5 Reactor Coolant System and Related Systems

5.0 Reactor Coolant System and Related Systems

This chapter describes the staff's review of the Advanced Power Reactor 1400 (APR1400) reactor coolant system (RCS).

5.1 Summary Description

5.1.1 Introduction

The 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 design control document (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 RCP, one pressurizer, four pressurizer pilot ~~operates~~ 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.

Please check if there is still an open issue. The RAI response was provided in Dec. 2015.

(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 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. Therefore, on October 2, 2015 the staff issued **RAI 233-8244, Question 05.02.02-1,** 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. **RAI 233-8244, Question 05.02.02-1, is being tracked as an open item.**

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 aforementioned open item will need be resolved before the staff can determine if the design of the POSRVs provide adequate overpressure protection for the RCS.

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, Section 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/cm^{2}A (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 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 ~~IWRST~~. These valves are Seismic Category I and designed in accordance with the ASME Code Section III (Class 2). DCD Tier 2, Section

IRWST

Please note that Inconel X-750 is used for none-pressure retaining items such as RV stabilizing shim and cap screws.

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 for the operating fleet.

used

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 has been ~~restricted~~ to spiral wound gaskets for the primary manways of SGs and the pressurizer. The staff issued **RAI 335-8351, Question 05.02.03-5** , dated December 14, 2015, requesting that the applicant provide supporting information regarding the use of X-750 for this application. In its response to RAI 335-8351, Question 05.02.03-5, dated January 13, 2016, the applicant replied that X-750 is used for this purpose in OPR1000 plants and that no cracks have been found to date. Furthermore, the gasket is replaced each time the manway is opened. The staff considered this reply adequate. 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, Revision 0, 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.

The EPRI Guidelines stipulate sampling three times/week for all but sulfate (once/week) and dissolved O (as stipulated in plant TS~~TS~~). However, they note that sampling frequencies may vary and that they will be determined in the plant TS~~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. **RAI 336-8367, Question 05.02.03-16 is being tracked as a confirmatory item pending the incorporation of the applicant's response into the next revision of the DCD.**

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, Si, and suspended solids. However APR1400 DCD Tier 2, Section 5.2.3.2.1 does not mention anything about diagnostic parameters. For this reason, in RAI 336-8367 Question 05.02.03-17, the staff requested information on the diagnostic parameters described in the EPRI Guidelines. In its response to RAIRAI 336-8367, Question 05.02.03-17, dated January 26, 2016, the applicant stated the following:

- 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 Si 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.

The staff will verify all of these changes in the next revision of the DCD. These changes are being tracked as a **confirmatory item**.

In APR1400 DCD Tier 2, Section 5.2.3.2.1, the applicant stated that a soluble zinc compound may be added to the reactor coolant for the purpose of radiation field reduction and mitigation of the PWSCC initiation. This is done because operating experience has proven that zinc addition in PWRs leads to thinner, more evenly distributed fuel crud. However the applicant does not provide an explanation on how this addition will take place nor the quantities to be added. Based on this, in RAI 336-836, Question 05.02.03-19, the staff requested a better description of the zinc addition to the RCS. In its response to RAI 336-836, Question 05.02.03-19, dated January 26, 2016, the applicant decided to not add Zn to the RCS. Although frequently added

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) relief requests; (9) Code cases; (10) other inspection programs; and the (11) PSI and testing program. In each of these areas, the application references the applicable ASME Code requirements. The application also provided seven 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 ~~TSTS~~ 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.
- 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

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 Section 5.2.4.1.1 states that the RPV, pressurizer, primary side of the ~~SGSG~~ SG, and associated piping, pumps, valves, bolting, and component supports are subject to inspection. The staff agreed that the aforementioned components are subject to inspection. However, it was unclear whether the system boundary encompasses all ASME Code Class 1 components that are subject to inspection. 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 clarify that all ASME Code Class 1 pressure-retaining components are subject to inspection. The staff determined that the applicant's response is acceptable because it is in compliance with 10 CFR 50.55a and ASME Section XI. The applicant also provided a markup copy of the proposed changes in a letter dated July 17, 2015. The staff determined that the applicant's proposed revisions to the DCD have appropriately addressed the issue and identifies a **confirmatory item** to confirm that the proposed changes are included in the next revision of the DCD.

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. The staff had concerns on whether APR1400 design includes provisions for two-sided access to dissimilar metal welds and austenitic welds to enable the performance of inservice examinations. 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 state that dissimilar metal welds and austenitic welds in piping will be examined from both sides. The DCD will also be revised to state 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 response is acceptable because it is in accordance with ASME Section XI. The applicant also provided a markup copy of the proposed changes in a letter dated July 17, 2015. The staff determined that the applicant's proposed

5.2.5(F) Conclusion

Based on the above, pending the resolution of three confirmatory items, 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

two

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 ~~four~~ 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 CRD 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.

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. 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. 10 CFR 50.60, as it relates to RCPB fracture toughness and material surveillance requirements of 10 CFR Part 50, Appendix G, and Appendix H.

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

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

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

"Delete"

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 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 ~~F316~~ (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.

F316LN

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 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 applicant also provided a markup copy of the proposed changes in a letter dated July 17, 2015. The staff determined that the applicant has appropriately addressed the issue and identifies a **confirmatory item** to confirm that the proposed changes are included in the next revision of the DCD.

ENiCrFe-7

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 ~~EnCriFe-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 also provided a markup copy of the proposed changes in a letter dated July 17, 2015. The staff determined that the applicant has appropriately addressed the issue and identifies this issue as a confirmatory item pending the incorporation of the proposed changes in the next revision of 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.

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

7. 10 CFR 50.60, as it relates to RCPB fracture toughness and material surveillance requirements of 10 CFR Part 50, Appendix G and Appendix H.

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

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

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

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 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 ~~ISIISI,~~ 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 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

ISIs.

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 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 ~~ISIISIs.~~ 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 its 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 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, 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 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," ~~SCSSCS~~ SCS in Section 5.4.7, "Shutdown Cooling System," pressurizer in Section 5.4.10, "Pressurizer," ~~PRTPRT~~ 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 Sections 5.4.4, 5.4.5, and 5.4.9, and the applicant noted that DCD 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 APR14000. 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.

Technical Reports: KHNP APR1400-A-M-NR-14001-P, (WCAP-17942-P), "KHNP APR1400 Flywheel Integrity Report," Revision 0, dated November 2014.

"Reactor Coolant System Component and Subsystem Design,"

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 Comp~~ and 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, preservice 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:

1. RG 1.14, ""Reactor Coolant Pump Flywheel Integrity"," as it relates to the RCP flywheel design, materials selection and fabrication, preservice inspection 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, "Pump Flywheel Integrity (PWR)." 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 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

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 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. Therefore, in **follow-up RAI 503-8641, Question 05.04.01.01-9,** dated July 6, 2016, the staff requested that the applicant 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 116 ~~ksi/in^2~~ psi that was in original flywheel analysis (dated August 13, 2014) in lieu of the current specified yield strength of 92.825 ~~ksi/in^2~~ psi 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. In **follow-up RAI 503-8641, Question 05.04.01.01-9,** dated July 6, 2016, the staff requested 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 Technical Report APR1400-A-M-NR-14001-P, "KHNP APR 1400 Flywheel Integrity Report," Revision 0, dated November 24, 2014, needs to be revised. **RAI 503-8641, Question 05.04.01.01-9 is being tracked as an open item.**

In its response to RAI 341-8410, Question 05.04.01.01-3c, dated April 29, 2016, 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. This was discussed in **follow-up RAI 503-8641, Question 05.04.01.01-10,** dated July 6, 2016, and the staff requested the applicant to revise, or explain why no revision is needed, the technical report 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. **RAI 503-8641, Question 05.04.01.01-10 is being tracked as an open item.**

Pre-service Inspection

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

2015

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. Therefore, in **RAI 341-8410, Question 05.04.01.01-4,** dated December 18, ~~2106~~, the staff requested that additional information in APR1400 DCD 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 Technical Report APR14001-A-M-NR-14001-P.

In its response to RAI 341-8410, Question 05.04.01.01-4, dated April 29, 2016, the applicant did not revise the APR1400 DCD to specify the maximum flaw size used as the acceptance criteria for the preservice inspection and that it is bounded by the flaw size used in determining the critical flaw size in Technical Report APR1400-A-M-NR-14001-P. This was discussed in **follow-up RAI 503-8641, Question 05.04.01.01-8,** dated July 6, 2016, which requested additional information on this subject. **RAI 503-8641, Question 05.04.01.01-11 is being tracked as an open item.**

In addition to the preservice inspection 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, Technical Report APR14001-A-M-NR-14001-P 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 Section 5.4.1.1.4 for the flywheel should also apply to the hub.

Therefore, in **RAI 341-8410, Question 05.04.01.01-5,** dated December 18, 2015, the staff requested that the applicant 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, dated April 29, 2016, the applicant did not revise the APR1400 DCD to include that the hub will be inspected for both pre-service inspection (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. In **follow-up RAI 503-8641, Question 05.04.01.01-12,** the staff discussed this issue and requested additional information in APR1400 DCD Section 5.4.1.1.4 to address whether the hub will be inspected for both PSI and ISI in the same manner. In addition, **follow-up RAI 503-8641, Question 05.04.01.01-12,** (and as discussed in the June 29, 2016, meeting) the staff requested that the applicant 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). **RAI 503-8641, Question 05.04.01.01-12 is being tracked as an open item.**

The staff cannot determine that the preservice inspection provides an acceptable initial flywheel condition until satisfactory resolution of RAI 341-8410, Questions 05.04.01.01-11 and 05.04.01.01-12. The initial flywheel condition, along with the flywheel analysis, provides a baseline for future ISIs to ensure that no flaws will propagate resulting in the fracture of the flywheel and generation of potential missiles.

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. Therefore, as stated above, in **follow-up RAI 503-8641, Question 05.04.01.01-12,** dated July 6, 2016, staff requested additional information on whether APR1400 DCD 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). **This is part of RAI 341-8410, Question 05.04.01.01-6, which is being tracked as an open item for this section concerning ISI and the previous section concerning PSI.**

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

As a result of RAI 341-8410, Questions **05.04.01.01-01 through 05.04.01.01-5 being open items,** the staff is unable to finalize its conclusion on Section 5.4.1.1 in accordance with the requirements of NRC regulations (GDCs 1 and 4).

5.4.1.2 RCP Design

5.4.1.2(A) 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(B) 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 top 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

The flywheel is located at the lower portion of the motor shaft.

the stated temperature margin. In **RAI 307-7835, Question 05.04-1**, Part 3, dated November 16, 2015, the staff requested additional information regarding a summary of the test results in DCD 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, dated January 7, 2016, 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-P, Rev. 0, "Justification that the APR1400 RCP Seals will not Exceed the Exit Temperature Specification Limit," November 2015, and 11A60-FS-DS480, Revision 03, "Design Specification for Reactor Coolant Pumps," dated 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 (~~-5.6~~ °C) margin for the APR1400 RCPs. Therefore, the staff determined that this response is acceptable, and RAI 307-7835, Question 05.04-1, Part 3, is considered resolved and closed.

12.2

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 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. In **RAI 307-7835, Question 05.04-1**, Part 4, dated November 16, 2015, the staff requested the applicant to provide the needed details. In its response to RAI 307-7835, Question 05.04-1, Part 4, dated January 7, 2016, the applicant proposed to add, in DCD 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. **RAI 307-7835, Question 05.04-1, Part 4, is being tracked as a confirmatory item pending incorporation of the proposed changes into the next revision of the DCD.**

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 Section 5.4.1.2,

The accelerometers are installed on the motor upper journal bearing assembly for the motor frame vibration.

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

thrust

A water lubricated radial bearing within the pump and radial and ~~thrush~~ 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 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. In **RAI 307-7835, Question 05.04-1**, Part 8, dated November 16, 2015, the staff requested the applicant to add those details in DCD Section 5.4.1 as parts of the pump design description. In its response to RAI 307-7835, Question 05.04-1, Part 8, dated January 7, 2016, the applicant proposed to add these clarifying details to DCD Section 5.4.1.2. The staff determined that this response is acceptable for the above discussed reason. **RAI 307-7835, Question 05.04-1, Part 8, is being tracked as a confirmatory item pending the incorporation of the proposed changes into the next revision of the DCD**.

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 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. In **RAI 307-7835, Question 05.04-1**, Part 6, dated November 16, 2015, the staff requested the applicant to add those details to DCD Section 5.4.1 as part of the pump design description. In its response to RAI 307-7835, Question 05.04-1, Part 6, dated January 7, 2016, 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. Therefore, the staff determined that this response is acceptable and RAI 307-7835, Question 05.04-1, Part 6 is considered 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 ~~temperature and~~ level (for motor bearings) "delete"
- Upper and lower journal bearing temperature (for motor bearing metal)
- Stator winding and motor bearing temperature recorder
- Motor ~~shaft~~ frame vibration transmitters ~~(in close proximity of motor bearings)~~ "delete"
- CCW flow and temperatures (outlets of motor oil coolers and high-pressure coolers)
- Upper and lower journal bearing temperature (for pump bearing metal)
- Upper ~~thrush~~ thrust bearing temperature (for pump ~~thrush~~ thrust bearing plate)
- Pump shaft vibration transmitter (on the rigid coupling)
- Frame vibration transmitter

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 (D) Technical Evaluation

5.4.2.1 (D)(a) 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, 2015, and December 10, 2015.

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 ~~SGSG~~s rather than stating they are identical. Verification of this change in the applicant's next revision of the DCD, is being tracked as a **confirmatory item**. 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."

SG

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 (B) 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 ~~TSTS~~ 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 (C) 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. 10 CFR 50.55a(g) requires that ISI programs meet the applicable inspection requirements in Section XI of the ASME Code.

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

4. 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. 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. ~~TSTS~~ 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 (D) 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, and May 3, 2016.

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 1.8, "Interfaces with Standard Designs," 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. Verification of the addition of this statement to DCD Subsection 5.4.2.2, "Steam Generator Program," in the applicant's next revision of the DCD is being tracked as a **confirmatory item.** 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. Verification of the addition of this statement to DCD Subsection 5.4.2.2, in the applicant's next revision of the DCD, will be tracked as a **confirmatory item.**

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. Therefore, the staff issued **RAI 299-8310, Question 05.04.02.02-1**, dated November 9, 2015, requesting consistency in the response in the 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, dated December 16, 2015, 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 acceptable because they delete unnecessary information and relocate information about the bases for the SG Program to a more appropriate location. **RAI 299-8310, Question 05.04.02.02-1 is being tracked as a confirmatory item pending incorporation of these changes into the next revision of the DCD.**

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. Verification of the deletion of this statement in DCD Subsection 5.4.2.2 in the applicant's next revision of the DCD is being tracked as part of a **confirmatory item.** The DCD markup for these changes included another wording change inconsistent with the STS as modified by ~~TSTS~~ TS Task Force (TSTF)-510, Revision 2, "Revision to Steam Generator Program Inspection Frequencies and Tube Sample Selection." Therefore, in **RAI 299-8310, Question 05.04.02.02-2**, dated November 9, 2015, the staff requested that the applicant change "degradation" to "indications" as originally proposed. In its response to RAI 299-8310, Question 05.04.02.02-2, dated December 16, 2015, 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. **RAI 299-8310, Question 05.04.02.02-2 is being tracked as a confirmatory item pending the incorporation of this change into the next revision of the DCD.** 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. Verification of this revision of DCD Subsection 5.4.2.2, in the applicant's next revision of the DCD, is being tracked as **a confirmatory item.** With respect to water chemistry programs, the CD 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 DDCD 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. Verification of this revision of DCD Subsection 5.4.2.2, in the applicant's next revision of the DCD, is being tracked as **a confirmatory item.**

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. Verification of this change to DCD, is being tracked as **a confirmatory item.** 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. Verification of these revisions in the applicant's next revision of the DCD are being tracked as part of **a confirmatory item.** However, several issues remained or were created in the applicant's proposed DCD revision, and on November 9, 2015, the staff issued

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 24	60 / 140 54.4 / 130 93.3 / 200	2 2 1
Safety S/D	28.1 / 400	~193.3 / ~380	18,927 / (5,000)			
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

should be rearranged to the row below.

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 Tier 1 DCD 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 ~~the~~ train to allow SCS operations with any single electrical failure.

"delete"

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 ~~RCS cold leg.~~ 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 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, ~~"Two CCW Divisions Shall Be OPERABLE."~~ "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."

staff concludes that the SCS components are adequately protected from overpressurization events.

safety

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 ~~shutdown~~ 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 overpressurized 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. Therefore, on September 23, 2015, the staff issued **RAI 221-8248, Question 05.04.07-2**, dated September 23, 2015, requesting 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, dated April 2, 2016, 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 Tier 2 DCD Table 5.4.7-1, "Shutdown Cooling System Design Parameters," to clarify the parameter as "Minimum NPSH Available." **RAI 221-8248, Question 05.04.07-2 is being tracked as a confirmatory item pending the incorporation of revisions to DCD Tier 2, Sections 5.4.7.3.1 and 14.2.12.4.22.**

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.

Natural Circulation Cooldown

In DCD 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 ~~SGSG~~ 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 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 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 Section 14.2.12.4.22. Therefore, on February 1, 2016, the staff issued **RAI 384-8100, Question 05.04.07-3**, dated February 1, 2016, requesting the applicant to provide: (1) additional information in DCD 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, dated March 17, 2016, 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 Tier 2 DCD, 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 Chapters 5, "Reactor Coolant System and Connecting Systems," and 14, "Verification Programs," to include a clarification of the power to flow ratio. **RAI 384-8100, Question 05.04.07-3 is being tracked as a confirmatory item pending the incorporation of revisions to DCD** Tier 2, Sections 5.4.7.3.1, "Performance Evaluation Assuming the Most Limiting Single Failure and Only Onsite Power Available," and 14.2.12.4.22, "Natural Circulation Test (First-of-a-Kind-Test)," since the staff finds the response acceptable because the applicant adequately demonstrated that the reactor remains stable during natural recirculation.

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

Therefore, the staff issued **RAI 42-7945, Question 19-2**, dated June 22, 2015, requesting 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 1) 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 2) to address the SCS pipe-slope.

In its response to RAI 42-7945, Question 19-2, dated July 28, 2015, 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.

07-4

Further, on July 19, 2016, the staff issued **RAI 492-8614, Question 05.04.~~07-7~~**, to clarify gas accumulation and gas entrainment requirements with respect to ITAAC. **RAI 492-8614, Question 05.04.~~07-7~~ is being tracked as an open item,** which will need to be resolved before the staff can determine if the issue of gas accumulation has been adequately addressed in the DCD.

07-4

Preoperational Testing and Post-Core Hot Functional Testing

In Tier 2 DCD 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