design complies with GDC 2, GDC 19, 10 CFR 52.47(a)(2)(iv), and 10 CFR 50.34(f)(2)(xxvi). Therefore, the staff determined that the intersystem interface design is acceptable because it can adequately terminate a potential ISLOCA transient.