5 REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

5.1 Summary Description

The AP1000 Design Control Document (DCD) Tier 2 Section 5.1, "Reactor Coolant System and Connected Systems," provides a summary description of the AP1000 reactor coolant system (RCS) and connected systems, as well as their design bases. DCD Tier 2 Sections 5.2 through 5.4 provide detailed design descriptions of reactor coolant pressure boundary integrity, reactor vessel (RV), and component and system design respectively. Therefore, in the following subsections the staff provides an overview of the AP1000 RCS and connected systems without an evaluation. Sections 5.2 through 5.4 of this report provide the staff evaluation.

DCD Tier 2 Figures 5.1-1 through 5.1-3 show the schematic and layout of the applicant AP1000 RCS and its principal auxiliary systems. The RCS consists of two heat transfer circuits (loops), each with a U-tube steam generator, two reactor coolant pumps, and a single hot leg pipe and two cold leg pipes for circulation of reactor coolant. The RCS also includes the pressurizer, interconnecting piping, valves, and instrumentation for operational control, actuation, and monitoring of plant safety systems. All RCS equipment is located in the reactor containment.

The reactor coolant pressure boundary (RCPB) provides a barrier against the release of radioactivity generated within the reactor. It is designed to provide a high degree of integrity throughout operation of the plant.

5.1.1 Design Bases

DCD Tier 2 Section 5.1.1, "Design Basis," lists the following design bases for the RCS and its major components:

- The RCS transfers to the steam and power conversion system the heat produced during power operation, as well as the heat produced when the reactor is subcritical, including the initial phase of plant cooldown.

- The RCS transfers to the normal residual heat removal system (RNS) the heat produced during the subsequent phase of plant cooldown and cold shutdown.

- During power operation and normal operational transients (including the transition from forced to natural circulation), the RCS [removes heat and maintains] fuel condition within the operating bounds permitted by the reactor control and protection systems.

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

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

- The RCS pressure boundary accommodates the temperatures and pressures associated with operational transients.

- The reactor vessel supports the reactor core and control rod drive mechanisms.

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

- The reactor coolant pumps supply the coolant flow necessary to remove heat from the reactor core and transfer it to the steam generators.

- The steam generators provide high-quality steam to the turbine. The tubes and tubesheet boundary prevent the transfer of radioactivity generated within the core to the secondary system.

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

- The RCS is monitored for loose parts, as described in [DCD Tier 2 Section 4.4.6].

- Applicable industry standards and equipment classifications of RCS components are identified in [DCD Tier 2 Tables 3.2-1 and 3.2-3].

- The reactor vessel head is equipped with suitable provisions for connecting the head vent system, which meets the requirements of 10 CFR 50.34(f)(2)(vi) (Three Mile Island [TMI] Action Item II.B.1). [See DCD Tier 2 Section 5.4.12]

- The pressurizer surge line and each loop spray line connected with the RCS are instrumented with resistance temperature detectors (RTDs) attached to the pipe to detect thermal stratification.

5.1.2 Design Description

The following components are included in the AP1000 RCS:

- the RV, including control rod drive mechanism housings
- the reactor coolant pumps, comprised of four canned motor pumps, which transfer fluid through the entire reactor coolant and reactor systems
- the primary portion of the steam generators containing reactor coolant, including the channel head, tubesheet, and tubes
- the pressurizer, which is attached by the surge line to one of the reactor coolant hot legs
- the pressurizer safety valves and automatic depressurization system valves
- the RV head vent isolation valves
- the interconnecting piping and fittings between the system components
- the piping, fittings, and valves leading to connecting auxiliary or support systems

DCD Tier 2 Tables 5.1-1 through 5.1-3 specify the principal system pressures, temperatures, flow rates, system design and operating parameters, and the thermal-hydraulic parameters of the RCS.

During operation, the reactor coolant pumps circulate pressurized water through the RV and the steam generators respectively. The water, which serves as coolant, moderator, and solvent for boric acid (chemical shim control), is heated as it passes through the reactor core. Heat is removed from the water and transferred to the main steam system in the steam generators. The water is then returned to the RV by the reactor coolant pumps to repeat the heat removal cycle.

RCS pressure is controlled by operation of the pressurizer, where water and steam are maintained in equilibrium by the activation of electrical heaters, a water spray, or both. Steam is formed by the heaters or condensed by the water spray to control pressure variations resulting from expansion and contraction of the reactor coolant.

Spring-loaded safety valves are connected to the pressurizer to provide overpressure protection for the RCS. These valves discharge into the containment atmosphere. Also attached to the pressurizer are two redundant sets of the first-three-stage automatic depressurization system (ADS) valves. These valves discharge steam and water (in three stages of operation) through spargers located in the in-containment refueling water storage tank (IRWST). The IRWST is part of the AP1000 passive core cooling system.

Two fourth-stage automatic depressurization valves are connected by two redundant paths to the RCS hot legs. These valves discharge directly to the containment atmosphere.

The RCS is also served by a number of auxiliary systems:

- the chemical and volume control system (CVS)
- the passive core cooling system (PXS)
- RNS
- the steam generator system (SGS)
- the primary sampling system
- the liquid radwaste system
- the component cooling water system (CCS)

5.1.3 System Components

DCD Tier 2 Section 5.1.3, "System Components," describes the major components of the RCS, below.

5.1.3.1 Reactor Vessel

The RV is cylindrical, with a hemispherical bottom head and a removable, flanged, hemispherical upper head. The vessel contains the core, core support structures, control rods, and other parts directly associated with the core. The vessel interfaces with the reactor internals, the integrated head package, and reactor coolant loop piping. It is supported by the containment building concrete structure.

The design of the AP1000 RV closely matches the existing vessel designs of other Westinghouse three-loop plants. New features have been incorporated into the AP1000 without departing from the proven features of existing vessel designs.

The RV has inlet and outlet nozzles positioned in two horizontal planes between the upper head flange and the top of the core. The nozzles are located in this configuration to provide an acceptable cross-flow velocity in the vessel outlet region, and to facilitate optimum layout of the RCS equipment. The inlet and outlet nozzles are offset, with the inlet positioned above the outlet, to allow mid-loop operation for removal of a main coolant pump without discharge of the core.

Coolant enters the vessel through the inlet nozzles and flows down the core barrel-vessel wall annulus, turns at the bottom, and flows up through the core to the outlet nozzles.

5.1.3.2 Steam Generators

Each steam generator (SG) is a vertical shell and U-tube evaporator with integral moisture separating equipment. The basic SG design and features are similar to previous Westinghouse SGs, including replacement SG designs.

The DCD describes several design enhancements to the AP1000 SGs. These include nickel-chromium-iron Alloy 690 thermally treated tubes on a triangular pitch, improved anti-vibration bars, single-tier separators, enhanced maintenance features, and a primary-side channel head design that allows easy access and maintenance by robotic tooling. The AP1000

SG employs tube supports utilizing a broached hole support plate design. All tubes in the SG are accessible for sleeving, if necessary.

The basic function of the AP1000 SG is to transfer heat from the single-phase reactor coolant water through the U-shaped heat exchanger tubes to the boiling, two-phase steam mixture in the secondary side of the SG. The SG separates dry, saturated steam from the boiling mixture, and delivers the steam to a nozzle from which it is delivered to the turbine. Water from the feedwater system replenishes the SG water inventory by entering the SG through a feedwater inlet nozzle and feedring.

In addition to its steady-state performance function, the SG secondary side provides a water inventory that is continuously available as a heat sink to absorb primary side high-temperature transients.

5.1.3.3 Reactor Coolant Pumps

Each reactor coolant pump (RCP) is a high-inertia, high-reliability, low-maintenance, hermetically sealed canned motor pump that circulates reactor coolant through the RV, loop piping, and SGs. The AP1000 design uses four RCPs. Two pumps are coupled with each SG. The pumps are integrated into the SG channel head.

The integration of the pump suction into the bottom of the SG channel head eliminates the cross-over leg of coolant loop piping; reduces the loop pressure drop; simplifies the foundation and support system for the SG, pumps, and piping; and reduces the potential for uncovering the core by eliminating the need to clear the loop seal during a small loss-of-coolant accident (LOCA).

Each AP1000 RCP is a vertical, single-stage centrifugal pump designed to pump large volumes of coolant at high pressures and temperatures. The pump impeller attaches to the rotor shaft of the driving motor, which is an electric induction motor. Both the stator and rotor are encased in corrosion-resistant cans constructed and supported to withstand full system pressure. Because of the RCPs canned design, shaft seals are eliminated in the AP1000 design. To provide the rotating inertia needed for flow coast-down, a uranium alloy flywheel is attached to the pump shaft.

The pump motor size is minimized through the use of a variable frequency drive to provide speed control in order to reduce motor power requirements during pump startup from cold conditions. The variable frequency drive is used only during heatup and cooldown when the RCS temperature is less than 232.2 ^{o}C (450 ^{o}F). During power operations the drive is isolated and the pump is run at constant speed.

5.1.3.4 Primary Coolant Piping

RCS piping is configured with two identical main coolant loops, each of which employs a single 78.34 cm (31 in.) inside diameter hot leg pipe to transport reactor coolant to a SG. The two RCP suction nozzles are welded directly to the outlet nozzles on the bottom of the SG channel head. Two 55.88 cm (22 in.) inside diameter cold leg pipes in each loop (one per pump) transport reactor coolant back to the RV to complete the circuit. The loop configuration and material have been selected such that pipe stresses are sufficiently low for the primary loop and large auxiliary lines to meet the requirements to demonstrate "leak-before-break" (LBB). Thus, pipe rupture restraints are not required, and the loop is analyzed for pipe ruptures only for small auxiliary lines that do not meet the LBB requirements.

5.1.3.5 Pressurizer

The pressurizer is the principal component of the RCS pressure control system. This is a vertical, cylindrical vessel with hemispherical top and bottom heads, where liquid and vapor are maintained in equilibrium, saturated conditions.

A 10.16 cm (4 in.) spray nozzle and two 35.56 cm (14 in.) nozzles for connecting the safety and depressurization valve inlet headers are located in the top head. Electrical heaters are installed through the bottom head. The heaters are removable for replacement. The bottom head contains the nozzle for attaching the surge line. This line, which connects the pressurizer to a hot leg, provides for the flow of reactor coolant into and out of the pressurizer during RCS thermal expansions and contractions.

5.1.3.6 Pressurizer Safety Valves

The two pressurizer safety valves are spring-loaded and self-actuated with back-pressure compensation. Valve set pressure is 17.23 MPa (2,485 psig). Their combined capacity is determined by the requirement to not exceed maximum RCS pressure limit during the Level B service condition loss-of-load transient., i.e., 110 percent of the RCS design pressure of 17.23 MPa (2485 psig), in compliance with the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section III.

5.1.3.7 Automatic Depressurization Valves

Several of the passive safety features of the AP1000 design are dependent on depressurization of the RCS. This is accomplished by the ADS valves located above the pressurizer (Stages 1 to 3) and attached to the RCS hot legs (Stage 4). The Stage 1 to 3 valves are arranged in six parallel sets (two valves in series) opening in three stages. The Stage 4 ADS valves consist of four paths, each path having two valves in series. To mitigate the consequences of the various accident scenarios, the ADS valves are arranged to open in a prescribed sequence determined by core makeup tank level and a sequence timer. A more detailed description of the ADS valves is included in DCD Tier 2 Section 5.4.6, “Automatic Depressurization System Valves,” and DCD Tier 2 Section 6.3, “Passive Core Cooling System.”

5.1.4 System Performance Characteristics

DCD Tier 2 Section 5.1.4, “System Performance Characteristics,” discusses the thermal-hydraulic parameters, system performance parameters and supporting design procedures used to establish the performance characteristics of the AP1000 RCS. The detailed design procedure establishes a best-estimate flow and conservatively high and low flows for the applicable mechanical and thermal design considerations. In establishing the range of design flows, the procedure accounts for uncertainties in the component flow resistance and in pump head-flow capability. The procedure also accounts for the uncertainties in the technique used to measure flow in the operating plant. DCD Tier 2 Section 5.1.4 also defines the four reactor coolant flows that are applied in plant design considerations, which are described as follows.

5.1.4.1 Best Estimate Flow

The best-estimate flow is the most likely value for the normal full-power operating condition. This flow value is determined by the best estimate of fuel, RV, SG, and piping flow resistance, and on the best estimate of the RCP head and flow capability. No uncertainties are assigned to either the system flow resistance or the pump head. The best-estimate flow provides the basis for the other design flows required for the system and component design. The best-estimate flow and head also define the performance requirement for the RCP. DCD Tier 2 Table 5.1-3 lists system pressure losses on the basis of best-estimate flow.

Although the best-estimate flow is the most likely value to be expected in operation, more conservative flow rates (such as thermal design flow rate and mechanical design flow rate) are applied in the thermal and mechanical designs.

5.1.4.2 Minimum Measured Flow

The minimum measured flow is specified in the technical specifications (TSs) as the flow that must be confirmed or exceeded by the flow measurements obtained during plant startup. This is the flow used in reactor core departure from nucleate boiling (DNB) analysis for the AP1000 thermal design procedure. In the thermal design procedure methodology for DNB analysis, flow measurement uncertainties are combined statistically with fuel design and manufacturing uncertainties. The measured reactor coolant flow will most likely differ from the best-estimate flow because of uncertainties in the hydraulics analysis and inaccuracies in the instrumentation used to measure flow. The measured flow is expected to fall within a range around the best-estimate flow. The magnitude of the expected range is established by statistically combining the system hydraulics uncertainty with the total flow rate within the expected range, less any excess flow margin that may be provided to account for future changes in the hydraulics of the RCS.

5.1.4.3 Thermal Design Flow

The thermal design flow is the conservatively low value used for thermal-hydraulic analyses where the design and measurement uncertainties are not combined statistically. Additional flow margin must therefore be explicitly included. The thermal design flow is derived by subtracting the plant flow measurement uncertainty from the minimum measured flow. The thermal design flow is approximately 4.5 percent less than the best-estimate flow. The thermal design flow is confirmed when the plant is placed in operation. DCD Tier 2 Table 5.1-3 presents important design parameters founded on the thermal design flow.

5.1.4.4 Mechanical Design Flow

Mechanical design flow is the conservatively high flow used as the basis for the mechanical design of the RV internals, fuel assemblies, and other system components. Mechanical design flow is established at 104 percent of best-estimate flow.

5.2 Integrity of Reactor Coolant Pressure Boundary

5.2.1 Compliance With Code and Code Cases

General Design Criteria 1, “Quality Standards and Records,” (GDC 1) requires that nuclear power plant structures, systems, and components (SSCs) important to safety be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety function to be performed. This requirement is applicable to both pressure-retaining and non-pressure-retaining SSCs that are part of the RCPB and other systems important to safety. Where generally recognized codes and standards are used, they must be identified and evaluated to determine their adequacy and applicability.

5.2.1.1 Compliance With 10 CFR 50.55a

Pursuant to Title 10 of the Code of Federal Regulations (10 CFR) 50.55a, “Codes and Standards,” components important to safety are subject to the following requirements:

(1) RCPB components must meet the requirements for ASME Class 1 (Quality Group (QG) A) components specified in ASME Code, Section III, except for those components that meet the exceptions of 10 CFR 50.55a(c)(2). Those RCPB components that meet these exceptions may be classified as Class 2 (QG B), or Class 3 (QG C).

(2) In accordance with 10 CFR 50.55a(d) and (e), components classified as QG B and C must meet the requirements for Class 2 and 3 components, respectively, as specified in ASME Code, Section III.

DCD Tier 2 Tables 3.2-1 and 3.2-3, and applicable piping and instrumentation diagrams collectively classify the mechanical and pressure-retaining components of the RCPB that do not meet the exclusion requirements discussed in (1) above, as ASME Code, Section III, Class 1 components. These Class 1 components are designated QG A in conformance with Regulatory Guide (RG) 1.26, Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants, Revision 3. The staff has compared the DCD Tier 2 Tables 3.2-1 and 3.2-3 with the corresponding tables in the AP600 DCD and found no significant changes.

The staff has evaluated the quality group classifications discussed in Section 3.2.2 of this report and concludes that AP1000 mechanical and pressure-retaining components in the RCPB have been acceptably classified as QG A in accordance with 10 CFR 50.55a, and are consistent with applicable portions of the NRC Standard Review Plan (SRP) Section 5.2.1.1.

In addition to the QG A components of the RCPB, certain lines that will perform a safety function and that meet the exclusion requirements of 10 CFR 50.55a(c)(2) are classified as QG B or C in accordance with Positions C.1 or C.2 of RG 1.26, Revision 3, and will be constructed as ASME Code, Section III, Class 2 or Class 3 components.

As discussed in the DCD Tier 2 Sections 5.2.1.1, “Compliance with 10 CFR 50.55a,” and DCD Tier 2 Section 5.2.1.3, “Alternate Classification,” the portion of the CVS inside containment that is defined as part of the RCPB uses an alternate quality group classification to that discussed above. This portion of the CVS is classified as non-safety, Class D. The safety-related classification of the RCPB ends at the third isolation valve between the RCS and the CVS (Reference DCD Tier 2 Figure 9.3.6-1). This is considered to be an alternate to the usual classification of the RCPB. Alternatives to 10 CFR 50.55a(c) requirements are allowed by 10 CFR 50.55a(a)(3) if the proposed alternative design provides an acceptable level of quality and safety. The applicant has provided the following design enhancements to the Class D portion of the CVS as an alternate design to meet an acceptable level of quality and safety:

- The isolation valves between the RCS and the CVS are ASME Class 1 valves designed and qualified for design conditions that include closing against blowdown full flow with full system differential pressure. In addition, although these valves are not classified as pressure isolation valves, DCD Tier 2 Table 3.9.16, provides a commitment that at each refueling outage, these valves will be leak tested to the same leak rate criteria that is specified in the AP1000 TSs for pressure isolation valves. Implementation of these additional leak rate tests will provide redundant leak tight barriers, when required, in each of the lines that connect the RCS and CVS.

- The AP1000 design also contains a third valve in each of the lines that connect the RCS and CVS. These third valves are in addition to the Class 1 valves discussed in the above design enhancement, and they will provide additional assurance that the RCS will be isolated in the event of a CVS failure.

- Although the Class D portions of the CVS are non-seismic, those portions inside containment will be analyzed to the same seismic design criteria as that accepted by the staff for Seismic Category II piping. The staff's acceptance of this criteria is discussed in Section 3.12.3.7 of this report. The seismic Category II analyses will provide adequate assurance that the loads resulting from an safe-shutdown earthquake (SSE) will not result in a loss of structural integrity of the CVS piping.

- All of the Class D portion of the CVS is constructed of or clad with corrosion-resistant material such as Type 304 or Type 316 stainless steel that is compatible with the reactor coolant. In addition, this portion of the CVS is designed to a design pressure of 21.4 MPa (3100 psi), which exceeds the RCS design pressure.

Based on the above design enhancements that have been added to the Class D portion of the CVS, the staff considers that the alternative design provides an acceptable level of quality and safety and is, therefore, acceptable.

DCD Tier 2 Section 5.2.1.1 states that the baseline code used to support the AP1000 DCD is the ASME Code, Section III, 1998 Edition up to and including the 2000 Addenda except that the ASME Code, Section III, 1989 Edition, 1989 Addenda will be used for Articles NB-3200, NB-3600, NC-3600, and ND-3600 in lieu of the later Edition and Addenda. The use of these Edition and Addenda meets 10 CFR 50.55a(b) and the associated modification in 50.55a(b)(1)(iii) and is, thus, acceptable. Any proposed change to the use of the ASME code editions or addenda by a Combined License (COL) applicant will require NRC approval prior to implementation.

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

To avoid this potential inconsistency for the AP1000 pressure-retaining components and their supports, proposed changes to the specific edition and addenda require NRC approval at the COL stage before implementation. This provides the COL applicant with the option to revise or supplement the referenced Code edition with portions of the later Code editions and addenda to ensure consistency between the design and construction practices. However, the staff finds that there might be a need to establish certain design parameters from a specific Code edition or addenda during its design certification review, particularly when that information is important for establishing a significant aspect of the design or is used by the staff to reach its final safety determination. Such considerations, if necessary, are reflected in the various sections of this report. Therefore, all ASME Code Class 1, 2, and 3 pressure-retaining components and their supports shall be designed in accordance with the requirements of ASME Code, Section III, using the specific edition and addenda given in the DCD.

The COL applicant should ensure that the design is consistent with the construction practices (including inspection and examination methods) of the ASME Code edition and addenda, as endorsed in 10 CFR 50.55a. DCD Tier 2 Section 5.2.6.1, "ASME Code and Addenda," contains a commitment that the COL applicant will address consistency of the design with the construction practices (including inspection and examination methods) of the later ASME Code edition and addenda. This is an acceptable commitment.

On the basis of the above evaluations, the staff concludes that the construction of all ASME Code, Class 1, 2, and 3 components and their supports for the AP1000 plant will conform to the appropriate ASME Code editions and addenda and the Commission’s regulations, and that component quality will be commensurate with the importance of the safety function of all such components and their supports. This constitutes an acceptable basis for satisfying GDC 1 and is acceptable.

5.2.1.2 Applicable Code Cases

The only acceptable ASME Code cases that may be used for the design of ASME Code Class 1, 2, and 3 piping systems in the AP1000 standard plant are those either conditionally or unconditionally approved in RG 1.84 in effect at the time of design certification, or determined to be conditionally acceptable as discussed above. However, the COL applicant may submit, with its COL application, future code cases that are endorsed in RG 1.84 at the time of the application provided they do not alter the staff’s safety findings on the AP1000 certified design. In addition, the COL applicant should submit those Code cases which are in effect at the time of the COL application that are applicable to RG 1.147, "Inservice Inspection Code Case Acceptability - ASME Section XI, Division 1," and RG 1.192, “Operation and Maintenance Code Case Acceptability, ASME OM [Operation and Maintenance] Code.”

It should be noted that ASME Code Case 2142-1, "F-Number Grouping for Ni-Cr-Fe, Classification UNS N06052 Filler Metal, Section IX," and CC 2143-1, "F-Number Grouping for Ni-Cr-Fe, Classification UNS W86152 Welding Electrode, Section IX" are also listed in the proposed Table 5.2-3. These cases will not be included in RG 1.84 because they are not ASME Section III Code Cases. However, these cases are acceptable because they include weld metal to be used in the welding of Ni-Cr-Fe Alloy 690, which the staff endorsed and accepted for use in its safety evaluation report (SER) for the Electric Power Research Institute (EPRI) advanced light water reactor Utility Requirements Document, Volume III.

On the basis of the above evaluation, the staff concludes that the ASME Code cases in DCD Tier 2 Table 5.2-3 either meet the guidelines of RG 1.84 or have been reviewed and endorsed by the staff and are acceptable for use on the AP1000 design. Compliance with the requirements of these Code Cases will result in a component quality that is commensurate with the importance of the safety functions of these components, constitutes the basis for satisfying GDC 1, and is acceptable.

5.2.2 Overpressure Protection

In the AP1000 design, overpressure protection for the RCS and steam system pressure boundaries is provided by the pressurizer safety valves (PSVs) and the SG safety valves (SGSVs) during normal power operation, and a relief valve in the suction line of the RNS during low temperature operation, in conjunction with the action of the reactor protection system. There are two PSVs, twelve SGSVs with six valves located in the safety-related portion of each main steam piping upstream of the main steam isolation valve, and one relief valve in the suction line of the RNS. Combinations of these systems provide compliance with the overpressure protection requirements of the ASME Code, Section III, Paragraphs NB-7300 and NC-7300, for pressurized-water reactor (PWR) systems. The ASME Code requires the total relieving capacity be sufficient to prevent a pressure rise of more than 10 percent above the design pressure of the RCS and SGs under any expected system pressurization transient conditions. The RNS suction relief valve for low-temperature over pressure protection (LTOP) prevents the RCS from exceeding the pressure-temperature limits determined from the ASME Code, Section III, Appendix G analyses.

General Design Criteria in 10 CFR Part 50, Appendix A, specify requirements regarding the RCS design:

- GDC 15, “Reactor Coolant System Design,” requires the RCS and associated auxiliary, control, and protection systems be designed with sufficient margin to assure that the design conditions of the RCPB are not exceeded during any condition of normal operation, including anticipated operational occurrences.

- GDC 31, “Fracture Prevention of Reactor Coolant Pressure Boundary,” requires the RCPB to be designed with sufficient margin to assure that boundary behaves in a nonbrittle manner and the probability of rapidly propagating fracture is minimized.

Section 5.2.2 of the SRP, including Branch Technical Position (BTP) Reactor Systems Branch 5-2, describes the acceptance criteria that demonstrate that a plant design complies with GDC 15 and 31. Therefore, the review of the AP1000 overpressure protection was performed in accordance with SRP Section 5.2.2 and BTP RSB 5-2. The staff reviewed the following DCD Tier 2 sections:

- 5.2.2, Overpressure Protection
- 5.4.5, Pressurizer
- 5.4.7, Normal Residual Heat Removal System
- 5.4.9, RCS Pressure Relief Devices
- 5.4.11, Pressurizer Relief Discharge System
- 10.3.2.2.2, Main Steam Safety Valves

5.2.2.1 Overpressure Protection During Power Operation

During power operation, overpressure protection for the RCS is provided by the two PSVs, twelve SGSVs, and the reactor protection system to maintain the primary and secondary pressures within 110 percent of their respective design pressures. The details of the SGSV design are discussed in DCD Tier 2 Section 10.3.2.2.2, “Main Steam Safety Valves,” with design data, including set pressures and relieving capacities, listed in DCD Tier 2 Table 10.3.2-2. The design parameters of the PSVs are specified in DCD Tier 2 Table 5.4-17. The minimum required relief capacity is 340,194 Kg/hr (750,000 lbm/hr) per valve at 3 percent accumulation, and the set pressure is 17.23 MPa ± 0.17 MPa (2485 psig ± 25 psi). The discharge of the PSV is routed through a rupture disk to containment atmosphere. The rupture disk, which has a pressure rating substantially less than the set pressure of the PSV, is to contain leakage past the PSV.

The PSVs are sized as determined by the analysis of a complete loss of steam flow to the turbine, with the reactor operating at 102 percent of rated power. This design-basis event bounds other events that could lead to overpressure of the RCS if adequate overpressure protection were not provided. Such overpressure events include loss of electrical load and/or turbine trip, uncontrolled rod withdrawal at power, loss of reactor coolant flow, loss of normal feedwater, and loss of offsite power to the station auxiliaries. The total PSV capacity is required to be at least as large as the maximum surge rate into the pressurizer during this transient. In this analysis, feedwater flow is also assumed to be lost, and steam relief through the SGSVs is considered when the secondary side pressure reaches 103 percent of the SG shell design pressure. No credit is taken for operation of the pressurizer level control system, pressurizer spray system, rod control system, steam dump system, or steamline power-operated relief valve. The reactor is maintained at full power with no credit taken for reactor trip or reactivity feedback during the transient. A 3 percent set pressure accumulation is also considered for the PSV relief. These assumptions meet the acceptance criteria of II.A of Section 5.2.2 of the SRP. With these assumptions, the results of the design basis safety analysis of a turbine trip event with a complete loss of steam load from full power, described in DCD Tier 2 Section 15.2.3, “Turbine Trip,” show that the actuation of the PSVs maintains the RCS pressure below the 110 percent of the design pressure. This analysis demonstrates that the capacities and setpoints of the PSVs and SGSVs are sufficient to ensure that the pressures of the RCS and the SGs remain below 110 percent of their design pressures. Design-basis safety analyses of the other overpressure events, described in DCD Tier 2 Sections 15.3, “Decrease in Reactor Coolant System Flow Rate, and DCD Tier 2 Section 15.4, “Reactivity and Power Distribution Anomalies,” also show the same conclusion. The PSV and SGSV setpoints and relieving capacities are, therefore, acceptable.

The PSV set pressure of between 17.06 MPa (2460 psig) and 17.41 MPa (2510 psig), i.e., 17.23 MPa (2485 psig) ± 1.0 percent tolerance, is specified in the limiting condition for operation (LCO) for AP1000 Technical Specification (TS) 3.4.6 (DCD Tier 2 Chapter 16, “Technical Specifications”). The PSVs are part of the RCPB and are ASME Code Class 1 components. These valves are tested and analyzed using the design transients, loading conditions, seismic considerations, and stress limits for Class 1 components discussed in DCD Tier 2 Section 3.9.1, “Special Topics for Mechanical Components,” DCD Tier 2 Section 3.9.2, “Dynamic Testing and Analysis,” and DCD Tier 2 Section 3.9.3, “ASME Code Classes 1, 2, and 3 Components, Component Supports, and Core Support Structures.” The staff evaluation of these subsections are discussed in the corresponding sections of this report. In addition, the PSVs are subjected to the verification program established by EPRI to address the requirements of 10 CFR 50.34(f)(2)(x) to qualify their operation for all fluid conditions expected under operating conditions, transients and accidents. This is addressed in Item II.D.1, "Performance Testing of PWR Safety and Relief Valves," in Chapter 20 of this report. The PSVs (i.e., RCS-PL-V005A and RCS-PL-V005B) are also subject to the surveillance requirement of AP1000 TS 3.4.6.1 and the inservice testing program (IST) requirements specified in DCD Tier 2 Table 3.9-16.

As discussed above, the overpressure protection design for the AP1000, at power operating conditions, complies with the guidelines of Section 5.2.2 of the SRP and the requirement of GDC 15, and is therefore acceptable.

5.2.2.2 Overpressure Protection During Low-Temperature Operation

Section 5.2.2 of the SRP specifies that the LTOP system be designed in accordance with the guidance of BTP RSB 5-2. The BTP specifies that the LTOP system be capable of relieving pressure during all anticipated overpressurization events at a rate sufficient to prevent exceeding the applicable TS and Appendix G limits for the RCS while operating at low temperatures. BTP RSB 5-2, the staff also specifies that the LTOP system meet the ASME Code Section III requirements, as well as RGs 1.26 and 1.29 regarding quality group and seismic design classifications. In addition, Section 5.2.2 of the SRP specifies that the LTOP system must be operable during startup and shutdown conditions below the enable temperature defined in BTP RSB 5-2. The enable temperature is defined as the water temperature corresponding to a metal temperature of at least the reference nil-ductility temperature plus 50°C (90°F) at the beltline location.

The LTOP system for the AP1000 is provided by the relief valve in the suction line of the RNS, which discharges to the containment sump. Administrative controls and plant procedures aid in controlling RCS pressure during low-temperature operation. Normal plant operating procedures maximize the use of a steam or gas bubble in the pressurizer during periods of low-pressure, low-temperature operation. For those low-temperature modes when operation with a water-solid pressurizer is possible, the RNS relief valve provides LTOP for the RCS. As discussed in DCD Tier 2 Section 5.4.7, “Normal Residual Heat Removal System,” the RNS relief valve and associated piping are safety-related. DCD Tier 2 Table 3.2-3 specifies that the RNS suction pressure relief valve (RNS-PL-V021) is an AP1000 Class B component, seismic Category I and meets the ASME Code, Section III, Class 2 requirements. Because the relief valve is connected to the piping between the containment isolation valves for the system, it also provides a containment boundary function and is subject to the containment isolation requirements discussed in DCD Tier 2 Section 6.2.3, “Containment Isolation System.” Also, the relief valve is subject to inservice test requirements as described in DCD Tier 2 Table 3.9-16. In addition, AP1000 TS LCO 3.4.14 requires operability of the RNS suction relief valve for low temperature overpressure protection during shutdown modes of operation, including MODE 4 operation when any cold leg temperature is below 135°C (275°F). When the LTOP is enabled, the relief valve will automatically open for overpressure protection when the RCS pressure exceeds the RNS relief valve setpoint. In response to a request for additional information (RAI) 440.036, the applicant stated that the LTOP enable temperature of 135° C (275°F) is based on the pressurizer safety valves for RCS overpressure protection when the RCS temperature is above 135°C (275°F). As indicated in Table 5.3-3 of the DCD, the end-of-life RT_{NDT} for the AP1000 RV is expected to be approximately -4°C (25°F). Therefore, the staff finds that the LTOP enable temperature of 135°C (275°F) is acceptable because it is significantly higher than the enable temperature defined by BTP RSB 5-2, i.e., nil-ductility reference temperature (RT_{NDT})+ 50°C (90°F) at the beltline location.

The sizing and setpressure of the RNS relief valve for LTOP are founded on sizing analysis performed to prevent the RCS pressure from exceeding the lower of either 110 percent of the RNS system design pressure or the applicable reactor vessel pressure/temperature (P/T) limits described in DCD Tier 2 Section 5.3.3, “Pressure-Temperature Limits.” In its response to RAI 440.036, the applicant stated that based on the nominal steady-state P/T limits applicable up to 54 effective full power years, the lowest Appendix G limit from DCD Tier 2 Figures 5.3.2 and 5.3.3 is 7.15 MPa (1,023 psig). Therefore, the RNS relief valve is sized to the system pressure limit of 990 psig, which is 110 percent of RNS design pressure of 6.31 MPa (900 psig). The RNS relief valve sizing is based on the following two types of events:

(1) the mass addition transient caused by a makeup/letdown mismatch
(2) the heat addition transient caused by an inadvertent start of one inactive RCP

These events result in bounding mass and energy input conditions relative to other credible events, such as inadvertent actuation of the pressurizer heaters, loss of residual heat removal with RCS heatup as a result of decay heat and pump heat, and inadvertent hydrogen addition. The design-basis analyses for the sizing of the RNS relief valve for LTOP protection assumes the transients occur while the pressurizer is in water-solid condition. The mass input event assumed the injection of water into the RCS from the operation of both makeup pumps and letdown isolated with a maximum makeup/letdown mismatch flow of 40.1 m^3/h (177 gpm), which is limited by the cavitating venturi located in the discharge header of the CVS system makeup pumps. The case of inadvertent restart of one RCP is postulated to occur over a range of reactor coolant temperatures between 37.8°C and 93.3°C (100°F and 200°F) and with the water in the SG secondary side 27.8°C (50°F) hotter than the primary side water. The assumption of a 27.8°C (50°F) temperature difference as the initial condition for the energy input transient conservatively bounds the cooldown operation controlled by the procedure. To prevent the possibility of a heat input transient, and thereby limit the required flow rate of the RNS suction relief valve, an administrative requirement is imposed in TS LCO 3.4.14 for the LTOP protection system that does not allow an RC pump to be started with the pressurizer level above 92 percent and the RCS temperature above 93.3°C (200°F).

The analysis is performed using the methodology described in the NRC-approved topical report, WCAP-14040-NP-A, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves," issued January 1998. The analysis does not consider single active failure of the RNS relief valve because it is a self-actuated spring relief valve, and the single active failure does not apply to passive valves. Based on the energy input transient, the minimum RNS relief valve capacity of 170 m^3/h (750 gpm) is calculated at an RCS pressure equivalent to the valve setpoint of 4.49 MPa (636 psig) plus 10 percent accumulation, is (4.93 MPa (700 psig)). With this setpoint, the peak pressure at the discharge of the RNS, the pump for the energy input transient is no higher than 6.75 MPa (979 psig), and the peak pressure in the RCS is approximately 5.89 MPa (840 psig). For the mass addition transient, the maximum flow rate is 40.1 m^3/h (177 gpm), which is much less than the RNS relief valve capacity therefore, the peak pressure at the inlet to the RNS relief valve will be no higher than the valve full open pressure of 4.93 MPa (700 psig).

Based on the information above, the relief valve would mitigate the limiting LTOP transient while maintaining the RCS pressure less than 110 percent of RNS design pressure. The minimum RNS relief valve capacity required is 170 m^3/h (750 gpm). DCD Tier 2 Table 5.4.17 provides the RNS relief valve design parameters, i.e., the nominal set pressure of 170 m^3/h (750 gpm), nominal setpressure of 4.49 MPa (636 psig), and full open pressure with 10 percent accumulation of 4.93 MPa (700 psig).

The RNS relief valve setpoint of 4.49 MPa (636 psig) was derived based on the lower of 110 percent of the RNS design pressure and the RCS P/T limit of 7.15 MPa (1023 psig), which was obtained from the bounding P/T heatup and cooldown curves specified in DCD Figures 5.3-2 and 5.3-3. These P/T limit curves are generic limiting curves for the AP1000 RV design on the basis of the copper and nickel material composition as described in DCD Tier 2 Table 5.3-1 and 54 effective full power years (EFPY). If the specific AP1000 P/T curves are not bounded by the curves of DCD Figures 5.3-2 and 5.3-3, either due to different RV material composition or plant operation greater than 54 EFPY, the RNS relief valve setpoint must be reevaluated.

Since the nil-ductility reference temperature of the RV material increases as exposure to neutron fluence increases as a result of neutron embrittlement effect, the operating P/T limit curves need to be periodically adjusted to accommodate the actual shift in the nil-ductility temperature. The RCS P/T limit curves are specified in the Pressure-Temperature Limits Report (PTLR) as required in the AP1000 TS LCO 3.4.3. The bases for AP1000 TS 3.4.14 notes that each time the PTLR curves are revised, the LTOP system must be re-evaluated to ensure its functional requirements can still be met using the RNS suction relief valve, or the depressurized and vented RCS condition. In DCD Tier 2 Section 5.3.6.1, the applicant requires that the COL applicant address the use of plant-specific P/T limit curves relative to the RV material composition during procurement of the RV, as well as the evaluation of the LTOP system, including evaluating the setpoint pressure for the RNS relief valve as noted in the basis of AP1000 TS 3.4.14.

Based on the above evaluation, the staff concludes that the appropriate set pressure will be used for the RNS relief valve to ensure the P/T limits are not exceeded. The AP1000 LTOP system meets BTP RSB 5-2 and is therefore acceptable.

5.2.3 Reactor Coolant Pressure Boundary Materials

The staff reviewed DCD Tier 2 Section 5.2.3, “Reactor Coolant Pressure Boundary Materials,” in accordance with Section 5.2.3, “Reactor Coolant Pressure Boundary Materials,” of the SRP to ensure that the materials are compatible with the primary coolant water.

The materials must meet the following:

- GDC 1 of Appendix A to 10 CFR Part 50 and Paragraph 50.55a(a)(1) of 10 CFR Part 50 require that structures, systems, and components important to safety shall be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety function to be performed;

- GDC 4 requires that structures, systems, and components important to safety shall be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents;

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

- GDC 30 requires that components that are a part of the RCPB shall be designed, fabricated, erected, and tested to the highest quality standards practical;

- GDC 31 requires that the RCPB shall be designed with sufficient margin to assure that, when stressed under operation, maintenance, testing, and postulated accident conditions, it will behave in a nonbrittle manner and with the probability of rapidly propagating fracture minimized;

- Appendix B, “Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants,” to 10 CFR Part 50 establishes the quality assurance requirements for the design, construction, and operation of those systems that prevent or mitigate the consequences of postulated accidents that could cause undue risk to the health and safety of the public; and

- Appendix G, “Fracture Toughness Requirements,” to 10 CFR Part 50 specifies the fracture toughness requirements for ferritic materials of the pressure-retaining components of the RCPB.

The following elements were reviewed: materials specifications compatibility of materials with reactor coolant, fabrication and processing of ferritic materials, and fabrication and processing of austenitic stainless steel. The acceptability of these elements is discussed in DSER sections 5.2.3.2 through 5.2.3.5 respectively.

5.2.3.1 Summary of Technical Information

In DCD Tier 2 Table 5.2-1, “Reactor Coolant Pressure Boundary Materials Specification,” the material specifications for the principal pressure-retaining applications in the Class 1 primary components and reactor coolant system piping are listed. This list includes the RV components, SG components, RCP, pressurizer, core makup tank (CMT), and the passive residual heat removal (RHR) heat exchanger.

The use of nickel-chromium-iron alloy in the RCPB design of AP1000 is limited to Alloy 690. The SG tubes are made of thermally treated Alloy 690. Alloy 600 is used in limited areas for welding or buttering and not in contact with the reactor coolant. The non-safety related portion of the CVS inside containment is constructed of materials compatible with the reactor coolant and is made of or clad with corrosion resistant material equivalent to the corrosion resistance of Types 304 and 316. Cast austenitic stainless steel (CASS) components do not exceed a ferrite content of 30 FN (ferrite number).

The RCS water chemistry is controlled to minimize corrosion and is routinely analyzed for verification. The design of the CVS allows for the addition of chemicals to the RCS to control pH, scavenge oxygen, control radiolysis reactions, and maintain corrosion product particulates below specified limits.

The ferritic low-alloy and carbon steels used in the principal pressure-retaining applications have corrosion resistant cladding material for surfaces exposed to the reactor coolant. This corrosion resistant material is at least equivalent to Types 304 and 316 austenitic stainless steel alloys or nickel-chromium-iron alloy, martensitic stainless steel, and precipitation hardened stainless steel. Austenitic stainless steel and nickel-chromium-iron alloy base materials with primary pressure-retaining applications are used in the solution-annealed or thermally treated conditions.

Hardfacing material in contact with the reactor coolant is primarily a qualified low or zero cobalt alloy equivalent to Stellite-6. The use of cobalt base alloy is minimized. Low or zero cobalt alloys used for hardfacing or other applications where cobalt alloys have been previously used are qualified using wear and corrosion tests. Cobalt-free wear resistant alloys considered for this design include those developed and qualified in nuclear industry programs.

The thermal insulation used on components subject to elevated temperature during system operation is made of the reflective stainless steel-type. In addition, compounded materials are silicated to provide protection of austenitic stainless steels against stress corrosion from accidental wetting from the environment.

The limiting SG and pressurizer reference temperatures for RT_{NDT} temperatures are guaranteed at 21.1°C (70°F) for the base materials and weldments. In addition, these materials meet the 67.7 newton-meter (50 foot-pound) absorbed energy and 0.089 cm (35 mils) lateral expansion requirements at -12.2°C (10°F).

Austenitic stainless steel materials used in the fabrication, installation, and testing of nuclear steam supply components and systems are handled, protected, stored, and cleaned to minimize contamination that could lead to stress corrosion cracking (SCC). Tools used in abrasive work operations on austenitic stainless steel do not contain and are not contaminated with ferritic carbon steel or other materials which could contribute to intergranular cracking or SCC.

The welding of austenitic stainless steel is controlled to mitigate the occurrence of microfissuring or hot cracking, in the weld.

5.2.3.2 Material Specifications

The specifications for pressure-retaining ferritic materials, nonferrous metals and austenitic stainless steels, including weld materials, that are used for each component in the RCPB must meet GDC 1, “Quality Standards and Records;” GDC 30, “Quality of Reactor Coolant Pressure Boundary;” and 10 CFR 50.55a, “Codes and Standards;” as these relate to quality standards for design, fabrication, erection and testing. These requirements are met for materials specifications by compliance with the appropriate provisions of the ASME Code and by applications of materials Code Cases in RG 1.85, “Materials Code Case Acceptability - ASME Code, Section III, Division 1.” In addition, by NRC letter from C.I. Grimes to D.J. Walters, *License Renewal Issue No. 98-0030, “Thermal Aging Embrittlement of Cast Austenitic Stainless Steel Components,”* May 19, 2000, the staff discusses an acceptable screening method based upon Molybdenum content, casting method, and ferrite content in the determination of the susceptibility of CASS components to thermal aging.

The staff reviewed DCD Tier 2 Section 5.2.3.1, “Materials Specifications,” to determine the suitability of the RCPB materials for this application. The AP1000 design conforms with the guidance provided in RG 1.85, “Materials Code Case Acceptability ASME Section III Division 1,” and appropriate provisions of the ASME Code.

The staff noted that the DCD states that the RCP pressure housing will be made from SA 351 or SA 352 CF3A material and that the RCP pressure boundary valve bodies may be castings of SA 351 CF3A. In addition, the DCD states that CASS will not exceed a ferrite content of 30 FN. Since CASS RCP pressure boundary components are subject to thermal embrittlement, the staff requested, in RAI 251.012, the applicant discuss the impact of this aging effect on the integrity of these components, how the thermal embrittlement mechanism has been considered in the design and material selection for the RCPB components, and the need to perform inspections to detect this aging effect. In its response, the applicant stated that, based on experience with casting materials, the selection of low carbon grade casting, i.e., CF3A, and control of the material specifications to below 20 FN, there should be no significant impact of thermal aging on the integrity of the components. The applicant responded further that the ASME Code inservice inspections will be relied on to detect the effects of any thermal aging. The proposed DCD change in the response to RAI 251.012 discusses the COL action items regarding these inspections in DCD Tier 2 Section 5.2.6, “Combined License Information Items.” The applicant also committed to revising the limit of the ferrite content of CASS to a maximum of 20 FN. This revised FN was provided in Revision 4 of DCD Tier 2 Section 5.2.3.1, “Materials Specifications.” The staff reviewed Revision 4 to the DCD and, subject to the clarification discussed below, finds it acceptable since it conforms with the guidance in RG 1.31, "Control of Ferrite Content in Stainless Steel Weld Metal," and criteria acceptable to the staff in the May 19, 2000, letter from C. Grimes to D. Walters.

The applicant needs to clarify in the DCD that the method used to calculate the δ-ferrite is based on Hull's equivalent factors or a method producing an equivalent level of accuracy; i.e., ± 6% deviation between the measured and calculated values, as discussed in the May 19, 2000, letter from C. Grimes to D. Walters. This is Open Item 5.2.3-1.

With the exception of Open Item 5.2.3-1, the staff finds that the materials specifications for the AP1000 design are acceptable and meet GDC-1 and GDC-30 because they meet the applicable provisions of the ASME Code, the applicable regulatory positions in RG 1.85, and criteria discussed in the May 19, 2000, letter from C. Grimes to D. Walters, in assuring quality standards of these materials for application in a nuclear power plant.

5.2.3.3 Compatibility of Materials with the Reactor Coolant

The materials of construction employed in the RCPB and in contact with the reactor coolant, contaminants, or radiolytic products must be compatible and must meet GDC 4, "Environmental and Dynamic Effects Design Bases," as it relates to compatibility of components with environmental conditions. The requirements of GDC 4 are met by compliance with the applicable provisions of the ASME Code and with the positions of RG 1.44, "Control of the Use of Sensitized Stainless Steel."

The staff reviewed DCD Tier 2 Section 5.2.3.2, "Compatibility with Reactor Coolant," to determine the compatibility of the RCPB components with various environments. The AP1000 design conforms with the guidance provided in RG 1.44, "Control of the Use of Sensitized Stainless Steel." In addition, ferritic low-alloy and carbon steels used in principal pressure-retaining components are clad with a layer of austenitic stainless steel.

The staff noted the discussion of safe-ends in DCD Tier 2 Section 5.2.3.2.2, "Compatibility of Construction Materials with Reactor Coolant." The staff requested, in RAI 252.002, that the applicant discuss the purpose of the safe-ends and the concern that if the purpose of the safe-ends is to protect the austenitic stainless steel from sensitization, then the A-8 weld (which is also austenitic stainless steel) may become sensitized during postweld heat treatment of the component at 593.3°C (1100°F). The applicant stated in its response that the purpose of the safe-ends is to protect the austenitic stainless steel from being heat treated during field installation, which may cause sensitization. The applicant further elaborated that based on experience with the safe-ends on current reactors, postweld heat treatment of the safe-ends at the fabrication shop does not cause a sensitization concern.

The staff reviewed this response and determined that it is not entirely acceptable since the A-8 welds may include austenitic stainless steels such as Types 304 and 316 that may become sensitized during postweld heat treatment. The staff reviewed Revision 4 of the DCD in which the applicant further clarified the purpose of the safe-ends and removed references to A-8 welds. The staff finds this clarification acceptable.

The staff finds that the materials for the AP1000 design are compatible with the reactor coolant and meet GDC-4 since they meet the guidance provided in RG 1.44 and provides for corrosion resistance of ferritic low alloy steel and carbon steel components through the use of austenitic stainless steel cladding.

5.2.3.4 Fabrication and Processing of Ferritic Materials

The fracture toughness properties of the ferritic materials in the RCPB must meet the requirements of 10 CFR Part 50, Appendix G, “Fracture Toughness Requirements,” GDC-1, “Quality Standards and Records,” as it relates to non-destructive testing (i.e., examination) to quality standards, GDC 14, “Reactor Coolant Pressure Boundary,” as it relates to extremely low probability of rapidly propagating fracture and gross rupture of the RCPB and GDC 31, “Fracture Prevention of Reactor Coolant Pressure Boundary,” as it relates to nonbrittle behavior of materials and the probability of rapidly propagating fracture being minimized.

The fracture toughness requirements of GDC-14 and GDC-31 are met through compliance with the acceptance standards in Article NB-2300 of the ASME Code, Section III and Appendix G, Article G-2000 of the ASME Code. The acceptance criteria for control of ferritic steel welding are met through compliance with the applicable provisions of the ASME Code and with positions in RG 1.34, “Control of Electroslag Weld Properties;” RG 1.43, “Control of Stainless Steel Weld Cladding of Low-Alloy Steel Components;” RG 1.50, “Control of Preheat Temperature for Welding of Low Alloy Steel;” and RG 1.71, “Welder Qualification for Areas of Limited Accessibility.”

The non-destructive examination (NDE) requirements of GDC-1 for the examination of ferritic components are met through compliance with the ASME Code, Section III, Subarticle NB-2500.

The staff reviewed DCD Tier 2 Section 5.2.3.3, “Fabrication and Processing of Ferritic Materials,” to ensure that the RCPB components satisfy the requirements regarding prevention of RCPB fracture, control of welding, and NDE.

The AP1000 design conforms with ASME Code, Section III, Subarticle NB-2300, to meet fracture toughness requirements in 10 CFR Part 50, Appendix G, and the following RGs to meet the controls for welding and material preservation:

- RG 1.34, “Control of Electroslag Weld Properties;”
- RG 1.43, “Control of Stainless Steel Weld Cladding of Low-Alloy Steel Components;”
- RG 1.50, “Control of Preheat Temperature for Welding of Low-Alloy Steel;” and
- RG 1.71, “Welder Qualification for Areas of Limited Accessibility,” with an acceptable alternative as discussed in this section below.

The pressure-retaining components of the RCPB that are made of ferritic materials are required, by 10 CFR Part 50, Appendix G, to meet the requirements for fracture toughness during system hydrostatic tests, and any condition of normal operation, including anticipated operational occurrences. For piping, pumps, and valves, this requirement is met through compliance with the requirements of the ASME Code, Section III, Paragraph NB-2331 or NB-2332, and the C_V values specified in Table NB-2332(a)-1. The AP1000 design complies with these Code requirements and therefore, satisfies the requirements of 10 CFR 50, Appendix G.

In addition, the AP1000 design meets the requirements of GDC-1 for NDE through its compliance with the ASME Code, Section III, 1998 Edition, 2000 addenda, as discussed in the DCD Tier 2 Section 5.2.1.1, "Compliance with 10 CFR 50.55a."

The staff noted the discussion of welding material control in DCD Tier 2 Section 5.2.3.3.2, "Control of Welding." The staff requested, in RAI 252.003, the applicant confirm that the storage and handling of the welding materials is also covered by ASME Code Section III, Subarticle NB-4400. In its response, the applicant responded that the requirements of NB-4400 will be implemented in the fabrication and installation of components. In addition, DCD Tier 2 Section 5.2.3.3.2, "Control of Welding," will be modified to include ASME Code, Section III, Subarticle NB-4400. The staff reviewed Revision 4 of the DCD and determined that it acceptably addresses this issue because the AP1000 design meets the appropriate ASME Code requirement for control of welding material.

The staff noted in DCD Tier 2 Appendix 1A, "Compliance with Regulatory Guides," the applicant states that the AP1000 design takes exception to RG 1.71, "Welder Qualification for Areas of Limited Accessability." Specifically, the AP1000 design does not require qualification or requalification of welders for areas of limited accessibility consistent with current practice as recommended in RG 1.71. The staff requested, in RAI 252.005, the applicant discuss, for welds which are not volumetrically examined, how the AP1000 design ensures that welds made in areas of limited accessibility and/or visibility will meet the fabrication requirements of ASME Section III. In its response, the applicant stated that, based on experiences in the fabrication of RCPB components, accessibility and visibility of welds which require only surface examination will not need welders qualified to RG 1.71 requirements. The applicant elaborated that all welds fabricated in shop can be set up using a mechanical positioner and that various tools are available to support the required inspection.

The staff reviewed this response and determined that it is acceptable because the AP1000 design includes ASME Code, Section III welder qualification requirements for the inspection of weld joints normally requiring a surface examination.

The staff finds that the AP1000 design meets Appendix G of 10 CFR Part 50, GDC-1, GDC-14, and GDC-31 because it includes appropriate controls for the fabrication and processing of ferritic materials to ensure fracture toughness of the RCPB components, control of welding, and NDE commensurate with the safety function of the RCPB.

5.2.3.5 Fabrication and Processing of Austenitic Stainless Steel

Process control techniques must be included during all stages of component manufacturing and reactor construction to meet GDC-1, "Quality Standards and Records," as it relates to non-destructive testing (i.e., examination) to quality standards, GDC 4, "Environmental and Dynamic Effects Design Bases," and 10 CFR Part 50, Appendix B, Criterion XIII, "Handling, Storing, and Shipping," by preventing severe sensitization of the material, by minimizing exposure of the stainless steel to contaminants that could lead to stress corrosion cracking, and by reducing the likelihood of component degradation or failure through contaminants.

The requirements of GDC 4 and 10 CFR Part 50, Appendix B, Criterion XIII, are met through compliance with the applicable provisions of the ASME Code and with the positions in RG 1.31, "Control of Ferrite Content in Stainless Steel Weld Metal;" RG 1.34, "Control of Electroslag Weld Properties;" RG 1.36, "Nonmetallic Thermal Insulation for Austenitic Stainless Steel;" RG 1.37, "Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water Cooled Nuclear Power Plants;" RG 1.44, "Control of the Use of Sensitized Stainless Steel;" and RG 1.71, "Welded Qualification for Areas of Limited Accessibility."

The NDE requirements of GDC-1 for the examination of austenitic components are met through compliance with the ASME Code, Section III, Paragraphs NB-2550 through NB-2570.

The staff reviewed DCD Tier 2 Section 5.2.3.4, "Fabrication and Processing of Austenitic Stainless Steel," to ensure that austenitic stainless steel RCPB components are compatible with environmental conditions to avoid sensitization and SCC, are compatible with thermal insulation, have appropriate controls on welding and material preservation, and have appropriate NDE.

The AP1000 design conforms with ASME Code, Section II, for the final heat-treatment of austenitic stainless steels; ASTM A 262, Practice A or E for materials testing; and the following guidance to meet the controls for welding and material preservation in conjunction with ASME Code, Section III:

- RG 1.31, "Control of Ferrite Content in Stainless Steel Weld Metal,"
- WCAP-8324-A, "Control of Delta Ferrite in Austenitic Stainless Steel Weldments," for δ-ferrite verification as alternative to RG 1.31, "Control of Ferrite Content in Stainless Steel Weld Metal,"
- RG 1.34, "Control of Electroslag Weld Properties,"
- RG 1.36, "Nonmetallic Thermal Insulation for Austenitic Stainless Steel,"
- RG 1.37, "Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water Cooled Nuclear Power Plants," with an acceptable exception in Section 17.3 of this SER,
- RG 1.44, "Control of the Use of Sensitized Stainless Steel," and
- RG 1.71, "Welder Qualification for Areas of Limited Accessibility," with an acceptable alternative as discussed in Section 5.2.3.4 of this SER.

The thermal insulation used in the AP1000 design of the RCPB is acceptable since it conforms with the guidance in RG 1.36 for nonmetallic insulation with respect to acceptable levels of leachable contaminants in these materials.

The AP1000 design takes an exception to quality standard ANSI N.45.2.1-1973 referenced in RG 1.37. The discussion of quality assurance documents is found in Section 17.3, "Quality Assurance During Design, Procurement, Fabrication, Inspection and/or Testing of Nuclear Plant Items," of this SER.

The AP1000 design meets the requirements of GDC-1 for NDE through its compliance with the ASME Code, Section III, 1998 Edition, 2000 Addenda, as discussed in the DCD Tier 2 Section 5.2.1.1, "Compliance with 10 CFR 50.55a."

The staff noted the discussion of the use of welding material that is not fully austenitic in DCD Tier 2 Section 5.2.3.4.6, "Control of Welding." The staff requested, in RAI 252.004, the applicant list the exact materials exempted from the delta ferrite requirement for the AP1000 design and the exact materials considered to be fully austenitic for welding applications. In its response, the applicant responded that the phrase “fully austenitic welding materials” refers to the regular austenitic stainless steel welding materials such as Types 308 and 309. In addition, these austenitic stainless steel welding materials do include ferrite and the DCD will require a minimum ferrite content of 5 FN.

The staff reviewed this response and determined that it is acceptable because the AP1000 design will specify the minimum ferrite content of 5 FN for fully austenitic welding materials as recommended in RG 1.31, “Control of Ferrite Content in Stainless Steel Weld Metal.” The staff reviewed Revision 4 of the DCD; specifically, Tier 2 of the DCD, Table 5.2-1, “Reactor Coolant Pressure Boundary Materials Specifications,” and found the specifications acceptable in addressing the staff’s concerns.

The staff concludes that the fabrication and processing of RCPB austenitic stainless steel meets GDC-1, GDC-4, and 10 CFR Part 50, Appendix B, Criterion XIII because it conforms with the applicable provisions of the ASME Code and the positions in or acceptable alternatives to RGs 1.31, 1.34, 1.36, 1.37, 1.44, and 1.71.

5.2.3.6 Conclusion

The staff concludes, with the exception of Open Item 5.2.3-1, that the design of the RCPB materials is acceptable and meets the requirements of GDC 1, 4, 14, 30, and 31; the requirements of 10 CFR Part 50, Appendices B and G; and the requirements of 10 CFR 50.55a.

5.2.4 RCS Pressure Boundary Inservice Inspection and Testing

The staff reviewed DCD Tier 2 Section 5.2.4, “Inservice Inspection and Testing of Class 1 Components,” in accordance with Section 5.2.4, “Reactor Coolant Pressure Boundary (RCPB) Inservice Inspection and Testing,” of the SRP. The requirements for periodic inspection and testing of the RCPB are acceptable if the inspection and test program satisfy Appendix A of 10 CFR Part 50, GDC 32 and meet 10 CFR 50.55a, "Codes and Standards.”

10 CFR 50.55a requires, in part, that ASME Code Class 1 components be designed and provided with access to enable the performance of inservice examination of such components and meet the preservice examination requirements set forth in Section XI of the ASME Code applied to the construction of the particular component.

10 CFR Part 50, Appendix A, GDC-32 requires, in part, that components that are part of the RCPB shall be designed to permit periodic inspection and testing of important areas and features to assess their structural and leaktight integrity.

Compliance with the preservice and inservice examinations of 10 CFR 50.55a, as detailed in Section XI of the ASME Code, constitutes an acceptable basis for satisfying the periodic inspection and testing requirements of GDC 32, “Inspection of reactor coolant pressure boundary.”

The staff evaluation of the inservice inspection and testing program for Class 1 components is divided into six sections, as described in the SRP. The six sections are: system boundary subject to inspection; accessibility; examination categories and methods; inspection intervals; evaluation of examination results; and system leakage and hydrostatic pressure tests. The acceptability of these elements is discussed in DSER Sections 5.2.4.2 through 5.2.4.7 respectively.

5.2.4.1 Summary of Technical Information

The DCD, for inservice inspection and testing of Class 1 components, states that preservice and inservice inspection and testing of ASME Code Class 1 pressure-retaining components (including vessels, piping, pumps, valves, bolting, and supports) within the RCPB will be performed in accordance with Section XI of the ASME Code including addenda according to 10 CFR 50.55a(g). This includes all ASME Code Section XI mandatory appendices.

The specific edition and addenda of the Code used to determine the requirements for the inspection and testing plan for the initial and subsequent inspection intervals is to be delineated in the inspection program. The Code includes requirements for system pressure tests and functional tests for active components. The requirements for system pressure tests and visual examinations are defined in Section XI, IWA-5000. These tests verify pressure boundary integrity in conjunction with inservice inspection.

DCD Tier 2 Section 3.9.6, “Inservice Testing of Pumps and Valves,” discusses the inservice functional testing of valves for operational readiness. Since none of the pumps in the AP1000 are required to perform an active safety function, the operational readiness test program for pumps is controlled administratively. This is evaluated in Section 3.9.6 of this SER.

In conformance with ASME Code and NRC requirements, the preparation of inspection and testing programs is the responsibility of the combined license applicant of each AP1000. DCD Tier 2 Section 5.2.4 indicates that these programs will comply with applicable inservice inspection provisions of 10 CFR 50.55a(2). However, the correct reference is 10 CFR 50.55a(b)(2). The applicant needs to correct this reference in the DCD. This is Confirmatory Item 5.2.4-1.

5.2.4.2 System Boundary Subject to Inspection

Consistent with the SRP, the applicant's definition of the RCPB is acceptable if it includes all pressure vessels, piping, pumps, and valves which are part of the reactor coolant system, or connected to the reactor coolant system, up to and including:

- The outermost containment isolation valve in system piping that penetrates the primary reactor containment.
- The second of two valves normally closed during normal reactor operation in system piping that does not penetrate primary reactor containment.
- The reactor coolant system safety and relief valves.

DCD Tier 2 Section 5.2.4.1 indicates Class 1 pressure-retaining components and their specific boundaries are included in the equipment designation list and the line designation list. Both of these lists are contained in the inspection program. ASME Code Class 1 components are designated AP1000 equipment Class A. The system boundary for pressure-retaining components is discussed in DCD Tier 2 Section 3.2.2, “AP1000 Classification System.” The applicant’s definition of the RCPB is consistent with SRP and is therefore acceptable.

5.2.4.3 Accessibility

The design and arrangement of system components are acceptable if adequate clearance is provided in accordance with Section XI, Subarticle IWA-1500, "Accessibility," of the ASME Code.

Accessibility for inspection is described in DCD Tier 2 Section 5.2.4.2. ASME Code Class 1 components are designed so that access is provided in the installed condition for visual, surface, and volumetric examination specified by the baseline ASME Code Section XI (1998 Edition, 2000 Addenda) and mandatory appendices. Design provisions, in accordance with Section XI, Subarticle IWA-1500, are incorporated in the design process for Class 1 components. Accessibility is acceptable because the AP1000 design incorporates the requirements of Subarticle IWA-1500.

5.2.4.4 Examination Categories and Methods

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

The applicant's examination techniques and procedures used for preservice inspection or inservice inspection of the system are acceptable if in agreement with the following criteria:

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

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

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

Examination techniques, categories and methods are discussed in DCD Tier 2 Sections 5.2.4.3, “Examination Techniques and Procedures,” and DCD Tier 2 Section 5.2.4.5, “Examination Categories and Requirements.” The visual, surface, and volumetric examination techniques and procedures agree with the requirements of Subarticle IWA-2200 and Table IWB-2500-1 of the ASME Code, Section XI. Examination categories and requirements are established according to Subarticle IWB-2500 and Table IWB-2500-1 of the ASME Code, Section XI. Qualification of the nondestructive examination personnel is in compliance with Subarticle IWA-2300 of the ASME Code, Section XI. The liquid penetrant method or the magnetic particle method is used for surface examinations. Radiography, ultrasonic, or eddy current techniques (manual or remote) are used for volumetric examinations. DCD Tier 2 Section 5.2.1.1 indicates the baseline used for the evaluation done to support the safety analysis report and the design certification is the 1998 Edition, 2000 Addenda of the ASME Code, Section XI. This edition and addenda of ASME Code, Section XI requires the implementation of Appendix VII for qualification of nondestructive examination personnel for ultrasonic examination and the implementation of Appendix VIII for performance demonstration for ultrasonic examination of reactor pressure boundary piping, RV welds and RV head bolts. Since the examination methods and categories applied to Class 1 components will be in accordance with the requirements of the ASME Code, Section XI, as discussed above, the examination categories and methods for the AP1000 for Class 1 components are acceptable.

In response to RAI 250.001, the applicant indicated that both the pressurizer and steam generator nozzle inside radius volumes are inspectable. The pressurizer inside radius volumes are intended to be examined from the outside diameter surface. The steam generator inside radius volumes are intended to be examined either from the outside or inside surfaces. This accessibility is therefore acceptable.

5.2.4.5 Inspection Intervals

The required examinations and pressure tests must be completed during each 10-year interval of service, hereinafter designated as the inspection interval. In addition, the scheduling of the program must comply with the provisions of Article IWA-2000, "Examination and Inspection," concerning inspection intervals of the ASME Code, Section XI.

Inspection intervals are discussed in DCD Tier 2 Section 5.2.4.4, “Inspections.” Inspection intervals are defined in Subarticles IWA-2400 and IWB-2400 of the ASME Code, Section XI. The inspection interval specified for the AP1000 Class 1 components are consistent with the definitions in Section XI of the ASME Code and are therefore acceptable.

5.2.4.6 Evaluation of Examination Results

- The standards for examination evaluation in the program for flaw evaluation are acceptable if in agreement with the requirements of ASME Code, Section XI, Article IWB-3000, "Acceptance Standards."

- The proposed program regarding repairs of unacceptable indications or replacement of components containing unacceptable indications is acceptable if in agreement with the requirements of ASME Code, Section XI, Article IWA-4000, "Repair/Replacement

Activities." The criteria that establish the need for repair or replacement are described in ASME Code, Section XI, Article IWB-3000, "Acceptance Standards."

Evaluation of examination results is discussed in DCD Tier 2 Section 5.2.4.6, “Evaluation of Examination Results.” Examination results are evaluated according to ASME Code, Section XI, IWA-3000 and IWB-3000, with flaw indications being evaluated according to IWB-3400 and Table IWB-3410-1. Repair procedures, if required, are according to the ASME Code, Section XI. Based on this method of evaluating examination results and the use of ASME Code rules for repair, the evaluation of examination results for AP1000 Class 1 components is acceptable.

5.2.4.7 System Leakage and Hydrostatic Pressure Tests

The pressure-retaining Code Class 1 component leakage and hydrostatic pressure test program is acceptable if the program agrees with the requirements of Section XI, Article IWB-5000, "System Pressure Tests," and the TS requirements for operating limitations during heatup, cooldown, and system hydrostatic pressure testing. In some cases, the TS limitations may be more severe than those in Article IWB-5000.

System leakage and hydrostatic pressure tests are discussed in DCD Tier 2 Section 5.2.4.7. System pressure tests will comply with IWA-5000 and IWB-5000 of the ASME Code, Section XI. Based on this method of performing pressure tests, the system leakage and hydrostatic pressure test for AP1000 Class 1 components is acceptable.

5.2.4.8 Conclusion

Based on the staff evaluation of the system boundary subject to inspection, accessibility, examination categories and methods, inspection intervals, evaluation of examination results, and system leakage and hydrostatic pressure tests, the staff concludes that the periodic inspection and testing of the RCPB are acceptable and the inspection and test program satisfy General Design Criterion 32, “Inspection of Reactor Coolant Pressure Boundary,” because they meet the applicable requirements of the ASME Code, Section XI, as endorsed in Section 50.55a to 10 CFR Part 50.

5.2.5 Reactor Coolant Pressure Boundary Leakage Detection

The staff reviewed the AP1000 design as it relates to its capability to detect and, to the extent practical, identify the source of RCPB leakage. The staff reviewed the RCPB leakage detection design in accordance with the guidelines provided in SRP Section 5.2.5. Staff acceptance of the leakage detection design is on the basis of the design meeting the requirements of GDC 2, "Design Basis for Protection Against Natural Phenomena," as it relates to the capability of the design to maintain and perform its safety function following an earthquake, and on the design meeting the requirements of GDC 30, "Quality of Reactor Coolant Pressure Boundary," as it relates to the detection, identification, and monitoring of the source of reactor coolant leakage. Conformance with GDC 2 is on the basis of the leakage detection design meeting the guidelines of RG 1.29, "Seismic Design Classification," Positions C.1 and C.2. Conformance with GDC 30 is on the basis of the leakage detection design meeting the guidelines of RG 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems," Positions C.1 through C.9. Leakage detection monitoring is also maintained in support of LBB criteria for high-energy fluid piping in containment. DCD Tier 2 Section 3.6.3 and Section 3.6.3 of this report addresses the application of LBB criteria.

The staff also reviewed the RCPB leakage detection design for compliance with the requirements of the TMI issue designated by 10 CFR 50.34(f)(2)(xxvi). The TMI issue states that applicants should provide for leakage control and detection in the design of systems outside containment that contain (or might contain) TID-14844 source term radioactive materials following an accident.

RCPB leakage detection is accomplished using instrumentation and other components of several systems. Diverse measurement methods including level, flow, and radioactivity measurements are used for leakage detection. The equipment classification for each of the systems and components used for leakage detection is generally determined by the requirements and functions of the system in which it is located. There is no requirement that leakage detection and monitoring equipment be safety-related.

RCPB leakage is classified as either identified or unidentified leakage. Identified leakage includes (1) leakage from closed systems such as RV seal or valve leakage that is captured and conducted to a collecting tank, and (2) intersystem leakage into auxiliary systems and secondary systems. (Intersystem leakage must be considered in the evaluation of the reactor coolant inventory balance.) Other leakage is unidentified leakage.

5.2.5.1 Identified Leakage Detection

Sources of identified leakage in containment include leaks from the RV head flange, pressurizer safety relief valves, and automatic depressurization valves. In the course of plant operations, various minor leaks of the RCPB may be detected by operating personnel. If these leaks can be subsequently observed, quantified, and routed to the containment sump, this leakage will be considered identified leakage.

Identified leakage other than intersystem leakage is collected in a closed reactor coolant drain tank (RCDT) located in the reactor cavity in containment. The RCDT vent is piped to the gaseous radwaste system to prevent release of radioactive gas to the containment atmosphere. Leakage detection alarms and indications are provided in the main control room (MCR). The RCDT, pumps, and sensors are part of the liquid radwaste system.

5.2.5.2 Intersystem Leakage Detection

DCD Tier 2 Section 5.2.5.2 states that possible intersystem leakage points across passive barriers or valves and their detection methods were considered. Auxiliary systems connected to the RCPB incorporate design and administrative provisions that limit leakage. Such leakage is detected by increasing auxiliary system level, temperature, flow, or pressure; by lifting relief valves; or increasing values of monitored radiation in the auxiliary system. The normal RNS and the CVS have the potential for intersystem leakage past closed valves.

An important potentially identifiable leakage path for reactor coolant is through the SG tubes into the secondary side of the SG. Identified leakage from the SG primary side is detected by one or a combination of the following methods:

- the condenser air removal radiation monitor
- the SG blowdown radiation monitor
- the main steamline radiation monitor
- the laboratory analysis of condensate

In addition, leakage from the RCS to the CCS is detected by the CCS radiation monitor, by increasing surge tank level, by high flow downstream of selected components, or by some combination of the preceding.

5.2.5.3 Unidentified Leakage Detection

DCD Tier 2 Section 5.2.5.3 states that to detect unidentified leakage in containment, three diverse methods may be utilized to quantify and assist in locating the leakage, including the following:

- containment sump level
- RCS inventory balance
- containment atmosphere radiation

In addition, other supplemental methods utilize containment atmosphere pressure, temperature, humidity, and visual inspection.

Position C.1 of RG 1.29 states that the SSCs listed in the RG, including their foundations and supports, should be designated as seismic Category I to ensure that they can withstand the effects of a SSE and remain functional. DCD Tier 2 Section 5.2.5.4 states that the containment sump level monitor and the containment atmosphere radiation monitor are classified as seismic Category I.

Position C.2 of RG 1.29 states that those parts of SSCs, whose continued function is not required but whose failure could reduce the functioning of any plant feature (identified in Position C.1) to an unacceptable safety level, or could result in an incapacitating injury to occupants of the MCR, should be designed and constructed so that an SSE would not cause such a failure. DCD Tier 2 Section 5.2.5 states that equipment classification for each of the systems and components used for leakage detection is generally determined by the requirements and functions of the system in which it is located. There is no requirement that leakage detection and monitoring equipment be safety-related.

On the basis of the above, the staff concludes that the design of systems and components used for leakage detection meets the guidelines of RG 1.29, Positions C.1 and C.2. Therefore, the design meets the requirements of GDC 2, as it relates to the capability of the systems and components to maintain and perform their safety function following an earthquake.

Position C.1 of RG 1.45 states that leakage to containment from identified sources should be collected or isolated so that flow rates are monitored separately from unidentified leakage and so that the total flow rate can be established and monitored. As stated in Section 5.2.5.1 above, identified leakage is monitored separately for the RV head flange, pressurizer safety relief valves, and automatic depressurization valves.

Position C.2 of RG 1.45 states that leakage to containment from unidentified sources should be collected and the flow rate monitored with an accuracy of 3.79 liters/min (1 gpm) or better. DCD Tier 2 Section 5.2.5.3 states that the sensitivity of leakage detection monitoring is such that the containment sump level monitoring can detect a change of 1.89 liters/min (0.5 gpm) in 1 hour.

Position C.3 of RG 1.45 states that at least three separate methods should be used for leakage detection. Two of these methods should include (1) sump level and flow monitoring and, (2) airborne particulate radioactivity monitoring. The third method may be selected from monitoring either (1) condensate flow from the containment air coolers or, (2) containment airborne gaseous activity. DCD Tier 2 Section 5.2.5.3 states that containment sump level monitoring, containment atmosphere radiation monitoring, and RCS inventory balance are utilized in the AP1000 design to detect and monitor leakage in containment. In particular, the applicant selected the gaseous N_{13}/F_{18} monitor for containment atmosphere radiation monitoring. No credit is taken for airborne particulate radioactivity monitoring. DCD Tier 2 Section 5.2.5.3 states that humidity, temperature, and pressure monitoring are also used for alarms and indirect indication of possible leakage in containment.

Position C.4 of RG 1.45 states that provisions should be made to monitor the systems connected to the RCPB for indications of intersystem leakage. Methods should include radioactivity monitoring and indicators to show abnormal water levels or flow in the affected systems. DCD Tier 2 Section 5.2.5.2 states that associated systems and components connected to the RCS have intersystem leakage monitoring devices. SG tube leakage is detected by the condenser air removal radiation monitor, the SG blowdown radiation monitor, the main steamline radiation monitor, or laboratory analysis of condensate. Leakage from the RCS to the CCS is detected by CCS radiation monitors, by increasing surge tank level, by high flow downstream of selected components, or by some combination of the preceding.

Position C.5 of RG 1.45 states that the sensitivity and response time of each method used to detect and monitor unidentified leakage in containment should be a minimum of 3.79 liters/min (1 gpm) in less than 1 hour. In DCD Tier 2 Section 5.2.5.3.3, the applicant states that the N_{13}/F_{18} radioactivity monitor can detect a 1.89 liters/min (0.5 gpm) within 1 hour when the plant is at full power. The monitor is operable when the plant is above 20-percent power.

Position C.6 of RG 1.45 states that the LDSs should be capable of performing their functions during and following an SSE. DCD Tier 2 Section 5.2.5.4 states that the containment sump level monitor and the containment atmosphere radiation monitor are classified as seismic Category I. Containment activity is monitored by the containment high-range radiation monitor, which is seismically qualified.

Position C.7 of RG 1.45 states that indicators and alarms for each LDS should be provided in the MCR. In addition, procedures for converting indications to a common leakage equivalent should be available to the operators. DCD Tier 2 Section 5.2.5.6 lists the alarms and/or indications for RCPB leakage provided in the MCR. The plant instrumentation system is a microprocessor-based system that accepts inputs from all RCPB leakage detection sensors and monitors. The containment sump level, containment atmosphere radioactivity, RCS inventory balance, and the flow measurements are provided as gallon per minute leakage equivalent.

Position C.8 of RG 1.45 states that the LDSs should be equipped with provisions for operability testing and calibration during plant operation. DCD Tier 2 Section 5.2.5.5 states that periodic testing of the leakage detection monitors verifies the operability and sensitivity of detection equipment. These tests include calibrations and alignments during installation, periodic channel calibrations, functional tests, and channel checks. The instrumentation for RCPB leakage detection can be tested for operability during plant operation.

Position C.9 of RG 1.45 states that the TS should include limits for both identified and unidentified leakage, and should address the availability of various instruments to assure coverage at all times. DCD Tier 2 Chapter 16 TS defines the operability requirements for the RCS leakage detection instrumentation. The instrumentation is designed to verify its operability at all times. Should a detector fail (e.g., signal outside the calibrated range or self-monitored trouble is detected), the plant instrumentation system will initiate a trouble alarm in the MCR, indicating that the readout of a specific monitor is questionable.

The staff compared AP1000 TS 3.4.8, "RCS Operational Leakage," and 3.4.10, "RCS Leakage Detection Instrumentation," with the the applicant Operating Group Standard TS (WOG STS) 3.4.13 and 3.4.15. AP1000 TS 3.4.10 requires that (a) one containment sump level channel and (b) one containment atmosphere radioactivity monitor to be operable for Modes 1, 2, 3, and 4. However, there are two notes, associated with this TS, allowing these two leakage detection instrumentation systems to not be required during certain conditions. The first note states that the containment atmosphere radioactivity monitor is only required to be Operable in Mode 1 with RTP [rated thermal power] > 20 percent. The second note states that containment sump level measurements cannot be used for leak detection if leakage is prevented from draining to the sump, such as by redirection to the in-containment refueling water storage tank (IRWST) by the containment shell gutter drains. In RAI 410.006, the staff requested an explanation that during Modes 1, 2, 3, and 4, if both notes are satisfied, what compensatory actions will be required to perform the function of RCS leakage detection.

In response to RAI 410.006, the applicant stated that when the conditions in both notes are satisfied, there are compensatory actions required for RCS leakage detection. The containment atmosphere radioactivity monitor is not required to be operable any time plant power is less than 20 percent of rated thermal power, and there are no additional compensatory leakage monitoring actions required when this instrument is not required to be operable. However, the containment sump level instrument is required to be operable in Modes 1, 2, 3, and 4 to provide RCS leakage detection, whether the containment radioactivity monitor is required or not.

The second note for the sump instrument does not eliminate the operability requirements for at least one containment sump level instrument channel in Modes 1, 2, 3, and 4 when the gutter drains are closed. The second note is intended to inform the operator that although the sump level instrument(s) may be operational, if the drain path for the containment shell gutter to the containment sump is closed, then the sump level measurement cannot perform its leak detection function. No condensate can return to the containment sump when the drain path is closed. Instead, the condensate will return to the IRWST. Condensate is able to drain to the sump as long as both series drain path isolation valves are open.

In the case with the drain path closed, the containment sump level instruments do not meet the TS definition of operable. Therefore, when the drain path is closed, both channels are inoperable (even though both may be operating) and Condition A for LCO 3.4.10 must be entered. The compensatory action is to perform SR 3.4.8.1 (RCS inventory balance) more frequently, once every 24 hours instead of once every 72 hours. In addition, at least one containment sump channel must be restored to operable status within 72 hours. This means that both gutter drain path isolation valves must be opened. Once both series isolation valves are open, then condensate will drain to the sump and the available containment sump level instrument is considered to be operable. This explanation addressed the staff's concern and identified acceptable compensatory actions. On the basis of the information above, the AP1000 design provides various instruments used to detect and monitor RCPB leakage and the TS assures that leakage detections are available at all times.

On the basis of the information provided by the applicant and evaluated above, the staff concludes that the RCPB leakage detection design conforms to the guidelines of RG 1.45, Positions C.1 through C.9. Therefore, the design meets the requirements of GDC 30 as it relates to the detection, identification, and monitoring of the source of reactor coolant leakage.

The TMI issue designated by 10 CFR 50.34(f)(2)(xxvi) (Item III.D.1.1 of NUREG-0737) states that applicants should provide for leakage control and detection in the design of systems outside of containment that contain (or might contain) total integrated dose (TID)-14844 source term radioactive materials following an accident. Applicants will submit a leakage control program, including an initial test program, a schedule for retesting these systems, and the actions to be taken for minimizing leakage from such systems. The goal is to minimize potential exposures to workers and the public, and to provide reasonable assurance that excessive leakage will not prevent the use of systems needed in an emergency. The applicant has addressed this TMI issue in DCD Tier 2 Section 1.9.3. The DCD states that the safety-related passive systems do not recirculate radioactive fluids outside containment following an accident. A non-safety-related system can be used to recirculate coolant outside of containment following an accident, but this systems is not operated when high containment radiation levels exist. This satisfies the requirements of 10 CFR 50.34(f)(2)(xxvi).

Systems and components utilized for RCPB leakage detection provide reasonable assurance that structural degradation, which may develop in pressure-retaining equipment of the RCPB and result in coolant leakage during service, will be detected on a timely basis. Thus, corrective actions may be taken before such degradation can become sufficiently severe to jeopardize the safety of the equipment, or before the leakage can increase to a level exceeding the capability of the makeup system to replenish the coolant loss.

On the basis of its review of information provided in the DCD, with clarification provided by the specified RAI responses, the staff concludes that the design of the systems and components for RCPB leakage detection is acceptable. The design meets the requirements of GDC 2 with respect to the capability of systems and components to maintain and perform their safety functions in the event of an earthquake, and meets the requirements of GDC 30 with respect to the detection, identification, and monitoring of the source of reactor coolant leakage. This conclusion is made on the basis of the following:

- The AP1000 design has met the requirements of GDC 2 with respect to the capability of systems and components to perform and maintain their safety functions in the event of an earthquake by meeting the guidelines of RG 1.29, Positions C.1 and C.2.

- The AP1000 design has met the requirements of GDC 30 with respect to the detection, identification, and monitoring of the source of reactor coolant leakage by meeting the guidelines of RG 1.45, Positions C.1 through C.9.

- The AP1000 design has met the requirements of 10 CFR 50.34(f)(2)(xxvi) with respect to minimizing leakage from systems outside containment that contain (or might contain) radioactive materials following an accident.

Therefore, the staff concludes that RCPB leakage detection for the AP1000 design conforms to the guidelines of SRP Section 5.2.5, and is acceptable.

5.3 Reactor Vessel

The AP1000 RV is described in DCD Tier 2 Section 5.3.1.2, “Safety Description.” The reactor vessel is cylindrical, with a hemispherical bottom head and a removable, flanged, hemispherical upper head. The vessel contains the core, core support structures, control rods, and other parts directly associated with the core. The vessel interfaces with the reactor internals, the integrated head package, and reactor coolant loop piping, and is supported on the containment building concrete structure.

5.3.1 Reactor Vessel Design

The design of the AP1000 RV closely matches the existing vessel designs of the applicant three-loop plants. New features for the AP1000 have been incorporated without departing from the proven features of existing vessel designs. The RV has inlet and outlet nozzles positioned in two horizontal planes between the upper head flange and the top of the core. The nozzles are located in this configuration to provide an acceptable crossflow velocity in the vessel outlet region and to facilitate optimum layout of the RCS equipment. The inlet and outlet nozzles are offset, with the inlet positioned above the outlet, to allow midloop operation for removal of a main coolant pump without discharge of the core.

Reactor coolant enters the vessel through the inlet nozzles and flows down the core barrel-vessel wall annulus, turns at the bottom, and flows up through the core to the outlet nozzles.

5.3.2 Reactor Vessel Materials

The staff reviewed DCD Tier 2 Section 5.3.2, “Reactor Vessel Materials,” in accordance with NRC SRP 5.3.1,” Reactor Vessel Materials.” The applicant’s RV materials are acceptable if they meet codes and standards and regulatory guidance commensurate with the safety function to be performed so that the relevant requirements of 10 CFR 50.55a, “Codes and Standards;” 10 CFR Part 50, Appendix G, “Fracture Toughness Requirements;” Appendix H, “Reactor Vessel Material Surveillance Program Requirements;” and GDC 1, 4, 14, 30, 31, and 32 are met. These requirements are discussed below.

GDC 1, "Quality Standards and Records," GDC 30, "Quality of Reactor Coolant Pressure Boundary," and 10 CFR 50.55a(a)(1) require that SSCs important to safety shall be designed, fabricated, erected and tested to quality standards commensurate with the importance with the safety function to be performed.

GDC 4, "Environmental and Dynamic Effects Design Bases," requires that SSCs important to safety shall be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents.

GDC 14, "Reactor Coolant Pressure Boundary," requires that the reactor coolant pressure shall be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture.

GDC 31, "Fracture Prevention of Reactor Coolant Pressure Boundary," requires that the RCPB shall be designed with sufficient margin to assure that, when stressed under operation, maintenance, testing, and postulated accident conditions, it will behave in a non-brittle manner and with the probability of rapidly propagating fracture minimized.

GDC 32, " Inspection of Reactor Coolant Pressure Boundary," requires, in part, that the RCPB components shall be designed to permit an appropriate material surveillance program for the reactor pressure vessel.

Appendix G, "Fracture Toughness Requirements," to 10 CFR Part 50 specifies the fracture toughness requirements for ferritic materials of the pressure-retaining components of the RCPB. The staff reviews the RV materials as they relate to the materials testing and acceptance criteria for fracture toughness contained in Appendix G. Pursuant to 10 CFR Part 50, Appendix G, the RV beltline materials must have Charpy upper shelf energy (USE) in the transverse direction for base material and along the weld for weld material of no less than 101.7 n-m (75 ft-lbs) initially and must maintain Charpy USE throughout the life of the vessel of no less than 67.8 n-m (50 ft-lbs).

Appendix H, "Reactor Vessel Material Surveillance Program Requirements," to 10 CFR Part 50 presents the requirements for a materials surveillance program to monitor the changes in fracture toughness properties of materials in the RV beltline region resulting from exposure to neutron irradiation and the thermal environment. These requirements include conformance with American Society of Testing Materials (ASTM) E-185, "Standard Recommended Practices for Surveillance Tests for Nuclear Reactor Vessels." Compliance with Appendix H satisfies the requirements of GDC 32 regarding the provision of an appropriate materials surveillance program for the RV. The staff reviewed the RV materials to determine that they meet the relevant requirements for Appendix H as they relate to the determination and monitoring of fracture toughness.

5.3.2.1 Summary of Technical Information

5.3.2.1.1 Material Specifications

The applicant indicated that the material specifications are in accordance with the ASME Code requirements. All ferritic RV materials comply with the fracture toughness requirements of 10 CFR 50.55a and Appendices G and H of 10 CFR Part 50.

The chemical composition of the ferritic materials of the RV beltline are restricted to maximum limits shown in DCD Tier 2 Table 5.3-1. Copper, nickel, and phosphorus content is restricted to reduce sensitivity to irradiation embrittlement in service.

5.3.2.1.2 Special Processes Used for Manufacturing and Fabrication

The RV is classified as AP1000 Class A. Design and fabrication of the RV is carried out in accordance with ASME Code, Section III, Class 1 requirements. The shell sections, flange, and nozzles are manufactured as forgings. The hemispherical heads are made from dished plates or forgings. The RV parts are joined by welding, using the single or multiple wire submerged arc and the shielded metal arc processes.

5.3.2.1.3 Special Methods for Nondestructive Examination

The NDE of the RV and its appurtenances is conducted in accordance with ASME Code, Section III, requirements; also, numerous examinations are performed in addition to ASME Code, Section III requirements.

5.3.2.1.3.1 Ultrasonic Examination

In addition to the required ASME Code straight beam ultrasonic examination, angle beam inspection over 100 percent of one major surface of plate material is performed during fabrication to detect discontinuities that may be undetected by the straight beam examination.

In addition to the ASME Code, Section III nondestructive examination, full penetration ferritic pressure boundary welds in the RV are ultrasonically examined during fabrication.

After hydrotesting, full penetration ferritic pressure boundary welds in the RV, as well as the nozzle to safe end welds, are ultrasonically examined. These inspections are performed in addition to the ASME Code, Section III, nondestructive examination requirements.

5.3.2.1.3.2 Penetrant Examinations

The partial penetration welds for the control rod drive mechanism head adapters and the top instrumentation tubes are inspected by dye penetrant after the root pass, in addition to ASME Code requirements. Additional information on the control rod drive mechanisms is provided in Section 4.5.1 of this SER.

5.3.2.1.3.3 Magnetic Particle Examination

Magnetic particle examination requirements below are in addition to the magnetic particle examination requirements of Section III of the ASME Code. All magnetic particle examinations of materials and welds are performed in accordance with the following:

- Prior to the final post weld heat treatment, only by the prod, coil, or direct contact method,
- After the final postweld heat treatment, only by the yoke method.

The following surfaces and welds are examined by magnetic particle methods. The acceptance standards are in accordance with Section III of the ASME Code.

- Magnetic particle examination of exterior vessel and head surfaces after the hydrostatic test.
- Magnetic particle examination of exterior closure stud surfaces and all nut surfaces after final machining or rolling.
- Magnetic particle examination of inside diameter surfaces of carbon and low alloy steel products that have their properties enhanced by accelerated cooling.

5.3.2.1.3.4 Weld Examination

Magnetic particle examination of the welds attaching the closure head lifting lugs and refueling seal ledge to the RV after the first layer and each 1.27 cm (0.5 in.) of weld metal is deposited. All pressure boundary welds are examined after back-chipping or back-grinding operations.

5.3.2.1.4 Special Controls for Ferritic and Austenitic Stainless Steels

Welding of ferritic steels and austenitic stainless steels is discussed in DCD Tier 2 Section 5.2.3, “Reactor Coolant Pressure Boundary Materials.”

5.3.2.1.5 Fracture Toughness

Assurance of adequate fracture toughness of ferritic materials in the RV is provided by compliance with the requirements for fracture toughness testing included in NB-2300 to Section III of the ASME Code, Appendix G of 10 CFR Part 50, and 10 CFR 50.61.

5.3.2.1.6 Material Surveillance

In the surveillance program, the evaluation of radiation damage is based on pre-irradiation testing of C_V and tensile specimens and post irradiation testing of C_V, tensile, and ½-T compact tension fracture mechanics test specimens. The program is directed toward evaluation of the effect of radiation on the fracture toughness of RV steels based on the transition temperature approach and the fracture mechanics approach. The program conforms to ASTM E-185-82, "Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels," and 10 CFR Part 50, Appendix H.

The RV surveillance program incorporates eight specimen capsules. The eight capsules contain 72 tensile specimens, 480 C_V specimens, and 48 compact tension specimens. Archive material sufficient for two additional capsules and HAZ materials will be retained. The applicant's program schedule for removal of the capsules for post-irradiation testing includes 5 capsules to be withdrawn which is in accordance with ASTM E-185-82 and Appendix H of 10 CFR Part 50.

5.3.2.1.7 Reactor Vessel Fasteners

The RV closure studs, nuts, and washers are designed and fabricated in accordance with the requirements of ASME Code Section III. The closure studs are fabricated from SA-540. The closure stud material meets the fracture toughness requirements of ASME Code, Section III and 10 CFR Part 50, Appendix G. Conformance with RG 1.65, "Materials and Inspections for Reactor Vessel Closure Studs," is discussed in Section 1.9. Nondestructive examinations are performed in accordance with ASME Code Section III.

5.3.2.2 Staff Evaluation

The staff reviewed DCD Tier 2 Section 5.3.2, "Reactor Vessel Materials," in accordance with Section 5.3.1, "Reactor Vessel Materials," of the SRP.

The NRC staff reviewed the AP1000 RV materials to ensure that the relevant requirements of GDC 1 and 10 CFR 50.55a(a)(1) and GDC 30 have been met as they relate to the material specifications, fabrication, and nondestructive examination to determine their adequacy to assure a quality product commensurate with the importance of the safety function to be performed. The material specifications for the AP1000 design are in accordance with ASME Code, Section III requirements and 10 CFR Part 50, Appendix G. In addition, the design and fabrication of the RV is performed in accordance with ASME Code, Section III, Class I requirements. Furthermore, the RV and its appurtenances are fabricated and installed in accordance with Section III of the ASME Code, Paragraph NB-4100. The nondestructive examination of the RV and its appurtenances is conducted in accordance with ASME Code, Section III requirements. Examination of the RV and its appurtenances by NDE are in compliance with Paragraph NB-5000, for normal methods of examination. The applicant identified other inspections, as stated above, in addition to the ASME Code requirements of NDE, i.e., angle beam inspections and dye penetrant examinations. The staff finds this acceptable because compliance with ASME Code, Section III, and 10 CFR Part 50, Appendix G, constitutes an acceptable basis for satisfying the requirements of GDC 1 and 10 CFR 50.55a(a)(1) and GDC 30 as they relate to the material specifications, fabrication, and nondestructive examination of RV materials.

The staff's evaluation of the welding of ferritic steels and austenitic stainless steels is provided in Section 5.2.3 of this SER, which addresses GDC 4.

The maximum limits for the elements in the materials of the RV beltline are provided in DCD Tier 2 Table 5.3-1. The sulfur and phosphorus content of welds and forgings are limited to a maximum of 0.01 percent. Nickel is limited to 0.85 percent, copper to 0.03 percent, and vanadium to 0.05 percent. Data compiled in EPRI Report NP-933, “Nuclear Pressure Vessel Steel Database,” indicate that this control on the level of material elements will provide the fracture toughness required to ensure the structural integrity of the RV as specified by Appendix G of 10 CFR Part 50. The staff finds this acceptable.

The tests for fracture toughness of RV materials specified in the DCD are in accordance with Paragraph NB-2300 of ASME Code, Section III, and 10 CFR Part 50, Appendix G. The staff confirmed that the applicant’s initial C_V minimum upper shelf fracture energy levels for the RV beltline base metal transverse direction and welds are 75 ft-lbs. DCD Tier 2 Table 5.3-3 indicates that the EOL values for the USE are greater than 50 ft-lbs for the beltline forgings and welds. The staff confirmed this by using the calculations of RG 1.99, Revision 2, “Radiation Embrittlement of Reactor Vessel Materials,” for the beltline forgings and welds. The predicted end-of-life Charpy USE and adjusted reference temperature for the RV materials are calculated in accordance with 10 CFR Part 50, Appendix G. The fracture toughness tests required by the ASME Code and Appendix G to 10 CFR Part 50 provide reasonable assurance that adequate safety margins against the possibility of nonductile behavior or rapidly propagating fracture can be established for all pressure-retaining components of the reactor coolant pressure boundary. This methodology will provide adequate safety margins during operating, testing, maintenance, and postulated accident conditions. Compliance with the provisions of Appendix G of 10 CFR Part 50 satisfies the requirements of GDCs 14 and 31 and 10 CFR 50.55a regarding the prevention of fracture of the reactor coolant pressure boundary. Therefore, the staff finds that the requirements of GDC 14, GDC 31, and 10 CFR 50.55a are adequately met. The staff’s evaluation of compliance with 10 CFR 50.61 (pressurized thermal shock) is provided in Section 5.3.3.2 of this SER.

The design of a RV must take into account the potential embrittlement of RV materials as a consequence of neutron irradiation and the thermal environment. GDC 32 requires, in part, that the RCPB components shall be designed to permit an appropriate material surveillance program for the reactor pressure vessel. The requirements for such a program are provided in Appendix H of 10 CFR Part 50.

The staff requested, in RAI 251.014, that the applicant describe the lead factors for the surveillance capsules. The staff requested that a commitment be made, in the AP1000 DCD, that an analysis will be performed for the COL application with regard to the capsule/holder model, in order to more accurately define the surveillance capsule lead factors and azimuthal locations. In its response to RAI 251.014, the applicant clarified its approach to define the surveillance capsule/holder location. In addition, the applicant revised the DCD to include an analysis that will be performed for the COL application with regard to the capsule/holder model, in order to confirm the proposed surveillance capsule lead factors and azimuthal locations. The staff found this approach acceptable because this analysis would more accurately define the surveillance capsule lead factors and azimuthal locations. This is COL Action Item 5.3.2-1.

To meet the requirements of GDC 32, the AP1000 design includes provisions to monitor changes in the fracture toughness, caused by exposure to neutron radiation of the RV beltline materials by the use of a materials surveillance program. Appendix H to 10 CFR Part 50 requires that the surveillance program for the AP1000 RV meet the recommendations of ASTM E-185. ASTM E-185 was prepared to be applicable to plants designed for a 40-year life, whereas the design life of AP1000 is 60 years. The recommended minimum number of surveillance capsules in ASTM E-185 for a RV with an end-of-life shift between 38°C and 93°C (100°F and 200°F) is four. The AP1000 surveillance capsule program includes eight specimen capsules, with archive materials available for at least two additional complete replacement capsules. The staff verified that the surveillance test materials will be prepared from samples taken from the actual materials used in fabricating the beltline of the RV. In addition, the staff verified that the base metal, weld metal, and HAZ materials included in the program shall be those predicted to be most limiting in regards to setting P-T limits for operation of the reactor to compensate for radiation effects during its lifetime. The staff found that the materials selection, withdrawal and testing requirements for the AP1000 design are in accordance with those recommended in ASTM E-185-82. Compliance with the materials surveillance requirements of Appendix H of 10 CFR Part 50 and ASTM E-185 satisfies the requirements of GDC 32 regarding an appropriate surveillance program for the RV. Thus, the AP1000 design meets GDC 32.

The applicant indicated that the material that the closure studs are fabricated from will meet the fracture toughness requirements of Section III of the ASME Code and Appendix G of 10 CFR Part 50. Nondestructive examination of the studs will be performed according to Section III of the ASME Code, Subarticle NB-2580. In addition, ISI will be performed according to Section XI of the Code, supplemented by paragraphs NB-2545 or NB-2546. The integrity of the AP1000 RV closure studs is assured by conformance with the recommendations of RG 1.65, “Materials and Inspections for Reactor Vessel Closure Studs,” thus satisfying the quality standards requirements of GDC 1 and 30, and 10 CFR 50.55a. Compliance with the recommendations of RG 1.65 also satisfies the prevention of fracture of the reactor coolant pressure boundary requirement of GDC 31, and the requirements of Appendix G of 10 CFR Part 50, as detailed in the provisions of Section III of the ASME Code.

5.3.2.3 Conclusion

The staff concludes that the AP1000 RV material specifications, RV manufacturing and fabrication processes, nondestructive examination methods of the RV and its appurtenances, fracture toughness testing, material surveillance and RV fasteners are acceptable and meet the material testing and monitoring requirements of Section III of the ASME Code, Appendices G and H of 10 CFR Part 50, and 10 CFR 50.55a, which provide an acceptable basis for satisfying the requirements of GDC 1, 14, 30, 31, and 32.

5.3.2.4 Reactor Vessel Material Surveillance Program

Appendix H, “Reactor Vessel Material Surveillance Program Requirements,” to 10 CFR Part 50 presents the requirements for a material surveillance programs for operating reactors. The purpose of the material surveillance program is to monitor changes in the fracture toughness properties of ferritic materials in the RV beltline region which result from exposure of these materials to neutron irradiation. Material surveillance is accomplished using surveillance capsules which are holders of archival beltline material and fast neutron ($E > 1.0$ MeV) dosimeters. Measurement of the irradiated material samples yields a measure of the embrittlement and measurement of the dosimeter activation allows the estimation of the irradiation exposure.

Regulatory Guide (RG) 1.190, which is based on GDC 14, 30, and 31, describes methods and practices acceptable to the staff regarding the calculational techniques and statistical practices using the dosimetry measurements. In addition the results of the dosimetry are used to benchmark and validate calculational methods to estimate vessel irradiation.

In the DCD and the response to RAI 440.037, Revision 1 the applicant clarified its methods and practices regarding the calculational techniques and statistical practices using the dosimetry measurements. These methods and practices are consistent with the guidance of RG 1.190. Therefore, the staff concludes that the AP1000 RV material surveillance program is acceptable.

5.3.3 Pressure Temperature Limits

The staff reviewed DCD Tier 2 Section 5.3.3, “Pressure-Temperature Limits,” in accordance with NRC SRP 5.3.2, “Pressure-Temperature Limits and Pressurized Thermal Shock.” The applicant’s pressure temperature (P-T) limit curves are acceptable if they meet codes and standards and regulatory guidance commensurate with the safety function to be performed so that the relevant requirements of 10 CFR 50.55a, Codes and Standards, 10 CFR Part 50, Appendix G, “Fracture Toughness Requirements,” and GDC 1, 14, 31, and 32 are met. These requirements are discussed below.

GDC 1, “Quality Standards and Records,” requires that SSCs important to safety shall be designed, fabricated, erected and tested to quality standards commensurate with the importance with the safety function to be performed.

GDC 14, “Reactor Coolant Pressure Boundary,” requires that the reactor coolant pressure shall be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture.

GDC 31, “Fracture Prevention of Reactor Coolant Pressure Boundary,” requires that the RCPB shall be designed with sufficient margin to assure that, when stressed under operation, maintenance, testing, and postulated accident conditions, it will behave in a non-brittle manner and with the probability of rapidly propagating fracture minimized.

GDC 32, “ Inspection of Reactor Coolant Pressure Boundary,” requires, in part, that the RCPB components shall be designed to permit an appropriate material surveillance program for the reactor pressure vessel.

The NRC has established requirements in 10 CFR Part 50 to protect the integrity of the RCPB in nuclear power plants. The staff evaluates the P-T limit curves based on the following NRC regulations and guidance: 10 CFR Part 50, Appendix G; RG 1.99, Rev. 2, “Radiation Embrittlement of Reactor Vessel Materials,” and SRP Section 5.3.2.

Appendix G to 10 CFR Part 50 requires that P-T limit curves for the RPV be at least as conservative as those obtained by applying the methodology of Appendix G to Section XI of the ASME Code.

RG 1.99, Revision 2, contains methodologies for determining the increase in transition temperature and the decrease in USE resulting from neutron radiation. SRP 5.3.2 provides an acceptable method of determining the P-T limit curves for ferritic materials in the beltline of the RPV based on the linear elastic fracture mechanics methodology of Appendix G to Section XI of the ASME Code. The basic parameter of this methodology is the stress intensity factor K_I, which is a function of the stress state and flaw configuration. Appendix G of the ASME Code requires a safety factor of 2.0 on stress intensities resulting from reactor pressure during normal and transient operating conditions; for hydrostatic testing curves, Appendix G of the ASME Code requires a safety factor of 1.5.

The methods of Appendix G of the ASME Code postulate the existence of a sharp surface flaw in the RPV that is normal to the direction of the maximum stress. This flaw is postulated to have a depth that is equal to 1/4 of the RPV beltline thickness and a length equal to 1.5 times the RPV beltline thickness. The critical locations in the RPV beltline region for calculating heatup and cooldown P-T curves are the 1/4 thickness (1/4T) and 3/4 thickness (3/4T) locations, which correspond to the depth of the maximum postulated flaw, if initiated and grown from the inside and outside surfaces of the RPV, respectively.

Appendix G of the ASME Code, Section XI, methodology requires that applicants determine the adjusted reference temperature (ART or adjusted RT_{NDT}). The ART is defined as the sum of the initial (unirradiated) reference temperature (initial RT_{NDT}), the mean value of the adjustment in reference temperature caused by irradiation (ΔRT_{NDT}), and a margin (M) term.

The ΔRT_{NDT} is a product of a chemistry factor (CF) and a fluence factor. The CF is dependent upon the amount of copper and nickel in the material and may be determined from tables in RG 1.99, Revision 2, or from surveillance data. The fluence factor is dependent upon the neutron fluence at the maximum postulated flaw depth. The margin term is dependent upon whether the initial RT_{NDT} is a plant-specific or a generic value and whether the CF was determined using the tables in RG 1.99, Revision 2, or surveillance data. The margin term is used to account for uncertainties in the values of the initial RT_{NDT}, the copper and nickel contents, the fluence and the calculational procedures. RG 1.99, Rev. 2, describes the methodology to be used in calculating the margin term.

Appendix H, "Reactor Vessel Material Surveillance Program Requirements," to 10 CFR Part 50 presents the requirements for a materials surveillance program to monitor the changes in fracture toughness properties of materials in the RV beltline region resulting from exposure to neutron irradiation and the thermal environment. These requirements include conformance with ASTM E-185-82, "Standard Recommended Practices for Surveillance Tests for Nuclear Reactor Vessels." Compliance with Appendix H satisfies the requirements of GDC 32 regarding the provision of an appropriate materials surveillance program for the RV. The staff reviewed the RV materials to determine that they meet the relevant requirements for Appendix H as they relate to the determination and monitoring of fracture toughness.

5.3.3.1 Summary of Technical Information

The AP1000 DCD for P-T limits indicates that the heatup and cooldown P-T limit curves are required as a means of protecting the RV during startup and shut down to minimize the possibility of fast fracture. The methods outlined in Appendix G of Section XI of the ASME Code are employed in the analysis of protection against nonductile failure. Beltline material properties degrade with radiation exposure, and this degradation is measured in terms of the ART, which includes a reference nil ductility temperature shift, initial RT_{NDT}, and margin.

The predicted ΔRT_{NDT} values are derived considering the effects of fluence and copper and nickel content for the RV steels exposed to the reactor coolant at temperatures between 273.8°C (525°F) to 301.7°C (575°F). RG 1.99, Revision 2, is used in calculating the adjusted reference temperature. The heatup and cooldown curves are developed considering a sufficient magnitude of radiation embrittlement so that no unirradiated ferritic materials in other components of the RCS will be limiting in the analysis.

The applicant stated that the P-T curves are developed considering a radiation embrittlement of up to 54 EFPYs consistent with an expected plant life of 60 years with 90 percent availability. The maximum limits for the copper and nickel elements of the RV are 0.03 percent copper and 0.85 percent nickel. The end of life RT_{NDT} will be determined for as built material. The end of life RT_{PTS} will also be determined for as-built material.

The operating curves are developed in accordance with 10 CFR Part 50, Appendix G, with the exception that the flange requirement is in accordance with WCAP 15315, "Reactor Vessel Closure Head/Vessel Flange Requirements Evaluation for Operating PWR and BWR [boiling water reactor] Plants." The curves are applicable up to 54 EFPYs. In DCD Tier 2 Figures 5.3-2 and 5.3-3, the applicant provided generic curves for the AP1000 RV design, which are limiting curves based on copper and nickel material composition.

The results of the material surveillance program will be used to verify the validity of ΔRT_{NDT} used in the calculation for the development of heatup and cooldown curves. The projected fluence, copper, and nickel contents, along with the RT_{NDT} calculation will be adjusted, if necessary, from time to time using the surveillance capsules.

The applicant also indicated that temperature limits for core operation, inservice leak and hydrotests, are calculated in accordance with the ASME Code, Section XI, Appendix G.

5.3.3.2 Staff Evaluation

The staff reviewed the P-T limits for AP1000 in accordance with Section 5.3.2 of the SRP to assure adequate safety margins of the structural integrity for the ferritic components of the RCPB.

The NRC staff reviewed the P-T limits imposed on the AP1000 RV materials to ensure that the relevant requirements of GDC 1 and 10 CFR 50.55a(a)(1) have been met as they relate to the selection of materials for the RV and their ability to assure adequate safety margins for the structural integrity of the RCPB ferritic components. The SRP indicates that P-T limits established for the RCPB in accordance with Appendix G, 10 CFR Part 50, and Section III of the ASME Boiler and Pressure Vessel Code, Appendix G, ensures that the RCPB material fracture toughness requirements are satisfied. The applicant indicated that the temperature limits for core operation, inservice leak and hydrotests, are calculated in accordance with Appendix G, 10 CFR Part 50, and ASME Code, Section XI, Appendix G. Therefore the staff finds that the applicant's RCPB meets the appropriate quality standards of the ASME Code, and thus the probability of the RCPB material failure and the subsequent effects on reactor core cooling and confinement are minimized; therefore the staff finds that the applicant adequately meets the relevant requirements of GDC 1 and 10 CFR 50.55a(a)(1).

The staff reviewed the P-T limits imposed on the RV to ensure that the materials selected for the RV meet the relevant requirements of GDC 14, in that they possess adequate fracture toughness properties to resist rapidly propagating failure and act in a nonbrittle manner. The applicant indicated that the P-T limit curves will be developed in accordance with the criteria of 10 CFR Part 50, Appendix G, thereby assuring a low probability of significant degradation or gross failure of the RCPB that could cause a loss of reactor coolant inventory and a reduction in the capability to confine fission products.

The staff reviewed the RV materials to ensure that the relevant requirements of GDC 31 have been met as they relate to behavior in a non-brittle manner and an extremely low probability of rapidly propagating fracture. In the DCD, the applicant indicated that RG 1.99, Rev. 2, is used in calculating the adjusted reference temperature. The staff requested, that the applicant discuss the effects of temperature on embrittlement of RV materials if a plant operates at a cold leg temperature below 273.8°C (525°F). The applicant, in its response dated October 18, 2002, indicated that the AP1000 cold leg temperature exceeds 273.8°C (525°F), and that the minimum steady state cold leg temperature is 279.4°C (535°F), which is the value that corresponds to the conditions of 100 percent power, thermal design flow, and 10 percent tube plugging; therefore the procedures of RG 1.99, Revision 2, for nominal embrittlement apply. The staff finds this acceptable because RG 1.99, Revision 2, provides methods for predicting radiation effects on fracture toughness properties that are applicable to compliance with requirements of GDC 31. In addition, the staff reviewed the P-T limits that will be imposed on the RCPB during preservice hydrostatic tests, inservice leak and hydrostatic tests, heatup and cooldown operations, and core operation-criticality, and verified that there will be adequate safety margins against nonductile behavior of rapidly propagating failure of ferritic components as required by GDC 31.

The staff reviewed the RV materials to ensure that the relevant requirements of GDC 32 have been met as they relate to the provision of a materials surveillance program. Compliance with Appendix H satisfies the requirements of GDC 32 regarding the provision of an appropriate materials surveillance program for the RV. The staff reviewed the RV materials to determine that they meet the relevant requirements for Appendix H as they relate to the determination and monitoring of fracture toughness. The staff's review as it relates to the materials surveillance program is provided in Section 5.3.2, "Reactor Vessel Materials," of this report.

The staff requested, in RAI 251.018, that the applicant demonstrate that the P-T limits are in accordance with Appendix G to 10 CFR Part 50. The applicant responded, that the AP1000 heatup and cooldown operating curves were generated using the most limiting adjusted reference temperature values and the NRC-approved methodology as documented in WCAP-14040-NP-A, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves," with staff approved exceptions.

One exception is that instead of using best estimate fluence values, the applicant is using fluence values that are calculated fluence values. The staff finds this acceptable because this is in compliance with RG 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence." The other exception is that the K_{Ic} critical stress intensities are used in place of the K_{Ia} critical stress intensities. This methodology is taken from staff approved ASME Code Case N-641. The staff found the applicant's responses acceptable because the AP1000 P-T limit curves were developed in accordance with 10 CFR Part 50, Appendix G, with the exception that the flange requirement is in accordance with WCAP 15315, "Reactor Vessel Closure Head/Vessel Flange Requirements Evaluation for Operating PWR and BWR Plants." Currently, the staff has not approved WCAP 15315. Any changes to the RV closure head requirements would be incorporated into Appendix G of 10 CFR Part 50. If a relaxation to 10 CFR Part 50, Appendix G is approved, this will allow the operating window to be wider. Since applicants using AP1000 are required to meet the requirements of 10 CFR Part 50, Appendix G, applicants using AP1000 must meet the closure head requirements of Appendix G of 10 CFR Part 50. However, the AP1000 DCD does not provide limitations (values of RT_{NDT}) for the closure flange region of the RV and head. The AP1000 design must include these limitations in order to satisfy Appendix G of 10 CFR Part 50. The applicant should provide these limitations that are consistent with the present TSs and 10 CFR Part 50, Appendix G, or provide closure flange limitations with new TSs that are consistent with 10 CFR Part 50, Appendix G. This is Open Item 5.3.3-1.

The P-T curves for the AP1000 are shown in DCD Tier 2 Figures 5.3-2 and 5.3-3. The applicant revised the DCD to indicate that these curves are generic curves for AP1000 RV design, and they are the limiting curves based on copper and nickel material composition. The applicant also indicated that use of plant-specific curves will be addressed by the COL applicant during procurement of the RV. The applicant also indicated that as noted in the bases to the TS 3.4.15, use of plant-specific curves requires evaluation of the LTOP system. This includes evaluating the setpoint pressure for the normal RHR system relief valve. Since TS will be developed by the applicant and reviewed by the staff, the applicant's LTOP setpoints will be reviewed at a later time.

The staff requested, in RAI 251.017, that the applicant provide details for the P-T limit calculations, including assumptions and margins. In response to RAI 251.017, the applicant provided the staff with details of the P-T limit calculations. The applicant indicated that the methodology of RG 1.99, Revision 2, is used to estimate the shift in reference temperature. The adjusted reference temperature is the sum of the following: the initial reference temperature for the material in the unirradiated condition, the shift in the reference temperature due to the irradiation of the material, and additional safety margins (margin values) to account for uncertainties in the RT_{NDT} measurements and calculation. The applicant indicated that the projected end-of-life fluence is 9.762×10^{19} n/cm^2 for the forging and 2.847×10^{19} n/cm^2 for the lower girth weld. The applicant further indicated that the margin values at the 1/4T and 3/4T locations for the forging are 45°F and 42°F, respectively. The margin values for the 1/4T and 3/4T locations of the lower girth weld are 66°F and 50°F, respectively.

The values of the copper and nickel composition and the initial RT_{NDT} values were provided in the AP1000 DCD. The applicant calculated the adjusted reference temperature values to be 63°F and 56°F at the 1/4T and 3/4T locations of the forging, respectively, and 93°F and 66°F at the 1/4T and 3/4T locations of the lower girth weld, respectively. The staff independently verified that the applicant's predicted shifts in the reference temperature for the RV materials were calculated using the methodology of RG 1.99, Rev. 2. This RG provides reasonably accurate and conservative predictions of adjusted reference temperatures for RV beltline materials that are produced domestically. The applicant's approach is, therefore, acceptable for domestically produced steels.

However, the staff believes that steels from nondomestic sources could have different characteristic responses to radiation embrittlement, particularly those steels with high phosphorus and sulfur contents. The methodology adopted in RG 1.99, Rev. 2, could possibly no longer apply to the steels with high phosphorus and sulfur contents. The applicant indicated that regardless of the source of material, the RV beltline material would be maintained to ASME Code specifications. In addition, DCD Tier 2 Table 5.3-1 indicates that restrictive maximum content limits would be imposed on the critical residual elements (copper, nickel, phosphorus, etc). The staff finds the applicant's approach acceptable because it is in compliance with the requirements of the ASME Code specifications and the chemical content controls imposed on the RV materials meet the guidelines for new plants as specified in RG 1.99, Rev. 2.

5.3.3.3 Conclusion

With the exception of Open Item 5.3.3-1, the staff concludes that the P-T limits imposed on the RCS for operating and testing conditions to ensure adequate safety margins against nonductile or rapidly propagating failure are in conformance with the fracture toughness criteria of Appendix G to 10 CFR Part 50. The change in fracture toughness properties of the RV beltline materials during operation will be determined through a surveillance program in conformance with Appendix H to 10 CFR Part 50. With the exception of Open Item 5.3.3.1-1, the use of operating limits, determined by the criteria defined in Section 5.3.2 of the SRP, provides reasonable assurance that nonductile or rapidly propagating failure will not occur, and constitutes an acceptable basis for satisfying the requirements of 10 CFR 50.55a, Appendix A of 10 CFR Part 50, and GDC 1, 14, 31, and 32.

5.3.4 Pressurized Thermal Shock

The staff reviewed DCD Tier 2 Section 5.3.4 as it applies to pressurized thermal shock (PTS) in accordance with SRP 5.3.2, "Pressure-Temperature Limits and Pressurized Thermal Shock." Section 50.61 of 10 CFR Part 50, "Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events," defines the fracture toughness requirements for protection against PTS events. Section 50.61 establishes the PTS screening criteria, below which no additional action is required for protection from PTS events. The screening criteria are given in terms of reference temperature (RT_{PTS}). These criteria are 148.9°C (300°F) for circumferential welds and 132.2°C (270°F) for plates, forgings, and axial welds.

5.3.4.1 Summary of Technical Information

The applicant indicated that the evaluation of the AP1000 RV materials showed that even at the fluence level which results in the highest RT_{PTS} value, this value is well below the screening criteria of 132.2°C (270°F) for forgings, and 148.9°C (300°F) for circumferential welds, as presented in 10 CFR 50.61. The screening criteria will not be exceeded using the method of calculation prescribed by the pressurized thermal shock rule for the vessel design objective.

The material properties, and initial RT_{NDT}, and end of life RT_{PTS} requirements and predictions are provided in DCD Tier 2 Tables 5.3-1 and 5.3-3. The materials that are exposed to high fluence levels at the beltline region of the RV are subject to the PTS rule.

5.3.4.2 Staff Evaluation

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

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

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

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

The applicant demonstrated that the AP1000 design meets the PTS screening criterion. The AP1000 reactor beltline design consists of two forgings and one circumferential weld. The AP1000 beltline forging material and weld metal will contain a maximum of 0.03 weight percent copper and 0.85 weight percent nickel. The initial RT_{NDT} for the forging is -23.3°C (-10°F) and for the circumferential weld it is -28.8°C (-20°F). In response to RAI 251.019, the applicant indicated that the maximum assumed neutron fluence is 9.76E19 n/cm^2 for the forgings and 2.85E19 n/cm^2 for the circumferential weld at end-of life (60 years). The margins, defined in 10 CFR 50.61, are 18.9°C (34°F) for the forgings and 31.1°C (56°F) for the circumferential weld.

Using the above values, the staff determined that after 60 years of operation, the RT_{PTS} values for the forgings and circumferential weld will be 30°C (54°F) and 48.8°C (88°F), respectively, well below the PTS screening criteria.

5.3.4.3 Conclusion

The staff concludes that the AP1000 RV meets the relevant requirements of 10 CFR 50.61. The staff's conclusion is based on the calculations that the RV beltline materials will be substantially below the PTS screening criteria after 60 years of operation.

5.3.5 Reactor Vessel Integrity

The staff reviewed DCD Tier 2 Section 5.3.4, "Reactor Vessel Integrity," in accordance with NRC SRP 5.3.3, "Reactor Vessel Integrity." The applicant's assessment of RV integrity is acceptable if it meets codes and standards and regulatory guidance commensurate with the safety function to be performed so that the relevant requirements of 10 CFR 50.55a, "Codes and Standards," 10 CFR 50.61, "Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events," 10 CFR Part 50, Appendix G, "Fracture Toughness Requirements," Appendix H, "Reactor Vessel Material Surveillance Program Requirements," and GDC 1, 4, 14, 30, 31, and 32. These requirements are discussed below.

GDC 1, "Quality Standards and Records," GDC 30, "Quality of Reactor Coolant Pressure Boundary," and 10 CFR 50.55a(a)(1) require that structures, systems and components important to safety shall be designed, fabricated, erected and tested to quality standards commensurate with the importance with the safety function to be performed.

GDC 4, "Environmental and Dynamic Effects Design Bases," requires that SSCs important to safety shall be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents.

GDC 14, "Reactor Coolant Pressure Boundary," requires that the reactor coolant pressure shall be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture.

GDC 31 requires that the RCPB shall be designed with sufficient margin to assure that, when stressed under operation, maintenance, testing, and postulated accident conditions, it will behave in a non-brittle manner and with the probability of rapidly propagating fracture minimized.

GDC 32, " Inspection of Reactor Coolant Pressure Boundary," requires, in part, that the RCPB components shall be designed to permit an appropriate material surveillance program for the RPV.

Section 50.61 of 10 CFR Part 50 defines the fracture toughness requirements for protection against PTS events. Section 50.61 establishes the PTS screening criteria, below which no additional action is required for protection from PTS events.

Appendix G, "Fracture Toughness Requirements," to 10 CFR Part 50 specifies the fracture toughness requirements for ferritic materials of the pressure-retaining components of the reactor coolant pressure boundary. The staff reviews the RV materials as they relate to the materials testing and acceptance criteria for fracture toughness contained in Appendix G.

Appendix H, "Reactor Vessel Material Surveillance Program Requirements," to 10 CFR Part 50 presents the requirements for a materials surveillance program to monitor the changes in fracture toughness properties of materials in the RV beltline region resulting from exposure to neutron irradiation and the thermal environment. These requirements include conformance with

ASTM E-185, “Standard Recommended Practices for Surveillance Tests for Nuclear Reactor Vessels.” Compliance with Appendix H satisfies the requirements of GDC 32 regarding the provision of an appropriate materials surveillance program for the RV. The staff reviewed the RV materials to determine that they meet the relevant requirements for Appendix H as they relate to the determination and monitoring of fracture toughness.

5.3.5.1 Summary of Technical Information

The applicant stated that the RV, including the closure head, is approximately 12.1 meters (40 feet) long and has an inner diameter at the core region of 398.8 cm (157 in). Surfaces which can become wetted during operation and refueling are clad to a nominal 0.56 cm (0.22 in) of thickness with stainless steel welded overlay which includes the upper shell top, but not the stud holes. The design objective for the AP1000 RV is to withstand the design environment of 17.34 MPa (2500 psi) and 343.3°C (650°F) for 60 years. The major factor affecting vessel life is radiation degradation of the lower shell.

As a safety precaution, there are no penetrations below the top of the core. The core is positioned as low as possible in the vessel to limit reflood time in an accident. To decrease outage time during refueling, access to the stud holes is provided to allow stud hole plugging with the head in place. The flange is designed to interface properly with a multiple stud tensioner device. By the use of a ring forging with an integral flange, the number of welds is minimized to decrease ISI time.

The vessel is manufactured from low alloy steel plates and forgings to minimize size. The chemical content of the core region base material is specifically controlled. A surveillance program is used to monitor the radiation damage to the vessel material.

The RV is designed and fabricated in accordance with the quality standards set forth in 10 CFR Part 50, GDC 1, GDC 30 and 50.55a, and the requirements of the ASME Code, Section III. The vessel design and construction enables inspection in accordance with the ASME Code, Section XI.

Cyclic loads are introduced by normal power changes, reactor trips, and startup and shutdown operations. These design base cycles are selected for fatigue evaluation and constitute a conservative design envelope for the design life. Thermal stratification during passive core cooling system operation and natural circulation cooldown is considered by performing a thermal/flow analysis using computational fluid dynamics techniques. This analysis includes thermally-induced fluid buoyancy, heat transfer between the coolant and the metal of the vessel and internals and uses thermal/flow boundary conditions based on an existing thermal/hydraulic transient analysis of the primary reactor coolant system.

The analysis verifies that the vessel is in compliance with the fatigue and stress limits of Section III of the ASME Code. The loadings and transients specified for the analysis are based on the most severe conditions expected during service. The heatup and cooldown rates imposed by plant operating limits are 37.8°C (100°F) per hour for normal operations.

The operating limitations for the RV are provided in DCD Tier 2 Section 5.3.3, “Pressure-Temperature Limits,” and in the AP1000 TS. In addition to the analysis of the primary components discussed in DCD Tier 2 Section 3.9.1.4, “Considerations for the Evaluation of the Faulted Conditions,” the RV is further qualified to ensure against unstable crack growth under faulted conditions. Safeguard actuation following a loss-of coolant, tube rupture, or other similar emergency or faulted event, produces relatively high thermal stresses in regions of the RV that come into contact with water from the passive core cooling system. Primary consideration is given to these areas, including the RV beltline region and the RV primary coolant nozzles, to ensure the integrity for the RV under these severe postulated transients. TMI Action Item II.K.2.13, is satisfied upon submittal of RT_{NDT} values which are below the PTS rule screening values. PTS is discussed further in Section 5.3.4 of this report.

The internal surfaces of the RV are accessible for periodic inspection. Visual and/or nondestructive techniques are used. During refueling, the vessel cladding is capable of being inspected in certain areas of the upper shell above the primary coolant inlet nozzles, and if deemed necessary, the core barrel is capable of being removed, making the entire inside vessel surface accessible.

Further details of the applicant’s inservice surveillance activities with regard to components of the RV is provided in DCD Tier 2 Section 5.3.4.7, “Inservice Surveillance.” Because radiation levels and remote underwater accessibility limits access to the RV, several steps, as indicated in the AP1000 DCD have been incorporated into the design and manufacturing procedures in preparation for the periodic nondestructive tests which are required by the ASME Code inservice inspection requirements.

The vessel design and construction enables inspection in accordance with the ASME Code, Section XI. The RV inservice inspection program is detailed in the technical specifications.

5.3.5.2 Staff Evaluation

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

The staff reviewed the information in each area to ensure that no inconsistencies exist that would reduce the certainty of vessel integrity. The areas reviewed and the sections of this report in which they are discussed are given below:

- pressure boundary materials (Section 5.2.3)
- ISI and testing of the RCPB (Section 5.2.4)
- reactor vessel materials fabrication methods (Section 5.3.2)
- pressure-temperature limits and operating conditions (Section 5.3.3)
- pressurized thermal shock (Section 5.3.4)

The integrity of the RV is assured because:

- The RV will be designed and fabricated to the high standards of quality required by the ASME Boiler and Pressure Vessel Code and the pertinent Code Cases.
- The RV will be fabricated from material of controlled and demonstrated quality.
- The RV will be subjected to extensive preservice inspection and testing to provide assurance that the vessel will not fall because of material or fabrication deficiencies.
- The RV will operate under conditions, procedures, and protective devices that provide assurance that the vessel design conditions will not be exceeded during normal reactor operation, maintenance, testing, and anticipated transients.
- The RV will be subjected to periodic inspection to demonstrate that the high initial quality of the RV has not deteriorated significantly under service conditions.
- The RV will be subjected to surveillance to account for neutron irradiation damage so that the operating limitation may be adjusted.
- The fracture toughness of the RV and RCPB materials will be sufficient to ensure that when stressed under operation, maintenance, testing, and postulated accident conditions, it will behave in a non-brittle manner and with the probability of rapidly propagating fracture minimized.

5.3.5.3 Conclusion

The staff concludes that the structural integrity of the AP1000 RV meets the requirements of GDC 1, 4, 14, 30, 31, and 32 of Appendix A to 10 CFR Part 50; Appendices G and H to 10 CFR Part 50; 10 CFR 50.61, and 10 CFR 50.55a; and is therefore acceptable. The basis for this conclusion is that the design, materials, fabrication, inspection, and quality assurance requirements of the AP1000 plant will conform to the applicable NRC regulations and RG set forth above, and the rules of ASME Code, Section III. The fracture toughness requirements of the regulations and ASME Code, Section III, will be met, including requirements for surveillance of vessel material properties throughout service life, in accordance with Appendix H to 10 CFR Part 50. In addition, operating limitations on temperature and pressure will be established for the plant in accordance with Appendix G, “Protection Against Nonductile Failure,” of ASME Code Section III, Appendix G, 10 CFR Part 50.

5.4 Component and Subsystem Design

In DCD Tier 2 Section 5.4, “Component and Subsystem Design,” the applicant describes the design of RCS components and subsystems for the AP1000.

5.4.1 Reactor Coolant Pump Assembly

The AP1000 Reactor Coolant Pumps (RCPs) are single-stage, hermetically sealed, high-inertia, centrifugal, canned-motor pumps. There are a total of four RCPs, two in each SG. Two pumps, rotating in the same direction, are directly connected to the two outlet nozzles on the SG channel heads. The RCPs are designed to pump large volumes of reactor coolant at high pressures and temperature. High volumetric flow rates are needed to ensure adequate core heat transfer so as to maintain a departure from nucleate boiling ratio (DNBR) greater than the acceptable limit established in the safety analysis. Rotational inertia of a flywheel and other rotating parts in the pump assembly results in continuous coastdown flow after an RCP trip.

The RCP is an integral part of the RCPB. Section 5.2 of this report discussed the requirements on the integrity of RCPB. A canned motor pump contains the motor and all rotating components inside a pressure vessel. The pressure vessel consists of the pump casing, thermal barrier, stator shell, and stator cap, which are designed for full RCS pressure. The stator and rotor are encased in corrosion-resistant cans that prevent contact of the rotor bars and stator windings with the reactor coolant. Because the shaft for the impeller and rotor is contained within the pressure boundary, seals are not required to restrict leakage out of the pump into containment. DCD Tier 2 Section 5.4.1.3.3 discusses the RCPB integrity of the reactor coolant pumps. Section 5.4.1.4 of this report describes the staff’s evaluation of the RCP for conformance to the RCPB requirements.

The RCP driving motor is a vertical, water-cooled, squirrel-cage induction motor with a canned rotor and a canned stator. It is designed for removal from the casing for inspection, maintenance, and replacement, if required. The motor is cooled by component cooling water circulating through a cooling jacket on the outside of the motor housing and through a thermal barrier between the pump casing and the rest of the motor internals. Inside the cooling jacket are coils filled with circulating rotor cavity coolant. This rotor cavity coolant is a controlled volume of reactor coolant that circulates inside the rotor cavity.

Each pump motor is driven by a variable speed drive, which is used for pump startup and operation until the RCS temperature has reached 232.2°C (450°F), above which the variable frequency drives are bypassed and the pump run at constant speed.

A flywheel, consisting of two separate assemblies, provides rotating inertia that increases the coastdown time for the pump.

5.4.1.1 Pump Performance

GDC 10 requires that the reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences. For PWR designs, SRP Section 4.4 specifies the criterion necessary to meet GDC 10 as that the hot rod in the core does not experience a departure from nucleate boiling, or the DNBR limit is not violated, during normal operation or anticipated operational occurrences.

The RCP is sized to deliver a flow rate that equals or exceeds that required to ensure adequate thermal performance under normal and anticipated transient conditions. Adequacy of the RCP design capacity of delivering the forced reactor coolant flow and coastdown flow rates after a RCP trip is verified through the safety analyses of the design basis transients to ensure that the DNBR limit is not violated during the transients. DCD Tier 2 Table 5.4-1 specifies the RCP design parameters with the design flow rate of 17,886 m^3/hr (78,750 gpm) per pump, the developed head of 111.25 m (365 ft), and the synchronous speed of 1800 gpm. DCD Tier 2 Table 4.4-1 provided the thermal and hydraulic data for the AP1000 design with the vessel minimum measured flow rate of 68,516 m^3/hr (301,670 gpm), and the vessel thermal design flow rate of 67,229 m^3/hr (296,000 gpm), representing the design and measurement flow uncertainties of 1.9 percent. DCD Tier 2 Table 15.0-3 lists the nominal values of pertinent plant parameters utilized in the accident analyses. With the assumption of 10 percent SG tube plugging, the minimum measured and thermal design flow rates of 68,500 m^3/hr (301,600 gpm) and 67,229 m^3/hr (296,000 gpm), respectively, are used in Chapter 15 safety analyses with or without the revised thermal design procedure. AP1000 TS LCO 3.4.1 requires the RCS flow to be greater than or equal to the minimum measured flow rate of 68,516 m^3/hr (301,670 gpm) for Mode 1 power operation, with a surveillance verification every 12 hours per TS surveillance requirement SR 3.4.1.3. This will ensure that the RCS flow rate used in the DCD Tier 2 Chapter 15 transient and accident analyses are conservative with respect to the actual RCS flow rate delivered by the RCPs. The staff has reviewed the safety analyses of the design-basis events described in DCD Tier 2 Chapter 15. With the minimum measured flow rate of the reactor coolant as the initial condition, and the flow coastdown (see Section 5.4.1.2 of this report) after the reactor trip, the DNBR limit is not violated for all the anticipated transients analyzed and, therefore, GDC 10 is met. Therefore, the staff concludes that the RCP design flow capacity is acceptable. The total delivery capability of the four RCPs will be verified per inspection, test, analysis, and acceptance criteria (ITAAC) DCD Tier 1 Table 2.1.2-4, Item 9.a.

The startup testing of the AP1000 initial test program requires the verification of adequacy of the RCS flow rate by (1) measurement prior to initial criticality, per Item 14.2.10.1.17, to verify adequacy of the RCS flow rate for power operation, and (2) measurement at approximately 100-percent rated thermal power condition, per Item 14.2.10.4.11, to verify that the RCS flow equals or exceeds the minimum value required by the plant technical specifications. The COL applicant is required by DCD Tier 2 Section 14.4.2, "Test Specifications and Procedures," to provide test specifications and test procedures for the pre-operational and startup tests for review by NRC. Therefore, the staff concludes that the AP1000 initial test program provides adequate verification of the total delivery capability of the reactor coolant pump for adequate core cooling.

As stated in DCD Tier 2 Section 5.4.1.3.1, to provide operational integrity and to minimize the potential for cavitation, ample margin is provided between the available net positive suction head (NPSH) and the required NPSH by conservative pump design and operation. The required NPSH is well within the operating RCS pressure during heatup, cooldown, and power operation with four pumps running. Since, the available NPSH is always larger than the required NPSH, cavitation is not a concern.

5.4.1.2 Coastdown Capability

For reactor fuel protection, each RCP has a high-density flywheel and high-inertia rotor. These provide rotating inertia to increase the pump's coastdown time following a pump trip and loss of electrical power. Continued coastdown flow of reactor coolant is important in ensuring that the fuel's DNBR limit will not be violated in the event of a partial or complete loss of the forced reactor coolant flow analyzed in DCD Tier 2 Chapter 15.3, "Decrease in Reactor Coolant System Flowrate." The adequacy of the RCP flywheel-rotor design to provide for sufficient rotating inertia, and thus flow coastdown capability following and RCP trip, is verified through the safety analyses of the loss of flow transients to demonstrate that the minimum DNBR limit is not violated. The staff has reviewed the safety analyses of the design-basis transients of partial and complete loss of forced reactor flow described in DCD Tier 2 Sections 15.3.1 and 15.3.2, respectively. The RCP coastdown flow rate is calculated on the basis of an RCP rotating moment of inertia of 695.3 kg-m^2 (16,500 lb-ft^2), which is specified in DCD Tier 2 Table 5.4-1, using the LOFTRAN computer code, which has been approved for the AP1000 transient analyses as discussed in Section 21.6.1 of this report. The analysis results of partial and complete loss of forced reactor coolant flow demonstrate that, with coastdown of the affected pumps, the DNBR does not decrease below the design basis limit value at any time during the transients. Therefore, the staff concludes that the RCP flywheel design provides adequate flow coastdown capability.

The acceptance criteria specified in DCD Tier 1 Table 2.1.2-4, Item 8b, for the calculated rotating moment of inertia of each RCP is no less than 695.3 kg-m^2 (16,500 lb-ft^2). Therefore, based on the above evaluation, the RCP coastdown capability is acceptable.

5.4.1.3 Rotor Seizure

In DCD Tier 2 Section 5.4.1.3.6.2, the applicant states that the design of the AP1000 RCP (and motor) precludes the instantaneous stopping of any rotating component of the pump or motor. However, a design-basis analysis of a postulated RCP rotor seizure is presented in DCD Tier Section 15.3.3. The analysis of thermal and hydraulic effects of the locked rotor event uses a nonmechanistic, instantaneous stop of the impeller. This conservative assumption bounds any slower stop. The transient analysis considers the effect of the locked rotor on the reactor core and RCS pressure to demonstrate that acceptable RV pressure boundary and radiological consequence limits are not exceeded. The staff reviewed the analysis of the pump rotor seizure event as part of the Chapter 15 design-basis analysis and found the result to be acceptable as discussed in Section 15.3.3 of this report.

5.4.1.4 Reactor Coolant Pump Flywheel Integrity

The following regulatory requirements are applicable with respect to the designs of the RCP flywheels for the AP1000 reactors:

- 10 CFR 50.55a(a)(1) and GDC 1 of Appendix A to 10 CFR Part 50; both require that SSCs important to safety be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety function to be performed.

- GDC 4 to Appendix A of 10 CFR Part 50, in part, requires that SSCs important to safety be protected against the dynamic effects, including missiles, pipe whipping, and discharging fluids that may result from equipment failures and from events and conditions outside the nuclear power plant unit.

5.4.1.4.1 Summary of Technical Information

DCD Tier 2 Section 5.4.1.3.6.3 provides the detailed technical description of the AP1000 RCP flywheel design. In this section of the design certification document, the applicant states that each RCP for the AP1000 reactor is designed with a high-density flywheel and high inertia rotor and that these components provide the RCP with a continual coast-down capability following an RCP trip. The applicant also states that, to ensure this coast-down capability, the RCP rotor must be designed against a sudden seizure. The RCP flywheel is also analyzed to demonstrate that integrity of the pressure boundary components will be maintained in the event of a postulated RCP flywheel missile.

5.4.1.4.2 Staff Evaluation

The RCP flywheels for the AP1000 are designed to provide the RCP with the ability to safely coast down from an RCP overspeed condition without resulting in a rupture of the RCP rotor. The applicant describes the design features of the AP1000 RCP flywheels in DCD Tier 2 Section 5.4.1.3.6.3 and WCAP-15994-P, Revision 0, "Structural Analysis for the AP1000 Reactor Coolant Pump High Inertia Flywheel (November 2002)." This WCAP addresses the fabrication, design, and structural integrity of the AP1000 RCP flywheel. The staff reviewed the information in DCD Tier 2 Section 5.4.1.3.6.3 and WCAP-15994-P, Revision 0, and Revision 1, to assess the AP1000 RCP flywheel design and whether the design for the flywheels had the potential to impact the structural integrity of the RCPB. The staff has evaluated the RCP rotors for protection against seizure in Section 5.4.1.3 of this report.

During the staff's review of the AP600 design certification, the staff issued RAIs (AP600 RAIs 251.2 through 251.23) to address questions on design aspects, materials of fabrication, fabrication practices, and structural integrity analyses used for the design of AP600 RCP flywheels. In RAI 251.21 for AP1000, the staff requested confirmation that WCAP-13474 and WCAP-13575 were still applicable to the design for the AP1000 RCP flywheels. In RAI 251.21, the staff also requested the applicant confirm the previous responses to AP600 RAIs 251.2 through 251.23 were applicable to the AP1000 RCP flywheel design, or else provide updated information to address the responses to AP600 RAIs 251.2 though 251.23 as they relate to the design of the AP1000 RCP flywheels and the structural integrity of the RCPB in the event of a postulated AP1000 RCP flywheel failure.

In the response to RAI 251.21, the applicant provided updated responses to the AP600 RAIs 251.2 through 251.23, as relevant to the design aspects, materials of fabrication, fabrication practices, stress analyses and missile generation analyses used for the design of AP1000 RCP flywheels and its assemblies. In addition, the applicant submitted Proprietary Class 2 Topical Report WCAP-15994-P, Revision 0, and indicated that the information in WCAP-15994-P, Revision 0, supercedes the information in WCAP-13474 and WCAP-13575 and updates the design information for the AP1000 RCP flywheels. The applicant submitted WCAP-15994-P, Revision 1, in order to update and clarify some of the design aspects for the RCP flywheel enclosure welds. The staff reviewed WCAP-15994-P, Revision 1, to assess the design aspects of the AP1000 RCP flywheel.

The AP1000 RCP flywheel assembly is fabricated from a high-quality, depleted uranium-molybdenum (U-2Mo) alloy casting. The uranium flywheel castings are made by a centrifugal casting process that minimizes casting defects. The flywheel is subjected to preservice volumetric and surface examinations. There is a lack of data regarding the fracture toughness of the uranium alloy used to fabricate the AP1000 RCP flywheel material. The lack of fracture toughness data for the depleted uranium alloy used in the design potentially diminishes the reliability aspect of the AP1000 RCP flywheel design. Therefore, the AP1000 RCP flywheel design basis is not predicated on precluding a stress-induced or fatigue-induced failure of the flywheel. Rather, the AP1000 RCP flywheel design is based on the limiting postulated AP1000 RCP flywheel missile fragment not having sufficient kinetic energy to penetrate the RCPB components associated with the RCP (i.e., RCP casing, stator shell/flange, and thermal barrier). Therefore, the potential for diminished fracture toughness reliability is not a factor in the staff’s assessment of the AP1000 RCP flywheel design.

The AP1000 RCP flywheels are located within an enclosure fabricated from Alloy 690. In contrast, the AP600 RCP flywheel enclosures were fabricated from Alloy 600. Alloy 690 should provide the AP1000 RCP flywheel enclosure with additional corrosion resistance. The staff considers this to be an improvement in the design of the AP1000 RCP flywheel enclosure.

The Alloy 690 enclosure is located within the RCP stator shell/flange and thermal barrier, which serve as part of the pressure boundary for the RCP. The flywheel enclosure is a welded design that is similar to the design used for other the applicant motor rotor designs. The flywheel enclosure isolates the RCP flywheel from exposure to the reactor coolant. The RCP flywheel enclosure is credited with minimizing the potential for corrosion of the flywheel and contamination of the reactor coolant by depleted uranium. However, the enclosure is not credited with retention of missile fragments that could potentially result from a postulated failure of the flywheel disc. There is no industry experience that demonstrates that the RCP flywheel enclosure is susceptible to fast-fracture-induced or fatigue-induced failures.

The applicant’s RCP flywheel design is fabricated from an alternative material and does not entirely conform to the guidelines in RG 1.14, Reactor Coolant Pump Flywheel Integrity, Revision 1. However, to meet the intent of RG 1.14, Revision 1, the applicant has performed the following three structural analyses for the AP1000 RCP flywheel designs:

- an analysis to evaluate the failure by ductile fracture of the uranium alloy RCP flywheel inserts (discs) using faulted stress limits in the ASME Code, Section III, Appendix F.

- a structural analysis of the flywheel enclosure under normal operating and design speeds (1800 rpm and 2250 rpm, respectively) using the ASME Code, Section III, Subsection NG limits.

- a kinetic energy assessment of the limiting RCP flywheel fragment that is postulated to occur with a failure of a flywheel disc (i.e., a RCPB safety analysis).

The applicant performed a ductile failure analysis of the RCP flywheel discs under rotational loading associated with normal operation (1800 rpm) and design overspeed operation (2250 rpm). Based on the applicant's analysis in WCAP-15994-P, Revision 1, the applicant has demonstrated that the primary stresses for the depleted uranium alloy discs are less than the stress limits under normal and design operating conditions, and are, therefore, acceptable.

The staff's basis for asking AP1000 RAI 251.021 was predicated, in part, on verifying the proper stresses associated with a limiting design basis accident of the AP1000 main coolant loop piping would be included as part of the applicant's ductile failure analysis for the RCP flywheel under design overspeed conditions. The applicant, as part of its response to AP1000 RAI 251.021 (and the applicant's response to AP600 RAI 251.8, as given in the attachment relative to the AP1000 RCP flywheel design) clarified that the AP1000 RCS coolant piping size 6 inches NPS or larger is qualified for leak-before-break, and therefore the stresses associated with the largest RCS pipe break analyzed for the flywheel integrity are those for a LOCA associated with a 4-inch NPS RCS pipe break. This provides additional information that clarifies the limiting stresses that were analyzed for the structural integrity assessment of the AP1000 RCP flywheel. Based on an acceptable review of AP1000 DCD Tier 2 Section 3.6.3 and DCD Tier 2 Appendix 3B on leak-before-break, the applicant has demonstrated that the stresses associated with a postulated LOCA for pipe sizes greater than 4 NPS in diameter need not be incorporated as inputs into the structural integrity assessments for the AP1000 RCP flywheels.

The applicant performed a structural analysis of the outer flywheel enclosure under both steady-state conditions (i.e., normal operating speeds at 1800 rpm) and design overspeed conditions (125 percent of normal operating speeds). The applicant's structural analyses for the flywheel enclosure under steady-state and design overspeed conditions were based on appropriate mechanical and thermal loading (stress) data. The applicant's analyses of the flywheel enclosure indicate that the stresses associated with the enclosure for both normal operating and design conditions are less than the allowable stress limits of the Alloy 690 used to fabricate the enclosure. This analysis demonstrates that the outer flywheel enclosures will not yield (plastically deform) under normal and design overspeed operations of the RCP flywheel. The applicant did not perform a stress analysis of the outer flywheel enclosure under critical overspeed conditions. This is acceptable to the staff because the applicant does not credit the flywheel enclosure with preventing a postulated flywheel fragment from reaching the pressure boundary components associated with the RCP.

The safety analysis for the AP1000 RCP flywheel design was evaluated in terms of whether or not the kinetic energy associated with a postulated failure of the limiting flywheel disc is capable of penetrating the pressure boundary components associated with the AP1000 RCP (i.e., the RCP casing, stator shell/flange, and thermal barrier). These pressure boundary components contain the RCP flywheel disc whose failure could generate a limiting flywheel fragment. The limiting RCP flywheel disc and the pressure boundary components associated with the RCP were analyzed to demonstrate that a failure of the flywheel would not penetrate the RCPB, even in the event of a postulated generation of a limiting RCP flywheel missile and breach of the

RCP flywheel enclosure. The theoretical worst-case flywheel failure analysis is analogous to the approach taken with the theoretical worst-case turbine disc failure analysis. The applicant has demonstrated, in WCAP-15994-P, Revision 1, that the highest amount of energy associated with an RCP flywheel missile constitutes only a small fraction (less than 15-percent) of the kinetic energy that would be required to penetrate the pressure boundary components associated with the RCP. This analysis provides an acceptable basis for not including the AP1000 RCP flywheels and their enclosures under an ISI program, as recommended by RG 1.14, Revision 1.

Since the applicant's safety analysis has demonstrated that a postulated RCP flywheel failure is not capable of penetrating the RCPB and will not result in a missile that could have adverse effects on the plant safety functions, the staff concludes that the requirement for an ISI program to preclude such failures is unnecessary from a safety standpoint.

5.4.1.4.3 Conclusion

The staff has reviewed the information in DCD Tier 2 Section 5.4.1.3.6.3, WCAP-15994-P, Revision 0 and Revision 1, and the applicant's response to AP1000 RAIs 251.20 and 251.21, as related to the applicant's design of the AP1000 RCP flywheels. On the basis of this review and acceptable conclusions on leak-before-break in Section 3.6.3 of this SER, the staff concludes that the applicant has demonstrated that the AP1000 RCP flywheels and their enclosures have been designed appropriately, considering the use of acceptable materials and fabrication processes, and that the integrity of the RCP pressure boundary will be maintained in the event of a postulated RCP flywheel missile. Based on this review, the staff concludes that the measures taken to assure the integrity of the RCP flywheels are acceptable and meet the safety requirements of GDC 1 and 4, and 10 CFR 50.55a(a)(1).

5.4.2 Steam Generators

The AP1000 design has two vertical-shell, U-tube Model Delta 125 steam generators. The basic function of these SGs is to transfer heat from the primary reactor coolant through the U-shaped heat exchanger tubes to the secondary side of steam generation. The design of the Model Delta 125 SGs, except for the configuration of the channel head, is similar to an upgraded Model Delta 75 SG, which have been placed in operation as replacement steam generators. In the channel head under the SG tube sheet, a divider plate is used to separate the inlet and outlet chambers. Two canned-motor RCPs are directly attached to the cold leg nozzles on the outlet channel head to provide the driving force for the reactor coolant flow. A passive residual heat removal (PRHR) nozzle is attached to the bottom of the channel head of the Loop 1 SG on the cold leg portion of the head. This nozzle provides recirculated flow from the PRHR heat exchanger (PRHRHX), which cools the primary side under emergency conditions.

The SG channel head, tubesheet, and tubes are a portion of the RCPB, and are designed to satisfy the criteria specified for Class 1 components. The tubes transfer heat to the secondary (steam) system while retaining radioactive contaminants in the primary system.

The SGs remove heat from the RCS during power operation, anticipated transients, and under natural circulation conditions. The SGs' heat transfer function and associated secondary water and steam systems are not required to provide a safety-grade safe shutdown of the AP1000. Safe shutdown is achieved and maintained by the safety-related passive core cooling systems.

For the SG operation, the reactor coolant flow from the RCS hot leg enters the primary side of inverted U-tubes, transferring heat to the secondary side during its traverse. The flow then returns to the cold leg side of the primary chamber, exits the SG via two cold leg nozzles and the canned RCPs, to the RV, thus completing a cycle.

If the PRHR system is activated, flow passes from the outlet of the PRHRHX, through the SG's PRHR nozzle connection into the SG channel head. Coolant then flows through the RCPs, into the cold legs and then into the RV.

On the secondary side, feedwater enters the SG at an elevation above the top of the U-tubes through a feedwater nozzle. The feedwater enters a feedring via a welded thermal sleeve connection, and leaves it through nozzles attached to the top of the feedring. This nozzle design minimizes the potential for trapping pockets of steam that can lead to water hammer in the feedwater piping, by discharging feedwater into the SG at an elevation above the top of the tube bundle and below the normal water level, thus reducing the potential for vapor formation in the feedring. After exiting the nozzles, the feedwater mixes with saturated water that has been mechanically separated from the steam flow exiting the SG by internal moisture separators. The combined feedwater/recirculation flow then enters the downcomer annulus between the tube wrapper and the shell. At the bottom of the tube wrapper, the water is directed toward the center of the tube bundle by the lowest tube support plate. This recirculation arrangement is designed to minimize low-velocity zones, which present the potential for sludge deposition. As the water passes the tube bundle, it is converted to a steam-water mixture, which, subsequently, rises into the steam drum section, where centrifugal moisture separators remove most of the entrained water from the steam. The steam continues to the secondary moisture separators, or dryers, for further moisture removal, increasing its quality to a designed minimum of 99.75 percent (0.25 percent by weight maximum moisture). Water separated from the steam combines with entering feedwater and recirculates through the SG. Dry steam exits the SG through the SG outlet nozzle, which has an installed steam-flow restrictor.

The startup feedwater system (SUFS) supplies water to the SGs during startup, shutdown and other times when the normal feedwater system is not needed or not operable. The SUFS is a non-safety grade system that will be used as a defense-in-depth system following a reactor trip or loss of main feedwater event. The SUFS thus provides investment protection for the plant. During startup and shutdown operations, the SG has enough surface area and a small enough primary-side hydraulic resistance to remove decay heat from the RCS by natural circulation (without operation of the RCPs).

The SG design requirements and design parameters are shown in DCD Tier 2 Tables 5.4-4 and 5.4-5, respectively. The evaluation of SG thermal performance, including required heat transfer area and steam flow, uses conservative assumptions for parameters such as primary flow rates and heat transfer coefficients. The effective heat transfer coefficient is determined by the physical characteristics of the AP1000 SG and the fluid conditions in the primary and secondary systems for the nominal 100 percent design case. It includes a conservative allowance for fouling and uncertainty.

As stated above, the SG heat transfer function is not required for safe shutdown. Because the secondary systems, such as the normal feedwater system and the SUFS are not safety-related systems, they cannot be credited in the SG heat transfer function for mitigation of transients and accidents in the design-basis analyses. The staff reviewed and confirmed that no credit of these non-safety-related systems is taken in the analyses of the design-basis transients and accidents in Chapter 15. However, in the evaluation of non-design-basis multiple SG tube rupture (MSGTR) events using realistic calculations, the heat transfer function as well as other accident-mitigating characteristics of the SG may be considered. The MSGTR/containment bypass issue is discussed in Section 5.4.2.2 of this report.

5.4.2.1 Steam Generator Materials

The staff reviewed DCD Tier 2 Section 5.4.2.4, "Steam Generator Materials," in accordance with Section 5.4.2.1, "Steam Generator Materials," of the SRP to ensure that this portion of the RCPB is maintained. The materials used in the fabrication of the SGs are acceptable if the following GDC are met:

- GDC 1 of Appendix A of 10 CFR Part 50 and Section 50.55a(a)(1) of 10 CFR Part 50 require that structures, systems, and components important to safety shall be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety function to be performed;

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

- GDC 15 requires that the component shall be designed with sufficient margin to assure that design conditions of the reactor coolant boundary are not exceeded during any condition of normal operation, including anticipated operational occurrences;

- GDC 31 requires that the component shall be designed with sufficient margin to assure that, when stressed under operation, maintenance, testing, and postulate accident conditions, it will behave in a nonbrittle manner and with the probability of rapidly propagating fracture minimized; and

5.4.2.1.1 Summary of Technical Information

The AP1000 SG is designed to the ASME Code with the pressure-retaining parts of the SG, including the primary and secondary pressure boundaries, classified as Class 1, and the secondary side of the SG classified as Class 2.

The pressure boundary materials used in the SG are selected and fabricated in accordance with the requirements of ASME Code, Sections II and III. The AP1000 design includes the use of Alloy 690, a Ni-Cr-Fe alloy (ASME SB-163), for the SG tubes. In addition, the channel head divider plate is made with Alloy 690 (ASME SB-168). The interior surfaces of the reactor coolant channel head, nozzles, and manways are clad with austenitic stainless steel while the primary side of the tubesheet is weld clad with Ni-Cr-Fe alloy (ASME SFA-5.14). The SG tubes are seal welded to the tubesheet cladding and comply with the ASME Code, Sections II and III. The welds are dye penetrant inspected and leak-tested before each tube is hydraulically expanded the full depth of the tubesheet bore. Ni-Cr-Fe alloys are used in areas where high velocities could lead to erosion corrosion; e.g., feedwater ring, feedwater sparger, and some primary separator parts. Heat and lot of tubing material for each SG tube are recorded and documented in addition to archive samples provided to the COL applicant for use in future materials testing programs or for use as ISI calibration standards.

To minimize crevice areas and deposition of contaminants, the following are considered in the design of the AP1000 SG:

- The portion of the tube within the tubesheet is expanded hydraulically to close the crevice between the tube and tubesheet.

- The SG tubes are supported by either an open lattice design (egg crate) or by a support plate. The support plates are made of Type 405 stainless steel alloy with a three-lobed (trifoil) tube hole design to provide flow adjacent to the tube outer surface.

- Anti-vibration bars are made from wide strips of Type 405 stainless steel to assist in the vibrational stability of the tube bundle.

- Wrapper design results in significant water velocities across the tubesheet to minimize dryout and sludge accumulation.

- Blowdown intake is at the periphery and is capable of continuous blowdown at a moderate volume and intermittent flow.

- A passive sludge collector (mud drum) provides a low flow settling zone and can be cleaned during plant shutdown.

- Four 6-inch access ports are available for sludge lancing; i.e., a method for cleaning the SG in which a hydraulic jet inserted through the access ports loosens deposits and flushes it out of the SG. These ports can also be used for inspection of the tube bundle and retrieval of loose objects. In addition, two 4-inch ports are located on the secondary shell to provide access to the U-Bend area of the tube bundle.

Corrosion tests performed on Alloy 690 TT ASME SB-163 have been conducted to simulate the effects of SG water chemistry on the tubes. Test results indicated that the loss of material due to general corrosion over the 60-year operating design objective is small compared to the tube wall thickness. In addition, tests have shown that the Alloy 690 TT provides as good or better corrosion resistance than Alloy 600 TT or Ni-Cr-Fe Alloy 800 in caustic and chloride aqueous solutions.

Laboratory tests also show that the Alloy 690 TT tubing is compatible with the AVT environment; i.e., a treatment program to minimize the possibility of tube wall thinning and intergranular corrosion in localized areas due to excessive levels of free caustic. Secondary side materials used in the AP1000 design are compatible with the secondary water chemistry.

5.4.2.1.2 Staff Evaluation

The staff reviewed DCD Tier 2 Section 5.4.2.4, "Steam Generator Materials," in accordance with Section 5.4.2.1, "Steam Generator Materials," of the SRP to ensure that the integrity of the SG materials is maintained and that the SG materials meet the requirements of GDCs 1, 14, 15, 31, and Appendix B to 10 CFR Part 50.

5.4.2.1.2.1 Selection and Fabrication of Materials

The materials selected (e.g., austenitic and ferritic stainless steels, ferritic low alloy steels, carbon steels, and high nickel alloys) for the SG are reviewed for adequacy, suitability and compliance with the ASME Code, Sections II and III. The requirements of GDC 1 are met for materials specifications by complying with the ASME Code; and for Code Cases, by meeting the appropriate provisions in RG 1.85, "Materials Code Case Acceptability - ASME Code Section III, Division 1." The fracture toughness requirements of GDC 14 and 31 for Class 1 ferritic materials are met by satisfying the requirements of Appendix G of 10 CFR Part 50, and the requirements of ASME Code, Section III, Subarticle NB-2300 and Appendix G, Article G-2000. The fracture toughness requirements of GDC 14 and 31 for Class 2 ferritic materials are met by satisfying the requirements of ASME Code Section III, Subarticle NC-2300.

The staff reviewed the materials selected for the SG and concludes the materials are acceptable since they meet the requirements/guidance of the ASME Code, Sections II and III, and RG 1.85, “Materials Code Case Acceptability - ASME Code, Section III, Division 1.”

The staff reviewed the AP1000 SG welding qualification, weld fabrication processes and inspection during fabrication and assembly and concluded they conform with the requirements of the ASME Code, Sections III and IX. In addition, the welds between the tube and the tubesheet conform with the requirements of the ASME Code, Sections III and IX.

The staff reviewed the tube material and its heat treatment and concluded that the thermally treated tubes are acceptable because of their improved corrosion resistance as observed in currently operating SGs.

Based on compliance with code requirements and RG 1.85 and based on the use of thermally treated tubing, the staff finds that the materials and fabrication processes used for the AP1000 SG design are acceptable and meet GDC-1.

The staff reviewed the fracture toughness of the RCPB materials and concluded they meet the requirements of Appendix G of 10 CFR Part 50 and the requirements of the ASME Code, Section II, Subarticle NB-2300 and Section III, Appendix G. In addition, the staff reviewed the fracture toughness of the Class 2 components of the SG and concluded they meet the requirements of the ASME Code, Section III, Subarticle NC-2300. Therefore, the AP1000 SG design satisfies the fracture toughness requirements of GDC 14 and 31.

5.4.2.1.2.2 Steam Generator Design

The design and fabrication of the SG is reviewed to determine the extent to which crevice areas are minimized and for sufficient corrosion allowance. The requirements of GDC 15 are met, in part, by designing the SG to avoid crevice areas and to promote high velocity flow along the tubes which minimize buildup of corrosion products and by meeting the appropriate provisions of the ASME Code, Section III.

The staff reviewed the design and fabrication of the SGs to determine the extent to which crevice areas are minimized. The staff notes that the AP1000 design includes features that minimize or eliminate the crevice areas that resulted in corrosion issues with earlier SG designs; specifically by expanding the tubes into the tubesheet for the entire length of the tubesheet and by using trifoil broached hole tube support plates.

The staff requested, in RAI 252.006, the applicant clarify which tube support plate design will be used in the AP1000 (i.e., open lattice (egg crate) or broached hole), since the discussion in DCD Tier 2 Section 5.4.2.3.3, "Mechanical and Flow-Induced Vibration under Normal Operating Conditions," only discusses the broached hole tube support plate design. In its response the applicant stated that the open lattice design is mentioned as a possible option for the tube support design; however, the design descriptions and evaluations in DCD Tier 2 Section 5.4.2.3.3 are based only on the broached hole support plate design. The staff reviewed this response and determined that if the open lattice (egg crate) tube support plate design is an option for the AP1000 design, then the SG design descriptions and evaluations in DCD Tier 2 Section 5.4.2.3.3 must be expanded to include this alternative design for the staff's approval. The staff reviewed Revision 4 to the DCD and found that references to the open lattice (egg crate) tube support plate design as an option have been removed. Based on this revision, the staff finds the discussions of the tube support plate design in Revision 4 to the DCD acceptable.

The staff requested, in RAI 251.022, the applicant provide the results of the flow-induced vibration (FIV) tests and calculations of the steam generators with special emphasis on fluid elastic vibration. In addition, the staff requested the criteria for establishing the instability threshold for ensuring that the fluid-elastic behavior does not contribute unacceptably to flow-induced vibration or alternating stresses. In its response, the applicant stated that the FIV analysis for the AP1000 SG is not complete; however, evaluation of the tube bundle designs for the Delta-109 and Delta-75 SGs have been performed. The Delta-109 tube bundle has a similar tube bundle configuration, including tube size and tube bundle diameter, as the AP1000 SG. Extensive testing and evaluation of the tube bundle designs for the the applicant SGs have been performed using analytical models to evaluate tube vibration. These results have been validated with a number of flow tests using various tube sizes and pitch geometries. The two regions of interest in the evaluation of FIV of SG tubes are the inlet area at the bottom of the tube bundle and the U-bend region at the top of the tube bundle.

In a follow up question to RAI 251.022, the staff requested the applicant to provide the basis for the 0.75 fluid-elastic stability ratio criterion and whether "time domain" analyses had been performed demonstrating that stresses associated with the criterion are negligible. The applicant's response does not address the staff's question. The staff's understanding of the applicant's response is that the 0.75 factor is based on judgement rather than being selected to address any specific uncertainty or time domain analysis. However, the applicant has not explained the rationale by which this judgement was reached. Furthermore, the applicant states that time-domain analyses, which include direct consideration of alternating stress and fatigue, have been performed in some cases and have sometimes led to U-bend support systems with more margin than is required to meet the 0.75 fluid-elastic stability ratio criterion. To reiterate, the staff is requesting the rationale for assuming the alternating stress and associated fatigue usage induced by fluid-elastic coupling is negligible for the case where the fluid-elastic stability ratio is 0.75. This is Open Item 5.4.2-1.

With the exception of open item 5.4.2-1, the staff finds that the AP1000 design meets GDC 15 by including appropriate design measures to minimize crevice areas prone to corrosion, by designing sufficient corrosion allowance, and by designing with sufficient margin to assure that design conditions are not exceeded as a result of tube vibration.

5.4.2.1.2.3 Compatibility of the Steam Generator Components with the Primary and Secondary Coolant

The design and fabrication of the SG is reviewed to ensure compatibility of austenitic and ferritic stainless steels, ferritic low alloy steels, carbon steels, and high nickel alloys with the primary and secondary coolants. The requirement of GDC 14 is met through proper maintenance of primary and secondary water chemistry such that the barrier between primary and secondary fluids maintains its integrity.

The staff reviewed the compatibility of austenitic and ferritic stainless steels, ferritic low alloy steels and carbon steels with the primary and secondary coolants. The AP1000 design includes primary and secondary water chemistry guidelines discussed and evaluated in Sections 9.3.3, “Primary Sampling System,” and 9.3.4, “Secondary Sampling System,” of this SER. Since using these guidelines in plant operation reduce the possibility of SCC, denting, pitting, and wastage of SG tubes through chemistry controls, the staff finds the AP1000 design acceptable in ensuring the compatibility of the SG components with the primary and secondary coolant. Thus, the staff finds that the AP1000 design meets GDC 14.

5.4.2.1.2.4 Cleanup of Secondary Side

The design and fabrication of the SG is reviewed to ensure access for removing surface deposits, sludge, and corrosion products which supplement the removal of sludge during blowdown. The requirement of GDC 14 is met by satisfying RG 1.37, “Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants,” and ANSI N45.2.1-1973, “Cleaning of Fluid Systems and Associated Components During Construction Phase of Nuclear Power Plants.”

The staff reviewed the AP1000 design for provisions to access the SG for removal of surface deposits, sludge, and corrosion products. These design features supplement sludge removal during blowdown of the SG. The staff finds that the AP1000 design is acceptable since it includes access ports for cleaning, inspection, and retrieval of loose objects. In addition, the primary and secondary sides of the AP1000 SGs are cleaned according to the guidance provided in RG 1.37. However, the AP1000 design takes an exception to quality standard ANSI N.45.2.1-1973 referenced in RG 1.37. The discussion of quality assurance documents is found in Section 17.3, “Quality Assurance During Design, Procurement, Fabrication, Inspection and/or Testing of Nuclear Plant Items,” of this SER.

5.4.2.1.3 Conclusion

The staff concludes, with the exception of open items 5.4.2-1 that the AP1000 SG materials are acceptable and meet the requirements of GDC 1, 14, 15, and 31; and the requirements of 10 CFR Part 50 Appendix G.

5.4.2.2 Steam Generator Inservice Inspection

The staff reviewed DCD Tier 2 Section 5.4.2.5, “Steam Generator Inservice Inspection,” in accordance with Section 5.4.2.2, “Steam Generator Tube Inservice Inspection,” of the SRP to ensure periodic inspection and testing of critical areas and features to assess their structural and leaktight integrity. The SG ISI program is acceptable if it complies with the following:

- 10 CFR 50.55a, “Codes and Standards,” as it relates to periodic inspection and testing of the RCPB as detailed in ASME Code, Section XI; and
- GDC 32, “Inspection of Reactor Coolant Pressure Boundary,” as it relates to accessibility of SG tubes for periodic testing.

The guidelines for periodic inspection and testing of the SG tube portion of the RCPB are specified in the applicable standard technical specifications (STS). The applicable STS for Westinghouse plants are found in NUREG 1431, Volume 1, Revision 2, "Standard Technical Specifications Westinghouse Plants." TS 5.5.9, "Steam Generator (SG) Tube Surveillance Program," notes that the current licensing basis SG tube surveillance requirements (i.e., technical specification SG surveillance) shall be included in this TS. This statement cannot be applied directly to the AP1000. However, the most recent NRC position on SG tube surveillance requirements of operating Westinghouse plants is found in NUREG-0452, Revision 4, "Standard Technical Specifications (STS) Pressurized Water Reactors." Thus, the staff reviewed the AP1000 SG ISI for consistency with the TS criteria in NUREG-0452, Revision 4.

5.4.2.2.1 Summary of Technical Information

The AP1000 design allows for inspection of pressure boundary parts, including individual tubes. In addition, the preservice and inservice inspection of the AP1000 SGs is performed according to the ASME Code and complies with the requirements of 10 CFR 50.55a.

The design of the AP1000 SGs includes the following openings to provide access to both the primary and secondary sides of the SG:

- Four 45.7 cm (18 in) diameter manways, one access to each chamber of the reactor coolant channel head and two in the steam drum;

- Two 10.2 cm (4 in) diameter inspection openings at each end of tubelane and above the top tube support plate;

- Additional access to the tube bundle U-bend through the internal deck plate at the bottom of the primary separators; and

- Deck plate openings welded with hatch plates that are removable through grinding or gouging.

5.4.2.2.2 Staff Evaluation

The staff reviewed DCD Tier 2 Section 5.4.2.5, “Steam Generator Inservice Inspection,” in accordance with Section 5.4.2.2, “Steam Generator Tube Inservice Inspection,” of the SRP 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 leaktight integrity of the tubes as required by 10 CFR Part 50, Appendix A, Criterion 32.

As part of its evaluation, the staff reviewed the requirements for the SG Surveillance Program contained in TS 5.5.5. The most recent generic technical specifications for SG ISI is NUREG-0452, Revision 4, “Standard Technical Specifications (STS) for the applicant Pressurized Water Reactors.” The TS surveillance requirements for all domestic SGs are very similar, if not identical, to those in NUREG-0452, Revision 4. These requirements include selecting and sampling of tubes, inspection intervals, sample expansion criteria, actions to be taken in the event defects are identified, and reporting requirements.

The staff requested, in RAI 250.003, the applicant revise the SG Tube Surveillance Program TSs to be consistent with the surveillance requirements contained in the NUREG-0452, Revision 4, STS. In its response, the applicant provided a revision to AP1000 TS 5.5.5. The staff reviewed this response and did not find it entirely acceptable. The staff noted the following issues that need to be addressed further:

- The proposed AP1000 TS indicate that the provisions of TS SR 3.0.2 are applicable. However, the staff position, as articulated in GL 91-04, “Changes in Technical Specification Surveillance Intervals to Accommodate a 24-Month Fuel Cycle,” is that the surveillance interval extension in TS SR 3.0.2 does not apply to SG inspection intervals. This is based on the conditions defined in TS 5.5.5.3.a. and b. under which the surveillance interval for SG tube inspections may be extended to a maximum of once per 40-months. In addition, TS 5.5.5.3.b. addresses when the SG tube inspection frequency shall be increased to at least once per 20 months. Therefore, the response was not acceptable until the TS 5.5.5 was revised to indicate that the provisions of SR 3.0.2 are not applicable. Revision 4 of the Steam Generator Tube Surveillance Program is contained in TS 5.5.4. TS 5.5.4.3.d indicates that the provisions of Specification 3.0.2 do not apply for extending the frequency of performing inservice inspections as specified in TS 5.5.4.3.a and 5.5.4.3.b. The staff finds this acceptable since it excludes the application of TS SR 3.0.2 to extend steam generator surveillance frequencies.

- The proposed TS included Table 5.5.5-1 which defines SG sample selection and inspection. However, there is strategy in NUREG-0452, Revision 4 for determining the minimum number of SGs to be inspected during first, second, and subsequent ISIs depending on the preservice inspection performed. The applicant did not apply this strategy or any acceptable alternative. Therefore, the response is not acceptable until TS Table 5.5.5-1 is revised to reflect the preservice inspection. Revision 4 of the Steam Generator Tube Surveillance Program includes a note to TS Table 5.5.4-1 that indicates that all steam generators shall be inspected during the first inservice inspection if no preservice inspection was conducted. The staff finds this acceptable since this note provides an appropriate strategy for determining the minimum number of SGs to be inspected during the first, second, and subsequent ISIs depending upon the preservice inspection performed.

5.4.2.2.3 Conclusion

The staff concludes that the AP1000 SG ISI program is acceptable and meets GDC-32. This conclusion is based on the design permitting access for periodic inspection and testing of critical areas for structural and leakage integrity and on the SG tube surveillance program TS being consistent with the TS requirements for Westinghouse domestic PWRs.

5.4.2.3 Containment Bypass Resulting From Steam Generator Tube Rupture

In SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Design," the staff identified a containment performance issue where rupture of one or more SG tubes could lead to actuation of the SG safety relief valves, thereby creating the potential for a stuck open safety relief valve, and an unisolable LOCA, with discharge of primary system radioactive inventory outside the containment. SECY-93-087 specifies that applicants for design certification for passive or evolutionary PWRs assess design features to mitigate containment bypass leakage during steam generator tube rupture (SGTR) events. The staff also recommends certain design features for consideration that could mitigate the release associated with an SGTR:

- a highly reliable (closed loop) SG shell-side heat removal system that relies on natural circulation and stored water sources

- a system that returns some of the discharge from the SG relief valve back to the primary containment

- increased pressure capacity on the SG shell side with a corresponding increase in the safety valve setpoints

DCD Tier 2 Appendix 1B provides a risk-reduction evaluation of severe accident mitigation design alternatives (SAMDA) for the AP1000 design. A total of fifteen design alternatives were selected for evaluation, including the three design features mentioned above. Each design alternative is evaluated to determine whether its safety benefit from risk reduction outweighs the costs of incorporating it in the plant. The applicant concluded that, because of the small initial risk associated with the AP1000, none of these SAMDAs are cost beneficial.

In response to a staff RAI 440.043, the applicant discussed the AP1000 design features that mitigate or prevent SG safety valve challenges during an event of rupture of multiple steam generator tubes and thus reduce the chance of containment bypass following a SGTR. This issue is discussed below.

5.4.2.3.1 AP1000 SGTR Mitigation Design Features

The AP1000 design incorporates several automatic protection actions and PXS for mitigation of the consequences of SGTR events. The automatic protection actions include reactor trip, actuation of the PXS, the RCP trip, termination of pressurizer heater operation, and isolation of the CVS flow and the SUFS. These protective actions result in automatic cooldown and depressurization of the RCS, termination of the break flow, stabilization of the RCS, prevention of SG overfill, and termination of release of steam to the atmosphere to minimize offsite radiation. The AP1000 PXS responds to the SGTR events by automatically terminating the loss of reactor coolant without actuating the ADS or overfilling the SG.

In the scenario of a SGTR, continued loss of RCS inventory to the SG secondary side through the ruptured tubes leads to a reactor trip on a low pressurizer pressure or over-temperature delta-T signal, and also causes the turbine trip. The core makeup tanks (CMTs) automatically actuate on a safeguards signal or low pressurizer level. The PRHR HX automatically actuates on the CMT actuation signal, high pressurizer pressure, or low SG level. The PRHR HX acts to reduce the RCS pressure below the pressure of the secondary system and isolate the break flow to the faulted SG. The heat is removed from the RCS through the PRHR instead of the intact SG power operated relief valve (PORV) to stop the leak to the faulted SG. The CMTs provide heat removal and coolant inventory makeup for shrinkage in the RCS. During a SGTR transient, the CMTs inject water in the recirculation mode, exchanging cold borated water for hot RCS water. Because the CMTs do not drain during recirculation injection, the CMT level remains above the ADS actuation setpoint and, therefore, the ADS is not actuated.

The AP1000 also provides additional defense-in-depth to mitigate multiple SGTRs. The active, non-safety-related systems can be used to mitigate the multiple SGTRs. The intact SG PORV is used to control the RCS pressure and isolate the break. The CVS auxiliary spray is used to reduce the RCS pressure to allow the pumped RNS to provide borated makeup flow to the system until the break is isolated. In case of failure of both the active non-safety-related mitigation and the safety-related PRHR HX mitigation, the AP1000 provides another defense-in-depth method of mitigation. This method uses the ADS and passive safety injection.

On the secondary side, a PORV is installed on the outlet piping from each SG to provide a means for plant cooldown by discharging steam to the atmosphere when the turbine bypass system is not available. The PORV automatically opens to release steam when the steam pressure exceeds its pre-determined set pressure, which is below the main steam safety valve (MSSV) set pressure; and will close and reseat at a pressure below the opening setpoint as the steam pressure decreases. A block valve, upstream of the PORV, with a safety-related operator closes automatically on low steam pressure to terminate steam release in the event of a PORV stuck open.

In the event that the PORV fails to open during a SGTR, this could result in the opening of the MSSVs. Because of the automatic SG overfill protection, which trips the CVS and SUFS flow, the SG is not overfilled and only steam is released through the MSSV. If the MSSV is assumed to fail open, the PRHR HX will not be able to terminate the loss of reactor coolant. The loss of primary system coolant through the SG tube and the stuck open valve eventually causes the CMTs to drain to the ADS actuation setpoint. Actuation of ADS depressurizes the RCS in a controlled, staged manner, and eventually, allows for gravity injection from the IRWST and the containment recirculation as the IRWST empties. The passive injection systems, CMTs, accumulators and IRWST gravity injection provide inventory makeup and boration throughout the depressurization. The core remains covered and cooled through out the sequence, and the plant achieves a safe, stable configuration without a release of fission products from the fuel matrix. Preventing the release of fission products from the core mitigates the beyond-design-basis containment bypass.

5.4.2.3.2 MSGTR Analysis

In DCD Tier 2 Section 15.6.3, “Steam Generator Tube Rupture,” the applicant provides the design-basis analysis for a single-tube SGTR. The design-basis analysis assumed no operator actions, and assumed a PORV fails to reseat after it opens with continued release through the PORV until the block valve closure at low steamline pressure. The results showed no fuel failure, no SG overfilling, and the resulting offsite radiological doses are within the dose acceptance limits.

In response to RAI 440.043, the applicant provided an analysis of the beyond-design-basis events of a multiple-tube rupture of five tubes. The intent of the analysis was to demonstrate the capability of the safety systems and automatic actions for mitigation of the multiple-tube rupture events. No operator actions were modeled in the analysis. The analysis was performed with the MAAP4 code. MAAP4 is a fast-running thermal-hydraulic computer code designed for severe accident analysis and was chosen by the applicant for the AP1000 probabilistic risk assessment (PRA) evaluation, as well as the evaluation of MSGTR. The staff’s evaluation of the use of MAAP4 for the AP1000 PRA evaluation is discussed in Chapter 19 of this report.

Two cases of five-tube rupture were analyzed using the MAAP4 accident analysis code: (1) multiple SGTR with passive system response, and (2) multiple SGTR with failed open MSSV. For both cases the accident is initiated by the simultaneous double-ended failure of five cold side tubes at the top of the tubesheet. Startup feedwater system and the CVS are conservatively assumed to function because they tend to make the accident worse.

Case 1 is a passive system mitigation case with PRHR heat exchanger operation. The SUFS controls operate normally and throttle the startup feedwater based on the normal SG operating level. The CVS provides RCS makeup until it is isolated on a High-2 SG narrow range level. The results show that the faulted SG does not overfill and the safety valves do not open. Therefore, bypass does not occur. Throughout the events, the core makeup tanks inject water in the recirculation mode, exchanging cold borated water for the hot reactor coolant. The CMTs do not drain, and therefore, the ADS does not actuate.

Case 2 is a passive system mitigation case with minimum PRHR heat removal. The SUFS controls is assumed to malfunction such that the SUFS flow continues when the SG level increases above the normal level until it is isolated on a High-2 SG narrow range level. The CVS provides RCS makeup until it is isolated on a High-2 SG narrow range level. The secondary system PORV is also conservatively assumed to not open. The combination of the low PRHR heat removal and the high SG level control causes the faulted SG pressure to exceed the MSSV lowest setpoint. When the MSSV opens, it is assumed to stick open although the SG is not predicted to overfill. Therefore, the SGTR scenario turns into a small break LOCA. Continued loss of coolant through the ruptured tubes and the stuck-open MSSV eventually leads to the voiding of the RCS, the draining of the CMT, and the actuation of ADS. The RCS is rapidly depressurized, which results in the actuation of the IRWST and eventual containment recirculation. The core remains covered and cooled. The maximum total release would be limited to the initial activity in the RCS. The results show that the loss of coolant from the RCS eventually drains the CMTs to the ADS actuation setpoint. The RCS depressurizes and gravity injection begins. The core remains covered and cooled, thus no significant fission product release occurs.

5.4.2.3.3 Conclusion

The AP1000 design has unique features for mitigation of the SGTR relative to the conventional PWRs. The analysis shows that the PORV will automatically open to release steam and reseat within a very short time. Throughout the accident, the core remains covered without voiding, and the SG is not overfilled. If the PORV fails to open, the MSSV will open and close within a short time. Because of the automatic overfill protection, the SG is not overfilled, and the MSSV will release steam only. In the extremely unlikely event of failure of the PORV to open coincident with failure of the MSSV to reseat, an unisolable small-break LOCA scenario occurs with release to the atmosphere. In this event, continued steam release and loss of reactor coolant through the ruptured tubes will result in draining of the CMTs. The ADS will be actuated as the CMT level falls below the ADS actuation setpoint. Rapid depressurization of the RCS eventually results in the gravity injection from the IRWST, as well as the containment recirculation as the IRWST empties. Eventually, the break flow through the ruptured tubes stops. The analysis indicates that, throughout the entire accident, the core remains covered and cooled without core damage.

The staff concludes that there is reasonable assurance that the unique design features of the AP1000 are capable of mitigating the consequences of a multiple tube rupture as specified by SECY-93-087. In the extremely unlikely event of PORV failure to open coincident with a stuck-open MSSV, no core damage will occur, and the total release to the atmosphere would be limited to the initial activity of the RCS. The staff concludes that there is reasonable assurance that the containment bypass as a result of multiple tube rupture poses no undue threat to the public health and safety, and the AP1000 design satisfies SECY-93-087.

5.4.3 RCS Piping

The RCS piping includes those sections of RCS hot leg and cold leg piping interconnecting the RV, SGs, and RCPs. It also includes piping connected to the reactor coolant loop piping and primary components. The RCS piping accommodates the system pressures and temperatures attained under all expected modes of plant operation or anticipated system interactions. The piping in the AP1000 RCS is AP1000 equipment Class A and fabricated according to ASME Code, Section III, Class 1 requirements, consistent with the requirements of 10 CFR 50.55a(c)(1). Lines with a 0.97 cm (3/8-inch) or less flow-restricting orifice qualify as AP1000 equipment Class B and are designed and fabricated with ASME Code, Section III, Class 2 requirements. Because the AP1000 CVS provides sufficient makeup of the reactor coolant in the event of a failure of a small line of 0.97 cm (3/8 inch) or less, Class B classification of small piping exempted from ASME Code, Section III, Class 1 requirements in accordance with the exception permitted in accordance with 10 CFR 50.55a(c)(2)(i).

In DCD Tier 2 Section 5.4.3.2.1, “Piping Elements,” the applicant provides a list of the piping connected to the RCS. The detailed RCS piping and instrument diagram is shown in DCD Tier 2 Figure 5.1-5. It includes the pressurizer surge, spray, and auxiliary spray lines; pressurizer safety valves; the ADS with the first three stages connected to the pressurizer and the fourth stage connected to the hot legs; the reactor system head vent line; the accumulator lines; the core makeup tank cold leg balance lines and injection lines; the PRHR system; the IRWST injection lines; the RNS pump suction line and discharge line; the CVS purification return lines to the SG channel head and the pressurizer spray; the CVS purification intake line from one RCS cold leg; and the drain, sample, and instrumentation lines. The RCS pressure boundary of these connecting lines start from their respective connections to the RCS and end at the second normally-closed isolation valves or check valves in the respective lines, or the code safety valves, as defined in 10 CFR 50.2. All the RCS-connecting piping that constitutes the RCPB is designed to meet the ASME Code Section III requirements (with one exception discussed below).

One exception to meeting the ASME Code Section III requirements is in the CVS. As discussed in DCD Tier 2 Section 3.9.6, the safety-related classification of the CVS ends at the third isolation valve in the purification loop intake line. The remainder of the purification subsystem of the CVS downstream of the third isolation valve inside containment consists of non-safety, Quality Group D components. Because the CVS purification intake line contains three isolation valves (CVS-PL-V001, -V002, -V003) that are maintained open during normal operation, the RCPB extends to the containment isolation valves of the CVS. However, because the portion of the CVS downstream of the three isolation valves can be isolated from the RCS, this portion need not be designed to ASME Class 1 in accordance with the exception criterion of 10 CFR 50.55a(c)(2)(ii). Regulatory Position C of RG 1.26 specifies the portion of RCPB that meets the exception criteria of 10 CFR 50.55a(c)(2) consist of safety-related quality Group B or C components. However, DCD Tier 2 Section 5.2.1.3 describes many design enhancements that have been added to the Class D portion of the CVS, such as use of three isolation valves of Class 1 design in the purification loop intake line and seismic design of piping in the Class D portion. These design enhancements result in an alternate design that provides an acceptable level of quality and safety. As discussed in Section 5.2.1 of this report, the staff evaluation has found this alternative design to be acceptable.

To minimize the potential for thermal stratification that could increase cyclic stresses and fatigue usage, the pressurizer surge line is specifically designed with various degrees of continuous slope up from the hot leg connection to the pressurizer, as shown in DCD Tier 2 Figure 5.4-4. The surge line is also instrumented with strap-on resistance temperature detectors at three locations, one on the vertical section of pipe directly under the pressurizer and the other two on the top and bottom of the pipe at the same diameter on a more horizontal section of pipe near the pressurizer, to monitor the temperature for indication of thermal stratification.

In DCD Tier 2 Table 5.4.7, the applicant lists the principal design data of the RCS piping, such as pipe sizes, thickness, and design pressure and temperature of the major RCS loop piping, pressurizer surge line, and other reactor coolant branch lines. All of the RCS piping and branch lines have a design pressure of 17.24 MPa (2485 psig). The loading combinations, stress limits, and analytical methods for the structural evaluation of the RCS piping and supports for design conditions, normal conditions, anticipated transients, and postulated accident conditions are discussed in DCD Tier 2 Section 3.9.3. The RCS piping construction is subject to a quality assurance program with the required testing specified in DCD Tier 2 Table 5.4-8, and meeting requirements established by the ASME Code. The staff finds that the RCS-connecting piping that constitutes the RCPB is designed to meet the ASME Code Section III requirements and, therefore, is acceptable.

The consequences of the RCS piping breaks, including postulated cold leg double-ended guillotine breaks, are analyzed in DCD Tier 2 Section 15.6 to demonstrate their compliance with the respective acceptance criteria. For those low-pressure systems and components outside the containment with connections directly or indirectly to the RCS, SECY-93-087 specifies that those low-pressure portions be designed with the ultimate rupture strength at least equal to the full RCS operating pressure. This is addressed in generic safety issue GSI 105, "Interfacing System LOCA for LWR," in Chapter 20 of this report, where the staff finds the design of the low-pressure piping to be acceptable.

5.4.4 Main Steamline Flow Restriction

Each SG contains a flow restrictor in its steam outlet nozzle. The flow restrictor consists of seven venturi inserts welded to the SG outlet nozzle forging. The inserts are arranged with one venturi at the centerline of the outlet nozzle, with the other six equally spaced around it. The steamline flow restrictor limits the steam flow rate from the secondary system to the choked flow of the venturi in the unlikely event of a break in the main steamline. This flow restriction is needed to perform the following functions:

- limit rapid rise in containment pressure
- limit the reactor cooldown rate within acceptable limits
- reduce thrust forces on the main steamline piping
- limit pressure differentials on internal SG components, particularly the SG tube support plates

The steamline flow restrictor is configured to minimize the unrecovered pressure loss across the restrictor during normal operation. The design data of the flow restrictors are specified in

DCD Tier 2 Table 10.3.2-1. The throat area of each venturi is 0.0186 m^2 (0.2 ft^2). With seven venturis in a flow restrictor, the equivalent throat area of the SG outlet is 0.13 m^2 (1.4 ft^2). The resultant pressure drop through the restrictor at 100-percent steam design flow rate of 3.40E+06 kg/hr (7.49E+06 lbm/hr) is approximately 55.2 kPa (8 psi).

The staff has reviewed the safety analysis of the design-basis event of steam system piping failure described in DCD Tier 2 Section 15.1.5. The analysis uses an effective nozzle flow area of 0.13 m^2 (1.4 ft^2) of the main steamline flow restrictors for each SG. The analysis results show that the acceptance criteria specified in SRP Section 15.1.5 are met. Therefore, the SG flow restrictor with an equivalent throat area of 0.13 m^2 (1.4 ft^2). Also, Item 8(b)(ii) in ITAAC DCD Tier 1 Table 2.2.4-4 requires a verification that the installed flow-limiting orifice within the SG main steamline discharge nozzle does not exceed 0.13 m^2 (1.4 ft^2). This is consistent with the safety analysis value and, therefore, is acceptable.

5.4.5 Pressurizer

The pressurizer is a vertical, cylindrical vessel having hemispherical top and bottom heads, and containing saturated water and vapor. The pressurizer is connected from its bottom to one of the RCS hot legs through a surge line, which allows continuous coolant volume and pressure adjustments between the RCS and the pressurizer. The pressurizer, with the liquid and vapor maintained in equilibrium under saturated conditions, controls the RCS pressure during steady-state operations and transients. Major components of the pressurizer include the pressurizer spray system, electrical heaters, code safety valves, ADS valves, and the surge line. The pressurizer is the principal component of the RCS pressure control equipment. It also accommodates changes in RCS liquid volume, and limits the changes in RCS pressure as a result of reactor coolant temperature changes during all modes of plant operation. The pressurizer also serves as a convenient source of reactor coolant makeup for minor RCS leakage, and is the initial source of water to keep the RCS full in the event of a small-break LOCA in the RCS piping.

During steady-state operation at 100-percent power, approximately 50 percent of the pressurizer volume is water and 50 percent is steam. Electric immersion heaters in the bottom of the vessel keep the pressurizer contents at saturation temperature. A small continuous spray flow is provided through a manual bypass valve around each power-operated spray valve to minimize the boron concentration difference between the liquid in the pressurizer and the reactor coolant. During transient events, pressure increases, caused by insurge of reactor coolant, are mitigated by the pressurizer spray such that the high pressurizer pressure reactor trip setpoint is not reached. Conversely, during pressure decreases, caused by outsurge of reactor coolant, water-to-steam flashing and automatic heater operation keep the RCS pressure above the low pressurizer pressure reactor trip setpoint. The heaters are also energized on the high water level during insurge to heat the subcooled surge water entering the pressurizer from the reactor coolant loop. The power to the pressurizer heaters are automatically blocked upon actuation of the core makeup tanks (see DCD Tier 2 Section 7.3.1.2.3). This action prevents the heaters from attempting to repressurize the RCS during passive safety injection and, therefore, reduces the potential for SG overfill for a SGTR event. This pressurizer heater trip function is credited as a backup protection in the design-basis analyses of a loss of feedwater event, and a SGTR event described in DCD Tier 2

Sections 15.2.7 and 15.6.3, respectively. In accordance with the technical specification screen criteria specified in 10 CFR 50.36, the pressurizer heater trip function is specified in DCD Tier 2 Table 3.3.2-1, Engineered Safeguards Actuation System Instrumentation, and subject to AP1000 TS LCO 3.3.2 and associated surveillance requirements.

The pressurizer safety valves provide overpressure protection of the RCS. This is discussed in Section 5.2.2 of this report. In addition, the pressurizer provides for high point venting of noncondensible gases from the RCS by remote manual operation of the first-stage ADS valves to vent the gas accumulated in the pressurizer following an accident. This is discussed in Section 5.4.12 of this report.

The AP1000 pressurizer has an internal volume of 59.5 m^3 (2100 ft^3), which is approximately 40 percent more volume than the pressurizers for current PWRs of similar thermal power level. This increased pressurizer volume provides plant operating flexibility, minimizes challenges to the safety/relief valves, and eliminates the need for PORVs. DCD Tier 2 Section 5.4.5.1 provides the design bases on the sizing of the AP1000 pressurizer to meet the following conditions without the need for a PORV:

- The combined saturated water volume and steam expansion volume is sufficient to provide the desired pressure response to system volume changes.

- The water volume is sufficient to prevent (1) a reactor trip during a step-load increase of 10 percent of full power, with automatic reactor control, and (2) uncovering the heaters following reactor trip and turbine trip, with normal operation of control systems and no failures of nuclear steam supply systems.

- The steam volume is large enough to (1) accommodate the surge resulting from a step load reduction from 100-percent power to house loads without reactor trip, assuming normal operation of control systems, and (2) prevent water relief through the safety valves following a complete loss of load with the high-water level initiating a reactor trip, without steam dump.

- A low pressurizer pressure safeguard actuation ("S") signal will not be activated because of a reactor trip and turbine trip, assuming normal operation of control and makeup systems and no failures of the nuclear steam supply systems.

The pressurizer performance during AOO and postulated accidents is reviewed as part of the design-basis accident analysis review discussed in Chapter 15 of this report. The results of the analyses demonstrate that the acceptance criteria specified in SRP Chapter 15 for the transients and accidents are met: i.e., the DNBR limit is met for all AOOs, the RCS pressure is within 110 percent of the RCS design pressure for the pressurization events, and the acceptance criteria of 10 CFR 50.46 are met for LOCAs. Therefore, the staff finds the pressurizer design to be acceptable.

5.4.6 Automatic Depressurization System Valves

The ADS valves are part of the RCS and interface with the PXS. The ADS is divided into two groups and four depressurization stages, with a total of 20 valves. These stages connect to the RCS at different locations. The first, second, and third stage valves are included as part of the PSARV module, which is connected to nozzles on top of the pressurizer. The two groups are on different elevations separated by a steel plate. The first stage ADS valves in each group are two motor-operated 10.2-cm (4-in.) valves in series. The second and third stage ADS valves each have two motor-operated 20.3-cm (8-in.) valves in series. The fourth stage ADS valves are 35.56-cm (14-in.) squib valves arranged in series with normally open, dc-powered, motor-operated valves. The outlets of the first three stages in each group are combined into a common discharge line to the IRWST. This discharge line has a vacuum breaker to help prevent water hammer following ADS operation by limiting the pressure reduction caused by steam condensation in the discharge line, and thus limiting the potential for liquid backflow from the IRWST. The fourth stage ADS valves connect to the RCS hot legs, and are interlocked so that they cannot be opened until RCS pressure has been substantially reduced.

DCD Tier 2 Section 6.3 discusses the operation of the PXS. DCD Tier 2 Section 7.3 describes the actuation logic and setpoints for opening various stages of the ADS valves. Opening of the ADS valves is necessary for the PXS to function as required to provide emergency core cooling following postulated accident conditions. The first stage valves may also be used to remove noncondensible gases from the steam space of the pressurizer, if necessary, following an accident.

The ADS functional performance (as part of the PXS performance) is evaluated in Chapter 6.3 of this report. The safety analyses of various design-basis accidents are evaluated in Chapter 15 of this report. The analysis results of design-basis accidents such as small-break LOCAs described in DCD Tier 2 Section 15.6.5 demonstrate that, with the ADS design and the passive core cooling system, the acceptance criteria specified in 10 CFR 50.46 are met. Therefore, the ADS design is acceptable.

5.4.7 Normal Residual Heat Removal System

The AP1000's normal residual heat removal system (RNS) is a non-safety-related system and is not required to operate to mitigate design-basis events. However, the RNS does perform the following safety-related functions:

- containment isolation of RNS lines penetrating containment using containment isolation valves according to the criteria specified in DCD Tier 2 Section 6.2.3

- preservation of the RCS pressure boundary integrity using pressure isolation valves according to the criteria specified in Subsection DCD Tier 2 Section 5.4.8

  provide a flow path for long-term post-accident makeup to the containment inventory

5.4.7.1 RNS Design Bases

The RNS performs the following non-safety-related functions. Their design bases are also described below.

- Shutdown Heat Removal

  The RNS is designed to remove both residual and sensible heat from the core and the RCS during shutdown operations, with the capability to (1) reduce the temperature of the RCS from 176.7°C (350°F) to 51.7°C (125°F) within 96 hours after shutdown during the second phase of plant cooldown (after the initial RCS cooldown is accomplished by the main steam system); and (2) maintain the reactor coolant temperature at or below 51.7°C (125°F) for the entire plant shutdown.

- Shutdown Purification

  The RNS is designed to provide RCS and refueling cavity purification flow to the CVS during refueling operations, with the purification flow rate consistent with that specified in DCD Tier 2 Table 9.3.6-1.

- In-Containment Refueling Water Storage Tank Cooling

  The RNS is designed to provide cooling for the IRWST during operation of the PRHRHX or during normal plant operations, when required. The RNS is designed to be manually initiated by the operator. During normal operation, the RNS with both subsystems of RNS pumps and heat exchangers available will limit the IRWST water temperature to no greater than 48.9 °C (120 °F). During extended operation of the PRHRHX, the RNS will limit the IRWST water temperature to less than the boiling temperature.

- Low-Pressure RCS Makeup and Cooling

  The RNS is designed to be manually initiated by the operator following the actuation of the ADS. The RNS provides low-pressure makeup from the cask loading pit to the RCS (once the pressure in the RCS falls below the shutoff head of the RNS pumps), and thus provides additional margin for core cooling.

- Low-Temperature Overpressure Protection

  The RNS is designed to provide LTOP for the RCS during refueling, startup, and shutdown operations to limit the RCS pressure within the limits specified in 10 CFR Part 50, Appendix G.

- Spent Fuel Pool Cooling

  The RNS is designed to have the capability to supplement or take over the cooling of the spent fuel pool when it is not needed for normal shutdown cooling.

5.4.7.2 RNS Design and Components

In DCD Tier 2 Section 5.4.7.2, the applicant describes the AP1000 RNS design, including specific design features to address the concerns related to mid-loop operation and interfacing system LOCA, respectively. The RNS consists of two mechanical trains of equipment; each consists of one pump and one heat exchanger. The two trains share a common suction line from the RCS and a common discharge header. The RNS is also comprised of piping, valves, and instrumentation necessary for system operation, as shown in DCD Tier 2 Figure 5.4-7.

Inside containment, the RNS suction header is connected to an RCS hot leg with a single step-nozzle connection. The suction header is comprised of two parallel lines with two sets of two normally closed motor-operated isolation valves in series for single failure consideration. These isolation valves comprise the RCS pressure boundary. The two lines are connected to a common suction header. This suction alignment is for reactor cooling during normal shutdown operation. A single line from the cask loading pit is connected to the suction header to provide a flow path for low-pressure makeup of the RCS.

Once outside containment, the suction header contains a single, normally closed, motor-operated isolation valve. Downstream of the isolation valve, the header branches into two separate lines, one to each pump. In each branch line is a normally open manual isolation valve upstream of the RNS pumps for pump maintenance.

The discharge of each RNS pump is routed directly to its respective RNS heat exchanger. A mini-flow line, which contains an orifice and is sized for a sufficient pump flow rate when the pressure in the RCS is above the RNS pump shutoff head, is routed from downstream of the heat exchanger to upstream of the pump suction. The outlet of each heat exchanger is routed to the common discharge header, which contains a normally closed motor-operated isolation valve before penetrating the containment.

Once inside containment, the common discharge header contains a check valve that acts as a containment isolation valve. Downstream of the check valve, the discharge header branches into two lines routed to the direct vessel injection (DVI) lines. These branch lines each contain two check valves in series that comprise the RCS pressure boundary. A line is branched from the common header to the CVS demineralizers for shutdown purification of the RCS. Another line is routed from the discharge header to the IRWST for cooling of the tank.

The RNS contains a single safety/relief valve, located off the RNS suction header inside containment that discharges to the IRWST. This relief valve is utilized for LTOP of the RCS.

In DCD Tier 2 Table 3.2-3, the applicant provides the safety classification and seismic categories of the RNS components. The portions of the RNS piping and components from the RCS up to and including the outer RNS suction isolation valve or outer RNS discharge check valve constitute the RCPB, and are designed with safety Class A requirements. The RNS RCPB valves include V001A, V001B, V002A, V002B, V015A, V015B, V017A, and V017B. DCD Tier 2 Section 5.4.8 states that these valves are manufactured to the requirements of ASME Code Class I. The portions from the RCPB to the containment isolation valves outside the containment are designed to safety Class B requirements. The RNS containment isolation valves include V002A, V002B, V011, V012, V013, V021, V022, V023, and V061. These valves (except for RCPB valves V002A and V002B which are ASME Code Class 1) are manufactured to ASME Code Class 2 requirements. The inside containment portions extending to the containment isolation valves outside containment are designed for full RCS pressure. The system piping and components outside containment, including the pumps, valves, and heat exchangers, are safety Class C, and have a design pressure and temperature such that full RCS pressure is below the ultimate rupture strength of the piping.

The design classifications of the RNS components discussed above comply with GDC 1 which specifies that SSCs important to safety shall be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed. The whole RNS system, except for the heat exchanger shell vents is designed for seismic Category I for pressure retention. This complies with GDC 2 which specifies the SSCs important to safety shall be designed to withstand the effects of natural phenomena, such as earthquakes, without loss of capability to perform their safety functions. This also complies with RG 1.29 which specifies that the SSCs that constitute the RCPB, are designated seismic category I and should be designed to withstand the effects of the SSE and remain functional. The staff finds that the RNS design for performing its safety-related functions of containment isolation and preservation of the RCPB integrity to be acceptable.

5.4.7.3 Shutdown Operation Design Features

In SECY-93-087, the staff specified that passive plants must have a reliable means of maintaining decay heat removal capability during all phases of shutdown activities, including refueling and maintenance. The staff review of the AP1000 design with respect to shutdown operations is based on the applicant’s systematic assessment of shutdown operation concerns identified in NUREG-1449, “Shutdown and Low-Power Operations at Commercial Nuclear Power Plants in the United States,” which encompasses mid-loop operation. This assessment is provided in DCD Tier 2 Appendix 19E, “Shutdown Evaluation.” The staff evaluation of the shutdown operation issues is addressed in Section 19.3 of this report. This section describes the RNS design features to address NUREG-1449 and Generic Letter 88-17 regarding mid-loop operation.

- Loop Piping Offset

  The levels of the RCS hot legs and cold legs are offset vertically with the hot leg nozzles 0.445 m (17.5 in) below the cold leg nozzles so that the RCS can be drained with the hot leg level remaining much higher than traditional designs for venting of the SGs prior to nozzle dam insertion. Furthermore, this loop piping offset allows a RCP to be replaced without removing a full core.

- Step-Nozzle Connection

  The RNS employs a step-nozzle connection to the RCS hot leg to minimize the likelihood of air ingestion into the RNS pumps during RCS mid-loop operations. The step-nozzle connection substantially lowers the RCS hot leg level at which a vortex occurs in the RNS pump suction line as a result of the lower fluid velocity in the hot leg nozzle.

- Self-Venting Suction Line

  The RNS pump suction line slopes continuously upward from the pump to the RCS hot leg with no local high points (where air could collect and cause a loss of RNS capability). This self-venting suction line will refill after a pump trip. The pumps can be immediately restarted once an adequate level is reestablished in the hot leg.

- Hot Leg Level Instrumentation

  The AP1000 RCS contains level instrumentation in each hot leg with a readout in the MCR. Alarms are also provided to alert the operator when the RCS level is approaching a low level. Additionally, the isolation valves in the RCS drain line are interlocked to close on a low RCS level during shutdown operations.

- Reactor Vessel Outlet Temperature

  Each hot leg is provided with a wide-range thermowell-mounted resistance temperature detector for measurement of reactor coolant fluid temperature in the hot leg when in reduced inventory conditions.

- ADS Valves

  The ADS valves of the first three stages are required to be open to provide a vent path to prevent RCS pressurization whenever the CMTs are blocked during shutdown conditions while the RV upper internals are in place.

- Other Features for Shutdown Operations

  The RNS contains instrumentation to monitor and control system performance. System parameters necessary for RNS system operation that are monitored in the MCR include the following instrumentation which also allow mid-loop operations to be performed from the MCR:

  - RNS pump flow discharge pressure
  - RNS heat exchanger inlet and outlet temperatures
  - RNS heat exchanger outlet flow and bypass flow
  - RCS wide-range pressure

The staff's evaluation of shutdown operations and AP1000 design features to support shutdown operations is based on DCD Tier 2 Appendix 19 E, and is provided in Section 19.3 of this report. The staff has concluded that the AP1000 design features which support shutdown operations, including those of the RNS, is acceptable.

5.4.7.4 Interfacing-Systems LOCA Design Features

In SECY-90-016 "Evolutionary Light Water Reactor (LWR) Certification Issues and Their Relationship to current Regulatory Requirements", as well as SECY-93-087, the staff specified that ALWR designs should reduce the possibility of a LOCA outside containment by designing, to the extent practicable, all systems and subsystems connected to the RCS to an ultimate rupture strength at least equal to full RCS pressure. SECY-90-016 "Evolutionary Light Water Reactor (LWR) Certification Issues and Their Relationship to Current Regulatory Requirements" also specified guidance for those systems that have not been designed to withstand full RCS pressure.

DCD Tier 2 Section 5.4.7.2.2 discusses the AP1000 design features to address the inter-system LOCA (ISLOCA). Section 3.1 of WCAP-15993, "Evaluation of the AP1000 Conformance to Inter-System Loss-of-Coolant Accident Acceptance Criteria," issued November 2002, provides a design evaluation of the RNS for conformance to the ISLOCA acceptance criteria. The AP1000 RNS design contains the following ISLOCA features:

- Increased Design Pressure

  The portions of the RNS from the RCS up to and including the containment isolation valves outside containment are designed to the full RCS operating pressure. The portions of the system downstream of the suction line containment isolation valve and upstream of the discharge line containment isolation valve, including the pumps, valves, flanges, fittings, and heat exchangers, have a design pressure of 6.21 MPa (900 psi), approximately 40 percent of the RCS operating pressure, so that its ultimate rupture strength is not less than the operating pressure of the RCS. An exception to this is the pump seal which does not meet this criterion. This is discussed in the staff evaluation of the ISLOCA in Chapter 20 of this report.

- Additional RCS Isolation Valve

  The RNS contains an additional isolation valve in the pump suction line from the RCS. This motor-operated containment isolation valve is designed to full RCS pressure, and provides an additional barrier between the RCS and lower pressure portions of the RNS.

- RNS Relief Valve

  The RNS relief valve is connected to the RNS pump suction line inside containment to provide LTOP of the RCS. It is connected to the high-pressure portion of the pump suction line; as such, it will reduce the risk of overpressurizing the low-pressure portions of the system.

- Features Preventing Inadvertent Opening of Isolation Valves

  The motor-operated isolation valves connected to the RCS hot leg are interlocked to prevent their opening at RCS pressures above 3.21 MPa (450 psig). These valves are also interlocked to prevent their being opened unless the isolation valve from the IRWST to the RNS pump suction header is closed. In addition, the power to these valves is administratively blocked at the valve motor control center to prevent their inadvertent opening.

- RCS Pressure Indication and High Alarm

  The RNS contains an instrumentation channel that indicates pressure in each RNS pump suction line. A high pressure alarm is provided in the MCR to alert the operator to a condition of rising RCS pressure that could eventually exceed the design pressure of the RNS.

The staff evaluation of the interfacing system LOCA is addressed in the discussion of GSI 105, "Interfacing System LOCA at LWRs," in Chapter 20 of this report. The staff finds that the RNS design features meet the ISLOCA specifications in SECY-90-016 and SECY-93-087.

5.4.7.5 RNS System Operation and Performance

In DCD Tier 2 Section 5.4.7.4, the applicant provides a general description of the RNS operation for the pertinent phases of plant operation (plant startup, plant cooldown, refueling, accident recovery operations, and spent fuel pool cooling). System operations are controlled and monitored from the MCR, even during mid-loop operations.

For accident recovery operations, the RNS can be employed to provide low-pressure RCS makeup upon actuation of ADS. The staff reviewed the AP600 emergency response guidelines (ERG), which are applicable to the AP1000, to evaluate a possible system interaction, caused by the RNS operation, which may adversely affect the performance of the passive safety systems. For post LOCA recovery, the ERGs instruct the operators to actuate the RNS and align the RNS pumps to take suction from the IRWST and inject into the RCS to provide additional core cooling if the CMT level begins to decrease. Operation in this mode provides additional injection flow to the RCS, thereby providing additional core cooling margin. Because the RNS pumps are aligned to inject into the RCS via the DVI lines, which are also the injection paths of the CMTs and IRWST, these shared connections can result in interactions with the PXS.

An evaluation of the potential for adverse system interactions of the RNS and the PXS is provided in WCAP-15992, Revision 1, "AP1000 Adverse System Interactions Evaluation Report." For a small break LOCA, the operation of the RNS pumps in the injection mode increases the backpressure on the CMT and prevents the CMT from draining to the ADS-4 actuation setpoint, thereby preventing the ADS-4 valves from actuating. Operation of the RNS pumps will refill the RCS and recover the water level in the pressurizer without the need to actuate ADS-4 valves. For a large break LOCA, the capacity of the RNS will not be sufficient to prevent the CMT from draining, and subsequent ADS-4 actuation. Therefore, RNS operation has no adverse impact.

However, because the RNS is aligned to the IRWST following drain down of the cask loading pit, continued long-term operation of the RNS pumps could result in the IRWST draining at a faster rate than if the RNS pumps were not operating. This is not a concern as long as the

RNS pumps continue to operate, and therefore provide higher injection rate than the gravity injection from the IRWST or the containment recirculation path. If the RNS pumps were to fail, the impact to post-accident RCS makeup by gravity injection from the IRWST and containment recirculation would be insignificant because of the use of the cask loading pit as a source of RCS makeup. For the AP1000 design, the RNS is initially aligned to the cask loading pit. The RNS aligns to the IRWST after the drain down of the cask loading pit. This operation delays the draining of the IRWST and extends the time at which containment recirculation is initiated, so that the core decay heat level is reduced at the time of containment recirculation initiation. In addition, the use of the cask loading pit provides additional post-accident water inventory, and thus increases the containment floodup level, which improves the driving head available for containment recirculation flow. Based on the discussion above, this system interaction is found to be acceptable.

5.4.7.6 Design Evaluation

The staff review of the RNS design is for compliance with the following requirements:

- GDC 1, as it relates to the quality standards of the SSCs important to safety
- GDC 2, as it relates to the seismic design of the SSCs important to safety by withstanding an SSE and remaining functional, with acceptability based on meeting RG 1.29
- GDC 4, as it relates to the dynamic effects associated with flow instability and loads
- GDC 5, as it relates to SSCs important to safety being prohibited from being shared among nuclear power units
- GDC 19, as it relates to a control room being provided from which actions can be taken to operate the nuclear power unit safely
- GDC 34, as it relates to the ability of the residual heat removal system to transfer fission product decay heat

The RNS is designed for a single nuclear power unit, and is not designed to be shared between units. The RCPB portion of the RNS is designed as safety Class A, and the containment isolation valves of the RNS are designed as Safety Class B, the remaining portions are designed as safety Class C. The pressure boundary is classified as seismic Category I and is designed to withstand a safe shutdown earthquake for pressure retention. The RNS is operated from the MCR. Also, the high energy piping of the RNS (i.e., the RNS suction and discharge portions that constitute the RCPB) are subject to LBB criteria for protection against dynamic effects. This is identified in Table 3B-1 and DCD Tier 2 Figure 3E-2. Therefore, the RNS meets GDC 1, 2, 4, 5, and 19. Because the RNS is not designed to provide safety-related decay heat removal function for mitigation of design-basis events, the safety-related heat removal function of GDC 34 is complied with by the safety-related PRHRHX. The evaluation of the PRHRHX is discussed in Section 6.3 of this report.

5.4.7.7 Inspection and Testing Requirements

DCD Tier 2 Section 5.4.7.6 describes inspection and testing requirements for the RNS. Proper operation of the RNS is verified through pre-operational tests, which include valve inspection and testing, flow testing, and verification of heat removal capability. The inspection and test requirements of the RNS valves are consistent with those identified in DCD Tier 2 Sections 5.2.4 and 6.6, respectively, for the valves that constitute the RCPB and the valves that isolate the line penetrating containment. In addition, these valves are included in DCD Tier 2 Table 3.9.16 and are subject to IST. The staff finds proper inspection and test requirements are made for the RNS valves performing safety-related functions of containment isolation and preserving RCPB integrity.

The set pressure and the relieving capacity of the relief valve, RNS-V021, which is provided for low-temperature overpressure protection, are verified to be consistent with the values specified in DCD Tier 2 Table 5.4-17. The relief valve relieving capacity will be certified in accordance with ASME Code Section III, NC-7000. The staff finds this acceptable.

The minimum flow rates to meet the functional requirements of cooling the RCS during shutdown operations and low pressure makeup to prevent 4th stage ADS actuation for small break LOCA, respectively, are specified in DCD Tier 2 Table 5.4-14. These shutdown cooling and low pressure makeup flow rates are confirmed through the tests with the RNS pump suction aligned to their respective operations, (i.e., with the suction aligned to the RCS hot leg and the cask loading pit, respectively). The RNS heat exchanger heat removal capability is specified in DCD Tier 2 Table 5.4-14, and is verified through the manufacturer’s test results and data. The staff finds these tests to confirm the RNS flow and heat transfer capabilities to be acceptable.

5.4.7.8 Regulatory Treatment of the RNS

The RNS is a non-safety-related system that is not required to operate to mitigate design-basis events. Therefore, the RNS is not required to meet safety-related system requirements. However, the RNS is a defense-in-depth system that provides the first line of defense during an accident to prevent unnecessary actuation of passive core cooling systems. Regulatory oversight of the active non-safety systems in passive plant designs is subject to a staff evaluation of the regulatory treatment of non-safety systems (RTNSS). A detailed evaluation of the RTNSS issue is described in Chapter 22 of this report.

In SECY-94-084, "Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety Systems in Passive Plant Designs," the staff describes the RTNSS process. The goal of the RTNSS process is to provide insights on the importance of non-safety related-systems to the overall safety of the passive advanced reactor design and assist in determining what, if any, additional regulatory controls should be applied to RTNSS-identified systems. The RTNSS process involves using both probabilistic and deterministic criteria to (1) determine whether regulatory oversight for certain non-safety-related systems is needed, (2) identify the risk significant SSCs for regulatory oversight, and (3) decide on an appropriate level of regulatory oversight for the various identified SSCs commensurate with their risk importance.

As the important non-safety-related SSCs identified through the RTNSS process do not meet the screening criteria specified in 10 CFR 50.36 for inclusion in the TS limiting conditions for operation, the applicant proposed a mechanism to provide for short-term availability control of these systems. DCD Tier 2 Section 16.3 provides short-term availability administrative controls for the RTNSS-identified important non-safety-related SSCs. For each RTNSS-identified SSC, the operability requirements for the required functions and system configurations are specified for various modes of operation, and the required actions and completion times are specified for conditions not meeting the operability requirements. Surveillance frequency requirements are also specified to confirm operability of the SSCs. A commitment is included in the DCD Tier 2 Section 16.3.2 for the COL applicant referencing the AP1000 design to develop and implement procedures consistent with the availability controls. These administrative availability controls will also be included in the AP1000 design control document.

In WCAP-15985, "AP1000 Implementation of the Regulatory Treatment of Non-Safety-Related Systems Process," the applicant provided the results of its evaluation on the basis of the RTNSS screening process. The RNS was identified as an important system needed for shutdown decay heat removal to support mid-loop operation with reduced reactor coolant inventory and, therefore, subject to additional regulatory controls. In addition, the RNS also provides a non-safety-related means of injecting the IRWST water into the RCS following ADS actuation to provide margin in the PRA sensitivity studies to mitigate at-power and shutdown events. The administrative short-term availability controls of the RNS functions at various modes of operation are specified in the DCD Tier 2 Table 16.3-2, and DCD Tier 2 Sections 2.1 and 2.2. In addition, the availability controls of the RNS supporting systems such as the CCS, the service water system, and the AC power supplies, are specified in DCD Tier 2 Table 16.3-2. The staff has reviewed DCD Tier 2 Table 16.3-2, and concluded that proper administrative controls are provided to ensure the short-term availability of the RCS to perform its required functions.

5.4.8 Valves

The design bases, design evaluation, qualification testing, ISI and IST of valves associated with the RCS and RCS-connected systems is collectively discussed in Sections 3.9.3, 3.9.6, 3.10, 5.2.3, 5.2.4, and 6.6 of this report.

5.4.9 Reactor Coolant System Pressure Relief Devices

The AP1000 design, which does not have a PORV in the reactor coolant system, relies on the PSVs connected to the pressurizer to provide overpressure protection of the RCS during power operation to comply with GDC 15 in Appendix A to 10 CFR Part 50. GDC 15 requires the RCS and associated auxiliary, control, and protection systems are to be designed with sufficient margin to assure that the design conditions of the RCPB are not exceeded during any condition of normal operation, including anticipated operational occurrences. The AP1000 also relies on the relief valve on the suction line of the RNS to provide LTOP consistent with the guidelines of Section 5.2.2 of the SRP. including BTP RSB 5-2.

It should be noted that the ADS valves, which provide a means to depressurize the RCS as part of the passive core cooling system, are not pressure relief devices for overpressure protection.

The first three stages of the ADS are connected to the pressurizer, and the first stage can also be used to vent non-condensable gases following an accident.

5.4.9.1 Pressurizer Safety Valves

The AP1000 has two PSVs, which are of the totally enclosed pop type, spring loaded, self-actuated by direct fluid pressure. There is no loop seal in the piping between the pressurizer and the PSVs to collect the steam condensate. The steam condensate will drain back to the pressurizer, and will not be discharged as a water slug during the initial opening of the valve. Each PSV discharge is directed through a rupture disk, located at the end of the discharge piping, to containment atmosphere. The rupture disk is provided to contain leakage past the valve, and is designed with a substantially lower set pressure than the PSV set pressure to ensure PSV discharge. A small pipe is connected to the discharge piping and directed to the reactor coolant drain tank to drain away condensed steam leaking past the safety valve. Positive position indication is provided for the PSVs, in accordance with the requirements of 10 CFR 50.34(f)(2)(xi), which requires direct indication of relief and safety valve position (open or closed) be provided in the MCR. Temperatures in the discharge lines are measured, and an indication and a high temperature alarm are provided in the control room for indication of any leakage or relief through the associated valve. The PSVs are designed to prevent RCS pressure from exceeding 110 percent of system design pressure. The design parameters of the PSVs are specified in DCD Tier 2 Table 5.4-17. As addressed in Section 5.2.2 of this report, the sizing of the PSVs with 3-percent accumulation meets GDC 15 and, therefore, is acceptable.

In 10 CFR 50.34(f)(2)(x), the NRC requires a test program and associated model development, as well as conducting of tests to qualify RCS relief and safety valves for all fluid conditions expected under operating conditions, transients, and accidents. This has been done through the tests of similar safety valves within the EPRI safety and relief valve test program, which found that the safety valves were adequate for steam flow and water flow, even though water flow is not anticipated through the PSVs. Item II.D.1, "Performance Testing of PWR Safety and Relief Valves," in Chapter 20 of this report addresses the resolution of the PS testing program. The PSVs are also subjected to preservice and inservice hydrostatic tests, seat leakage tests, operational tests, and inspections. This is done through the IST specified in DCD Tier 2 Table 3.9-16, as well as the IST for ASME Code Class 2 and 3 components in DCD Tier 2 Section 6.6. The test program for the safety valves complies with the requirements of ANSI/ASME Code of Operations and Maintenance, Part 1, “Requirements for Inservice Testing of Nuclear Power Plant Pressure Relief Devices,” and is therefore acceptable.

5.4.9.2 RNS Relief Valve

The RNS relief valve on the RNS pump suction line is spring loaded, self-actuated by direct fluid pressure, and is designed for water relief with an accumulation of 10-percent of the set pressure. The set pressure (setpoint) is the lower of the values determined on the basis of the RNS design pressure or the RV low temperature pressure limit. The design parameters of the RNS relief valve, including the set pressure and relieving capacity, are specified in DCD Table 5.4-17. The determination of the set pressure and relieving capacity, is discussed in Section 5.2.2 of this report. The lowest permissible lift set pressure is determined by the required NPSH for the RCPs. Position indication for the RNS relief valve is provided in accordance with the requirements of 10 CFR 50.34(f)(2)(xi), which requires direct indication of relief and safety valve position (open or closed) be provided in the MCR. Therefore, this is acceptable.

RCS pressure relief devices are required by 10 CFR 50.34(f)(2)(x) to be subjected to tests to qualify for all fluid conditions expected under operating conditions, transients, and accidents. DCD Tier 2 Section 5.4.9.4 states that the RNS relief valve is designed for water relief and is not a RCS pressure relief device since it has a set pressure less than RCS design pressure. Therefore, the valve selected for the RNS relief valve is independent from the EPRI safety and relief valve test program. Since the RNS relief valve is not a RCPB valve, and is designed for low-temperature overpressure protection, the staff agrees it need not be included in the EPRI test program for the safety and relief valves. As specified in DCD Tier 2 Table 3.2-3, the RNS relief valve is an AP1000 Class 2 component, and will be designed, manufactured, and tested to ASME Section III, Class 2 requirements. In addition, the RNS relief valve is also subject to IST as specified in DCD Tier 2 Table 3.9-16 for its safety related missions and functions. The staff finds these test requirements for the RNS relief valve complies with the ASME Code Section III requirements and are therefore acceptable.

5.4.10 RCS Component Supports

The design bases and design evaluation of the RCS component supports are described in Sections 3.9.3.3 and 3.12.6 of this report. Inservice inspection of RCS components is discussed in Sections 5.4.2.2 and 6.6 of this report.

5.4.11 Pressurizer Relief Discharge

The AP1000 design does not have a pressurizer relief discharge system. The AP1000 employs neither power-operated pressurizer relief valves nor a pressurizer relief discharge tank. Some of the functions provided by the pressurizer relief discharge system in previous nuclear power plants are provided by portions of other systems in the AP1000.

The staff reviewed the AP1000 pressurizer relief discharge using SRP Section 5.4.11, "Pressurizer Relief Tank," for guidance. The SRP acceptance criteria specify that the design meet GDC 2, "Design Basis for Protection Against Natural Phenomena," as it relates to the protection of safety-related systems from the effects of earthquakes, and GDC 4, "Environmental and Dynamic Effects Design Bases," as it relates to a failure of the system resulting in missiles or adverse environmental conditions that could result in damage to safety-related systems or components. Conformance with GDC 2 is on the basis of meeting the guidelines of RG 1.29, "Seismic Design Classification," Positions C.2 and C.3. Position C.2 addresses those portions of SSCs which should be designed and constructed such that an SSE could not cause their failure and result in reduced functioning of any seismic Category I equipment or incapacitating injury to occupants in the MCR. Position C.3 addresses the extension of seismic Category I design requirements to the first seismic restraint beyond the defined boundaries. Conformance with GC 4 is on the basis of meeting the acceptance criteria of SRP 5.4.11, as applicable.

The systems and components for AP1000 pressurizer relief discharge are discussed in DCD Tier 2 Section 5.2.2, “Overpressure Protection,” Section 5.4.6, “Automatic Depressurization System Valves,” Section 5.4.9, “Reactor Coolant System Pressure Relief Devices,” Section 5.4.11, “Pressurizer Relief Discharge,” Section 5.4.12, “Reactor Coolant System High Point Vents,” and Section 6.3, “Passive Core Cooling System.” This equipment is located inside containment and is designed to provide overpressure protection for the RCS during power operation. Two pressurizer safety valves are located on top of the RCS pressurizer. DCD Tier 2 Table 3.2-3 and DCD Tier 2 Section 3.2 state that the pressurizer safety valves are classified as AP1000 equipment Class A (ANS safety Class 1), seismic Category I, and ASME Code Class 1. These valves are tested in accordance with requirements of the ASME Code, Section XI.

The pressurizer safety valves are spring loaded, self-actuated by direct fluid pressure, and have backpressure compensation features. They are the totally enclosed pop type, and are designed to reclose and prevent further flow of fluid after normal conditions have been restored. Because loop seals are not installed between the pressurizer and safety valves, steam condensation flows back into the pressurizer instead of forming a water slug that would blow out during initial safety valve actuation. Although the valves are designed for the flow of both steam and water, water is not expected to flow through the valves. The normal residual heat removal relief valve is designed for water relief.

The pressurizer safety valves are sized on the basis of the analysis of a complete loss of steam flow to the turbine with the reactor operating at 102 percent of rated power. In the analysis, no credit is taken for the operation of the pressurizer level control system, pressurizer spray system, rod control system, steam dump system, steamline PORVs, or direct reactor trip on turbine trip. The feedwater system is also assumed to be lost. Under these conditions, the total pressurizer safety valve capacity is at least as large as the maximum surge rate into the pressurizer during this postulated event. This results in a safety valve capacity that prevents system pressure from exceeding 110 percent of system design pressure.

Pressurizer safety valve discharge is routed through a rupture disk to the containment atmosphere. The rupture disk is designed to contain any leakage past the safety valves and has a pressure rating much lower than the set pressure of the safety valve. Leakage past the safety valve during normal operation is collected and routed to the reactor coolant drain tank (RCDT). Each safety valve discharge line includes a temperature indicator and alarm in the MCR.

Pressurizer safety valve discharge is directed away from SSCs inside containment, which could be damaged by the discharge. The containment pressure resulting from a safety valve discharge is significantly less than the containment design pressure (the containment design pressure is determined by LOCA considerations), and the resulting heat load is well within the capacity of the normal fan coolers and the PCS.

5.4.11.1 Automatic Depressurization System

The ADS is shown in DCD Tier 2 Figure 5.1-5 (sheet 1 of 3 and sheet 2 of 3). The system is not a pressure relief system. It is designed to depressurize the RCS under emergency plant operations and to vent noncondensable gases from the pressurizer steam space following an accident. Operation of the ADS valves is required for the PXS to function following postulated accident conditions. The first stage valves are used to vent noncondensable gases from the pressurizer steam space. In DCD Tier 2 Table 3.2-3 and Section 3.2, the applicant states that the valves are classified as AP1000 equipment Class A (ANS safety Class 1), seismic Category I, and ASME Code Class 1. The valves are tested in accordance with requirements of ASME Code, Section XI.

The ADS consists of 20 valves divided into 2 divisions, and further divided into 4 depressurization stages. These valves are connected to the RCS at three locations. The two divisions of the first-, second-, and third- stage valves are connected to the top of the pressurizer while one division of the fourth-stage valves is connected to the hot leg of each RCS loop and vents directly to a SG compartment. The fourth-stage valves are designed such that they cannot open against full system pressure.

The discharge from the first-, second-, and third- stage ADS valves is routed to the IRWST by way of two depressurization spargers (one per division). The spargers are classified as AP1000 equipment Class C (ANS safety Class 3) and seismic Category I, and are designed to distribute steam inside the IRWST to ensure effective steam condensation. The IRWST also receives discharges from the relief valve of the RNS, and steam and gas discharges from the PRHR high point vents and the RV high point vents (discussed in DCD Tier 2 Section 5.4.12).

As described in DCD Tier 2 Sections 5.4.6. and 6.3, the ADS, consisting of four stages, is part of the RCS and interfaces with the PXS. Two valves are located in each discharge path to prevent inadvertent ADS valve discharges should a valve accidentally open. Diverse and redundant features are provided in the ADS control system to ensure that valves do not inadvertently open. Following ADS actuation, steam can condense in the discharge line creating a vacuum condition that could result in a reverse flow of water from the IRWST. To prevent this, vacuum breakers are provided in the discharge lines to limit the pressure drop that may occur following ADS actuation and thus prevent backflow.

5.4.11.2 In-Containment Refueling Water Storage Tank

The IRWST is a stainless steel-lined compartment inside containment that is integrated into the containment structure underneath the operating deck. The tank is classified as AP1000 equipment Class C (ANS safety Class 3) and seismic Category I. The tank is designed to absorb the pressure increase and heat input from the discharge from a first-stage ADS valve (including the water seal, steam, and gases) when venting noncondensable gases from the pressurizer following an accident.

As stated above, the first-, second-, and third-stage ADS valves are divided into two divisions that connect to two separate spargers below the water level of the IRWST. The discharge from the spargers does not result in pressures in excess of the design pressure of the IRWST during a first stage ADS valve discharge of steam, water, and noncondensable gases during an accident. In addition, the IRWST has covered vents that provide tank overpressure protection. The IRWST does not use a cover gas or a spray system, and does not have a connection to the waste gas processing system. The IRWST is cooled by the RNS and includes level and temperature indicators and alarms.

Conformance with GDC 2 is on the basis of meeting the guidelines of Positions C.2 and C.3 of RG 1.29. Position C.2 states that those portions of the system whose function is not required, but whose failure could reduce the functioning of any seismic Category I system or could result in incapacitating the occupants of the MCR., should be designed and constructed so that an SSE would not cause this failure. As stated above, the pressurizer relief discharge components are seismic Category I, and discharge is directed away from any safety-related SSCs inside containment, that could be damaged by the discharge. Also, the discharges from the ADS valves are routed to the IRWST, which is designed to accommodate these discharges and therefore will not pose a hazard to nearby safety-related SSCs. These processes occur inside containment and therefore do not affect the MCR. In addition, the applicant has stated in DCD Tier 2 Appendix 1A that the AP1000 design will conform to the guidelines of this position.

Position C.3 states that seismic Category I design requirements should extend to the first seismic restraint beyond the defined boundaries. Those portions of the system that form interfaces between seismic Category I and non-seismic Category I features should be designed to seismic Category I requirements. The applicant has stated in DCD Tier 2 Appendix 1A that the system design will conform to the guidelines of this position.

The pressurizer safety valve discharge is directed away from safety-related SSCs inside containment, that could be damaged by the discharge. In addition, discharges from the ADS valves are routed to the IRWST, which is designed to accommodate these discharges. On the basis of this information, the staff concludes that the pressurizer relief discharge equipment is adequately protected from the dynamic effects associated with failed SSCs inside containment, and also will not pose a hazard to other safety-related SSCs inside containment should any of the pressurizer relief discharge equipment fail.

Considering the evaluation of information and commitments provided by the applicant in the DCD, the staff concludes that equipment used for AP1000 pressurizer relief discharge meets the requirements of GDC 2 on the basis of conformance with Positions C.2 and C.3 of RG 1.29, and also meets the requirements of GDC 4 on the basis of the protection of safety-related SSCs from effects associated with a failure of the equipment. Therefore, the staff concludes that systems and components used for AP1000 pressurizer relief discharge conform to the appropriate guidelines of SRP 5.4.11, and are acceptable.

5.4.12 Reactor Coolant System High Point Vents

RCS high-point vents are provided to exhaust noncondensIble gases accumulated in the primary system that could inhibit natural circulation core cooling. 10 CFR 50.34(f)(2)(vi) requires that the RCS be provided with high-point vents to maintain adequate core cooling, and that systems to achieve this capability be capable of being operated from the MCR. and that their operation not lead to an unacceptable increase in the probability of a LOCA or an unacceptable challenge to containment integrity.

In the AP1000 design, noncondensible gases from the RCS are vented using either a reactor head vent or, following an accident, the first-stage valves of the ADS connected to the pressurizer. In addition, the PRHRHX piping and the CMT inlet piping in the PXS also include a high-point vent and are, therefore, in compliance with 50.34(f)(2)(vi).

The review of the AP1000 RCS high-point vent design was performed in accordance with Section 5.4.12 of the SRP as discussed below.

5.4.12.1 Reactor Vessel Head Vent System

The RV head vent system (RVHVS) is designed to remove noncondensible gases or steam from the RCS, with a capacity to vent a volume of hydrogen at system pressure and temperature equivalent to approximately 40 percent of the RCS volume in one hour. The primary function of the RVHVS is for use during plant startup to properly vent air from the RV head and fill the RCS. The RVHVS valves also provides an emergency letdown path with a letdown flow rate within the capabilities of the normal makeup system to prevent pressurizer overfill following long-term loss of heat sink events.

The RVHVS consists of two parallel flow paths. Each contains two redundant, 2.54 cm (1 in) open/close, solenoid-operated isolation valves in series, and a flow-limiting orifice downstream. The system discharges to the IRWST.

The solenoid-operated isolation valves are fail-closed, normally closed valves, powered by the safety-related Class 1E dc and uninterruptible power supply system. The RVHVS is operated from the MCR, which has individual positive valve position indication and alarm. These valves are included in the AP1000 operability program with the IST requirements specified in DCD Tier 2 Table 3.9-16, and are qualified to IEEE-323, IEEE-344, and IEEE-382.

The RVHVS is designed so that a single failure of the remotely operated vent valves, power supply, or control system does not prevent isolation of the vent path. The two redundant isolation valves in series minimize the possibility of RCPB leakage, and ensure that the failure of any one valve does not inadvertently open a vent path.

The flow-limiting orifices limit the flow rate from the head vent path. Acceptance criteria II.5 in Section 5.4.12 of the SRP specifies that the size of the vent line should be kept smaller than the size corresponding to the definition of a LOCA to avoid unnecessary challenges to the emergency core cooling system. Although the size of the vent pipe of 2.54 cm (1 in) is larger than the size corresponding to the definition of a LOCA, the use of the orifices to restrict the flow rate of the head vent to within the capabilities of the normal makeup capability of the CVS allows the AP1000 to meet the intent of this criterion.

In the event of a break of the RVHVS line, it would result in a small break LOCA no greater than 2.54 cm (1 in) diameter. Such a break is similar to the hot leg break LOCA analyzed in DCD Tier 2 Section 15.6.5. The analysis results indicating no core uncovery also apply to a RVHVS line break.

The acceptance criteria of Section 5.4.12 of the SRP specifies that procedures should be developed for use of the vent paths to remove gases that may inhibit core cooling from the U-tubes of the SGs; and that the procedures to operate the vent system should consider when venting is needed, and when it is not needed, with consideration of a variety of initial conditions, operator actions, and necessary instrumentation. The SG tube venting procedures are described in the applicant's response to RAI 440.049.

The primary function of the RVHVS is for use during plant startup to properly vent air from the RV head and fill the RCS. During plant startup operations when the RV head is in place and the RCS is filled water solid, the air in the RCS is vented through repeated procedures of (1) starting a RCP in each SG for a short time with the high-point vents closed to allow collection of air in the RCS high points, and (2) opening the vents to allow air trapped in the high points to be vented.

In addition to the normal venting procedures during startup, the AP1000 RVHVS could also be used under a design basis accident scenario. During an accident, the AP1000 design relies on the passive safety-related systems such as the PRHRHX to provide the safety-related function of core cooling, and therefore does not require the SG U-tubes to be vented to provide coolability of the core. However, the RVHVS is used under loss of heat sink events where the pressurizer level can increase and eventually become water solid following long-term operation of the CMTs. To avoid this occurrence, the functional restoration guidelines for high pressurizer level in the ERG requires that the RV vent flow be established to provide a bleed path in response to high pressurizer level conditions to reduce the RCS inventory and prevent pressurizer overfill. When the pressurizer level is sufficiently reduced, the operator recloses the head vent valves. In this case, the operator uses pressurizer level as the primary indication to control operation of the RV head vent.

The RV head vent system consists of safety-grade equipment. The piping and equipment from the vessel head vent up to and including the second solenoid valve constitute the RCPB, and are designed and fabricated to ASME Code Section III, Class 1 requirements. The remainder of the piping and equipment are design and fabricated in accordance with ASME Code Class 3 requirements. The piping stresses meet the requirements of ASME Code, Section III, NC-3600, with a design temperature of 343.3°C (650°F) and a design pressure of 17.23 MPa (2485 psig). The RVHVS can be operated from the control room or the remote shutdown workstation. Each solenoid-operated isolation vent valve has a position sensor with indication in the control room. Inservice inspection and testing of the RVHVS is in accordance with DCD Tier 2 Section 3.9.6 for valves and DCD Tier 2 Section 5.2.4 for ASME Code Class 1 components that are part of the RCPB. The RVHVS meets the acceptance criteria specified in Section II of Section 5.4.12 of the SRP and is therefore acceptable. The resolution of TMI Action Item II.B.1, RCS High-Point Vent, is addressed in Chapter 20 of this report.

5.4.12.2 ADS First-Stage Valves

As discussed in Section 5.4.6 above, the first-stage valves of the AP1000 ADS provide the capability to remove noncondensible gases from the pressurizer steam space following an accident. Gas accumulations are removed by remote manual operation of the first-stage ADS valves. The discharge of the ADS valves is directed to the IRWST.

The ADS is primarily designed to function as a part of the PXS. The ADS piping up to and including the second isolation valve in series also constitutes the RCPB, and both the piping and valves are designed, constructed, and inspected to ASME Code Class 1 and seismic Category I requirements. The ADS valves are active valves required to provide safe shutdown or to mitigate the consequences of postulated accidents. However, venting of noncondensible gases from the pressurizer steam space is not required to provide safety-related core cooling following a postulated accident. Therefore, the acceptance guidelines of the SRP Section 5.4.12 do not apply to the ADS.

5.4.12.3 Passive RHR Heat Exchanger and Core Makeup Tank High-Point Vents

The PRHRHX inlet piping and the CMT pressure balance line piping in the PXS include high-point vents that provide the capability for removing and preventing the accumulation of noncondensible gases that could interfere with heat exchanger or CMT operation. These gases are normally expected to accumulate when the RCS is refilled and pressurized following refueling. There are level indicators to indicate when gasses have collected in the vent line. Any noncondensible gases that collect in this high point can be manually vented. The discharge of the PRHRHX high-point vent is directed to the IRWST, and the discharge of the CMT high-point vent is directed to the RCDT.

These high-point vent lines contain two manual isolation valves in series, so that a single failure of either valve to reseat following venting operation does not prevent isolation of the flow path. The isolation valves in the vent line have position sensors with position indication in the MCR. Each vent line also contains a 0.95 cm (0.375 in.) flow restrictor such that the break flow is within the makeup capability of the CVS and, therefore, would not normally require actuation of the passive safety systems. The vent lines downstream of the flow restrictors and 2.54 cm (1 in.) lines designed to ASME Code Section III, Class 2 requirements. Inservice inspection of the PRHR HX and CMT high point vents are in accordance with DCD Tier 2 Section 6.6 for ASME Code Class 2 components, and Section 5.2.4 fir ASME Code Class 1 components that are part of the RCPB. The staff concludes that the PRHR HX and CMT high point vents are acceptable since they provide a means to prevent accumulation of noncondensable gases from the RCS that could interfere with operation of the PXS, and are designed in accordance with the ASME Code Section III requirements.

5.4.13 Core Makeup Tank

There are two CMTs in the AP1000 design as part of the passive core cooling system (PXS). In the CMTs, cold borated water, under system pressure, is stored to provide high-pressure reactor coolant makeup and boration for LOCA and for non-LOCA events, when the normal makeup system is unavailable or insufficient. DCD Tier 2 Section 6.3 describes the operation of the CMTs in the PXS and the connections to the CMTs.

5.4.13.1 Design Description

The AP1000 CMT is a low-alloy steel vessel with a minimum free internal volume of 70.75 m^3 (2500 ft^3), and is supported on columns. DCD Tier 2 Table 6.3-4 provides the CMT design data. The CMT injection line connects from one nozzle on the lower head to the RV DVI piping.

The discharge line contains two normally closed, fail-open, parallel isolation valves, and two check valves in series. The CMT pressure balance line connects from the top nozzle in the center of the upper head to one of the RCS cold legs. The pressure balance line with the open flow path to the cold leg maintains system pressure. The top nozzle incorporates a diffuser inside the tank. The bottom of the diffuser, which has the same diameter and thickness as the connecting piping, is plugged and holes are drilled in the side to force the steam flow to turn 90 degrees, which limits the steam penetration into the coolant in the CMT. The diffuser is designed to reduce steam and hot water velocities entering the CMT, thereby minimizing potential water hammer and reducing the amount of mixing that occurs during initial CMT operation. Two sample lines in the upper and lower head, respectively, are provided for sampling the solution in the CMT. A fill connection is provided for makeup water from the CVS.

5.4.13.2 Design Bases

The CMT is a part of the RCPB and AP1000 Class A equipment, and is designed and fabricated according to ASME Code, Section III, Class 1 component requirements. Materials of construction are specified to minimize corrosion/erosion and to provide compatibility with the operating environment, including the expected radiation level. DCD Tier 2 Section 5.4.13.4 states, and the staff agrees, that erosion is not an issue because there is normally no flow in the CMT. Those portions of the CMT in contact with reactor coolant are fabricated from or clad with stainless steel. Contamination of stainless steel and nickel-chromium-iron alloys by copper, low-melting-temperature alloys, mercury, and lead is prohibited. The material selection and water chemistry specification, and test and inspections of CMT are discussed in Sections 5.2.3 and 5.2.4, respectively, of this report.

5.4.13.3 Design Evaluation

The loading combinations, stress limits, and analytical methods for the structural evaluation of the CMT for various plant conditions are discussed in DCD Tier 2 Section 3.9.3. The requirements for dynamic testing and analysis are discussed in DCD Tier 2 Section 3.9.2. The transients used to evaluate the CMT are founded on the system design transients described in DCD Tier 2 Section 3.9.1.1. In addition to normal RCS transients, the evaluation of component cyclic fatigue of the CMT also assumes 30 occurrences in the plant 60-year lifetime where a small leak draws in hot RCS fluid, and 10 occurrences of increasing containment temperature above normal operating range.

The mechanical component design evaluation with respect to the RCS design transients; requirements for dynamic testing and analysis; and loading combinations; stress limits; and analytical methods for structure evaluation; are discussed in DCD Tier 2 Sections 3.9.1, 3.9.2, and 3.9.3, respectively. The staff evaluation of these sections are discussed in their respective sections of this report.

The functional performance of the CMTs is evaluated in Chapter 6.3 of this report, as part of the PXS performance, as well as the safety analyses of various design basis transients and accidents in Chapter 15 of this report, to demonstrate its capability to comply with respective acceptance criteria. The staff has reviewed the PXS function performance and the design basis analyses of transients and accidents, as described in Section 6.3 and Chapter 15 of this report. In addition, in support of the AP600 design certification application, the applicant performed various separate effects and integral system tests to study thermal-hydraulic behavior and the phenomena of the AP600 PXS and components, and to validate the codes used for the design basis analysis of transients and accidents for the AP600. The same computer codes used for the AP600 are used for the AP1000. In Chapter 21 of this report, the staff discusses the applicability of the AP600 test program and the computer codes to the AP1000 design. On the basis of the acceptability of these evaluations referenced above the staff concludes that the CMT design meets the guidelines of SRP 6.3 and GDC 2, 4, 5, 17, 36, and 37, and the PXS as a whole meets GDC 27, 34, and 35. Therefore the CMT design is acceptable.

5.4.14 Passive Residual Heat Removal Heat Exchanger

The AP1000 passive residual heat removal heat exchanger (PRHRHX) is part of the passive core cooling system (PXS). Its function is to remove core decay heat for any postulated non-LOCA event where a loss of cooling capability via the SGs occurs. Section 6.3 of this report discusses the operation of the PRHRHX in the PXS.

5.4.14.1 Design Description

The PRHRHX consists of a top and lower tubesheet mounted through the wall of the IRWST. A series of 1.9 cm (0.75 in) outer diameter C-shaped tubes connect to the tubesheets, with the top of the tubes located several feet below the IRWST water surface. DCD Tier 2 Table 6.3-4 provides the AP1000 PRHRHX design data. An inlet channel head mounted to the top tube sheet is connected through piping to one of the RCS hot legs. An outlet channel head mounted to the bottom tube sheet is connected through piping to the SG cold-side channel head. The primary coolant passes through the tubes, transferring decay heat to the IRWST water. Sufficient thermal driving head is generated in the process to maintain natural circulation flow through the heat exchanger. The design minimizes the diameter of the tubesheets and allows ample flow area between the tubes in the IRWST. The horizontal lengths of the tubes and lateral support spacing in the vertical section allow for the potential temperature difference between the tubes at both cold and hot conditions. The PRHRHX is welded to the IRWST. The tubes are supported in the IRWST interior with a frame structure. The top of the structure supports a cover that traps and condenses steam during initial activation of the PRHRHX, and helps to minimize the amount of humidity in containment.

5.4.14.2 Design Bases

The PRHRHX, in conjunction with the PCS, is designed to be able to automatically remove core decay heat for an unlimited period of time. This capability requires a closed-loop mode of operation where the condensate from steam generated in the IRWST is returned to the tank. The PRHRHX and the IRWST are designed to delay significant steam release to the containment for at least one hour. The PRHRHX will keep the reactor coolant subcooled and prevent water relief from the pressurizer. In addition, the PRHRHX will cool the RCS to 204.4°C (400°F) in 72 hours, with RCPs operating or, if required, in the natural circulation mode, so that the RCS can be depressurized to reduce stress levels in the system.

The PRHRHX is designed to withstand the design environment of 17.24 MPa (2500 psia) and 343.3°C (650°F) for 60 years. The PRHRHX is part of the RCPB, is AP1000 class A equipment, and is designed and fabricated according to the ASME Code, Section III, as a Class 1 component. Material specifications, compatibility with the operating environment, including the expected radiation level, as well as the fabrication and processing of the stainless steel for the PRHRHX as the RCPB are discussed in DCD Tier 2 Section 5.2.3, with the staff evaluation provided in the Section 5.2.3 of this report. DCD Tier 2 Section 5.2.4 discusses the ISI and testing of Class 1 components, which are applicable to the PRHRHX.

5.4.14.3 Design Evaluation

The loading combination, stress limits, and analytical methods for the evaluation of structural integrity of the PRHRHX, and the transients used to evaluate the PRHRHX under various plant conditions, are discussed in DCD Tier 2 Sections 3.9.1 through 3.9.3. During normal plant operation, the PRHRHX, without flow through it, is pressurized to the RCS hot leg pressure at the IRWST temperature. Operation of the PRHRHX is evaluated using Service Levels B, C, and D plant conditions, as described in DCD Tier 2 Section 3.9.1.1. In addition to loads resulting from normal RCS transients and the PRHRHX operation, the evaluation also considers hydraulic loads due to discharge of steam from the ADS valves into the sparger in the IRWST. Seismic, LOCA, sparger activation, and flow-induced vibration loads are derived using dynamic models of the PRHRHX. The dynamic analysis considers the hydraulic interaction between the coolant and system structural elements. The evaluation of component cyclic fatigue also assumes two additional Service Level B transients that affect only the PRHRHX:

- 30 occurrences in the plant 60-year lifetime where a small leak in the manway cover draws in hot RCS fluid

- 10 occurrences of increasing IRWST temperature as a result of an event that activates passive core cooling

The staff evaluation of the mechanical component design with respect to the design transients; requirements for dynamic testing and analysis; and loading combinations, stress limits, and analytical methods for structure evaluation; are discussed in Sections 3.9.1, 3.9.2, and 3.9.3, respectively, of this report.

The PRHRHX functional performance is evaluated in Chapter 6.3 of this report, as part of the PXS performance. The safety analyses of various design-basis transients and accidents is presented in Chapter 15 of this report to demonstrate the PXS capability to comply with applicable acceptance criteria. In addition, in support of the AP600 design certification application, the applicant had performed various separate effects and integral system tests to study thermal-hydraulic behavior and the phenomena of the AP600 PXS and components, and to validate the codes used for the design basis analysis of transients and accidents for the AP600. The same computer codes used for the AP600 are used for the AP1000. In Chapter 21 of this report, the staff discusses the applicability of the AP600 test program and the computer codes to the AP1000 design. On the basis of the acceptability of these evaluations referenced above, the staff concludes that the PRHR HX design meets the guidelines of SRP 6.3 and GDC 2, 4, 5, 17, 34, 36, and 37. Therefore, the PRHR HX design is acceptable.