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. Following is an overview of the AP1000 RCS and connected systems. Sections 5.2 through 5.4 of this report provide the staff's evaluation of these systems.

DCD Tier 2, Figures 5.1-1 through 5.1-3 show the schematic and layout of the AP1000 RCS and its principal auxiliary systems. The RCS consists of two heat transfer circuits (loops), each with a U-tube steam generator (SG), two reactor coolant pumps (RCPs), 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 tube sheet 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 4.4.6.4.
- 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 meet the requirements of Title 10, Section 50.34 (f)(2)(vi), of the *Code of Federal Regulations* (10 CFR 50.34(f)(2)(vi)) (Three Mile Island [TMI] Action Item II.B.1, "Reactor Coolant System Vents"), as described in 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 reactor vessel, including the control rod drive mechanism housings
- the reactor coolant pumps, comprising 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, tube sheet, 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 reactor vessel 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 thermal-hydraulic parameters of the RCS.

During operation, the RCPs circulate pressurized water through the RV and the SGs 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 SGs. The RCPs then return the water to the RV to repeat the heat removal cycle.

RCS pressure is controlled by the operation of the pressurizer, which maintains water and steam in equilibrium through 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. Two redundant sets of the first-three-stage automatic depressurization system (ADS) valves are also attached to the pressurizer. 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 ADS 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)
- the residual heat removal system (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. The following sections discuss each of the components in detail.

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 the 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 midloop 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 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, including nickel-chromium-iron (Ni-Cr-Fe) Alloy 690 thermally treated (TT) tubes on a triangular pitch; improved antivibration bars (AVBs); 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 which in turn delivers the steam 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 the 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. Shaft seals are eliminated in the AP1000 design because of the RCPs canned design. To provide the rotating inertia needed for flow coastdown, a uranium alloy flywheel is attached to the pump shaft.

A variable frequency drive provides speed control and minimizes the pump motor size. This reduces the requirements for motor power 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 °C (450 °F). During power operations, the drive is isolated and the pump runs 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-centimeter (cm) (31-inch (in.)) inside diameter hot leg pipe to transport reactor coolant to an 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 ensure that pipe stresses are sufficiently low for the primary loop and large auxiliary lines to meet the “leak-before-break” (LBB) requirements. Thus, pipe rupture restraints are not required, and the loop is only analyzed for pipe ruptures in the 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 at saturated conditions.

The top head includes 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. 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 MegaPascal (MPa) (2485 pounds per square inch gauge (psig)). The combined capacity of the two valves is determined by the requirement not to exceed the maximum RCS pressure limit during the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section III defined Level B service condition, loss-of-load transient, (i.e., 110 percent of the RCS design pressure of 17.23 MPa (2485 psig)). Thus, the design of the AP1000 pressurizer safety valves complies with the requirements of the ASME Code, Section III.

5.1.3.7 Automatic Depressurization Valves

Several of the passive safety features of the AP1000 design depend 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 the core makeup tank (CMT) level and a sequence timer. DCD Tier 2, Sections 5.4.6, “Automatic Depressurization System Valves,” and 6.3, “Passive Core Cooling System,” include a more detailed description of the ADS valves.

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 design procedure accounts for uncertainties in the component flow resistance and in the 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 the best-estimate flow.

Although the best-estimate flow is the most likely value to be expected in operation, more conservative flow rates (i.e., 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 (TS) as the flow that must be confirmed or exceeded by the flow measurements obtained during plant startup. The AP1000 thermal design procedure uses this flow rate in the reactor core departure from nucleate boiling (DNB) analysis. 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-flow value used for thermal-hydraulic analyses when 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 value 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 the best-estimate flow.

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

5.2.1 Compliance With Code and Code Cases

General Design Criteria (GDC) 1, "Quality Standards and Records," 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, as well as other systems important to safety. Where generally recognized codes and standards are used, they must be identified and evaluated to determine their adequacy and applicability.

5.2.1.1 <u>Compliance With 10 CFR 50.55a</u>

Pursuant to Title 10 of the <u>Code of Federal Regulations</u> (10 CFR) 50.55a, "Codes and Standards," components important to safety are subject to the following requirements:

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

- In accordance with 10 CFR 50.55a(d) and (e), components classified as QG B and QG 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 drawings (P&ID) collectively classify the mechanical and pressure-retaining components of the RCPB that do not meet the exclusion requirements, discussed 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 DCD Tier 2, Tables 3.2-1 and 3.2-3, to the corresponding tables in the AP600 DCD and found no significant differences.

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

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 QG C, in accordance with Positions C.1 or C.2 of RG 1.26, Revision 3. These will be constructed as ASME Code, Section III, Class 2 or Class 3 components.

As discussed in DCD Tier 2, Sections 5.2.1.1, “Compliance with 10 CFR 50.55a,” and 5.2.1.3, “Alternate Classification,” the portion of the chemical and volume control system (CVS) inside containment that is defined as part of the RCPB uses an alternate quality group classification. 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 (see DCD Tier 2, Figure 9.3.6-1). This is considered to be an alternate to the usual classification of the RCPB. Title 10, Section 50.55a(a)(3), of the Code of Federal Regulations allows alternatives to 10 CFR 50.55a(c) requirements, if the proposed alternative design provides an acceptable level of quality and safety. The applicant provided the following design enhancements to the Class D portion of the CVS as an alternate design:

- 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 TS for pressure isolation valves. Implementation of these additional leak rate tests will provide redundant leaktight 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 nonseismic, those portions inside containment will be analyzed to the same seismic design criteria as that accepted by the staff for seismic Category II piping. Section 3.12.3.7 of this report discusses the staff's acceptance of this criteria. The seismic Category II analyses will provide adequate assurance that the loads resulting from a safe-shutdown earthquake 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 to the Class D portion of the CVS, the staff finds that the alternative design provides an acceptable level of quality and safety and, therefore, is acceptable.

DCD Tier 2, Section 5.2.1.1, states that the baseline code used to support the AP1000 DCD is ASME Code, Section III, 1998 Edition, up to and including the 2000 Addenda. However, 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 the requirements of 10 CFR 50.55a(b) and the associated modification in 10 CFR 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 the 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 later Code editions and addenda to ensure consistency between the design and construction practices. However, the staff acknowledges that a need may exist to establish certain design parameters from a specific Code edition or addenda during its design certification review, particularly when the information is important to developing a significant aspect of the design or is used by the staff to reach its final safety determination. Various sections of this report reflect such considerations, as necessary. 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. The staff finds this to be an acceptable commitment. This is COL Action Item 5.2.1.1-1.

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

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, “Design, Fabrication, and Materials Code Case Acceptability, ASME Section III, Division 1,” and that are 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 that 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.”

The staff notes 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 DCD Tier 2, Table 5.2-3. RG 1.84 will not include these cases because they are not ASME Section III Code cases. However, the staff considers these cases to be acceptable because they include the 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 in the AP1000 design. The applicant’s compliance with the requirements of these Code cases will result in a component quality that is commensurate with the importance of the safety functions of the components. This satisfies the requirements of GDC 1 and, therefore, is acceptable.

5.2.2 Overpressure Protection

In the AP1000 design, overpressure protection for the RCS and the 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 normal residual heat removal system (RNS) during low-temperature operation, in conjunction with the action of the reactor protection system. There are 2 PSVs, 12 SGSVs with 6 valves located in the safety-related portion of the main steam piping running upstream of the main steam isolation valve (MSIV), and 1 relief valve in the suction line of the RNS. Combinations of these systems provide compliance with the overpressure protection requirements of ASME Code, Section III, Paragraphs NB-7300 and NC-7300, for pressurized-water reactor (PWR) systems. The ASME Code requires that the total relieving capacity be sufficient to prevent a pressure rise of more than 10 percent above the design pressure of the RCS and the SGs under any expected system pressurization transient conditions. The RNS suction relief valve for low-temperature overpressure protection (LTOP) prevents the RCS from exceeding the pressure-temperature limits determined by the analyses described in ASME Code, Section III, Appendix G.

The following 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 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.

- GDC 31, “Fracture Prevention of Reactor Coolant Pressure Boundary,” requires the RCPB to be designed with sufficient margin to assure that the 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 (RSB) 5-2, describes the acceptance criteria that demonstrate that a plant design complies with GDC 15 and 31. Therefore, the staff reviewed the AP1000 overpressure protection in accordance with SRP Section 5.2.2 and BTP RSB 5-2. In particular, the staff reviewed the following DCD Tier 2 sections:

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

5.2.2.1 Overpressure Protection during Power Operation

During power operation, 2 PSVs, 12 SGSVs, and the reactor protection system provide overpressure protection for the RCS to maintain the primary and secondary pressures within 110 percent of their respective design pressures. DCD Tier 2, Section 10.3.2.2.2, “Main Steam Safety Valves,” discusses the details of the SGSV design. DCD Tier 2, Table 10.3.2-2, lists the design data, including set pressures and relieving capacities. DCD Tier 2, Table 5.4-17, specifies the design parameters of the PSVs. The minimum required relief capacity is 340,194 kilograms per hour (Kg/hr) (750,000 pounds mass per hour (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 the containment atmosphere. The rupture disk, which has a pressure rating substantially less than the set pressure of the PSV, is designed to contain leakage past the PSV.

The size of the PSVs is determined by an 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 Section 5.2.2, II.A 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 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 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 15.4, “Reactivity and Power Distribution Anomalies,” also demonstrate the same conclusion. The PSV and SGSV setpoints and relieving capacities, therefore, are 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) in AP1000 TS 3.4.6, “Pressurizer Safety Valves,” described in 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, Sections 3.9.1, “Special Topics for Mechanical Components,” 3.9.2, “Dynamic Testing and Analysis,” and 3.9.3, “ASME Code Classes 1, 2, and 3 Components, Component Supports, and Core Support Structures.” The corresponding sections of this report discuss the staff’s evaluation of these sections. In addition, the PSVs are subjected to the EPRI verification program established to address the requirements of 10 CFR 50.34(f)(2)(x) to qualify PSVs for operation in all fluid conditions expected under operating conditions, transients, and accidents. Chapter 20, Item II.D.1, “Testing Requirements,” of this report addresses this issue. 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 requirements of GDC 15 and, therefore, is 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 the applicable TS and Appendix G limits for the RCS from being exceeded while the plant is operating at low temperatures. BTP RSB 5-2 also specifies that the LTOP system meet the ASME Code, Section III requirements, as well as RGs 1.26 and 1.29, “Seismic Design Classification,” 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 (RT_{NDT}) 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, which 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.” The relief valve is also subject to inservice test requirements described in DCD Tier 2, Table 3.9-16. In addition, AP1000 TS LCO 3.4.14, “Low-Temperature Overpressure Protection (LTOP) System,” 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 the staff’s request for additional information (RAI) 440.036, the applicant stated that the LTOP enable temperature of 135 °C (275 °F) is based on the PSVs for RCS overpressure protection when the RCS temperature is above 135 °C (275 °F). As indicated in DCD Tier 2, Table 5.3-3, the end-of-life RT_{NDT} for the AP1000 RV is expected to be approximately 10 °C (50 °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.,RT_{NDT}+ 50 °C (90 °F) at the beltline location).

The sizing and set pressure of the RNS relief valve for LTOP are founded on the sizing analysis performed to prevent the RCS pressure from exceeding the lower of either 110 percent of the RNS system design pressure of 6.31 MPa (900 psig), 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 (Revision 2), the applicant stated that based on the nominal steady-state P/T limits applicable up to 54 effective full-power years (EFPYs), the lowest Appendix G limit from DCD Tier 2, Figures 5.3-2 and 5.3-3, is 4.38 MPa (621 psig). Therefore, the RNS relief valve is sized to the Appendix G limit of 4.38 MPa (621 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 the 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 that transients occur while the pressurizer is in water-solid condition. The mass input event assumes 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 cubic meters per hour (m^3/h) (177 gallons per minute (gpm)), which is limited by the cavitating venturi located in the discharge header of the CVS system makeup pumps. The case of an 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 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, thereby limiting the required flow rate of the RNS suction relief valve, TS LCO 3.4.14 imposes an administrative limit for the LTOP protection system that does not allow an RCP to be started with the pressurizer level above 92 percent and the RCS temperature above 93.3 °C (200 °F).

This 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 192.7 m^3/h (850 gpm) is calculated at an RCS pressure equivalent to the valve setpoint of 3.55 MPa (500 psig) plus 10 percent accumulation, or 3.89 MPa (550 psig). With this setpoint, the peak pressure at the discharge of the RNS is no higher than 5.52 MPa (786 psig), and the peak pressure in the RCS is approximately 4.33 MPa (614 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’s full-open pressure of 3.89 MPa (550 psig).

Based on the information above, the relief valve would mitigate the limiting LTOP transient while maintaining the RCS pressure at less than the Appendix G limit. The minimum required capacity of the RNS relief valve is 192.7 m^3/h (850 gpm). DCD Tier 2 Table 5.4-17 provides the RNS relief valve design parameters (i.e., the nominal set pressure of 192.7 m^3/h (850 gpm), nominal set pressure of 3.55 MPa (500 psig), and full-open pressure, with 10 percent accumulation, of 3.89 MPa (550 psig)).

The RNS relief valve setpoint of 3.55 MPa (500 psig) was derived based on the lower of 110 percent of the RNS design pressure or the RCS P/T limit of 4.38 MPa (621 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 based on the copper and nickel material composition, as described in DCD Tier 2, Table 5.3-1, and 54 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 EFPYs, the RNS relief valve setpoint must be reevaluated.

Since the RT_{NDT} of the RV material increases as exposure to neutron fluence increases, as a result of the 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 by AP1000 TS LCO 3.4.3, “RCS Pressure and Temperature (P/T) Limits.” The bases for AP1000 TS 3.4.14 notes that each time the PTLR curves are revised, the LTOP system must be reevaluated to ensure that 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 its procurement of the RV. In addition, the COL applicant should address the use of these curves during the evaluation of the LTOP system, including the setpoint pressure for the RNS relief valve, as noted in AP1000 TS BASES B3.4.14. This is COL Action Item 5.2.2.2-1.

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

- GDC 1 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 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.

- 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 reviewed the materials specifications; compatibility of materials with the reactor coolant; fabrication; and processing of ferritic materials; and fabrication and processing of austenitic stainless steel. Sections 5.2.3.2 through 5.2.3.5 of this report discuss the acceptability of these elements.

5.2.3.1 Summary of Technical Information

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

The use of Ni-Cr-Fe alloy in the RCPB design of the AP1000 is limited to Alloy 690. The SG tubes are made of thermally treated Alloy 690. 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 alloys. Cast austenitic stainless steel (CASS) components do not exceed a ferrite content of 30 ferrite number (FN).

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 Ni-Cr-Fe alloy, martensitic stainless steel, and precipitation-hardened stainless steel. Austenitic stainless steel and Ni-Cr-Fe alloy base materials with primary pressure-retaining applications are used in 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 in which 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 reflective stainless steel. In addition, compounded materials are silicated to provide protection of austenitic stainless steels against stress corrosion caused by 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 (N-m) (50 foot-pounds (ft-lbs)) 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 they relate to quality standards for design, fabrication, erection, and testing. These requirements are met for materials specifications by complying with the appropriate provisions of the ASME Code, and by applying the Code cases included in RG 1.84. In addition, in the 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,” dated May 19, 2000, the staff discusses an acceptable screening method, based upon Molybdenum content, casting method, and ferrite content, for determining 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 1.84, as well as the 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 that the applicant discuss (1) the impact of this aging effect on the integrity of these components, (2) the consideration of the thermal embrittlement mechanism in the design and material selection for the RCPB components, and (3) the need for inspections to detect this aging effect. In its response, the applicant stated that based on its experience with casting materials, the selection of low-carbon-grade casting, i.e., CF3A, and control of the material specifications to below 20 FN, thermal aging should not have a significant impact on the integrity of the components. The applicant responded further that the ASME Code inservice inspections (ISI) will be relied on to detect the effects of any thermal aging. The response to RAI 251.012 discusses the COL action items regarding these inspections in a proposed change to 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. The applicant revised DCD Tier 2, Section 5.2.3.1, “Materials Specifications,” to include this revised FN. The staff reviewed the revised DCD and, subject to the clarification discussed below, finds it acceptable because it conforms to the guidance in RG 1.31, “Control of Ferrite Content in Stainless Steel Weld Metal,” and the criteria discussed in the May 19, 2000, letter from C.I. Grimes to D.J. Walters.

The staff requested the applicant 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 percent deviation between the measured and calculated values), as discussed in the May 19, 2000, letter from C.I. Grimes to D.J. Walters. This was Open Item 5.2.3-1. Subsequent to the issuance of the DSER, the applicant revised the DCD to indicate that the calculation of ferrite content is based on Hull’s equivalent factors. Therefore, since the application is consistent with the staff’s position on a method the staff finds acceptable for calculating the δ-ferrite in CASS, Open Item 5.2.3-1 is resolved.

5.2.3.3 Compatibility of Materials with the Reactor Coolant

The materials of construction employed in the RCPB that are in contact with the reactor coolant, contaminants, or radiolytic products must be compatible and must meet the requirements of GDC 4, “Environmental and Dynamic Effects Design Bases,” as they relate to the compatibility of components with environmental conditions. The requirements of GDC 4 are met by complying with the applicable provisions of the ASME Code and with the regulatory 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 to the guidance provided in RG 1.44. In addition, ferritic low-alloy and carbon steels used in the 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 address 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 because 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 the revised 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 to be acceptable.

In addition, the staff finds that the materials for the AP1000 design are compatible with the reactor coolant and meet the requirements of GDC 4, "Environmental and Dynamic Effects Design Bases," because they meet the guidance provided in RG 1.44, which 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, GDC 1, as it relates to nondestructive 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 31 are met through compliance with the acceptance standards in Article NB-2300 of ASME Code, Section III, and Appendix G, Article G-2000, to 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 the regulatory positions of 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 nondestructive 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. The design also meets the controls for welding and material preservation by conforming to RGs 1.34, 1.43, 1.50, and an acceptable alternative to RG 1.71, as discussed below.

Appendix G to 10 CFR Part 50 requires the pressure-retaining components of the RCPB to be made of ferritic materials 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 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 Part 50, Appendix G.

In addition, the AP1000 design meets the requirements of GDC 1 for NDE through its compliance with ASME Code, Section III, 1998 Edition, 2000 Addenda, as discussed in 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, that 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 stated 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, will be modified to include ASME Code, Section III, Subarticle NB-4400. The staff reviewed the revised DCD and determined that it acceptably addressed 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," that the applicant stated that the AP1000 design takes exception to RG 1.71. 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 to explain how the AP1000 design will ensure that welds made in areas of limited accessibility and/or visibility meet the fabrication requirements of ASME Code, Section III, for welds that are not volumetrically examined. 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 to 10 CFR Part 50 and GDC 1, 14, and 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, as it relates to nondestructive testing (i.e., examination) to quality standards; GDC 4; and 10 CFR Part 50, Appendix B, Criterion XIII, "Handling, Storing, and Shipping." These requirements prevent severe sensitization of the material, by minimizing exposure of the stainless steel to contaminants that could lead to SCC, and reduce 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 regulatory positions of RG 1.31, “Control of Ferrite Content in Stainless Steel Weld Metal”; RG 1.34; 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.

The NDE requirements of GDC 1 for the examination of austenitic components are met through compliance with 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 to ASME Code, Section II, for the final heat treatment of austenitic stainless steels; American Society for Testing and Materials (ASTM) A 262, Practice A or E for materials testing; and the following guidance for 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 an alternative to RG 1.31
- 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 as discussed in Section 17.3 of this report
- RG 1.44, “Control of the Use of Sensitized Stainless Steel”
- RG 1.71, “Welder Qualification for Areas of Limited Accessibility,” with an acceptable alternative as discussed in Section 5.2.3.4 of this report

The thermal insulation used in the AP1000 design of the RCPB is acceptable because it conforms to 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 American National Standards Institute (ANSI) N.45.2.1-1973, as referenced in RG 1.37. Section 17.3 of this report provides a discussion of relevant quality assurance documents.

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

The staff noted the discussion about 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 to 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 stated 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 a minimum ferrite content of 5 FN for fully austenitic welding materials, as recommended in RG 1.31. The staff reviewed the revised DCD, specifically, DCD Tier 2, Table 5.2-1, “Reactor Coolant Pressure Boundary Materials Specifications,” and found that the specifications acceptably address the staff’s concerns.

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

5.2.3.6 <u>Welding of Alloy 52/152/690 Materials</u>

The staff has identified concerns related to the difficulties associated with welding of Alloy 690 material and the partial penetration J-groove welds of Alloy 690 materials, and the potential for welding defects to occur in Ni-Cr-Fe Alloy 52/152 weld materials. The staff requested Westinghouse to provide information on the steps it has taken to address this potential. This is Open Item 5.2.3-2.

Gas Tungsten Arc Welding (GTAW) and Shielded Metal Arc Welding (SMAW) processes are commonly used for the fabrication of Alloy 690 materials. Alloy 690 material can be welded with GTAW using Alloy 52 weld filler material. SMAW processes can be used with Alloy 152 electrodes for the fabrication of Alloy 690 material. Previous experience indicates that hot cracks are observed when Alloy 690 material is welded with 52/152 weld materials. When the SMAW process was used with an Alloy 152 electrode, oxide inclusions were observed in the weld, which necessitated removal by interpass grinding. To address potential weld quality concerns with Alloy 52/152 welds, Westinghouse proposes to control hot cracking by using proper welding techniques when welding is done using the GTAW and SMAW processes. Westinghouse also proposes additional weld inspections to ensure high-quality welds.

By letter dated December 12, 2003, Westinghouse indicated that liquid penetrant (PT) examination will be performed on the first weld layer, every one-quarter inch of the intermediate weld layers, and the final weld layer of the partial penetration J-groove welds of Alloy 690 materials. This requirement exceeds the inspection criteria specified in ASME Section III, Paragraph NB-5245, which requires PT examination after the first weld layer, successive PT examinations for every half-inch of weld metals, and PT examination of the final layer. The staff agrees with Westinghouse's proposal to increase the frequency of PT examination, which enhances the ability to detect hot cracks in the weld metal, thereby increasing the weld quality.

The staff requested that Westinghouse address the issue of pressure boundary leaks which occurred, in part, due to a lack of fusion in the partial penetration J-groove weld in the lower vessel head penetrations at South Texas Project, Unit 1. Westinghouse proposed new welding inspection requirements for ultrasonic testing (UT) of the interface where the partial penetration J-groove welds join the penetration tube. This examination will be performed to ensure that lack of fusion in the weld area is detected. The staff agrees with Westinghouse's proposed UT examination of partial penetration J-groove welds, which would be expected to detect lack of fusion defects in the weld. Westinghouse also indicated that the proposed PT examinations, followed by UT examination, are applicable to all partial penetration J-groove welds of Alloy 690 penetrations in ASME Section III, Class 1, RCS components. Westinghouse proposes to use the acceptance criteria specified in ASME Section III for PT examinations of the partial penetration J-groove welds. Westinghouse's proposal also includes the application of ASME Section XI acceptance criteria for the UT examinations of the partial penetration J-groove weld interfaces. The staff agrees that these steps will provide reasonable assurance that any weld defects will be detected and repaired during fabrication, and will enhance the quality of the partial penetration J-groove welds.

Westinghouse's proposal also discusses the NDE requirements for fabrication of full penetration butt welds. Westinghouse proposes to use the radiography and final pass PT requirements specified in ASME Section III, Paragraph NB-5000. Unlike the partial penetration J-groove welds, the fabrication of full penetration butt welds is less cumbersome, and the occurrence of weld defects is less frequent. Therefore, the staff agrees that Westinghouse's proposed NDE requirements will provide reasonable assurance that the quality of the RCPB butt welds will be maintained.

Based on the information provided by Westinghouse and discussed above, the staff concludes that Open Item 5.2.3-2 is resolved.

5.2.3.7 Low-Temperature Crack Propagation

The staff identified concerns regarding the susceptibility of high-chromium, nickel-based alloys (i.e., Alloy 690/52/152) to significantly lowered fracture toughness when tested in relatively low-temperature (e.g., less than 120 °C (248 °F)) hydrogenated water. This is Open Item 5.2.3-3. By letter dated December 31, 2003, Westinghouse provided information to address this concern.

Westinghouse cited a number of research studies in its December 31, 2003, letter (including references produced by Bettis and French researchers), and noted that reductions to fracture toughness and the resistance to crack propagation have been observed for certain high-chromium, nickel-based alloys, including Alloy 690 and weld metal Alloy EN52, in the presence of substantial concentrations of hydrogen in the service environment. This mode of material property degradation is referred to as low-temperature crack propagation (LTCP). LTCP has been observed to occur only under rising loads at relatively low temperatures and slow to moderate loading rates (less than 1000 MPa√meter/hr (909 Kip (1000 pounds) per inch squared √in./hr) for welded components). LTCP has been attributed to the stress-assisted accumulation of hydrogen adjacent to, and immediately ahead of, an advancing crack, leading to a decreasing resistance to intergranular crack propagation. The occurrence of the LTCP phenomenon requires time for the diffusion of hydrogen to the crack tip and, therefore, LTCP is not an issue for welded components subjected to rapid transients. LTCP has not been observed to occur from machined notches or at free surfaces in the absence of a sharp crack, because hydrogen does not concentrate in the diffuse strain field associated with such features. It should be noted, however, that the morphology of hot cracks (which have been shown to be potentially common in Alloy 52/152 welds) consists of rounded, interdendritic pores, and LTCP would not be expected to occur at such defects.

Based on the preceding discussion, Westinghouse concluded that four conditions are necessary for the occurrence of LTCP:

(1) relatively high concentrations of hydrogen in the environment and in the metal
(2) relatively low temperatures
(3) the presence of a sharp crack tip
(4) the presence of loads which rise at a moderate rate to levels great enough to fail the flawed material

Westinghouse conducted a review of the AP1000 primary system considering various cold temperature transients. Westinghouse concluded that the accident scenario that is most likely to result in the conditions necessary to support LTCP is the inadvertent opening of the isolation valve that separates the large-volume CMTs from the RV. This transient results in a large volume of room-temperature water flowing through the direct vessel injection (DVI) nozzle and, in particular, the Alloy 52/152 safe end butt welds used to connect the nozzle safe ends to the nozzle body. If it is assumed that the operators take no action, the temperature of the weld metal would quickly fall to near the temperature of the injection water. Thermal stresses would be induced in the pipe wall as a result of this cooling transient, which would be added to stresses from the primary system pressure.

The hydrogen concentration in a PWR primary water environment is in the range of 25 to 50 H_2/kg H_2O. Based on the research data applicable to Alloy 690 and Alloy EN52, this concentration of hydrogen is believed to be insufficient for LTCP. Although the available research did not include testing of Alloy EN152, for the purposes of this safety evaluation, it is presumed that Alloy 152 would exhibit similar properties to EN52. Therefore, Westinghouse concluded that the first condition for LTCP (i.e., the presence of relatively high concentrations of hydrogen in the environment and in the metal) is unlikely to be satisfied.

From the evaluation of the aforementioned accident scenario, Westinghouse concluded that the second condition for LTCP (i.e., the presence of low temperature) could be satisfied. Westinghouse noted that the third condition for LTCP (i.e., the presence of a sharp crack tip) is believed to be unlikely, except for the possible existence of a “lack-of-fusion” condition arising from the welding.

Westinghouse performed finite element analyses of the nozzle safe end using the thermal conditions discussed above. The results of this analysis indicate that the axial and hoop stresses were significantly below the yield stresses and, therefore, do not approach the level where LTCP would occur. Hence, Westinghouse concluded that the fourth condition for LTCP (i.e., the presence of loads which rise at a moderate rate to levels great enough to fail the flawed material) was unlikely to be satisfied.

Based on the information above, Westinghouse concluded that the conditions necessary for the occurrence of LTCP cannot take place in the AP1000 design. The staff agrees with Westinghouse’s analysis and, on this basis, concludes that Open Item 5.2.3-3 is resolved.

5.2.3.8 Conclusions

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

5.2.4 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 to 10 CFR Part 50, GDC 32, “Inspection of Reactor Coolant Pressure Boundary,” and meet 10 CFR 50.55a, “Codes and Standards.”

10 CFR 50.55a requires that ASME Code Class 1 components be designed with sufficient access to enable the performance of inservice examination of such components. The design must also 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 that components that are part of the RCPB 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.

The staff's evaluation of the inservice inspection and testing program for Class 1 components is divided into six sections, as described in the SRP. These include (1) the system boundary subject to inspection, (2) accessibility, (3) examination categories and methods, (4) inspection intervals, (5) evaluation of examination results, and (6) system leakage and hydrostatic pressure tests. Sections 5.2.4.2 through 5.2.4.7 of this report, respectively, discuss the acceptability of these elements.

5.2.4.1 Summary of Technical Information

The DCD 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. Section XI, IWA-5000, defines the requirements for system pressure tests and visual examinations. These tests verify pressure boundary integrity in conjunction with ISI.

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 the pumps is controlled administratively. Section 3.9.6 of this report evaluates this program.

Consistent with ASME Code and NRC requirements, the preparation of inspection and testing programs is the responsibility of the COL applicant for each AP1000. DCD Tier 2, Section 5.2.4, indicates that these programs will comply with the applicable ISI provisions of 10 CFR 50.55a(2). However, this reference should be 10 CFR 50.55a(b)(2). This is part of COL Action Item 5.2.4.1-1.

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 RCS, or connected to the RCS, up to and including the following:

- 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 that 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 as AP1000 equipment Class A. DCD Tier 2, Section 3.2.2, "AP1000 Classification System," discusses the system boundary for pressure-retaining components. The applicant's definition of the RCPB is consistent with the SRP and, therefore, is acceptable.

5.2.4.3 Accessibility

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

DCD Tier 2, Section 5.2.4.2, describes accessibility for inspection. ASME Code Class 1 components are designed so that access is provided in the installed condition for visual, surface, and volumetric examination, as specified by the baseline ASME Code Section XI (1998 Edition, 2000 Addenda) and mandatory appendices. Design provisions, in accordance with ASME Code, 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 ASME Code, Section XI, Article IWB-2000, "Examination and Inspection." 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 requirements of Article IWB-2000 also list the methods of examination for the components and parts of the pressure-retaining boundary.

The applicant's examination techniques and procedures used for preservice inspection or ISI of the system are acceptable, if they agree 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 ASME Code, Section XI.

- The methods, procedures, and requirements regarding qualification of nondestructive 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 ASME Code, Section XI. In addition, the performance demonstration for ultrasonic examination systems reflects the requirements provided in Appendix VIII, "Performance Demonstration for Ultrasonic Examination Systems," to Division 1 of ASME Code, Section XI.

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

5.2.4.5 <u>Inspection Intervals</u>

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

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

5.2.4.6 <u>Evaluation of Examination Results</u>

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.”

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

5.2.4.7 System Leakage and Hydrostatic Pressure Tests

The pressure-retaining ASME Code Class 1 component leakage and hydrostatic pressure test program is acceptable if the program agrees with the requirements of ASME Code, 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.

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

5.2.4.8 Conclusions

Based on its 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. In addition, the inspection and test program satisfies GDC 32 because it meets the applicable requirements of ASME Code, Section XI, as endorsed in 10 CFR 50.55a.

5.2.5 Reactor Coolant Pressure Boundary Leakage Detection

The staff reviewed the capability of the AP1000 design 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 they relate 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 they relate 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, "Leak-Before-Break," of this report address 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) total integrated dose (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. Leakage detection and monitoring equipment is not required to 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, operating personnel may detect various minor leaks of the RCPB. If these leaks can be subsequently observed, quantified, and routed to the containment sump, this leakage is 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 the applicant considered possible intersystem leakage points across passive barriers or valves and their detection methods. 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; 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 steam generator blowdown radiation monitor
- the main steamline radiation monitor
- laboratory analysis of condensate

In addition, leakage from the RCS to the component cooling water system (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, the following three diverse methods may be utilized to quantify and assist in locating the leakage:

(1) containment sump level
(2) reactor coolant system (RCS) inventory balance
(3) 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 safe-shutdown earthquake (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 the 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. The leakage detection and monitoring equipment is not required to be safety-related.

Based on 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 they relate 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 the total flow rate can be established and monitored. As stated in Section 5.2.5.1 of this report, 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/minute (L/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 L/min (0.5 gpm) in 1 hour. The containment sump level is monitored by three sensors. The third sensor is provided for redundancy in detecting main steamline leakage. The minimum detectable leak is 0.03 gpm, which has sufficient sensitivity to detect 0.25 gpm.

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 provide an 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 L/min (1 gpm) in less than 1 hour. In DCD Tier 2, Section 5.2.5.3.3, the applicant stated that the N_{13}/F_{18} radioactivity monitor can detect a 1.89 L/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 leakage detection systems (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 liters per minute (gallons 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, 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., a 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 3.4.7, “RCS Operational Leakage,” and 3.4.9, “RCS Leakage Detection Instrumentation,” to the Westinghouse Owners Group standard technical specification (WOG STS) 3.4.13, “RCS Operational Leakage,” and 3.4.15, RCS Leakage Detection Instrumentation.” AP1000 TS 3.4.9 requires one containment sump level channel and one containment atmosphere radioactivity monitor to be operable for Modes 1, 2, 3, and 4. However, there are two notes associated with this TS which relax the requirements for these two leakage detection instrumentation systems under certain conditions. The first note states that the containment atmosphere radioactivity monitor is only required to be Operable in Mode 1 with rated thermal power (RTP) greater than 20 percent. The second note states that containment sump level measurements cannot be used for leak detection, if the leakage is prevented from draining to the sump (e.g., such as by redirection to the IRWST by the containment shell gutter drains). In RAI 410.006, the staff asked what compensatory actions will be required to perform the function of RCS leakage detection if both notes are satisfied during Modes 1, 2, 3, and 4.

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 that the plant power is less than 20 percent of the RTP; no additional compensatory leakage monitoring actions are needed 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.

If a drain path is 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 3.4.9 must be entered. The compensatory action is to perform surveillance requirement (SR) 3.4.7.1 (RCS water inventory balance) more frequently (i.e., 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.

From 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 they relate 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) 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 system 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 system 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 severe enough 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 the AP1000 systems and components to maintain and perform their safety functions in the event of an earthquake. The design also 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 fulfilled 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 fulfilled 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 fulfilled 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, therefore, is acceptable.

5.3 <u>Reactor Vessel</u>

DCD Tier 2, Section 5.3.1.2, "Safety Description," describes the AP1000 RV. 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 the 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's 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 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 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 SRP Section 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. This will ensure 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 and 30, and 10 CFR 50.55a(a)(1) require 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 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 the reactor coolant pressure boundary 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 the RCPB shall be designed with sufficient margin to ensure that, when stressed under operation, maintenance, testing, and postulated accident conditions, it will behave in a nonbrittle manner, and minimize the probability of rapidly propagating fracture.

- GDC 32 requires the RCPB components shall be designed to permit an appropriate material surveillance program for the reactor pressure vessel.

- Appendix G to 10 CFR Part 50 specifies the fracture toughness requirements for ferritic materials of the pressure-retaining components of the RCPB. The staff reviewed 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 to 10 CFR Part 50 presents the requirements for a materials surveillance program to monitor the changes in the fracture toughness properties of materials in the

RV beltline region resulting from exposure to neutron irradiation and the thermal environment. These requirements include conformance to 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 whether they meet the relevant requirements of Appendix H for determining 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 to 10 CFR Part 50.

The chemical composition of the ferritic materials of the RV beltline are restricted to the 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. Numerous examinations are also performed in addition these requirements.

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 not be found by the straight-beam examination.

In addition to the ASME Code, Section III, NDE, 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, NDE requirements.

Penetrant Examinations. The partial penetration welds for the control rod drive mechanism head adapters and the top instrumentation tubes are inspected by PT after the root pass, in addition to ASME Code requirements. Section 4.5.1 of this report provides additional information on the control rod drive mechanisms.

Magnetic Particle Examination. The magnetic particle examination requirements described 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:

- Only the prod, coil, or direct contact method is used prior to the final postweld heat treatment.

- Only the yoke method is used after the final postweld treatment.

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 the inside diameter surfaces of carbon and low-alloy steel products that have their properties enhanced by accelerated cooling

Weld Examination. Magnetic particle examination of the welds attaching the closure head lifting lugs and refueling seal ledge to the RV is conducted after the first layer and again after 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

DCD Tier 2, Section 5.2.3, “Reactor Coolant Pressure Boundary Materials,” discusses welding of ferritic steels and austenitic stainless steels.

5.3.2.1.5 Fracture Toughness

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

5.3.2.1.6 Material Surveillance

In the surveillance program, the evaluation of radiation damage is based on preirradiation testing of C_V and tensile specimens, and postirradiation testing of C_V, tensile, and ½-T compact tension fracture mechanics test specimens. The program evaluates the effect of radiation on the fracture toughness of RV steels based on the transition temperature and fracture mechanics approaches. 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 heat-affected zone (HAZ) materials will be retained. The applicant’s schedule for removing the capsules for postirradiation testing includes the withdrawal of five capsules which is in accordance with ASTM E-185-82 and Appendix H to 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. DCD Tier 2, Section 1.9 discusses the conformance of the AP1000 design with RG 1.65, “Materials and Inspections for Reactor Vessel Closure Studs.” Nondestructive examinations are performed in accordance with ASME Code Section III.

5.3.2.2 <u>Staff Evaluation</u>

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 also reviewed the AP1000 RV materials to ensure that the relevant requirements of GDC 1 and 30 and 10 CFR 50.55a(a)(1) have been met as they relate to the material specifications, fabrication, and NDE. Compliance with these requirements will determine whether the RV materials are adequate 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 the ASME Code, Section III, Paragraph NB-4100. The NDE 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 complies with Paragraph NB-5000, for normal methods of examination. The applicant identified other inspections, as previously stated, 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 30 and 10 CFR 50.55a(a)(1) as they relate to the material specifications, fabrication, and NDE of RV materials.

Section 5.2.3 of this report provides the staff’s evaluation of the welding of ferritic steels and austenitic stainless steels, as well as addresses GDC 4.

DCD Tier 2, Table 5.3-1, provides the maximum limits for the elements in the materials of the RV beltline. 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 to 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 ASME Code, Section III, Paragraph NB-2300, 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 101.7 N-m (75 ft-lbs). DCD Tier 2, Table 5.3-3, indicates that the end-of-life (EOL) values for the USE are greater than 67.8 N-m (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 EOL 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 RCPB. This methodology will provide adequate safety margins during operating, testing, maintenance, and postulated accident conditions. Compliance with the provisions of Appendix G to 10 CFR Part 50 satisfies the requirements of GDC 14 and 31 and 10 CFR 50.55a regarding the prevention of fracture of the RCPB. Therefore, the staff finds that the applicant has adequately met the requirements of GDC 14 and 31, and 10 CFR 50.55a. Section 5.3.3.2 of this report provides the staff's evaluation of compliance with 10 CFR 50.61 (pressurized thermal shock).

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

The staff requested, in RAI 251.014, that the applicant describe the lead factors for the surveillance capsules. The staff requested that the applicant commit, in the AP1000 DCD, that an analysis will be performed for the COL application with regard to the capsule/holder model 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 defining 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 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 5.3.2.2-1.

To meet the requirements of GDC 32, the AP1000 design includes provisions for a material surveillance program to monitor changes in the fracture toughness caused by exposure of the

RV beltline materials to neutron radiation. 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. ASTM E-185 recommends a minimum of four surveillance capsules for an RV with an EOL shift between 38 °C and 93 °C (100 °F and 200 °F). 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 will be those predicted to be most limiting in terms of 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 to 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 the requirements of GDC 32.

The applicant indicated that the material used to fabricate the closure studs will meet the fracture toughness requirements of Section III of the ASME Code and Appendix G to 10 CFR Part 50. NDE of the studs will be performed 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 Paragraph 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 requirement in GDC 31 for the prevention of fracture of the RCPB and the requirements of Appendix G to 10 CFR Part 50, as detailed in the provisions of Section III of the ASME Code.

5.3.2.3 Conclusions

The staff concludes that the AP1000 RV material specifications, RV manufacturing and fabrication processes, NDE 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 ASME Code, Section III, Appendices G and H to 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 Materials Surveillance Program

Appendix H, “Reactor Vessel Material Surveillance Program Requirements,” to 10 CFR Part 50 presents the requirements for a material surveillance program 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 is greater than 1.0 million electron volts) dosimeters. Measurement of the irradiated material samples yields a measure of the embrittlement, and measurement of the dosimeter activation estimates the irradiation exposure.

RG 1.190, “Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence,” which is based on GDC 14, 30, and 31, describes methods and practices acceptable to the staff regarding calculational techniques and statistical practices using the dosimetry measurements. In addition, the results of the dosimetry are used to benchmark and validate calculational methods for estimating vessel irradiation.

In the DCD and its response to RAI 440.037, Revision 1, the applicant clarified its methods and practices regarding 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. This is COL Action Item 5.3.2.4-1.

5.3.3 Pressure-Temperature Limits

The staff reviewed DCD Tier 2, Section 5.3.3, “Pressure-Temperature Limits,” in accordance with SRP Section 5.3.2, “Pressure-Temperature Limits and Pressurized Thermal Shock.” The applicant’s P/T limit curves are acceptable if they meet codes and standards, and regulatory guidance commensurate with the safety function to be performed. This will ensure 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 requires that SSCs important to safety shall be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety function to be performed.

- GDC 14 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 nonbrittle manner and minimize the probability of rapidly propagating fracture.

- GDC 32 requires 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 on10 CFR Part 50, Appendix G; RG 1.99, Revision 2, and SRP Section 5.3.2.

Appendix G to 10 CFR Part 50 requires that P/T limit curves for the reactor pressure vessel (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 Section 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 to 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 to the ASME Code requires a safety factor of 1.5.

The methods of Appendix G to 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 one-quarter 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 one-quarter thickness (1/4T) and three-quarters 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 to 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, Revision 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. 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 whether they meet the relevant requirements for Appendix H as they relate to determining and monitoring 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 shutdown 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 to calculate 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. This is 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 EOL RT_{NDT} will 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 consistent 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. These 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 ensure adequate safety margins for the structural integrity of the ferritic components of the RCPB.

The 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 ensure adequate safety margins for the structural integrity of the RCPB ferritic components. The SRP indicates that P/T limits established for the RCPB consistent with the requirements of 10 CFR Part 50, Appendix G, and to ASME Code, Section III, Appendix G ensure 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 with10 CFR Part 50, Appendix G, 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. 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 has adequately met 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 to 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 ensuring 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 nonbrittle manner and an extremely low probability of rapidly propagating fracture. In the DCD, the applicant indicated that RG 1.99, Revision 2, is used to calculate the ART. 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). This value 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 the effects of radiation on fracture toughness properties that are applicable to the 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. The staff verified that adequate safety margins against nonductile behavior of rapidly propagating failure of ferritic components will exist, 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 10 CFR Part 50, 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 whether they meet the relevant requirements for Appendix H as they relate to determining and monitoring fracture toughness. Section 5.3.2, "Reactor Vessel Materials," of this report provides the staff's review of the material surveillance program.

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 ART 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 calculated fluence values. The staff finds this acceptable because it complies with RG 1.190. Another exception is that the plane strain facture toughness (K_{Ic})critical stress intensities are used in place of the crack arrest facture toughness (K_{Ia}) critical stress intensities. This methodology is taken from the staff-approved ASME Code Case N-641. The staff found the applicant's responses to be 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 to 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. Because applicants using the AP1000 are required to meet the requirements of 10 CFR Part 50, Appendix G, they must meet the closure head requirements of Appendix G to 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 to 10 CFR Part 50. The applicant was asked to provide those limitations that are consistent with the present TS and 10 CFR Part 50, Appendix G, or provide closure flange limitations with new TS that are consistent with 10 CFR Part 50, Appendix G. This was Open Item 5.3.3-1.

By letter dated October 10, 2003, the applicant provided the revised sections of the AP1000 DCD, which also included revised P/T curves to address the reactor vessel closure head flange requirements of 10 CFR Part 50, Appendix G. The closure flange limitation is based on an RT_{NDT} of 10 °F, in accordance with DCD Tier 2, Table 5.3-3. The staff confirmed that the proposed curves are in accordance with 10 CFR Part 50, Appendix G for the closure flange. In addition, the applicant indicated that the RNS relief valve setpoint and capacity would also be revised as a result of a revised LTOP evaluation based on the new P/T curves. The staff found that the applicant adequately revised the appropriate sections of the AP1000 DCD to address the staff's open item with respect to the reactor vessel closure flange limitations. Therefore, the staff considers Open Item 5.3.3-1 to be closed because the applicant adequately meets the reactor vessel closure head flange requirements of 10 CFR Part 50, Appendix G.

As stated above, the applicant also provided revised P/T curves for the AP1000 design which 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 the AP1000 RV design. In addition, they are the limiting curves based on copper and nickel material composition. The applicant also indicated that the COL applicant will address the use of plant-specific curves during procurement of the RV. As noted in the bases to TS 3.4.14, the applicant indicated that use of plant-specific curves requires evaluation of the LTOP system. This includes evaluating the setpoint pressure for the normal RNS 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. This is COL Action Item 5.2.2.2-1.

The staff requested, in RAI 251.017, that the applicant provide details for the P/T limit calculations, including their 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 it used the methodology of RG 1.99, Revision 2, to estimate the shift in reference temperature. The ART is the sum of 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 EOL fluence is 9.762×10^{19} neutrons/centimeter squared (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 7.2 °C (45 °F) and 5.6 °C (42 °F), respectively. The margin values for the 1/4T and 3/4T locations of the lower girth weld are 18.9 °C (66 °F) and 10 °C (50 °F), respectively.

The AP1000 DCD provides the values of the copper and nickel composition and the initial RT_{NDT} values. The applicant calculated the adjusted reference temperature values to be 17.2 °C (63 °F) and 13.3 °C (56 °F) at the 1/4T and 3/4T locations of the forging, respectively, and 33.9 °C (93 °F) and 18.9 °C (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, Revision 2. This RG provides reasonably accurate and conservative predictions of adjusted reference temperatures for RV beltline materials that are produced domestically. The staff finds the applicant's approach 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 of RG 1.99, Revision 2, may not apply to 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, and the chemical content controls imposed on the RV materials meet the guidelines for new plants, as specified in RG 1.99, Revision 2.

5.3.3.3 Conclusions

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 material surveillance program developed in conformance with Appendix H to 10 CFR Part 50. The use of operating limits, as 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. This constitutes an acceptable basis for satisfying the requirements of 10 CFR 50.55a, Appendix A to 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 Section 5.3.2, "Pressure-Temperature Limits and Pressurized Thermal Shock." Title 10, Section 50.61, "Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events," of the Code of Federal Regulations defines the fracture toughness requirements for protection against PTS events. This section also 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 reference pressurized thermal shock temperature (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 PTS rule for the vessel design objective. DCD Tier 2, Tables 5.3-1 and 5.3-3 provide the material properties, initial RT_{NDT}, and EOL RT_{PTS} requirements and predictions. Materials 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 to serve as a limiting level of RV material embrittlement beyond which operation cannot continue without further plant-specific evaluation. The screening criteria are given in terms of reference temperature, RT_{PTS}. The screening criteria are 132.2 °C (270 °F) for plates and axial welds, and 148.9 °C (300 °F) for circumferential welds. The RT_{PTS} is defined by the following equation:

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

where:

$RT_{NDT\,(U)}$ = initial reference temperature

ΔRT_{PTS} = mean value in the adjustment in reference temperature caused by irradiation

M = margin to be added to cover uncertainties in the initial reference temperature, copper and nickel contents, fluence and calculational procedures

The applicant demonstrated that the AP1000 design meets the PTS screening criteria. 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} is -23.3 °C (-10 °F) for the forging, and -28.8 °C (-20 °F) for the circumferential weld. In response to RAI 251.019, the applicant indicated that the maximum assumed neutron fluence is 9.7E19 n/cm^2 for the forgings, and 2.85E19 n/cm^2 for the circumferential weld at EOL (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 Conclusions

The staff concludes that the AP1000 RV meets the relevant requirements of 10 CFR 50.61 because calculations show that the RV beltline materials will be substantially below the PTS screening criteria after 60 years of operation. The COL applicant will address verification of RT_{PTS} values based on plant-specific material properties and projected neutron fluences for the plant design objective of 60 years. This is COL Action Item 5.3.4.3-1.

5.3.5 Reactor Vessel Integrity

The staff reviewed DCD Tier 2, Section 5.3.4, “Reactor Vessel Integrity,” in accordance with SRP Section 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. This will ensure 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”; 10 CFR Part 50, 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 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, “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 boundary 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 nonbrittle manner and will minimize the probability of rapidly propagating fracture.

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

Title 10, of the Code of Federal Regulations, Section 50.61, defines the fracture toughness requirements for protection against PTS events. This section also 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 reviewed 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 material 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 determining and monitoring 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, no penetrations are made 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 and 30, 10 CFR 50.55a, and the requirements of ASME Code, Section III. The vessel design and construction enables inspection in accordance with 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 and heat transfer between the coolant and the metal of the vessel and internals. The analysis also 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.

DCD Tier 2, Section 5.3.3, “Pressure-Temperature Limits,” and the AP1000 TS provide the operating limitations for the RV. 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 of the RV under these severe postulated transients. TMI Action Item II.K.2.13, “Thermal-Mechanical Report on Effect of HPI [High Pressure Injection] on Vessel Integrity for Small-Break Loss-of-Coolant Accident with no AFW [Auxiliary Feedwater],” is satisfied upon submittal of RT_{NDT} values which are below the PTS rule screening values. Section 5.3.4 of this report further discusses PTS.

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.

DCD Tier 2, Section 5.3.4.7, “Inservice Surveillance,” provides further details of the applicant’s inservice surveillance activities with regard to the components of the RV. Because radiation levels and remote underwater accessibility limit 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 ISI requirements of the ASME Code.

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

5.3.5.2 Staff Evaluation

Although the staff reviewed most areas separately in accordance with the other SRP 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. Section 5.3.3 of the SRP provides the acceptance criteria and references that form the bases for this evaluation.

The staff reviewed the information in each area to ensure that inconsistencies did not exist that would reduce the certainty of vessel integrity. The areas reviewed and the sections of this report in which they are discussed are as follows:

- reactor coolant pressure boundary materials (Section 5.2.3)
- reactor coolant system pressure boundary ISI and testing (Section 5.2.4)
- reactor vessel materials (Section 5.3.2)
- pressure-temperature limits (Section 5.3.3)
- pressurized thermal shock (Section 5.3.4)

The integrity of the RV is ensured for the following reasons:

- 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 ensure that the vessel will not fail because of material or fabrication deficiencies.

- The RV will operate under conditions, procedures, and protective devices that ensure that the vessel design conditions will not be exceeded during normal reactor operation, maintenance, testing, and anticipated transients.

- The RV will be subjected to periodic inspection to demonstrate that its 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, they will behave in a nonbrittle manner and will minimize the probability of rapidly propagating fracture.

5.3.5.3 Conclusions

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. Therefore, the staff finds the structural integrity of the AP1000 RV to be 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, as well as 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 its service life, in accordance with Appendix H to 10 CFR Part 50. In addition, operating limitations on temperature and pressure will be established for the plant in accordance with Appendix G, “Protection Against Nonductile Failure,” of ASME Code, Section III, and Appendix G to 10 CFR Part 50.

5.4 Component and Subsystem Design

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

5.4.1 Reactor Coolant Pump Assembly

The AP1000 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 coast-down flow after an RCP trip.

The RCP is an integral part of the RCPB. Section 5.2 of this report discusses 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 RCPs. 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 frequency drive, which is used only during heatup and cooldown when the reactor trip breakers are open. During power operations, the variable frequency drives is isolated and the pump are run at constant speed.

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

5.4.1.1 Pump Performance

GDC 10 requires that the reactor core and associated coolant, control, and protection systems 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 states that to meet the requirements of GDC 10, the hot rod in the core must not experience a departure from nucleate boiling, or the DNBR limit must not be 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 to deliver the forced reactor coolant flow and coast-down flow rates after an 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; the design flow rate is 17,886 m^3/hr (78,750 gpm) per pump, the developed head is 111.25 m (365 ft), and the synchronous speed is 1,800 revolutions per minute (rpm). DCD Tier 2, Table 4.4-1, provides the thermal and hydraulic data for the AP1000 design; the vessel minimum measured flow rate is 68,516 m^3/hr (301,670 gpm), and the vessel thermal design flow rate is 67,229 m^3/hr (296,000 gpm), representing an uncertainty in the design and measurement flow 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 the Chapter 15 safety analyses, with or without the revised thermal design procedure. AP1000 TS LCO 3.4.1, "RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits," 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, in accordance with TS 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. Therefore, GDC 10 is met. The staff concludes that the RCP design flow capacity is acceptable. The total delivery capability of the four RCPs will be verified using the inspections, tests, analyses, and acceptance criteria (ITAAC) described in DCD Tier 1, Table 2.1.2-4, Item 9.a.

The startup testing of the AP1000 contained in DCD Chapter 14, "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, "Reactor Coolant System Flow Measurement," to verify the 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, "Reactor Coolant System Flow Measurement at Power Conditions," to verify that the RCS flow equals or exceeds the minimum value required by the plant TS. DCD Tier 2, Section 14.4.2, "Test Specifications and Procedures," requires the applicant to provide test specifications and test procedures for the preoperational and startup tests for review by the NRC. Therefore, the staff concludes that the AP1000 initial test program provides adequate verification of the total delivery capability of the RCP for adequate core cooling.

As stated in DCD Tier 2, Section 5.4.1.3.1, ample margin is provided between the available net positive suction head (NPSH) and the required NPSH to provide operational integrity and to minimize the potential for cavitation 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. Because the available NPSH is always larger than the required NPSH, cavitation is not a concern.

5.4.1.2 Coast-Down 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 coast-down time following a pump trip and loss of electrical power. Continued coast-down 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, as analyzed in DCD Tier 2, Chapter 15.3, "Decrease in Reactor Coolant System Flowrate." The ability of the RCP flywheel-rotor design to provide sufficient rotating inertia, and thus flow coast-down capability following an RCP trip, is verified through the safety analyses of the loss-of-flow transients which demonstrate that the minimum DNBR limit is not violated. The staff has reviewed the safety analyses of the design-basis transients for partial and complete loss of forced reactor flow, as described in DCD Tier 2, Sections 15.3.1 and 15.3.2, respectively. The RCP coast-down 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. The staff has approved this computer code for the AP1000 transient analyses, as discussed in Section 21.6.1 of this report. The analysis results for 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 coast-down capability.

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

5.4.1.3 Rotor Seizure

In DCD Tier 2, Section 5.4.1.3.6.2, the applicant stated that the design of the AP1000 RCP (and motor) precludes the instantaneous stopping of any rotating component of the pump or motor. However, DCD Tier 2, Section 15.3.3, "Reactor Coolant Pump Shaft Seizure (Locked Rotor)," presents a design-basis analysis of a postulated RCP rotor seizure. 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 the 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 results to be acceptable, as discussed in Section 15.3.2 of this report.

5.4.1.4 Reactor Coolant Pump Flywheel Integrity

The following regulatory requirements are applicable 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 shall be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety function to be performed.

- GDC 4 in Appendix A to 10 CFR Part 50 requires that SSCs important to safety shall 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 stated that each RCP for the AP1000 reactor is designed with a high-density flywheel and high-inertia rotor. These components provide the RCP with a continual coast-down capability following an RCP trip. The applicant also stated 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 the 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 a rupture of the RCP rotor resulting. The applicant described 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." This report 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, Revisions 0 and 1, to assess the AP1000 RCP flywheel design, and whether the design had the potential to impact the structural integrity of the RCPB. Section 5.4.1.3 of this report describes the staff's evaluation of the RCP rotors for protection against seizure.

During the staff's review of the AP600 design certification, the staff asked the applicant (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 the 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 of the AP1000 RCP flywheels. In RAI 251.21, the staff also requested the applicant to confirm that its previous responses to AP600 RAIs 251.2 through 251.23 were applicable to the AP1000 RCP flywheel design. If not, the staff asked the applicant to 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 response to RAI 251.21 for AP1000, 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 this report 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, to update and clarify some of the design aspects of 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 the applicant's other 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. No industry experience 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 the faulted stress limits found in 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 also 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 analysis presented 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. Therefore, the staff finds them to be acceptable.

AP1000 RAI 251.21 was predicated, in part, on verifying that 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 251.21 (and its 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 of 6-inch nominal pipe size (NPS) or larger is qualified for leak before break. Therefore, the stresses associated with the largest RCS pipe break analyzed for the flywheel integrity for a LOCA associated with a 4-inch NPS RCS pipe break. This provides additional information that clarifies the limiting stresses that the applicant 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-inch NPS 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 material 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. The staff finds this acceptable 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 in 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.

Because 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's 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 Conclusions

The staff has reviewed the information in DCD Tier 2, Section 5.4.1.3.6.3, WCAP-15994-P, Revisions 0 and 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 the applicant's acceptable conclusions on LBB described in Section 3.6.3 of this report, 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. In addition, the staff finds 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 ensure 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 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 HX, which cools the primary side under emergency conditions.

The SG channel head, tube sheet, 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 heat transfer function of the SGs, 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 the 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, and enters the RV, thus completing a cycle.

If the PRHR system is activated, flow passes from the outlet of the PRHR HX, through the SG PRHR nozzle connection, and 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 exits 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. Centrifugal moisture separators then 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).

DCD Tier 2, Tables 5.4-4 and 5.4-5, respectively, detail the SG design requirements and design parameters. 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 of the DCD. 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. Section 5.4.2.3 of this report discusses the MSGTR/containment bypass issue.

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 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 of the safety function to be performed.

- GDC 14 requires that the steam generator 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 steam generator 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 steam generator shall be designed with sufficient margin to assure that when stressed under operating, maintenance, testing, and postulated accident conditions, it will behave in a nonbrittle manner and minimize the probability of rapidly propagating fracture.

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, designated as Class 1, and the secondary side of the SG designated 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 tube sheet is weld clad with Ni-Cr-Fe alloy (ASME SFA-5.14). The SG tubes are seal-welded to the tube sheet 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 tube sheet 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). The heat and lot of tubing material for each SG tube are recorded and documented. In addition, archive samples are available to the COL applicant for use in future materials testing programs or 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 tube sheet is expanded hydraulically to close the crevice between the tube and tube sheet.

- 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.

- Antivibration 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 tube sheet 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 15.2–cm (6-inch) access ports are available for sludge lancing, which is a method for cleaning the SG in which a hydraulic jet inserted through the access ports loosens deposits and flushes them out of the SG. These ports can also be used for inspection of the tube bundle and retrieval of loose objects. In addition, two 10.2-cm (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 simulated the effects of SG water chemistry on the tubes. Test results indicate 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 all-volatile treatment 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 GDC 1, 14, 15, and 31, and Appendix B to 10 CFR Part 50.

Selection and Fabrication of Materials. The staff reviewed the materials selected (e.g., austenitic and ferritic stainless steels, ferritic low-alloy steels, carbon steels, and high nickel alloys) for the SG in terms of their adequacy, suitability, and compliance with ASME Code, Sections II and III. The requirements of GDC 1 are met for materials specifications by complying with the ASME Code. The requirements of the Code cases are fulfilled by meeting the appropriate provisions in RG1.84. The fracture toughness requirements of GDC 14 and 31 for Class 1 ferritic materials are met by satisfying the requirements of Appendix G to 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 that the materials are acceptable because they meet the requirements/guidance of the ASME Code, Sections II and III, and RG 1.84.

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

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

Based on their compliance with code requirements and RG 1.84, as well as 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 concludes that they meet the requirements of Appendix G to 10 CFR Part 50 and the requirements of 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 ASME Code, Section III, Subarticle NC-2300. Therefore, the staff finds that the AP1000 SG design satisfies the fracture toughness requirements of GDC 14 and 31.

Steam Generator Design. The staff reviewed the design and fabrication of the SG to determine the extent to which crevice areas are minimized and whether sufficient corrosion allowance exists. 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 to minimize buildup of corrosion products, and by meeting the appropriate provisions of 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, the AP1000 design expands the tubes into the tube sheet for the entire length of the tube sheet and uses trifoil broached hole tube support plates.

The staff requested, in RAI 252.006, that the applicant clarify which tube support plate design will be used in the AP1000 (i.e., open lattice (egg crate) or broached hole), because 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 on the broached hole support plate design only. 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 the revised DCD and found that the applicant had removed references to the open lattice (egg crate) tube support plate design as an option. Based on this revision, the staff finds the discussions of the tube support plate design in the DCD to be acceptable.

The staff requested, in RAI 251.022, the applicant to provide the results of the flow-induced vibration (FIV) tests and calculations on the SGs, with special emphasis on fluid-elastic vibration. In addition, the staff requested the criteria for establishing the instability threshold for ensuring that fluid-elastic behavior does not contribute unacceptably to FIV 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 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 in 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 followup question to RAI 251.022, the staff requested the applicant to provide the basis for the 0.75 fluid-elastic stability ratio criterion, and describe whether "time domain" analyses had been performed demonstrating that stresses associated with the criterion are negligible. The applicant's response did not address the staff's question. The staff's understanding of the applicant's response is that the 0.75 factor is based on judgment rather than being selected to address any specific uncertainty or time domain analysis. However, the applicant did not explain the rationale by which this judgment was reached. Furthermore, the applicant stated 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 requested the rationale for assuming that the alternating stress and associated fatigue usage induced by fluid-elastic coupling is negligible for the case in which the fluid-elastic stability ratio is 0.75. This was Open Item 5.4.2-1 in the DSER.

By letter dated July 3, 2003, Westinghouse stated that the design stability ratio criterion of 0.75 is based on judgment and is intended to bound the onset of instability due to uncertainties in the analytically derived flow fields, mathematical models of the tube/support system, and material properties. Conservative instability constants and damping values are employed such that the stability ratio criterion need not be adjusted to reflect uncertainties in these parameters. Assuming ideal pinning of the tubes at the tube support plates, tube response is driven entirely by turbulence and vortex shedding when below the fluid-elastic instability threshold. Westinghouse analytical models conservatively bound the turbulence and vortex shedding response.

Westinghouse noted that actual SG tube supports are not ideal pinned supports. Nominally, these supports have widths and clearances that permit a range of possible tube/support interaction conditions. Among the possibilities is that the tube may initially respond to the flow field within the clearances as if one or more of the supports is not present. Thus, the tube could begin to respond to turbulence, vortex shedding, or fluid-elastic excitation earlier than would be expected from a model which assumes ideally pinned supports until the magnitude of vibration causes intermittent interaction with the support across the clearance. Westinghouse stated that some early model steam generators experienced moderate tube wear in the u-bends resulting from fluid-elastic rattling within loosely fitting AVB supports. Westinghouse considers the alternating stresses from vibration from within tube supports as part of its fatigue analyses.

The opposite extreme to loose supports are supports that become effectively clamped as a result of a buildup of deposits in the clearance region or denting. Such a condition can lead to fluid-elastic excitation earlier than would be expected from a model that assumes pinned supports. The 0.75 stability ratio criterion is not intended to address such off nominal conditions. Instead, Westinghouse considers a range of support assumptions and corresponding damping models in its analyses.

Based on the above, the staff concludes that Westinghouse has provided an adequate description of the basis for the 0.75 stability ratio design criterion and the context of its usage. Therefore, Open Item 5.4.2-1 is resolved.

<u>Compatibility of the Steam Generator Components with the Primary and Secondary Coolant</u>. 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 requirements of GDC 14 are met through proper maintenance of primary and secondary water chemistry to ensure the integrity of the barrier between primary and secondary fluids.

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 report. When used in plant operations, these guidelines reduce the possibility of SCC, denting, pitting, and wastage of SG tubes through chemistry controls. Therefore, 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 the requirements of GDC 14.

<u>Cleanup of Secondary Side</u>. 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 requirements of GDC 14 are 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 to verify access to 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 because 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 report.

5.4.2.1.3 Conclusions

The staff concludes that because the AP1000 SG materials satisfy staff criteria regarding materials selection, fabrication, and compatibility with the environments, the materials are acceptable and meet the requirements of GDC 1, 14, 15, and 31, as well as 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

- GDC 32, "Inspection of Reactor Coolant Pressure Boundary," as it relates to the 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 for the SG tube surveillance requirements (i.e., technical specification—SG surveillance) will 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 for 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 inspections of the AP1000 SGs are performed according to the ASME Code and comply 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 providing 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 the 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

- 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, GDC 32.

As part of its evaluation, the staff reviewed the requirements for the SG Tube Surveillance Program contained in TS 5.5.4. The most recent generic technical specifications for SG ISI are found in 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 methods for selecting and sampling 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 to revise the SG Tube Surveillance Program TS to be consistent with the surveillance requirements contained in NUREG-0452, Revision 4. 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 Generic Letter (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 5.5.4.3.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 5.5.4.3.b. addresses when the SG tube inspection frequency shall be increased to at least once per 20 months. Therefore, the staff finds the response was unacceptable until the TS 5.5.4 is revised to indicate that the provisions of TS SR 3.0.2 are not applicable. TS 5.5.4 includes Revision 4 of the SG Tube Surveillance Program. TS 5.5.4.3.d indicates that the provisions of TS SR 3.0.2 do not apply to extending the frequency of performing ISIs, as specified in TS 5.5.4.3.a and 5.5.4.3.b. The staff finds this to be acceptable because it excludes the application of TS SR 3.0.2 to extend SG surveillance frequencies.

- The proposed TS is included in AP1000 TS Table 5.5.4-1 which defines SG sample selection and inspection. However, NUREG-0452, Revision 4, includes a strategy 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 the applicant revises TS Table 5.5.4-1 to reflect the preservice inspection. The applicant revised the SG Tube Surveillance Program to include a note to TS Table 5.5.4-1 that indicates that all SGs shall be inspected during the first ISI if no preservice inspection was conducted. The staff finds this acceptable because the note to TS Table 5.5.4-1 provides an appropriate strategy for determining the minimum number of SGs to be inspected during the first, second, and subsequent ISIs, depending on the preservice inspection performed.

DCD Section 5.4.15, “Combined License Information,” indicates that the COL applicant will address steam generator tube integrity with a Steam Generator Tube Surveillance Program and will address the need to develop a program for periodic monitoring of degradation of steam generator internals. This commitment is acceptable because such a program is needed to implement the TSs and because it addresses concerns addressed in Generic Letter 97-06, “Degradation of Steam Generator Internals.” This is COL Action Item 5.4.2.2.3-1.

5.4.2.2.3 Conclusions

The staff concludes that the AP1000 SG ISI program is acceptable and meets the requirements of GDC 32. This conclusion is based on the accessibility of the AP1000 design for periodic inspection and testing of critical areas for structural and leakage integrity, and on the consistency of the SG Tube Surveillance Program TS with the TS requirements for Westinghouse domestic PWRs.

5.4.2.3 <u>Containment Bypass Resulting from Steam Generator Tube Rupture</u>

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 in which 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 the following 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 15 design alternatives were selected for evaluation, including the 3 design features mentioned above. Each design alternative was 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 RAI 440.043, the applicant discussed the AP1000 design features that mitigate or prevent SG safety valve challenges during a rupture of multiple SG tubes, thus reducing the chance of containment bypass following an SGTR. This issue is discussed below.

5.4.2.3.1 AP1000 Steam Generator Tube Rupture Mitigation Design Features

The AP1000 design incorporates several automatic protection actions and the passive core cooling system (PXS) for mitigating 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 steam release 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 SGTR scenario, continued loss of RCS inventory to the SG secondary side through the ruptured tubes leads to a reactor trip upon a low-pressurizer pressure or over-temperature delta-T signal. This will also cause a turbine trip. The CMTs automatically actuate upon a safeguards signal or low-pressurizer level. The PRHR HX automatically actuates upon 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 an 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 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. If both the active non-safety-related mitigation and the safety-related PRHR HX mitigation fails, 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 predetermined set pressure, which is below the main steam safety valve (MSSV) set pressure. The PORV 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 which is stuck open.

In the event that the PORV fails to open during a SGTR, the MSSVs could open. 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 the 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 throughout 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 Multiple Steam Generator Tube Rupture Analysis

In DCD Tier 2, Section 15.6.3, “Steam Generator Tube Rupture,” the applicant provided 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 closed at low steamline pressure. The results showed no fuel failure, no SG overfilling, and the resulting offsite radiological doses 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 MSGTR 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. Section 19.1.10.5 of this report discusses the staff's evaluation of the use of MAAP4 for the AP1000 PRA evaluation.

The applicant analyzed two cases of five-tube rupture using the MAAP4 accident analysis code. The first assumed multiple SGTR with passive system response, and the second assumed multiple SGTR with failed open MSSV. In both cases, the accident was initiated by the simultaneous, double-ended failure of five cold side tubes at the top of the tube sheet. Startup feedwater system and the CVS were conservatively assumed to function because they tend to make the accident worse.

Case 1 is a passive system mitigation case with PRHR HX operation. The SUFS controls operate normally and throttle the SUFS 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 CMTs 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 are assumed to malfunction allowing SUFS flow to continue 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 the ADS. The RCS is rapidly depressurized, which results in the actuation of the IRWST and eventual containment recirculation. The analysis results showed that the core remains covered and cooled with the collapsed liquid level well above the top of the active fuel throughout the entire transient. The maximum total release would be limited to the initial activity in the RCS, thus no significant fission product release occurs.

The staff notes that the MAAP4 code was used in the AP600 PRA success criteria analysis. WCAP-14896 provided benchmark studies with the NOTRUMP code for a series of small and medium LOCA event sequences to support its use for AP600 PRA success criteria. The staff found that, in most cases MAAP4 and NOTRUMP predicted similar trends for system behavior in the base cases and sensitivity analyses. On the basis of the benchmark study comparisons, the staff determined MAAP4 to be an adequate screening tool for addressing thermal-hydraulic uncertainties and determining PRA success criteria for the AP600, subject to certain limitations as discussed in WCAP-14869. The use of MAAP4 for AP1000 is the same as was used for AP600. The applicant evaluated the limitations discussed in WCAP-14869 and concluded that MAAP4 could be used as a screening tool for evaluating PRA success criteria for AP1000. The staff agrees with this conclusion with the limitation that those success paths that give marginal results with MAAP4 should be verified using a computer code which the NRC staff has reviewed. Since the results of the MSGTR events show the core is covered with the collapsed liquid level well above the top of the fuel and therefore have large margins to core uncovery, the staff concludes that no additional analyses are required and that no core damage will occur.

5.4.2.3.3 Conclusions

The AP1000 design has unique features for mitigating an SGTR as compared to the design of conventional PWRs. The analysis shows that the PORV will automatically open to release steam and will 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 that the PORV fails to open coincident with a 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 that the PORV fails 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 resulting from a multiple tube rupture poses no undue threat to public health and safety, and the AP1000 design satisfies the provisions of SECY-93-087.

5.4.3 RCS Piping

The RCS piping includes those sections of the 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 is 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 to 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 in.) or less, Class B classification of small piping is exempted from ASME Code, Section III, Class 1, requirements in accordance with the exception permitted by 10 CFR 50.55a(c)(2)(i).

In DCD Tier 2, Section 5.4.3.2.1, “Piping Elements,” the applicant provided a list of the piping connected to the RCS. The detailed RCS P&ID 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 CMT 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, “Definitions.” All the RCS-connecting piping that constitutes the RCPB is designed to meet the ASME Code Section III requirements (with the 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 requirements, in accordance with the exception criterion of 10 CFR 50.55a(c)(2)(ii). Regulatory Position C of RG 1.26 specifies that the portion of the RCPB that meets the exception criteria of 10 CFR 50.55a(c)(2) must 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 the 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 finds 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. These detectors monitor the temperature for indications of thermal stratification.

In DCD Tier 2, Table 5.4-7, the applicant listed 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). DCD Tier 2, Section 3.9.3, discusses 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. The RCS piping construction is subject to a quality assurance program with the required testing specified in DCD Tier 2, Table 5.4-8. This quality assurance program must also meet the 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.

DCD Tier 2, Section 15.6, analyzes the consequences of the RCS piping breaks, including postulated cold leg double-ended guillotine breaks, 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,” as discussed in Chapter 20 of this report.

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, and 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. DCD Tier 2, Table 10.3.2-1, specifies the design data of the flow restrictors. 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 lb/hr) is approximately 55.2 kPa (8 psi).

The staff 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 has 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, the staff finds it to be 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 so 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 CMTs (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 an 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 an SGTR event, as described in DCD Tier 2, Sections 15.2.7 and 15.6.3, respectively. In accordance with the TS screening criteria specified in 10 CFR 50.36, the pressurizer heater trip function is described in AP1000 TS Table 3.3.2-1, “Engineered Safeguards Actuation System Instrumentation,” and is subject to AP1000 TS LCO 3.3.2, “Engineering Safety Feature Activation System (ESFAS) Instrumentation,” and associated surveillance requirements.

The pressurizer safety valves provide overpressure protection of the RCS, as discussed in Section 5.2.2 of this report. In addition, the pressurizer provides for high-point venting of noncondensable 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. Section 5.4.12 of this report discusses this in more detail.

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 a 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 for the sizing of the AP1000 pressurizer. These will ensure that the plant can 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 the 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 staff reviewed the pressurizer performance during anticipated operational occurrences (AOO) and postulated accidents as part of its 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 pressurizer safety and relief valve (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, 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 provide emergency core cooling following postulated accident conditions. The first stage valves may also be used to remove noncondensable gases from the steam space of the pressurizer, if necessary, following an accident.

Section 6.3, “Passive Core Cooling System,” of this report evaluates the ADS functional performance (as part of the PXS performance). Chapter 15 of this report evaluates the safety analyses of various design-basis accidents. The analysis results of design-basis accidents, such as the small-break LOCAs described in DCD Tier 2, Section 15.6.5, demonstrate that the ADS design and the passive core cooling system meet the acceptance criteria specified in 10 CFR 50.46. Therefore, the staff finds the ADS design to be acceptable.

5.4.7 Normal Residual Heat Removal System

The AP1000 normal 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 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 <u>Residual Heat Removal System Design Bases</u>

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 PRHR HX 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 PRHR HX, 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), thus providing 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 to the specifications found 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 <u>Residual Heat Removal System Design and Components</u>

In DCD Tier 2, Section 5.4.7.2, the applicant described 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 also comprises 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 comprises 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. A normally open manual isolation valve is located 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 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 provided 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 that 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 will perform its safety-related functions of containment isolation and preservation of the RCPB integrity and, therefore, is 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's 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. DCD Tier 2, Appendix 19E, "Shutdown Evaluation," provides this assessment. Section 19.3 of this report discusses the staff's evaluation of the shutdown operation issues. This section describes the RNS design features that address the issues raised in NUREG-1449 and GL 88-17, "Loss of Decay Heat Removal," regarding mid-loop operation. Each design feature of the RNS and its design bases are described below:

- 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 in traditional designs for venting of the SGs prior to nozzle dam insertion. Furthermore, this loop piping offset allows an 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.

- Automatic Depressurization System 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 allows 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, and is provided in Section 19.3 of this report. The staff concludes that the AP1000 design features, including those of the RNS, are adequate to support operations at shutdown and low power conditions, as discussed in Section 19.3 of this report, and therefore, are 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 in SECY-93-087, the staff specified that ALWR designs should reduce the possibility of a LOCA outside containment by designing, to the extent practical, all systems and subsystems connected to the RCS to an ultimate rupture strength at least equal to full RCS pressure. SECY-90-016 also specifies 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 that 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. However, the pump seal does not meet this criterion. This exception is discussed in the staff's evaluation of GSI 105, "Interfacing Systems LOCA at Light Water Reactors," in Chapter 20 of this report.

- Additional Reactor Coolant System 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.

- Residual Heat Removal System 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, thereby reducing 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.

- Reactor Coolant System 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's evaluation of the interfacing system LOCA is addressed in the discussion of GSI 105, 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 and, therefore, are acceptable.

5.4.7.5 Residual Heat Removal System Operation and Performance

In DCD Tier 2, Section 5.4.7.4, the applicant provided 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 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.

WCAP-15992, Revision 1, "AP1000 Adverse System Interactions Evaluation Report," provides an evaluation of the potential for adverse system interactions of the RNS and the PXS. 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 draindown 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 provide a 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 draindown 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, thus increasing the containment floodup level, which improves the driving head available for containment recirculation flow. Based on the discussion above, the staff finds this system interaction to be acceptable.

5.4.7.6 Design Evaluation

The staff reviewed the RNS design 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 to withstand an SSE and remain 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 the prohibition on SSCs important to safety from being shared among nuclear power units
- GDC 19, as it relates to the provision of a control room 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 an SSE for pressure retention. The RNS is operated from the MCR. 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 DCD Tier 2, Table 3B-1 and 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 PRHR HX complies with the heat removal function of GDC 34. Section 6.3 of this report discusses the evaluation of the PRHR HX.

5.4.7.7 Inspection and Testing Requirements

DCD Tier 2, Section 5.4.7.6, describes inspection and testing requirements for the RNS. Preoperational tests, which include valve inspection and testing, flow testing, and verification of heat removal capability, verify the proper operation of the RNS. 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 that the applicant has set proper inspection and test requirements for the RNS valves performing the safety-related functions of containment isolation and RCPB integrity preservation.

The staff verified that the set pressure and the relieving capacity of the relief valve, RNS-V021, which provides low-temperature overpressure protection, are 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 to be acceptable.

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

5.4.7.8 Regulatory Treatment of the Residual Heat Removal System

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 the 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). Chapter 22 of this report provides a detailed evaluation of the RTNSS issue.

The staff describes the RTNSS process in SECY-94-084, “Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety Systems in Passive Plant Designs.” The goal of the RTNSS process is to (1) provide insights on the importance of non-safety-related systems to the overall safety of the passive advanced reactor design, and (2) 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.

Because 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 important non-safety-related SSCs identified in the RTNSS. 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. DCD Tier 2, Section 16.3.2, includes a commitment that the COL applicant referencing the AP1000 design will develop and implement procedures consistent with the availability controls. The AP1000 DCD will also include these administrative availability controls.

In WCAP-15985, “AP1000 Implementation of the Regulatory Treatment of Non-Safety-Related Systems Process,” the applicant provided the results of its evaluation based on the RTNSS screening process. The RNS was identified as an important system, necessary for shutdown decay heat removal to support mid-loop operation with reduced reactor coolant inventory. Therefore, the RNS is subject to additional regulatory controls. In addition, the RNS 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. DCD Tier 2, Table 16.3-2, Sections 2.1 and 2.2, specify the administrative short-term availability controls of the RNS functions at various modes of operation. In addition, DCD Tier 2, Table 16.3-2, specifies the availability controls of the RNS supporting systems, such as the CCS, the service water system, and the alternating current power supplies. The staff has reviewed DCD Tier 2, Table 16.3-2, and concludes 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, and 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, in accordance with GDC 15 in Appendix A to 10 CFR Part 50. GDC 15 requires the RCS and its associated auxiliary, control, and protection systems 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 PXS, are not pressure relief devices for overpressure protection. The first three stages of the ADS are connected to the pressurizer; the first stage can also be used to vent noncondensable 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, and self-actuated by direct fluid pressure. No loop seal in the piping exists 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 the containment atmosphere. The rupture disk is provided to contain leakage past the valve, and is designed with a substantially lower set pressure than that of the PSV. This will ensure PSV discharge. A small pipe is connected to the discharge piping and directed to the

RCDT 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 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. DCD Tier 2, Table 5.4-17, specifies the design parameters of the PSVs. As addressed in Section 5.2.2 of this report, the sizing of the PSVs with 3-percent accumulation meets the requirements of GDC 15. Therefore, the staff finds this to be acceptable.

In 10 CFR 50.34(f)(2)(x), the NRC requires a test program and associated model development, as well as the conduct 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, “Testing Requirements,” in Chapter 20 of this report addresses the resolution of the PSV 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 ISI for ASME Code Class 2 and 3 components, as specified 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.” Therefore, the staff finds the applicant’s test program to be acceptable.

5.4.9.2 Residual Heat Removal System Relief Valve

The RNS relief valve on the RNS pump suction line is spring-loaded and 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. DCD Tier 2, Table 5.4-17, specifies the design parameters of the RNS relief valve, including the set pressure and relieving capacity. Section 5.2.2 of this report discusses the determination of the set pressure and relieving capacity. 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 10 CFR 50.34(f)(2)(xi), which requires that direct indication of relief and safety valve position (open or closed) be provided in the MCR. Therefore, the staff finds this to be acceptable.

RCS pressure relief devices must be subjected to qualification tests for all fluid conditions expected under operating conditions, transients, and accidents, as required by 10 CFR 50.34(f)(2)(x). DCD Tier 2, Section 5.4.9.4, states that the RNS relief valve is designed for water relief and is not an RCS pressure relief device because it has a set pressure less than the RCS design pressure. Therefore, the valve selected for the RNS relief valve is independent from the EPRI safety and relief valve test program. Because the RNS relief valve is not an RCPB valve, and is designed for LTOP, the staff agrees it need not be included in the EPRI test program for the safety and relief valve test program. 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 that these test requirements for the RNS relief valve comply with the ASME Code, Section III, requirements and, therefore, are acceptable.

5.4.10 RCS Component Supports

Sections 3.9.3.3 and 3.12.6 of this report describe the design bases and design evaluation of the RCS component supports. Sections 5.4.2.2 and 6.6 of this report discuss ISI of RCS components.

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, as it relates to the protection of safety-related systems from the effects of earthquakes, and GDC 4, as it relates to a failure of the system resulting in missiles or adverse environmental conditions that could result in damage to the safety-related systems or components. Conformance with GDC 2 is demonstrated by meeting the guidelines of RG 1.29, “Seismic Design Classification,” Positions C.2 and C.3. Position C.2 addresses those portions of the 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 cause an 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 GDC 4 is demonstrated by meeting the acceptance criteria of SRP Section 5.4.11, as applicable.

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”; discuss the systems and components for AP1000 pressurizer relief discharge. 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 Section 3.2, state that the pressurizer safety valves are classified as AP1000 equipment Class A (American Nuclear Society (ANS) safety Class 1), seismic Category I, and ASME Code Class 1. These valves are tested in accordance with requirements of 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 RNS 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 the 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 RCDT. Each safety valve discharge line includes a temperature indicator and alarm in the MCR.

Pressurizer safety valve discharge is directed away from the 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 passive containment cooling system (PCS).

5.4.11.1 <u>Automatic Depressurization System</u>

DCD Tier 2, Figure 5.1-5 (sheet 1 of 3 and sheet 2 of 3), details the ADS. 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 stated 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 the requirements of ASME Code, Section XI.

The ADS consists of 20 valves divided into 2 divisions, which are then 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 an 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 (see 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, thus preventing 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 of a first-stage ADS valve (including the water seal, steam, and gases) when the venting of noncondensable gases from the pressurizer occurs 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 covered 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 demonstrated by 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 incapacitate 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 nonseismic 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 will also not pose a hazard to other safety-related SSCs inside containment, should any of the pressurizer relief discharge equipment fail.

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

5.4.12 Reactor Coolant System High-Point Vents

RCS high-point vents are provided to exhaust noncondensable gases accumulated in the primary system that could inhibit natural circulation core cooling. As required by 10 CFR 50.34(f)(2)(vi), the RCS must include high-point vents to maintain adequate core cooling, and the systems available to achieve this must be capable of being operated from the MCR. In addition, the operation of these systems must not lead to an unacceptable increase in the probability of a LOCA, or an unacceptable challenge to containment integrity.

In the AP1000 design, noncondensable 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 PRHR HX piping and the CMT inlet piping in the PXS also include a high-point vent and, therefore, are in compliance with 10 CFR 50.34(f)(2)(vi).

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

5.4.12.1 <u>Reactor Vessel Head Vent System</u>

The RV head vent system (RVHVS) is designed to remove noncondensable 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 1 hour. The RVHVS is primarily used during plant startup to properly vent air from the RV head and to fill the RCS. The RVHVS valves also provide 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; they are qualified to Institute of Electrical and Electronics Engineers (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. This will 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.

A break of the RVHVS line would result in a small-break LOCA no greater than 2.54 cm (1 in.) 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 specify that procedures should be developed to use the vent paths to remove gases that may inhibit core cooling from the U-tubes of the SGs. In addition, the procedures to operate the vent system should consider when venting is needed and when it is not needed, taking into account a variety of initial conditions, operator actions, and necessary instrumentation. The applicant’s response to RAI 440.049 describes the SG tube venting procedures.

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 an 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 PRHR HX, to provide the safety-related function of core cooling. Therefore, the design 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 a 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 designed 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, therefore, is acceptable. Chapter 20 of this report addresses the resolution of TMI Action Item II.B.1.

5.4.12.2 Automatic Depressurization System First-Stage Valves

As discussed in Section 5.4.6 of this report, the first-stage valves of the AP1000 ADS provide the capability to remove noncondensable 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 noncondensable gases from the pressurizer steam space is not required to provide safety-related core cooling following a postulated accident. Therefore, the acceptance guidelines of SRP Section 5.4.12 do not apply to the ADS.

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

The PRHR HX inlet piping and the CMT pressure balance line piping in the PXS include high-point vents that provide the capability to remove and prevent the accumulation of noncondensable gases that could interfere with the operation of the heat exchanger or CMT. These gases are normally expected to accumulate when the RCS is refilled and pressurized following refueling. Level indicators identify when gases have collected in the vent line. Any noncondensable gases that collect in this high point can be manually vented. The discharge of the PRHR HX 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 are 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 is in accordance with DCD Tier 2, Section 6.6, for ASME Code Class 2 components, and DCD Tier 2, Section 5.2.4, for 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 because they provide a means to prevent accumulation of noncondensable gases from the RCS that could interfere with operation of the PXS. Further, these high-point vents are designed in accordance with the ASME Code Section III requirements.

5.4.13 Core Makeup Tank

The AP1000 design includes two CMTs as part of the PXS. In the CMTs, cold borated water, under system pressure, is stored to provide high-pressure reactor coolant makeup and boration for LOCA and 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). It is supported on columns. DCD Tier 2, Table 6.3-2, 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. This 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 Ni-Cr-Fe alloys by copper, low-melting-temperature alloys, mercury, and lead is prohibited. Sections 5.2.3 and 5.2.4 of this report discuss the material selection and water chemistry specification, and the test and inspections of the CMT, respectively.

5.4.13.3 Design Evaluation

DCD Tier 2, Section 3.9.3, discusses the loading combinations, stress limits, and analytical methods for the structural evaluation of the CMT for various plant conditions. DCD Tier 2, Section 3.9.2, discusses the requirements for dynamic testing and analysis. 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 the component cyclic fatigue of the CMT also assumes 30 occurrences in the plant's 60-year lifetime in which a small leak draws in hot RCS fluid, and 10 occurrences of increasing containment temperature above normal operating range.

DCD Tier 2, Sections 3.9.1, 3.9.2, and 3.9.3, respectively, discuss 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. The staff's evaluation of these sections are discussed in the related sections of this report.

Chapter 6.3 of this report evaluates the functional performance of the CMTs, as part of the PXS performance, as well as the safety analyses of various design-basis transients and accidents described in Chapter 15 of this report, to demonstrate the capability of the CMTs to comply with the 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, as well as 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 were used for the AP1000 design. In Chapter 21 of this report, the staff discusses the applicability of the AP600 test program and computer codes to the AP1000 design. Because it finds the evaluations referenced above to be acceptable, the staff concludes that the CMT design meets the guidelines of SRP Section 6.3 and GDC 2, 4, 5, 17, 36, and 37. In addition, the staff finds that 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 PRHR HX is part of the PXS. Its function is to remove core decay heat for any postulated non-LOCA event in which a loss-of-cooling capability via the SGs occurs. Section 6.3 of this report discusses the operation of the PRHR HX in the PXS.

5.4.14.1 Design Description

The PRHR HX consists of a top and lower tube sheet mounted through the wall of the IRWST. A series of 1.9-cm (0.75-in.) outer diameter C-shaped tubes connect to the tube sheets, with the top of the tubes located several feet below the IRWST water surface. DCD Tier 2, Table 6.3-2, provides the AP1000 PRHR HX 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 tube sheets 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 in both cold and hot conditions. The PRHR HX 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 PRHR HX. This also helps to minimize the amount of humidity in containment.

5.4.14.2 Design Bases

The PRHR HX, 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 PRHR HX and the IRWST are designed to delay significant steam release to the containment for at least 1 hour. The PRHR HX will keep the reactor coolant subcooled and prevent water relief from the pressurizer. In addition, the PRHR HX will cool the RCS to 204.4 °C (400 °F) in 72 hours with RCPs operating or, if required, in the natural circulation mode, to allow the RCS to be depressurized to reduce stress levels in the system.

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

5.4.14.3 <u>Design Evaluation</u>

DCD Tier 2, Sections 3.9.1 through 3.9.3 discuss the loading combination, stress limits, and analytical methods for evaluating the structural integrity of the PRHR HX, as well as the transients used to evaluate the PRHR HX under various plant conditions. During normal plant operation, the PRHR HX, without flow through it, is pressurized to the RCS hot-leg pressure at the IRWST temperature. Operation of the PRHR HX is evaluated using ASME, Section III defined 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 PRHR HX 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 PRHR HX. The dynamic analysis considers the hydraulic interaction between the coolant and system structural elements. The evaluation of component cyclic fatigue also assumes the following two additional Service Level B transients that affect only the PRHR HX:

(1) 30 occurrences in the plant's 60-year lifetime in which a small leak in the manway cover draws in hot RCS fluid

(2) 10 occurrences of increasing IRWST temperature as a result of an event that activates passive core cooling

Sections 3.9.1, 3.9.2, and 3.9.3, respectively, of this report discuss the staff's 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.

Chapter 6.3 of this report evaluates the PRHR HX functional performance, as part of the PXS performance. Chapter 15 of this report presents the safety analyses of various design-basis transients and accidents 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 the thermal-hydraulic behavior and the phenomena of the AP600 PXS and components, as well as 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 design. In Chapter 21 of this report, the staff discusses the applicability of the AP600 test program and the computer codes to the AP1000 design. Because it finds the evaluations referenced above to be acceptable, 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.