5.2.1.2.5 *Combined License Information Items*

DCD Tier 2, Table 1.8-2 and Section 5.2.6 contain the following three COL information items pertaining to Code Cases. The COL items, as modified by the applicant's RAI response described above, are acceptable. The inclusion of these modifications in the DCD are subject to confirmation as part of the confirmatory item RAI 26-7948, Question 05.02.01.02-1.

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Table 5.2-1. APR1400 combined license information items for Section Error! Reference source not found..

at the time of the application

Item No.	Description	DCD Tier 2 Section
COL 5.2(1)	The COL applicant is to address the addition of ASME Code Cases that are approved in NRC RG 1.84 at the time of the application.	5.2.1.2
COL 5.2(2)	The COL applicant is to address the ASME Code Cases approved in NRC RG 1.147 and invoked for the ISI program of a specific plant.	5.2.1.2
COL 5.2(3)	The COL applicant is to address the ASME Code Cases approved in NRC RG 1.192 at the time of the application and invoked for operation and maintenance activities of a specific plant.	5.2.1.2

A COL applicant may identify within its COL application, the planned use of additional Code Cases provided they do not alter the staff's safety findings on the APR1400 certified design. The COL information item is sufficient to alert a COL applicant who seeks to use additional code cases to identify them to the staff. No additional COL information items need to be included in DCD Tier 2, Table 1.8-2 for Code Cases.

5.2.1.2.6 *Conclusion*

The staff determined that the ASME Code Cases identified in the DCD are acceptable as specified in the applicable NRC RGs, with conformance to conditions in the applicable RGs, as discussed above. Pending completion of the confirmatory items, the staff concludes that the information provided in the DCD, with respect to the use of ASME Code Cases, is acceptable and sufficient to support compliance with the requirements of 10 CFR Part 50, Appendix A, GDC 1, and 10 CFR 50.55a that nuclear power plant SSCs important to safety be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety function to be performed.

5.2.2 Overpressure Protection

5.2.2.1 Introduction

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

APR1400 includes pilot operated safety relief valve (POSRV) to provide overpressure protection of the RCS. Four POSRVs are connected to the top of the pressurizer by separate inlet lines. The use of shutdown cooling system (SCS) suction line relief valves provide sufficient pressure addition transient analysis methodology, assumptions, computer code, and input parameters designed to determine the limiting event in regards to overpressure protection. The applicant identified OVERP as the computer code used to provide the pressure response of the water-solid system of an energy addition transient. OVERP code, assumptions, initial conditions, and input data are described in technical report WCAP-15688, "CE-NSSS LTOP Energy Addition Transient Analysis Methodology." The staff reviewed the response information with respect to the technical report and found it compatible and acceptable. The staff determined that the response is acceptable because the assumptions, input parameters, and initial conditions were adequately conservative to perform the LTOP energy addition transient analysis pressure response.

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The applicant did provide information on the secondary-to-primary heat transfer, using a secondary temperature 230 degrees Fahrenheit (°F) (110 degrees Celsius (°C)) greater than the RCS, which the staff concludes is substantially more conservative than the 100 °F (37 °C) difference allowed by technical specifications, and is therefore acceptably conservative.

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

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

The pressurizer is designed to maintain RCS operating pressure so that the minimum pressure during operating transients is above the setpoint for the SIAS and low pressure reactor trip, and the maximum pressure is below the high pressure reactor trip setpoint. As indicated in DCD Tier 2, Section 5.4.10.3, "Design Evaluation," it is demonstrated by analysis in accordance with requirements for ASME Section III Class 1 vessels that the pressurizer is adequate for all normal operating and transient conditions expected during the life of the facility. The staff issued RAI 233-8244, Question 05.02.02-1 (ML15296A004), requesting the applicant to provide additional details regarding the analysis of the pressurizer size and spray capacity. In its

response to RAI 233-8244, Question 05.02.02-1 (ML15348A083), the applicant contained the results of an analysis of the turbine trip event, which was identified by the applicant as the limiting AOO. The applicant's analysis demonstrated that the pressurizer and associated spray are sized such that the pressurizer POSRVs and RPS are not actuated for the limiting AOO.

5.2.2.5 Combined License Information Items

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

5.2.2.6 Conclusion

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

5.2.3 Reactor Coolant Pressure Boundary Materials

5.2.3.1 Summary of Application

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

"Processing and Use of Stainless Steel," Revision 1, issued March 2011;"

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

issued October 2013; and, RG 1.65, "Materials and Inspections for Reactor Vessel Closure Studs," Revision 1, issued April 2010.

control element drive mechanism (CEDM) component,

Material Specification

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

Compatibility with Reactor Coolant

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

"martensitic stainless steel;" (e.g., SG divider plate)

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

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

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

Fabrication and Processing of Ferritic Materials

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

Fabrication and Processing of Austenitic Stainless Steel

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

Welding procedures are based on testing on stainless steel mockups to establish procedures that do not produce sensitized structures in unstabilized Type 300 stainless steels. Confirmation of this is provided by application of ASTM A262 Practices A or E. The primary parameters from this testing are: weld heat input less than 23.6 kJ/cm, interpass temperature 176.6 °C (349.9 °F) maximum, and carbon content 0.065 percent maximum. Welds produced under these conditions are considered to be adequate for service coupled with the oxygen content limits on the reactor coolant chemistry.

"176.7 °C (350 °F)"

"427 ºC (800 ºF) to 816 ºC (1500 ºF)"

The unstabilized stainless steel to be used in APR1400 consists of Type 304 and Type 316 material. These materials are not to be exposed to temperatures from 427 °C (800.6 °F) to 816 °C (1500.8 °F) to prevent sensitization. In addition the ferrite content and temperature exposure of cast stainless steels and stainless steel weld filler metals are controlled to prevent sensitization and other degradation. The fracture toughness of cast stainless steels is to be ensured by control of ferrite content based on the operating temperature of the component and the specific type of cast steel. "except for bolting or pin materials" (e.g, refer to DCD Sections 4.5.1.2 and 4.5.2.1)

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

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

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

Prevention of PWSCC for Nickel-Based Alloys

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

Threaded Fasteners

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

5.2.3.2 Regulatory Basis

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

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

5.2.3.3.4 *Fabrication and Processing of Austenitic Stainless Steel*

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

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

"(350 °F),"

The DCD indicates that all austenitic stainless steels are supplied in the annealed condition as specified by the pertinent ASME Code and are to be treated consistent with RG 1.44 including use of ASTM A262 Practice A or E tests to confirm and ensure proper heat treatment. The applicant stated that welding procedures were extensively tested on mockups, fabricated using production techniques, to select only the procedures and/or practices demonstrated not to produce a sensitized structure. ASTM A262, Practice A or E is used as the standard for acceptability consistent with RG 1.44. Based on this testing the applicant stated that a weld heat input of less than 23.6 kJ/cm, maximum interpass temperature of 176.7 °C (350.1 °F), and a maximum carbon content of 0.065 percent would produce adequate non-sensitized results. While the staff cannot determine whether the use of the weld heat input, maximum interpass temperature, and maximum carbon content specified by the applicant will ensure non-sensitization of the material, that staff determined that this is acceptable as conformance with RG 1.44 through the use of ASTM A262 Practice A or E will ensure adequate results.

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

"except for bolting or pin materials" (e.g, refer to DCD Sections 4.5.1.2 and 4.5.2.1)

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

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

DCD Tier 2, Section 5.2.3.4.4 details the controls on welding. The applicant listed adherence to RGs 1.31 with ferrite limits, 1.34, and 1.71. The ferrite limits are consistent with the staff provides threshold values and operator actions for primary water chemistry that are in compliance with the latest version of the EPRI Guidelines (COL 9.3(7)). These COL Items ensure the APR1400 primary water chemistry will conform to the latest EPRI Guidelines and thus will follow industry best practices, and the staff therefore determined that the APR1400 RCS water chemistry values, are acceptable.

EPRI Guidelines also specify certain water chemistry concentrations as control parameters, which require strict adherence to limits in order to achieve material protection. In APR1400 DCD Tier 2, Section 5.2.3.2.1, “Reactor Coolant Chemistry,” the applicant provided a detailed discussion of the control parameters for dissolved oxygen, ammonia, lithium, dissolved hydrogen, fluoride and sulfate. For each of these chemical species, except Li, the applicant provided limiting values equal to, or more stringent than, the Action Level 2 values from the EPRI Guidelines. (For chloride and fluoride ion concentrations, these values are also mentioned in RG 1.44 as strict limits.) For Li concentration, no exact limits are provided in the EPRI Guidelines, as this component is determined by the pH control. The EPRI Guidelines only state Action Level 1 limits for H_2 and O_2, and DCD Tier 2, Table 5.2-55, “Reactor Coolant Design Specification,” standard values are consistent with, or more stringent than, these limits. The EPRI Guidelines stipulate sampling three times/week for all but sulfate (once/week) and dissolved O (as stipulated in plant TS). However, they note that sampling frequencies may vary and that they will be determined in the plant TS. The staff concluded that the applicant has provided appropriate limits for the RCS water chemistry control parameters since the limits are the same, or more stringent that the limits recommended by the EPRI Guidelines.

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

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

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

"silica"

- 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.

"5.2-7"

DCD Tier 2, Table 9.3.4-1B, "Reactor Coolant Detailed Plant Startup Operation Specifications," states that the standard value for pH at 25 °C (77 °F) is between 4.6 and 7.3. The applicant provided a description of its pH control program that complies with the EPRI Guidelines. The applicant also stated that during prior to initial criticality, the boron will range from near zero to 4,400 ppm, and lithium from 0 to 3.5 ppm. This will give a pH range of 4.2 to 10.7. During operation boron will range from near zero at end of cycle to 4,400 ppm at refueling. Lithium, the alkalizing additive employed, will vary from 0.2-3.5 ppm. In addition the applicant agreed to rewrite Table 9.3.4-1B. The staff finds this acceptable as it conforms to the EPRI Guidelines.

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

5.2.3.4 Combined License Information Items

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

Table 5.2.3-1 Combined License Information Items

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

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

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

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

5.2.5.4.6 *Intersystem Leakage*

The regulatory positions in RG 1.45 states that the plant should monitor intersystem leakage for systems connected to the RCPB. SRP Section 5.2.5, "Reactor Coolant Pressure Boundary Leakage Detection," indicates that the applicant should identify all potential intersystem leakage paths and the instrumentation to monitor the intersystem leakage.

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

The staff issued 552-9083, Question 05.02.05-4 (ML17228A993), requesting the applicant revise DCD Tier 2, Section 5.2.4 to identify (1) the CVCS as a potential intersystem leakage path and (2) the instrumentation provided to monitor intersystem leakage between the RCS and the CVCS. In its response to RAI 552-9083, Question 05.02.05-4, (ML17248A376), the applicant revised DCD Tier 2, Section 5.2.5.4 to address these issues. The staff determined the response acceptable because it adequately described the intersystem leakage between the RCS and the CVCS. **RAI 552-9083, Question 05.02.05-4 is tracked as a confirmatory item.**

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

5.2.5.4.7 *Technical Specifications*

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

In DCD Tier 2, Chapter 16, "Technical Specifications," the applicant provided LCO 3.4.12 and LCO 3.4.14 for the leakage detection system that specify allowable leakage limits and operability requirements for instruments of diverse monitoring principles during plant operating modes 1, 2, 3, and 4.

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

a) No pressure boundary leakage

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

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

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

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

a) One containment sump level monitor

b) One containment atmosphere radioactivity (particulate) monitor

c) One containment atmosphere humidity monitor

The adequacy of the specified TS limit of 0.5 gpm for unidentified leakage and leakage detection systems are evaluated above in SER Section 5.2.5 (D)(b). The limit of 10 gpm for identified leakage is consistent with Combustion Engineering Standard TS. The limit for the primary-to-secondary leakage in TS LCO 3.4.12 (d) is reviewed in SER Subsection 5.4.2.2, "Steam Generator Program." The adequacy of three diverse leakage detection monitors in TS LCO 3.4.14, which include sump level monitor, radioactivity (particulate) monitor, and humidity monitor, are evaluated in SER Section 5.2.5 (D)(b) above. Additional review on TS LCO 3.4.14 is in SER Section 16.0(D)(g).

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

of Table 2.4.7-1, “Leakage Detection System ITAAC.” In addition, the containment atmosphere humidity monitor will be deleted from Subsection 2.4.7.1, “Design Description,” and the Table 2.4.7-1 as a result of the applicant’s response to RAI 80-8040, Question 05.02.05-1. The humidity sensor is designed to indicate a sudden and significant increase of humidity level by annunciating an alarm in the MCR due to unidentified leakage.” The accuracy and sensitivity of the humidity sensor used in the APR1400 design is not sufficient enough to establish a quantitative correlation between the unidentified RCS leakage and the humidity sensor to detect the level of detail of 0.5 gpm within one hour. The staff determined that the response to RAI 369-8486, Question 05.02.05-3 is acceptable because the criteria should be a leakage rate instead of a “change in” leakage rate. In addition, a quantitative ITAAC criterion for the humidity monitor can be removed, because the quantitative acceptance criterion of the response time for the leakage detection is not applicable as discussed in previously in Section 5.2.5 (D)(b) of this SER.

Based on the above, the staff determined that the ITAAC for the APR1400 RCS leakage detection is adequate because it has addressed the capability of all the RCS leakage detection methods listed in the plant TS and the associated display and alarms in the MCR. Therefore, RAI 369-8486, Question 05.02.05-3, was being tracked as a confirmatory item. The staff confirmed that the DCD Tier 2, Revision 1, dated March 10, 2017, was revised as committed in the response to RAI 369-8486, Question 05.02.05-3. Therefore, RAI 369-8486, Question 05.02.05-3, is resolved and closed.

5.2.5.5 Combined License Information Items

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

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

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

5.2.5.6 Conclusion

Based on the above, pending the resolution of two 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

"CED mechanism"

The reactor vessel (RV) is a vertically mounted cylindrical shell with a hemispherical lower head welded to the cylindrical shell and a removable hemispherical upper closure head. The RV contains the reactor fuel and the vessel internals, which direct the flow of reactor coolant. The RV has four inlet and two outlet nozzles located in a horizontal plane just below the RV flange, but above the top of the fuel. The reactor coolant enters the RV through the inlet nozzles, is guided downward into the annulus between the RV shell and the core barrel, and then upward through the core, acquiring thermal energy. The reactor coolant leaves the RV through the outlet nozzles. The RV closure head contains penetrations for 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.

5.3.1.4.4 *Special Controls for Ferritic and Austenitic Stainless Steels*

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

"alloy"

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

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

5.3.1.4.5 *Fracture Toughness*

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

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

5.3.1.4.6 *Material Surveillance*

GDC 32 requires that the RCPB components shall be designed to permit an appropriate material surveillance program for the RV. Title 10 CFR Part 50, Appendix H states that RVs that are projected to have a peak neutron fluence exceeding 10^{17} n/cm^2 (E > 1.0 MeV) must have their beltline materials monitored by a surveillance program complying with ASTM

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International (formerly the American Society for Testing and Materials) (ASTM) E-185. The latest edition of ASTM E-185 incorporated by reference into 10 CFR Part 50, Appendix H, "Reactor Vessel Material Surveillance Program Requirements," is the 1982 Edition (ASTM E-185-82).

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

"with twenty-four longitudinal standard charpy test specimens for base metal"

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

"with 12 longitudinal precracked charpy specimens for base metal"

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

DCD Section 5.3.1.6.3, "Surveillance Capsules," describes the design and layout of the surveillance capsule assemblies and DCD Section 5.3.1.6.5, "Irradiation Locations," describes their location within the RV and associated lead factors. A diagram of the capsule assemblies is provided in DCD Figure 5.3-1, "Typical Surveillance Capsule Assembly." The applicant stated that the capsule assemblies are corrosion resistant and that all compartments are sealed with helium. The surveillance capsules are equipped with lock assemblies that fix the locations of

"(transverse)"

"(transverse)"

"approximately 1.4." (refer to DCD Sec. 5.3.1.6.5)

the capsules within the holders and prevent relative motion. The lock assemblies also serve as a point of attachment for the tooling used to remove the capsules from the reactor. DCD Section 5.3.1.6.5 states that the design of the capsule assemblies and holders also permits the remote installation of replacement capsule assemblies. The capsule holders are welded to the RV cladding on the inside surface, and the welds are subject to inspection in accordance with

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ASME Section III and Section XI. The surveillance capsules are located to produce a lead factor (ratio of the neutron flux at the location of the capsule to that at the RV inner surface at the peak neutron fluence location) of 1.5. The axial and vertical position of the capsule assemblies within the RV are illustrated on DCD Figures 5.3-5, “Location of Surveillance Capsule Assemblies (Plan View),” and 5.3-6, “Location of Surveillance Capsule Assemblies (Elevation View).” The azimuthal locations of the capsule assemblies are also provided in DCD Table 5.3-7,, “Capsule Assembly Removal Schedule.” The design, location, and associated lead factors of the surveillance capsule assemblies and holders are acceptable because they meet the requirements of ASTM E-185-82, and therefore comply with 10 CFR Part 50, Appendix H.

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

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

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

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

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

"Irradiation Effects Prediction Basis,"

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

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

5.3.1.4.7 *Reactor Vessel Fasteners*

DCD Section 5.3.1.7, "Reactor Vessel Fasteners," states that the bolting material for the RV closure head is fabricated from SA-540, Grade B24, Class 3. The applicant indicated that the bolting material meets the fracture toughness requirements of 10 CFR Part 50, Appendix G and the intent of RG 1.65. The DCD also states that the bolting materials are tested in accordance with ASME Section III, NB-2220 and NB-2300. NDE is performed in accordance with ASME Section III, NB-2580 during the manufacturing process. Manganese phosphate coating is used on the studs, nuts and washers to improve anti-galling properties and corrosion resistance. Also, nickel-based anti-seize lubricant is added to the threads and bearing surfaces to prevent 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," SCS in Section 5.4.7, "Shutdown Cooling System," pressurizer in Section 5.4.10, "Pressurizer," PRT in Section 5.4.11, "Pressurizer Relief Tank," RCS high-point vents in Section 5.4.12, "Reactor Coolant System High Point Vents," pressurizer pilot operated safety relief valves in Section 5.4.13, "Main Steamline Flow Restrictor," and RCS supports in Section 5.4.14,, "Safety and Relief Valves." According to RG 1.206, the NRC reserved DCD 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

APR1400

the design stress limit in lieu of the ultimate strength of the flywheel material. The staff verified that the calculated stresses of the flywheel during normal operating conditions are below one-third of the yield strength, and that the stresses during design overspeed conditions are less than the two-thirds of the yield strength of the flywheel material.

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

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

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

0.08128 mm (0.0032 inch)

Technical Report APR1400-A-M-NR-14001-P, Revision 3 also provided a fatigue crack growth that was determined from the crack growth rate in Appendix A of Section XI to the ASME Code. An initial crack length of 13 mm (0.50 inch) was assumed, with an assumed loading cycle of 6000 starts and stops for the life of the pump. A crack growth of 0.8128 (0.032 inch) was calculated. Since the fatigue crack growth of 0.8128 (0.032 inch) for a 60-year period is minimal, fatigue is not a major contributor for crack growth. Also, Technical Report APR1400-A-M-NR-14001-P, Revision 3 demonstrated that at design overspeed, the critical crack length is

over 152.4 mm (6 inches). Since, the inspection technique is capable of detecting flaws of 13 mm (0.50 inch), the staff concludes that the structural integrity of the flywheel is ensured because the ISI performed at regular intervals, and will detect the flaw before it will reach the critical crack size.

In its response to RAI 341-8410, Question 05.04.01.01-3c (ML16120A475), the applicant did not provide an analysis of the hub nor an acceptable justification for the fatigue crack growth rates used for the flywheel. The use of fatigue crack growth rates from ASME Code, Section XI, Appendix A, Paragraph A-4300 for the proposed flywheel material is unacceptable, as those fatigue crack growth rates are for SA-533 Grade B, Class 1 and SA-508, Class 3 steels, and no

III and XI of the ASME Code, in that the probability of a flywheel failure is sufficiently small, thereby minimizing the potential of generating missiles from the RCP flywheel.

5.4.1.2 RCP Design

5.4.1.2.1 *Introduction*

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

5.4.1.2.2 *Summary of Application*

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

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

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

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

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

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 departure from nucleate boiling (DNB) margin during the gradual loss of forced RCS flow that occurs during RCP coastdown following a LOOP event.

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

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

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

The shaft seal assembly consists of two face-type, mechanical seals in series, with a controlled leakage bypass network to provide differential pressure equally across each seal, and also a third low-pressure vapor seal designed to withstand system operating pressure only when the pumps are not operating. The applicant designed each of the mechanical seals for the full the proposed DCD revisions. The staff issued RAI 299-8310, Question 05.04.02.02-1, (ML15314A024), requesting consistency in the response of the applicant's August 4, 2015, letter and the proposed DCD revision. Specifically, the response incorrectly stated that only one paragraph would be deleted.

In its response to RAI 299-8310, Question 05.04.02.02-1 (ML17233A369), the applicant confirmed the intent to delete the two unnecessary paragraphs from DCD Subsection 5.4.2.2.2.12 and move a third paragraph to DCD Subsection 5.4.2.2.2.3. The response included the proposed DCD revisions submitted in the initial response. The staff determined that the proposed DCD revisions are acceptable because they delete unnecessary information and relocate information about the bases for the SG Program to a more appropriate location. The staff confirmed that DCD Tier 2, Revision 1, dated March 10, 2017, was revised as committed in the response to RAI 299-8310, Question 05.04.02.02-1. Subsequently, in its revised response to RAI 299-8310, Question 05.04.02.02-1 (ML17233A369), the applicant replaced the term "repair criteria" with "plugging criteria" in two places in DCD Subsection 5.4.2.2.2.3. The staff determined this is acceptable because "plugging criteria" is consistent with the STS as modified by TSTF-510. Therefore, **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. The staff found that some references to repairs and sleeves remained in Revision 1 of the DCD and TS. Verification of the deletion of these references in the applicant's next revision of DCD Subsection 5.4.2.2 and the TS 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 TS Task Force (TSTF)-510, Revision 2, "Revision to Steam Generator Program Inspection Frequencies and Tube Sample Selection." The staff issued RAI 299-8310, Question 05.04.02.02-2, (ML15314A024), requesting that the applicant change "degradation" to "indications" as originally proposed. In its response to RAI 299-8310, Question 05.04.02.02-2, (ML15352A301), the applicant proposed changing the last word in DCD Item 5.4.2.2.2.12.d from "degradation" back to "indications" as originally proposed. The staff determined that this is acceptable because the proposed wording is consistent with the STS. Therefore, **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. The staff confirmed that the statement was deleted in DCD Tier 2, Revision 1, dated March 10, 2017. With respect to water chemistry programs, the DCD stated they are based on the EPRI programs (primary and secondary) but did not describe them or DCD 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. The staff confirmed that these revisions were incorporated into DCD Tier 2, Revision 1, dated March 10, 2017.

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