5.2 Integrity of the Reactor Coolant Pressure Boundary

This section describes the measures employed to provide and maintain the integrity of the reactor coolant pressure boundary (RCPB) for the plant design lifetime. Consistent with the definition in 10 CFR 50.2, the U.S. EPR RCPB includes all pressure-containing components, such as pressure vessels, piping, pumps, and valves which are part of the reactor coolant system (RCS) or connected to the RCS, up to and including these:

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

Section 3.9 presents the design transients, loading combinations, stress limits, and evaluation methods used in the design analyses of RCPB components and supports to demonstrate that RCPB integrity is maintained.

5.2.1 Compliance with Codes and Code Cases

5.2.1.1 Compliance with 10 CFR 50.55a

The RCPB components are designed and fabricated as Class 1 components in accordance with Section III of the ASME Boiler and Pressure Vessel Code (Reference 1), except for components that meet the exclusion requirements of 10 CFR 50.55a(c) which are designed and fabricated as Class 2 components. The RCPB component classification complies with the requirements of GDC 1 and 10 CFR 50.55a. Table 3.2.2-1—Classification Summary lists the RCPB components, including pressure vessels, piping, pumps, and valves, along with the applicable component codes. Other safety-related plant components are classified in accordance with RG 1.26, as specified in Section 3.2.

The code of record for the design of the U.S. EPR is the 2004 edition of the ASME Boiler and Pressure Vessel Code (no addenda).

The application of Section XI of the 2004 edition of the ASME Boiler and Pressure Vessel Code to the U. S. EPR is described in Section 5.2.4 and Section 6.6 The application of the ASME Code for Operation and Maintenance of Nuclear Power Plants (OM Code) (Reference 2) is described in Section 3.9.6.

5.2.1.2 Compliance with Applicable Code Cases

ASME Section III Code Cases acceptable for use in the U.S. EPR design, subject to the limitations specified in 10 CFR 50.55a, are listed in RG 1.84. Table 5.2-1—ASME Code

Cases lists the specific Code Cases used in the U.S. EPR design. A COL applicant that references the U.S. EPR design certification will identify additional ASME Code Cases to be used. Code Cases pertaining to ASME Code Section III, Division 2 are addressed in Section 3.8.

ASME Section XI Code Cases acceptable for use for inservice inspection (ISI), subject to the limitations specified in 10 CFR 50.55a, are listed in RG 1.147 and described in Section 5.2.4 and Section 6.6. ASME OM Code Cases acceptable for use for inservice testing (IST), subject to the limitations specified in 10 CFR 50.55a, are listed in RG 1.192 and described in Section 3.9.6.

5.2.2 Overpressure Protection

Pressurizer safety relief valves (PSRV) protect the RCPB from overpressure during power operation and during low temperature operation. Auxiliary and emergency systems connected to the RCS are not utilized for RCPB overpressure protection.

Main steam safety valves (MSSV) and main steam relief trains protect the secondary side of the steam generators from overpressure. Secondary side overpressure protection is addressed in Section 10.3.

5.2.2.1 Design Bases

Component design bases for the PSRVs and the secondary side overpressure protection devices are addressed in Section 5.4.13 and Section 10.3, respectively.

The PSRVs are part of the RCPB and are designed to meet the requirements for ASME Section III, Class 1 components (GDC 1, GDC 30, 10 CFR 50.55a). Component classifications are presented in Section 3.2.

The opening set pressures and capacity of the PSRVs are sufficient to limit the RCS pressure to less than 110 percent of the RCPB design pressure during any condition of normal operation, including anticipated operational occurrences (AOO) (GDC 15). The bounding design transient for RCPB overpressure is a turbine trip at full power. This transient bounds all upset, emergency and faulted conditions identified in Section 3.9.1.

The PSRVs maintain the RCS pressure below brittle fracture limits when the RCPB is stressed under operating, maintenance, testing, and postulated accident conditions, including low temperature operation, so that the RCPB behaves in a non-brittle manner and the probability of rapidly propagating fracture is minimized (GDC 31).

The PSRVs can perform their overpressure protection functions at power and low temperature operations assuming a single failure or malfunction of an active component.

Direct indication of PSRV main disk position is provided in the main control room (MCR) (10 CFR 50.34(f)(2)(xi)).

5.2.2.2 Design Evaluation

Overpressure protection of the RCPB at power and during low temperature operation is provided by three pilot-operated PSRVs installed on top of the pressurizer (PZR) that discharge to the pressurizer relief tank (PRT) through a common header. At power, a spring-operated pilot valve actuates the main safety valve of the PSRV assembly. During low temperature conditions, two solenoid-operated pilot valves in series actuate the main safety valve. The PSRV assemblies are described in Section 5.4.13. The PRT and associated discharge piping is described in Section 5.4.11.

A surge line allows unobstructed flow between the PZR and the RCS. The surge line is sized to provide an allowable pressure drop between the RCS loops and the PZR during overpressure transients. The design of the surge line is described in Section 5.4.3.

The relief capacities of the PSRVs are determined from the postulated overpressure transient conditions, as described in Section 5.2.2.2.1 and Section 5.2.2.2.2.

5.2.2.2.1 Overpressure Protection at Power

As noted in Section 5.4.10, the pressurizer is sized to preclude actuation of the PSRVs in conjunction with the normal spray during normal operational transients. The turbine trip transient establishes the design requirements for RCPB overpressure protection. The event was analyzed in accordance with the methodology described in the Codes and Methods Applicability Report for the U.S. EPR (Reference 7). The plant response to this transient, including assumptions used in the analysis, initial plant conditions, and system parameters, is described in Section 15.2.2. Chapter 15 also describes the analytical model used in the transient analysis and addresses the bases for its validity.

The analysis assumes a loss of offsite power and a single active failure of a main steam relief train. For conservatism, five percent steam generator tube plugging is assumed. After the turbine trip occurs, high PZR pressure generates a reactor trip signal in both subsystems “A” and “B” of the protection system. Failure of one of the two protection subsystems is assumed as failure of the first reactor trip signal, and the remaining subsystem fulfills the reactor trip function.

The PSRVs operate in response to PZR pressure at power without any external energy source or control signal. The analysis confirms that three PSRVs have sufficient capacity to limit the RCS pressure to less than 110 percent of the RCPB design pressure, consistent with ASME Section III, NB-7000. Specific PSRV information, including discharge capacity, is presented in Section 5.4.13.

5.2.2.2.2 Low Temperature Overpressure Protection

Features that provide low temperature overpressure protection (LTOP) for the RCPB are designed in accordance with BTP 5-2 (Reference 8). Each PSRV is equipped with two solenoid-operated pilot valves in series for LTOP of the RCPB, although only two PSRVs are used during low temperature operations. During reactor cool down, an alarm is generated from the wide range cold leg temperature instrumentation once conditions for permissive P17 are met, allowing the operator to validate the permissive to enable LTOP. Refer to Section 7.2.1.3.12 for a description of the P17 permissive. The operator manually validates permissives to enable LTOP. Low temperature overpressure protection is operable during startup and shutdown conditions at or below the reactor pressure vessel (RPV) brittle fracture protection function enable temperature.

The LTOP enable temperature and PSRV set points are selected so that the peak RPV pressure does not exceed the 10 CFR Part 50, Appendix G limits. PSRV setpoints are presented in Section 5.4.13. Section 5.3.2 addresses pressure-temperature limits for the RPV.

Low temperature RCPB overpressure events include mass input events and heat input events. These events are considered and are presented with the interlock or equipment lock-out that could prevent the event, where applicable:

- Mass input events.
  - Unplanned start of four medium head safety injection (MHSI) pumps simultaneously with a failure of one large miniflow line to open.
    - P17 holds open the large miniflow lines to prevent overpressurizing the RCS if this scenario occurs, however one large miniflow line is assumed to fail closed.
  - Both charging pumps running and the control valve failed open (a maximum runout flow of 112.66 lb_m/second, total for both pumps).
  - Unplanned start of two extra borating system pumps.
  - Release of accumulators into the RCS.
- Heat input events.
  - Startup of a reactor coolant pump (RCP) with the secondary temperature 50°F higher than the primary temperature.
  - PZR heaters energize.
  - Residual heat removal system (RHRS) connecting valves fail closed.

Two mass input events – start of four MHSI pumps with one large miniflow line closed, and both charging pumps running with control valve failed open – and one heat input event – startup of an RCP with the secondary side hotter than the primary side – were selected for analysis. The other overpressure events are bounded by the analyzed events. The conservative analyses assume water solid conditions with letdown lines isolated and the most limiting single failure. Set point uncertainties are added to the nominal PSRV open and close set points. For the charging pump and RCP events, the limiting single failure is a failure of one PSRV. For the MHSI event, the limiting single failure is the failure of one large miniflow line to open.

Analyses demonstrate that the low temperature PSRV set points yield peak pressures within 10 CFR Part 50, Appendix G limits for the corresponding temperatures for the design events over a range of initial RCS temperatures. The capacity of the PSRVs at the reduced (LTOP) setpoionts is sufficient to provide overpressure protection.

5.2.2.3 Piping and Instrumentation Diagrams

Figure 5.1-4—RCS Piping and Instrumentation Diagram, displays the piping and instrumentation diagram for the RCS, including the PSRVs.

5.2.2.4 Equipment and Component Description

Section 5.4.13 presents the PSRV design parameters, including capacities and set pressures, and addresses the operation of the PSRVs. Figure 5.4-8 includes a schematic representation of the PSRVs. Section 3.9 and Section 3.11 present component stress analyses and environmental conditions for which the components are designed.

5.2.2.5 Mounting of Pressure Relief Devices

The PSRVs are mounted to the PZR nozzles. The stress analyses and the associated acceptance criteria for the PZR nozzles and the PSRV is performed in accordance with the requirements for ASME class 1 components and piping. Section 3.9.3.2 addresses stress and load combination requirements for the PSRVs.

5.2.2.6 Applicable Codes and Classification

The PSRV design is in accordance with ASME Section III, NB-3500. The PSRVs function to meet the requirements for overpressure protection as presented in ASME Section III, NB-7000. Section 5.2.1 identifies the code edition and addenda applicable to the design.

Section 3.2 identifies the classifications applied to the overpressure protection equipment and components.

5.2.2.7 Material Specification

Section 5.2.3 describes RCPB material requirements and fabrication controls. The selected PSRV materials are consistent with specifications identified in ASME Section II in accordance with ASME Section III, NB-2000, and can be found in Table 5.2-2—Material Specifications for RCPB Components. Code cases which may be applied in the selection of PSRV materials are listed in Section 5.2.1.2.

5.2.2.8 Process Instrumentation

Temperature sensors in each PSRV discharge line provide indications and alarms in the MCR to alert the operator to steam discharge through the PSRVs from valve operation or leakage. Discharge flow is also indicated by an increase in the PRT temperature or level. Valve stem position indication for each PSRV is displayed in the MCR, in accordance with 10 CFR 50.34(f)(2)(xi).

The operator is alerted when the LTOP function is to be enabled during plant cool down. Positive indication is provided in the MCR when the low temperature protection function is enabled.

Refer to Section 7.3 for further details on process instrumentation associated with overpressure protection.

5.2.2.9 System Reliability

During hot RCS conditions, the PSRVs are considered a passive device. The spring-operated pilot valves are designed in accordance with the requirements of ASME Section III, NB-7511.1. With successful operation of the pilot valve, a large differential pressure reliably opens the main relief disk, which relieves RCS pressure. Detailed operation of the PSRV is described in Section 5.4.13.

The two solenoid-operated pilot valves in series provide single failure protection against spurious opening or failure to close the PSRVs during low temperature operation. The series pilot valves are powered from separate electrical divisions that are backed by uninterruptible power supplies. Although only one PSRV is required for LTOP, at least two PSRVs are in service to meet the single failure criteria. The U.S. EPR can cope with an inadvertent opening of a PSRV. Section 15.6.1 presents the analysis for inadvertent opening of a PSRV.

The inservice inspection and testing of the PSRVs, described in Section 5.2.2.10, provide reasonable assurance of reliability and proper operation at power and during low temperature conditions.

5.2.2.10 Testing and Inspection

Refer to Section 14.2 (Test #037) for initial plant testing. Refer to Chapter 16 Technical Specifications 3.4.10 and 3.4.11 for surveillance requirements.

Prior to entering the LTOP mode of PSRV operation during plant shutdown, the solenoid-operated pilot valves to be placed in service are tested for operability. This testing is performed on only one solenoid-operated pilot valve at a time by transmitting a simulated actuation signal to it and verifying valve opening through remote position indication. Each solenoid-operated pilot valve is returned to closed and operable status following successful testing to prevent unnecessary operation of the relief valve during subsequent testing of the remaining solenoid-operated pilot valves.

The PSRVs are subjected to a qualification and testing program to demonstrate acceptable performance for all fluid conditions expected under operating conditions, transients, and accidents, including anticipated transients without scram (ATWS) conditions, in accordance with 10 CFR 50.34(f)(2)(x).

5.2.3 Reactor Coolant Pressure Boundary Materials

RCPB materials are fabricated and selected to maintain pressure boundary integrity for the plant design lifetime.

The RCPB materials are selected from ASME Section II in accordance with ASME Section III, NB-2120. Materials Code Cases approved for use by RG 1.84 and applied to the U.S. EPR design are identified in Section 5.2.1.2. (GDC 1, GDC 30, 10 CFR 50.55a)

The RCPB materials and the reactor coolant chemistry are specified for compatibility to avoid degradation or failure in environmental conditions associated with normal operations, maintenance, testing, and postulated accidents. Ferritic low alloy and carbon steel RCPB components have either austenitic stainless steel or nickel-base alloy cladding on surfaces exposed to the reactor coolant. (GDC 4)

Ferritic RCPB materials comply with the fracture toughness requirements of 10 CFR Part 50, Appendix G. Complying with Appendix G requirements minimizes the probability of rapidly propagating fracture and gross rupture of the RCPB. (GDC 14, GDC 31)

RCPB materials are handled, protected, stored, and cleaned according to recognized and accepted methods that are designed to prevent damage or deterioration. Process specifications stipulate the procedures covering these controls in compliance with 10 CFR 50, Appendix B, Criterion XIII.

5.2.3.1 Material Specifications

Table 5.2-2 lists the materials incorporated into the design of the RCPB (excluding the reactor pressure vessel), including grade or type and final metallurgical condition. Table 5.2-2 includes the materials specified for the steam generators, PZR, RCPs, RCPB piping, and control rod drive mechanism. ASME Boiler and Pressure Vessel Code, Section II material specifications are used for materials in the RCPB, including weld materials.

The weld filler materials used for joining the ferritic base materials of the RCPB conform to ASME Section II Part C material specifications SFA 5.5, 5.17, 5.18, 5.20, 5.23, 5.28, and 5.29. The weld filler materials used for joining the austenitic stainless steel base materials of the RCPB conform to ASME Section II Part C material specifications SFA 5.4, 5.9, and 5.22. The weld filler materials used for joining nickel-chromium-iron (NiFeCr) alloys in similar base material combination and in dissimilar ferritic or austenitic base material combination conform to ASME Section II Part C material specifications SFA 5.11 and 5.14.

Low alloy steel pressure boundary forgings have limited sulfur content not exceeding 0.008 wt%, (wt = weight). Clad low alloy steel pressure boundary materials have ASTM grain size 5 or finer.

Austenitic stainless steel base metal conforms to RG 1.44. Austenitic stainless steel base metal and weld metal have limited carbon content not exceeding 0.03 wt%. Austenitic stainless steel base metal and weld filler metal in contact with RCS primary coolant has limited cobalt content not exceeding 0.05 wt%. Austenitic stainless steel base metal in contact with RCS primary coolant has limited sulfur content not exceeding 0.02 wt%. When supplementary chemical analysis is performed which would be more complete than the analysis used to check the content of specific elements, the results will show that the sample contains no more than residual antimony. In addition, the carbon portion of the reactor coolant pump journal bearings will have no antimony.

Austenitic stainless steel welds in RCS piping, including surge line piping, have delta ferrite content limited to a ferrite number (FN) between 5 and 10, measured as determined by ASME Section III, NB-2433. Austenitic stainless steel weld materials for stainless steel welds joints in the balance of the RCPB system have delta content ferrite limited to an FN between 5 and 20, as determined by ASME Section III, NB-2433.

NiCrFe Alloy 600 base metal or Alloys 82/182 weld metal is not used in RCPB applications. NiCrFe Alloy 690 base metal has controlled chemistry, mechanical properties, and thermo-mechanical processing requirements that produce an optimum microstructure for resistance to intergranular corrosion. Alloy 690 is solution annealed and thermally treated to optimize the resistance to intergranular corrosion.

Alloy 690 and its weld filler metals (Alloy 52/52M/152) in contact with RCS primary coolant have limited cobalt content not exceeding 0.05 wt%. Alloy 690 in contact with RCS primary coolant has limited sulfur content not exceeding 0.02 wt%.

5.2.3.2 Compatibility with Reactor Coolant

5.2.3.2.1 Reactor Coolant Chemistry

The RCS water chemistry is controlled to minimize negative impacts of chemistry on materials integrity, fuel rod corrosion, fuel design performance, and radiation fields, and is routinely analyzed for verification. The water chemistry parameters are based on industry knowledge and industry experience as summarized in the EPRI PWR Primary Water Chemistry Guidelines (Reference 3).

The chemical and volume control system (CVCS) provides the primary means for maintaining the required volume of water in the RCS and for the addition of chemicals. The design of the CVCS allows for the addition of chemicals to the RCS to control pH, scavenge oxygen, control radiolysis reactions, and maintain corrosion product particulates within a specified range. Table 5.2-3—Reactor Coolant Water Chemistry - Control Parameters shows the control values for the reactor coolant chemistry parameters and impurity limitations during power operation. These criteria conform to the recommendations of RG 1.44 and the EPRI PWR Primary Water Chemistry Guidelines report.

Enriched boric acid (EBA) is added to the RCS as a soluble neutron poison for core reactivity control. Lithium hydroxide enriched in lithium 7 is used as a pH control agent to maintain a slightly basic pH at operating conditions. This chemical is chosen for its compatibility with the materials and water chemistry of borated water/stainless steel/zirconium/nickel-base alloy systems. Lithium-7 is also produced in solution from the neutron irradiation of the dissolved boron in the coolant.

In addition to degasification during startup, two chemicals are added to the reactor coolant to control oxygen: (1) hydrazine during startup operations below 250°F; and (2) hydrogen immediately prior to and following criticality. Dissolved hydrogen is added to maintain a reducing environment by scavenging oxidizing molecular products formed by the radiolysis of water and with any oxygen introduced into the RCS with makeup water.

Suspended solids (corrosion product particulates) in the reactor coolant are minimized by the coordinated boron-lithium chemistry program and by filtration during shutdown operations. Other impurity concentrations are maintained below specified limits through the control of the chemical quality of makeup water and chemical additives and by purification of the reactor coolant through the mixed bed ion exchangers. Section 9.3.4 addresses RCS water chemistry control.

5.2.3.2.2 Compatibility of Construction Materials with Reactor Coolant

Ferritic low alloy and carbon steels used in principal pressure retaining applications have either austenitic stainless steel or nickel-base alloy corrosion resistant cladding on all surfaces that are exposed to the reactor coolant. The cladding of ferritic type base material receives a post-weld heat treatment, as required by ASME Section III.

Unstabilized austenitic stainless steel base materials with primary pressure retaining applications are used in the solution annealed and water quenched (or rapidly cooled) condition in accordance with RG 1.44. Unstabilized austenitic stainless steels are not heated above 800°F, other than locally by welding operations, after the final heat treatment.

Stabilized austenitic stainless steels have a stabilizing heat treatment above 800°F; the stabilizing element combines with the carbon to form carbide. Chromium carbides are prevented from precipitating if a subsequent heat treatment in the 800°F to 1500°F temperature range occurs.

Due to the control of oxygen, chlorides, and fluorides in the reactor coolant, any unstabilized stainless steel locally sensitized at the high temperatures used during fabrication are not expected to experience stress corrosion cracking during normal plant operation. Precipitation hardenable stainless steel (SA-453 Grade 660) is used as a necked-down bolt for the control rod drive mechanism; because of its location it will not have contact with reactor coolant. The RCP bolting is external to the wetted pressure boundary. Alloy 690 base materials with primary pressure retaining applications are used in the solution annealed and thermally treated condition to optimize resistance to intergranular corrosion. Alloy 600 base and weld filler materials are not used in the RCS including any RCPB applications.

5.2.3.3 Fabrication and Processing of Ferritic Materials

5.2.3.3.1 Fracture Toughness

The fracture toughness properties of the RCPB components including pumps, piping, and valves comply with the requirements of 10 CFR 50, Appendix G and ASME Section III, NB-2300, NC-2300, and ND-2300 as appropriate. Section 5.3.1 provides a specific description of the reactor vessel materials and Section 5.4.1 provides a specific description of the RCP flywheel. The maximum reference temperature RT_{NDT} for steam generator and PZR RCPB components and their weldments is limited to -4°F; actual fracture toughness data are supplied on material test reports for each component at the time of shipment. An RT_{NDT} of -4°F is sufficient to comply with the above stated fracture toughness requirements for the US EPR. Forgings currently used in replacement components and new plant construction have been shown to easily meet this maximum -4°F RT_{NDT} requirement.

Calibration of temperature instruments and Charpy impact test machines are performed to meet the requirements of ASME Section III, NB-2360, NC-2360, and ND-2360 as appropriate. Impact test procedures comply with the requirements of ASME Section III, NB-2320, NC-2320, and ND-2320 as appropriate.

5.2.3.3.2 Control of Welding

Welding is conducted utilizing procedures qualified according to the rules of ASME Sections III and IX. Control of welding variables, as well as examination and testing during procedure qualification and production welding, is performed in accordance with ASME Code requirements.

Electroslag welding performed on RCPB components conforms to the requirements of RG 1.34, "Control of Electroslag Weld Properties." The procedure qualification for electroslag welding includes a requirement that the process variables selected will produce a solidification pattern with a joining angle of less than 90 degrees in the weld center. This procedure qualification includes a requirement for a macro-etch test to be performed in the longitudinal weld direction of the center plane across the weld from base metal to base metal, and a requirement that the test verify the desired solidification pattern has been obtained and that the weld is free of unacceptable fissures or cracks. The results of the tests are included in the certified qualification test report. For the longitudinal production welds of low alloy steel vessels, material containing base metal and weld metal taken from weld prolongations are tested as follows: tensile and impact tests similar to those required for the base metal by paragraph NB-3211(d) of Section III are made to determine the mechanical properties of the quenched and tempered weld metal. To verify that the specified weld solidification pattern has been obtained and that the weld center is sound, either a macro-etch test or an impact test with the specimen notch located at the weld center is used. The tests specified are applied to each of the welds. In the event that properties obtained from tests identified are not acceptable, additional procedures qualification is performed.

Stainless steel corrosion resistant weld overlay cladding of low alloy steel components conforms to the requirements of RG 1.43, "Control of Stainless Steel Weld Cladding of Low-Alloy Steel Components." Controls to limit underclad cracking of susceptible materials conform to the requirements of RG 1.43.

Procedure Qualification Records and Welding Procedure Specifications performed to support welding of low alloy steel welds in the RCPB conform to the requirements of RG 1.50, "Control of Preheat Temperature for Welding of Low-Alloy Steel" and the guidelines of ASME Section III, Division 1, Nonmandatory Appendix D.

Interpass temperatures to support welding of low alloy steel welds in the RCPB are qualified per ASME Sections III and IX. The typical minimum preheat temperature is 200°F and the typical maximum interpass temperature is 600°F.

Welders and welding operators are qualified in accordance with ASME Section IX and RG 1.71, "Welder Qualification for Areas of Limited Accessibility."

The practices for storing and handing welding electrodes and fluxes comply with ASME Code, Section III, Paragraphs NB-2400 and NB-4400.

5.2.3.3.3 Nondestructive Examination for Ferritic Steel Tubular Products

Nondestructive examinations performed on ferritic steel tubular products to detect unacceptable defects will comply with ASME Section III, NB-2550 through NB-2570, and ASME Section XI examination requirements.

5.2.3.4 Fabrication and Processing of Austenitic Stainless Steels

5.2.3.4.1 Prevention of Sensitization and Intergranular Corrosion of Austenitic Stainless Steels

Austenitic stainless steels are susceptible to different forms of intergranular corrosion in aggressive environments when sensitized. Grain boundary carbide sensitization occurs when metal carbides precipitate on the grain boundaries when the material is heated in the temperature range of 800°F to 1,500°F.

Avoidance of intergranular attack in austenitic stainless steels is accomplished by five main methods:

- Use of low carbon (less than 0.03 wt% carbon) unstabilized austenitic stainless steels.
- Monitoring of the ferrite number of weld filler metals to ensure correct ferrite content.
- Utilization of materials in the solution annealed plus rapid cooled condition and the prohibition of subsequent heat treatments in the 800°F and 1,500°F temperature range.
- Control of primary water chemistry to maintain an environment which does not promote intergranular attack.
- Control of welding processes and procedures to avoid heat affected zone sensitization as given in RG 1.44.

The water chemistry in the RCS is controlled to the ranges specified in Table 5.2-3 and by plant procedures to prevent the intrusion of aggressive species. Section 9.3.4 addresses RCS water chemistry control. Precautions are taken to prevent the intrusion of chlorides and other contaminants into the system during fabrication, shipping, and storage. The use of hydrogen in the reactor coolant inhibits the presence of oxygen during operation. The effectiveness of these controls has been demonstrated by tests and operating experience.

Measures are taken to prevent sensitization of unstabilized austenitic stainless steel materials during component fabrication; the wrought products listed in Table 5.2-2 are used in the solution annealed condition and rapidly cooled. Heat treatment parameters comply with ASME Section II. The material is either cooled by water quenching or cooled quickly enough through the sensitization temperature range to avoid carbide formation at the grain boundaries and sensitization. Non-sensitization of the base materials can be verified by a corrosion test – in accordance with ASTM A-262 (Reference 4), Practice A or E – as required by RG 1.44. When testing of the weld heat affected zone (HAZ) of materials is required, the tests are performed in accordance with ASTM A-262, Practice E. Low carbon austenitic stainless steel materials and their welds in product forms which do not have inaccessible cavities or chambers that would preclude rapid cooling when water quenching need not be corrosion tested, provided that the solution heat treatment is followed by water quenching or rapid cooling so as to avoid chromium carbide precipitation.

All unstabilized austenitic stainless steel material, including weld material, has a carbon content of less than 0.03 wt%. RG 1.44 requires that any material subjected to sensitizing temperatures subsequent to solution heat treatment should be material with a carbon content of less than 0.03 wt%.

Stabilized austenitic stainless steels have a stabilizing heat treatment above 800°F where chromium carbides are prevented from precipitating after the stabilizing element combines with the carbon. Due to the stabilizing heat treatment, stabilized austenitic stainless steels are not expected to experience sensitization. The lack of sensitization in these alloys, in addition to the five points listed above, negates the concern of intergranular corrosion in stabilized austenitic stainless steels. Stabilized austenitic stainless steel is solution annealed and rapidly cooled so that the material is cooled through the sensitization temperature range rapidly to prevent sensitization. If means other than rapid cooling are used, the material is tested in accordance with Practice E of ASTM A262 to demonstrate the material is in the unsensitized condition.

Due to necessary welding, the unstabilized austenitic stainless steel in the HAZ is heated in the sensitized temperatures range (800°F to 1500°F) during fabrication. Welding practices and material composition are controlled to manage the sensitization while the material is in this temperature range and all weld metals have a carbon content of less than 0.03 wt% to prevent undue sensitization.

The unstabilized austenitic stainless steel casting material used in the RCP is used for the RCP casing. The maximum carbon content of this material, as with other austenitic stainless steel materials, is 0.03 wt%.

No cold-worked grade austenitic stainless steels are used for manufacture of the RCPB components. Inservice inspections follow the requirements of ASME Section XI, industry materials reliability programs, and NRC guidance to check for intergranular corrosion from sensitization.

Actual yield strength values for austenitic stainless steel materials are supplied on material test reports for each component at the time of shipment.

Forged stainless steel components within the RCPB that are subject to ASME Section XI volumetric examinations are specified to have a sufficiently large grain size to allow for inspection through ultrasonic methods, while continuing to meet the specified mechanical properties of the ASME Code.

5.2.3.4.2 Cleaning and Contamination Protection Procedures

Austenitic stainless steel materials used in the fabrication, installation, and testing of nuclear steam supply components and systems are handled, protected, stored, and cleaned according to recognized and accepted methods that are designed to minimize contamination which could lead to stress corrosion cracking.

Procedures are developed to provide cleanliness controls during all phases of manufacture and installation including final flushing. As applicable, these procedures supplement the equipment specifications and purchase order requirements of individual austenitic stainless steel components procured for RCPB applications and follow the guidance of RG 1.37, Revision 1, “Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants.” Controls are established to minimize the introduction of potentially harmful contaminants including chlorides, fluorides, and low melting point alloys on the surface of austenitic stainless steel components. In accordance with RG 1.44, all cleaning solutions, processing equipment, degreasing agents, and other foreign materials are completely removed at any stage of processing prior to elevated temperature treatments. Pickling of austenitic stainless steel is avoided.

Tools for abrasive work such as grinding, polishing, or wire brushing do not contain, and are not contaminated by previous usage on, ferritic carbon steel or other materials that could contribute to intergranular cracking or stress-corrosion cracking.

5.2.3.4.3 Compatibility of Construction Materials with External Insulation and Reactor Coolant

The thermal insulation used on the RCPB is the reflective stainless steel type, wherever clearances permit. Areas of little clearance are insulated with high performance compounded materials which yield low leachable chloride and/or fluoride concentrations in accordance with RG 1.36, "Nonmetallic Thermal Insulation for Austenitic Stainless Steel." The martensitic stainless steel forming the CRDM pressure housing is not insulated. Calcium silicate is not used as an insulating material. The insulation is designed to prevent the ingress and retention of liquid to reduce contamination of the insulation and the components of the RCS.

In the event of coolant leakage, the ferritic materials will show increased general corrosion rates. Where minor leakage is anticipated from service experience, such as valve packing and pump seals, only materials that are compatible with the coolant are used. Ferritic materials exposed to coolant leakage can be readily observed as part of the plant specific boric acid corrosion control (BACC) program utilizing in-service visual and/or other nondestructive inspections to assure the integrity of the component for subsequent service.

5.2.3.4.4 Control of Welding

Welding is conducted utilizing procedures qualified according to the rules of ASME Sections III and IX. Control of welding variables, as well as examination and testing during procedure qualification and production welding, is performed in accordance with ASME Code requirements.

Welding on RCPB components conforms to the guidance contained in RG 1.31, “Control of Ferrite Content in Stainless Steel Weld Metal,” RG 1.34, "Control of Electroslag Weld Properties," and RG 1.71, "Welder Qualification for Areas of Limited Accessibility." The procedure qualification for electroslag welding includes a requirement that the process variables selected will produce a solidification pattern with a joining angle of less than 90 degrees in the weld center. This procedure qualification includes a requirement for a macro-etch test to be performed in the longitudinal weld direction of the center plane across the weld from base metal to base metal, and a requirement that the test verify the desired solidification pattern has been obtained and that the weld is free of unacceptable fissures or cracks. The results of the tests are included in the certified qualification test report. For the longitudinal production welds of low alloy steel vessels, material containing base metal and weld metal taken from weld prolongations are tested as follows: tensile and impact tests similar to those required for the base metal by paragraph NB-3211(d) of Section III are made to determine the mechanical properties of the quenched and tempered weld metal. To verify that the specified weld solidification pattern has been obtained and that the weld center is sound, either a macro-etch test or an impact test with the specimen notch located at the weld center is used. The tests specified are applied to each of the welds. The austenitic stainless steel production welding is monitored to verify compliance with limits for the process variables specified in the procedure qualification. In the event that properties obtained from tests identified are not acceptable, additional procedures qualification is performed.

5.2.3.4.5 Nondestructive Examination for Wrought Austenitic Stainless Steel Tubular Products

Nondestructive examinations performed on austenitic stainless steel tubular products to detect unacceptable defects will comply with ASME Section III, NB-2550 through NB-2570, and Section XI examination requirements.

5.2.3.4.6 Cast Austenitic Stainless Steel Materials used in the RCPB

The RCP casing is made from ASME SA-351 Grade CF3 material with additional restrictions on silicon (1.5% maximum) and niobium (restricted to trace elements). In addition, the ferrite content of cast austenitic stainless components in the RCPB will be limited to a ferrite content of less than 20 percent. These restrictions reduce susceptibility to thermal aging (Section 3.6.3.3.6). For cast austenitic stainless steel material used in the RCPB, the percent ferrite is calculated using Hull's equivalent factors as indicated in NUREG/CR-4513 Rev. 1 (May 1994).

5.2.3.5 Prevention of Primary Water Stress-Corrosion Cracking for Nickel-Base Alloys

Nickel-base alloy components in the RCS are protected from primary water stress-corrosion cracking (PWSCC) by:

- Using only Alloy 690 and Alloys 52/52M/152 weld metals in NiCrFe applications (Alloy 600 base metal and Alloys 82/182 weld metal is not used).
- Controlled chemistry, mechanical properties, and thermo-mechanical processing requirements that produce an optimum microstructure for resistance to intergranular corrosion for NiCrFe Alloy 690 base metal.
- Limiting the sulfur content of NiCrFe base metal in contact with RCS primary fluid to maximum 0.02 wt%.

The NiCrFe materials that are used in the RCPB, including weld materials, conform to the fabrication, construction, and testing requirements of ASME Section III. Material specifications comply with ASME Section II Parts B and C.

Inservice inspections follow the requirements of ASME Section XI, industry materials reliability programs, and NRC guidance to confirm PWSCC does not occur in Alloy 690 materials.

EPRI report MRP-111, “Resistance to Primary Water Stress Corrosion Cracking of Alloys 690, 52, and 152 in Pressurized Water Reactors” (Reference 5), details the prevention of and resistance of PWSCC in Alloy 690, 52/52M, and 152 in pressurized water reactors. This document concludes that wrought Alloy 690 and its weld metals (Alloys 52/52M and 152) are highly corrosion resistant materials deemed acceptable for replacing Alloy 600 in pressurized water reactor applications. No stress corrosion degradation of Alloy 690 materials had been observed in any replacement application as of the time MRP-111 was written (early 2004) and since the first use of Alloy 690 in pressurized water reactors (approximately 14 years).

Reference 5 summarizes a comprehensive review of laboratory test data of stress corrosion cracking of Alloy 690 in simulated primary water environments which provides reasonable assurance of the high resistance to PWSCC for Alloy 690 and its weld metals. Alloy 690 and its weld metals have been used in numerous PWR replacement component items and are unlikely to experience stress corrosion cracking under standard operating conditions and in monitored primary water.

5.2.3.6 Threaded Fasteners

Threaded fasteners used in the RCS conform to the applicable requirements of ASME Section III. Materials used in threaded fasteners are selected for their compatibility with the RCS and refueling water. Only proven materials for the specific application and environment are used after evaluation of the potential for degradation. Bolting materials which have no contact with the primary water in the RCS, such as the bolting for the RCP casing, are at much less risk for stress corrosion cracking or intergranular attack than those which come in contact with primary water in the RCS. Section 3.13 provides more description of the design of threaded fasteners for the RCS.

5.2.4 Inservice Inspection and Testing of the RCPB

In accordance with GDC 32, components of the U.S. EPR that are part of the RCPB are designed to permit periodic inspection and testing of important areas and features to assess structural and leaktight integrity. The inservice inspection (ISI) and preservice inspection (PSI) program for Class 1 RCPB components is fully described, as that term is defined in SRM-SECY-04-0032 (Reference 6), in this section. The program complies with the applicable requirements of 10 CFR 50.55a.

The preservice testing (PST) and inservice testing (IST) of pumps and valves is performed in accordance with the OM Code, as described in Section 3.9.6. Inservice inspection of threaded fasteners is addressed in Section 3.13.2.

Preservice inspections and periodic inservice inspections are required for Quality Group A components of the U.S. EPR. These components are defined as Class 1 components by the ASME Boiler and Pressure Vessel Code, Section III. The ASME Code Class 1 boundary subject to inspection is comprised of the RCPB components (other than steam generator tubes, addressed in Section 5.4.2.2) and associated supports to include pressure vessels, piping, pumps, valves, and bolting that meet the definition for Quality Group A components presented in Regulatory Guide 1.26, "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants." Subsection NB of Section III of the ASME Code presents the construction requirements for Class 1 components, and Subsection IWB of Section XI presents their preservice and inservice inspection requirements. Section 3.2.2 includes a list of the ASME Code Class 1 pressure retaining components and addresses application of the 10 CFR 50.55a regulatory and Section III code criteria to their classification. Class 1 pressure retaining components and their welded attachments are pressure tested and are inspected by visual, surface, and volumetric examination methods, as required by Subsection IWB of Section XI of the ASME Code.

U. S. EPR design standards include provisions for placement of Class 1 piping and components, and establishing minimum structural clearances around them, such that adequate access for inservice inspection is maintained. These provisions preclude locating welds or portions of welds such that they would otherwise be exempt from examination due to their inaccessibility because they are encased in concrete, buried underground, located inside a penetration, or encapsulated by a guard pipe.

The preservice and inservice inspections meet the requirements set forth in Section XI of the ASME and Pressure Vessel Code as specified in 10 CFR 50.55a(g) with exceptions as permitted in 10 CFR 50.55a(g)(6)(i). The code of record (ASME Code edition) for the design of the U.S. EPR is identified in Section 5.2.1.1. The PSI program for Class 1 components consists of inspecting Class 1 components initially selected for the ISI program with specific exceptions and acceptance criteria as described in Section 5.2.4.2.

A COL applicant that references the U.S. EPR design certification will identify the implementation milestones for the site-specific ASME Section XI preservice and inservice inspection program for the RCPB, consistent with the requirements of 10 CFR 50.55a (g). The program will identify the applicable edition and addenda of the ASME Code Section XI, and will identify additional relief requests and alternatives to Code requirements.

5.2.4.1 Inservice Inspection and Testing Program

5.2.4.1.1 Arrangement and Accessibility of Systems and Components

The U.S. EPR design provides ready access to systems, structures, and components (SSC) to accommodate comprehensive inspection using currently available inspection equipment and techniques. The accessibility incorporated into the design conforms with IWA-1500 and the requirements of 10 CFR 50.55a(g)(3)(i). This readily accessible configuration allows enhanced flaw detection and reliable flaw characterization, and also lowers occupational radiation exposure through reduced inspection times.

Factors such as examination requirements, examination techniques, accessibility, component geometry, and material selection are used in evaluating component designs for ease of inspection. The components and welds requiring ISI have design features that allow ready inspection, including clearances for personnel and favorable materials, weld-joint simplicity, elimination of geometrical interferences, and proper weld surface preparation. Removable insulation is used on piping and components requiring volumetric and surface inspection. Pipe hangers and supports are positioned to accommodate weld inspection. The surfaces of welds within the inspection boundary are finished to permit effective examination.

The design of the RCPB provides accessibility to the internal surfaces of the reactor vessel, including the reactor vessel nozzle interior surfaces, and most external zones of the vessel, including the nozzle-to-reactor coolant piping welds, the top and bottom heads, and external surfaces of the reactor coolant piping, except for the area of pipe within the primary shield concrete.

Permanent and temporary platforms are provided to facilitate access to pumps, valves, and pipe welds. Space is also provided to handle and store insulation, structural members, shielding, and similar materials related to the inspection. Hoists and other handling equipment are provided, and the lighting and power sources needed for the inspection equipment are installed at appropriate locations.

These design features permit inspection of the RCPB in accordance with GDC 32.

5.2.4.1.2 Examination Categories and Methods

Examination and pressure testing categories and requirements for Class 1 components and piping, including the method of examination for the components and parts of the pressure retaining boundaries, are in accordance with IWA-2200 and Table IWB-2500-1 of the ASME Code.

Review of the inspection requirements is part of the design process, and this review results in component designs that allow examination by existing methods, and also results in recommendations for enhanced inspections. The visual, surface, and volumetric examination techniques and procedures are performed in accordance with Articles IWA-2000 and IWB-2000 of Section XI of the ASME Code. The acceptance standards for the results from these examinations are in accordance with Article IWB-3000 of Section XI.

Three different visual examination methods are used for detecting imperfections that are open to the surface. The VT-1 examinations detect discontinuities and imperfections on the surface of components, including cracks, wear, corrosion, or erosion. The VT-2 examinations detect evidence of leaks from pressure retaining components during system pressure tests. The VT-3 examinations determine the general condition of components and their supports by verifying parameters such as clearances, settings, and physical displacements. These latter examinations also detect discontinuities and imperfections, such as loss of integrity at bolted or welded connections; loose or missing parts; or debris, corrosion, wear, or erosion. Visual examination by remote viewing techniques is performed in accordance with Subsubarticle IWA-2210 of Section XI and Article 9 of Section V of the ASME Code.

Surface examinations are performed using either the liquid penetrant, magnetic particle, or eddy current method. Articles 6 and 7 of Section V of the ASME Code present the performance requirements for liquid penetrant and magnetic particle examinations, respectively. Mandatory Appendix IV of Section XI of the ASME Code presents the requirements for performing eddy current examination for detecting surface flaws. Mechanized surface examination techniques are verified to provide results at least equivalent to manual surface examination techniques.

Volumetric examinations may be performed using radiography, ultrasonic, or eddy current techniques (manual or remote). Due to logistical and administrative control issues associated with radiography, ultrasonic examination is generally preferred for regularly scheduled volumetric examination of process component welds, while an eddy current examination is generally preferred for inspecting heat exchanger tubes and other small diameter or limited-access components. Radiography is, however, a permissible volumetric examination technique and may be incorporated in the ISI program. Performance requirements for these three volumetric examination techniques are in accordance with these sections of the ASME code:

- Ultrasonic—Section XI, Mandatory Appendices I, VII, and VIII.
- Eddy Current—Section V, Article 8.
- Radiography—Section V, Article 2.

The methods, procedures, and requirements for qualification of personnel performing ultrasonic testing comply with the guidance provided in Appendix VII of Section XI of the ASME code. In addition, performance demonstration for ultrasonic examination procedures, equipment, and personnel used to detect and size flaws is in accordance with the requirements of Appendix VIII of ASME Section XI. Use of Appendix VIII and the supplements to Appendix VIII and Article I-3000 of Section XI must be in accordance with the 2001 edition of the code, until use of Appendix VIII and the supplements to Appendix VIII and Article I-3000 in accordance with a later edition and addenda of the code is approved in accordance with 10 CFR 50.55a. In the event that methods, procedures, and requirements not qualified in accordance with the requirements of Appendix VIII are used for ultrasonic examination of reactor-vessel-to-flange welds, closure-head-to-flange welds, and integral attachment welds, such activities will conform to the regulatory positions of RG 1.150.

Acoustic emission may be used to monitor the growth of flaws initially detected by other non-destructive examination methods, in accordance with ASME code Section V, Article 13, and the requirements of IWA-2234 of Section XI of the ASME code.

Alternative examination methods, a combination of methods, or newly developed techniques may be substituted for the methods specified in division 1 of the ASME code, provided the requirements of IWA-2240 of Section XI of the ASME code are met. Use of the provision for alternative examination methods in IWA-2240 must be in accordance with the 1997 addenda of the code until use of IWA-2240 in accordance with a later edition and addenda of the code is approved in accordance with 10 CFR 50.55a.

5.2.4.1.3 Inspection Intervals

Inspection scheduling for Class 1 components is in accordance with the requirements of Subarticles IWA-2400 and IWB-2400 of Section XI of the ASME code, and is generally established so that all required inspections are completed during successive ten year intervals. Inservice examinations are intended to be performed during normal plant outages, such as refueling or maintenance shutdowns, that occur during the inspection interval. Thus, each inspection interval may be reduced (except the first interval of inspection program A) or extended by as much as one year to enable an inspection to coincide with a plant outage. It is not necessary that the inspection intervals for the IWB (Class 1) portion of the ISI program conform to the same inspection programs as those for the IWC (Class 2) and the IWD (Class 3) inspections.

5.2.4.1.4 Evaluation of Examination Results

Evaluation of the examination results for Class 1 components is in accordance with Article IWB-3000 of Section XI. Article IWB-3000 presents parametric flaw size limits that may be used to determine the acceptability of returning affected components to immediate or continued service. The IWB-3000 further specifies the process for resolving unacceptable results so that an affected component may be returned to service. Depending upon the type of examination and flaw characteristics, such resolution may be accomplished by supplemental examination, repair, replacement, or acceptance by analytical evaluation.

Components whose inservice volumetric and surface examination(s) either reconfirm the absence of flaws or detect flaws that are acceptable under the provisions of IWB-3131(b) are acceptable for continued service. Components whose inservice volumetric and surface examination(s) detect flaws that exceed the acceptance standards of Table IWB-3410-1 are acceptable for continued service without a repair/replacement activity if an analytical evaluation, as described in IWB-3600, meets the acceptance criteria of IWB-3600. The area containing the flaw shall be subsequently re-examined. Alternatively, correction by a repair/replacement activity, to the extent necessary to meet the acceptance standards of IWB-3000, and satisfactory performance of the additional examination requirements of IWB-2430, is required to qualify such components for return to service.

Repairs and replacements of Class 1 components are in accordance with the repair/replacement program, which is implemented in accordance with Article IWA-4000 of Section XI of the ASME code.

5.2.4.1.5 System Pressure Tests

Class 1 systems and components subject to pressure testing are tested using reactor coolant as the pressurizing medium. Testing is performed in accordance with Articles IWA-5000 and IWB-5000 of the ASME code. The tests are conducted at a system pressure not less than that corresponding to 100 percent rated reactor power. The system test pressure and temperature is attained at a rate in accordance with the heat-up limitations specified for the limiting system. The test pressure shall not exceed the limiting conditions specified in the technical specifications.

The pressure retaining boundary during the system leakage test corresponds to the reactor coolant boundary, with all valves in the position required for normal reactor operation startup. The visual examination, however, extends to and includes the second closed valve at the boundary extremity. The pressure-retaining boundary during the system leakage test conducted at or near the end of each inspection interval extends to all Class 1 pressure retaining components within the system boundary.

In accordance with IWA-5212, a hydrostatic test and visual examination may be performed in lieu of the system pressure test and visual examination. For hydrostatic testing, the test pressure is as specified in IWB-5230. The initial hydrostatic test pressure is expected to be 1.10 times the pressure corresponding to 100 percent rated reactor power, maintained at a test temperature of 100°F or below. However, this pressure, and the pressure for subsequent hydrostatic tests, may be further limited by consideration for such variables as test temperature, fracture toughness of ferritic materials, and the limiting conditions specified in the plant technical specifications.

These pressure tests are included in the list of transients, presented in Section 3.9.1, used for design and fatigue analysis of all ASME Code Class 1 components.

5.2.4.1.6 Code Exemptions

No exceptions from code required examinations for Class 1 PSI or ISI are required for the U.S. EPR.

Certain Class 1 components are exempt from surface and volumetric examination in accordance with Subarticle IWB-1220. These include:

- Components that are connected to the RCS and are part of the RCPB, and that are of such a size and shape so that upon postulated rupture the resulting flow of coolant from the RCS under normal plant operating conditions is within the capacity of makeup systems that are operable from on-site emergency power. The emergency core cooling systems are excluded from the calculation of makeup capacity.
- Components and piping segments of nominal pipe size (NPS) 1 and smaller, except for steam generator tubing, including those:
  - That have one inlet and one outlet, both of which are NPS 1 and smaller.
  - Those that have multiple inlets or multiple outlets whose cumulative cross-sectional area does not exceed the cross-sectional area defined by the OD of NPS 1 pipe.
- Reactor vessel head connections and associated piping, NPS 2 and smaller, made inaccessible by control rod drive penetrations.

5.2.4.1.7 Relief Requests

No relief from Class 1 PSI or ISI requirements is required for the U.S. EPR.

5.2.4.1.8 Code Cases

No code cases applicable to Class 1 PSI or ISI requirements are invoked for U.S. EPR design. However, supplemental inservice inspections for the reactor pressure vessel head are required, in accordance with 10 CFR 50.55a. Compliance with the requirements of this order may be accomplished with conditional implementation of code case N-729-1, "Alternative Examination Requirements for PWR Reactor Vessel Upper Heads with Nozzles Having Pressure-Retaining Partial-Penetration Welds." COL applicants that reference the U.S. EPR design certification may invoke code case N-729-1, with conditions, in accordance with 10 CFR 50.55a.

5.2.4.1.9 Augmented ISI to Protect Against Postulated Piping Failures

No Class 1 piping penetrates the Reactor Building. Therefore, augmented ISI to protect against postulated failures of Class 1 piping between containment isolation valves is not required for the U.S. EPR. Refer to Section 6.6 for a description of augmented ISI for Class 2 high energy piping.

5.2.4.1.10 Other Inspection Programs

The ISI program includes provisions to detect and correct potential RCPB corrosion caused by boric acid leaks, as described in NRC generic letter 88-05. The ISI program for the U.S. EPR regarding potential RCPB corrosion due to boric acid leaks includes regular visual inspections as specified in Sections 5.2.4.1.1 through 5.2.4.1.4 and use of RCPB leakage detection systems as described in Section 5.2.5.

The ISI program includes supplemental inservice inspections for the reactor pressure vessel head consistent with those inspections required by NRC order EA-03-009 and first revised order EA-03-009.

5.2.4.2 Preservice Inspection and Testing Program

The PSI program for Class 1 components conforms to the guidelines of Article NB-5280 of Section III, Division I, of the ASME code. The program consists of inspecting Class 1 components initially selected for the ISI program described in Section 5.2.4.1, with the exceptions of pressure testing the pressure retaining components (examination category B-P of Table IWB-2500-1) and visual VT-3 examination of the internal surfaces of the pump casings and valve bodies (examination categories B-L-2 and B-M-2 of Table IWB-2500-1).

In accordance with the provisions of IWB-3112(b), a component whose preservice volumetric or surface examination (in accordance with IWB-2200) detects flaws that meet the nondestructive examination standards of NB-2500 and NB-5300 is acceptable for service. A component whose preservice volumetric or surface examination (in accordance with IWB-2200) detects flaws, other than those determined acceptable by the provisions of IWB-3112(b), that exceed the standards of Table IWB-3410-1 must be corrected by a repair/replacement activity to the extent necessary to meet the acceptance standards prior to placing the component in service.

A component whose preservice visual examination detects the relevant conditions described in the standards of Table IWB-3410-1, unless such components are shown by supplemental volumetric or surface examinations to meet the requirements for those supplemental examinations (IWB-3110), must be corrected by a repair/replacement activity or by corrective measures to the extent necessary to meet the acceptance standards of Table IWB-3410-1.

The preservice testing (PST) program for Class 1 pumps and valves consists of testing of the Class 1 pumps and valves selected for the IST program, in accordance with the OM Code, as described in Section 3.9.6.

5.2.5 RCPB Leakage Detection

The RCPB leakage detection systems are designed to detect and, to the extent practical, identify the source of reactor coolant leakage. Diverse measurement methods include monitoring of sump level and flow, containment airborne radioactivity, and containment air cooler condensate flow.

The RCPB leakage detection systems are designed and classified in accordance with RG 1.29 (GDC 2). Section 3.2 identifies the seismic and system quality group classifications for the leakage detection systems.

The RCPB leakage detection systems conform to the guidance of RG 1.45, Revision 1 regarding detection, monitoring, quantifying, and identification of reactor coolant leakage (GDC 30).

The RCPB leakage detection systems are sufficiently reliable, redundant, and sensitive to support the application of LBB analyses to eliminate the need to consider the dynamic effects of main reactor coolant loop and PZR surge line ruptures from the design basis. LBB analyses are addressed in Section 3.6.3.

Reactor coolant leakage is categorized as either identified leakage or unidentified leakage. Identified leakage includes:

- Leakage into closed systems (e.g., pump seal or valve packing leaks). The leakage is captured, quantified, and directed to a sump or collection tank.
- Leakage into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of unidentified leakage monitoring systems or not to be from a flaw in the RCPB.
- Intersystem leakage into connected systems, including leakage through steam generator tubes.

All other leakage is categorized as unidentified leakage.

The design leakage rates are:

- No pressure boundary leakage. (Seal and gasket leakage is not “pressure boundary leakage” in this context.)
- 1 gpm unidentified leakage.
- 10 gpm identified leakage.
- 150 gallons per day primary to secondary leakage through any one steam generator.

5.2.5.1 Detecting, Monitoring and Collecting Unidentified Leakage

These methods are used to detect and monitor unidentified leakage inside containment for the U.S. EPR:

- Containment sump level and discharge flow monitoring.
- Containment atmosphere radiation monitoring.

- Containment air cooler condensate monitoring.

These additional methods also indicate leakage inside containment:

- RCS inventory balance.
- Localized humidity and temperature monitoring.

5.2.5.1.1 Containment Sump Level and Discharge Flow Monitoring

The nuclear island drain and vent system (NIDVS) leakage detection function consists of water level measurements provided within the system sumps and collection tanks. The NIDVS instrumentation is credited for main reactor coolant loop and PZR surge line LBB monitoring and can reliably detect a leakage rate of 0.5 gpm in one hour.

Increased frequency of sump pump actuation may be an indication of RCS leakage. An alarm is provided to the operator in the MCR when a pump is running. An alarm is also generated if the pump continues to run for an extended period without reaching the low level, indicating that there is a large continuous flow towards the reactor building sump.

The NIDVS is designed and equipped with provisions to permit testing for operability and calibration.

The reactor building floor drains collect leakage from contaminated spaces in the Reactor Building and from process drains that cannot be recycled. The RCPB leakage drains to the floor drains system and ultimately to the sump where it is identified and quantified by the sump instrumentation. The reactor building floor drains have five small intermediate collection sumps where separate branches of the drain system intersect. The total volume of all five of these intermediate sumps is less than 0.5 gallons, so that they have no significant effect on the flow from an unidentified leakage source, or the prompt identification of it by the sump instrumentation.

5.2.5.1.2 Containment Atmosphere Radiation Monitoring

Gaseous and airborne particulate radiation monitors continuously monitor radioactivity levels in the containment atmosphere. Radiation levels are indicated in the MCR and alarms alert the operator to elevated levels of radioactivity.

The airborne particulate radiation monitors can detect a 1.0 gpm leakage rate within one hour at full power operation. The gaseous radiation monitors can detect a 1.0 gpm leakage rate within one hour at full power operation. The sensitivity of the containment atmosphere radiation monitors is sufficient for detection of the limiting leakage based on the realistic source terms analysis presented in the environmental report as addressed in Section 11.1. Section 11.5 addresses radiation monitors in more detail.

The airborne particulate radioactivity monitors are designed to withstand the effects of the safe shutdown earthquake and remain functional.

5.2.5.1.3 Containment Air Cooler Condensate Monitoring

Condensate level and flow sensors are installed in the collection tank and drain line of each containment air cooler unit. The system is credited for main reactor coolant loop and PZR surge line LBB monitoring and can reliably detect a leakage rate of 0.5 gpm in one hour. The condensation rate can be determined by rate of level change or by direct flow indication and is indicated in the MCR. An alarm is generated when the threshold is reached.

5.2.5.1.4 RCS Inventory Balance

The RCS inventory balance is a quantitative inventory or mass balance calculation to measure RCS leakage. This approach allows both the type and magnitude of leakage to be determined.

To perform an inventory balance accurately, the plant must be in a steady-state condition. This condition is defined as stable RCS pressure, temperature, and power level. The PZR, PRT, reactor coolant drain tank (RCDT), and in-containment refueling water storage tank (IRWST) levels must be known and trended for the calculation. The mass balance calculation may also require temporary isolation of interconnected support systems from the RCS. The leakage rate is determined by observing the rate of change of reactor coolant inventory as indicated by the change in volume control tank (VCT) level.

Identified leakage is monitored using the RCDT level to calculate a leakage rate and by monitoring the intersystem leakage. The unidentified leakage rate is calculated by subtracting the identified leakage rate from the total RCS leakage rate.

The inventory balance calculation is performed using data from display and processing computer systems with additional input from sensors in the protection system, the CVCS, the NIDVS, and other systems.

5.2.5.1.5 Localized Humidity and Temperature Monitoring

Humidity and air temperature measuring instruments are installed in these locations:

- Near each of the RCS hot legs next to the steam generator.
- Near each of the RCS cold legs next to the RCP.
- Each upper steam generator compartment.
- Near the PZR surge line.

- The PZR compartment.

Containment atmospheric conditions are determined from these humidity and temperature measurements which indicate in the MCR. Alarms are generated by increasing humidity.

5.2.5.2 Detecting, Monitoring and Collecting Identified Leakage

Provisions are incorporated into the U. S. EPR design to isolate, capture, and quantify leakage from known potential sources, such as flanges and relief valves, so that such leakage may be monitored separately from unidentified leakage. Minor leakage of the RCS may also be identified by operating personnel during normal plant operation. Such leakage is also classified as identified leakage if it can be quantified.

5.2.5.2.1 Reactor Pressure Vessel Flange Leak-Off Monitoring

The reactor pressure vessel flange is equipped with two concentric O-rings. A seal leak-off line drains from the space between these two O-rings and is routed to the Reactor Coolant Drain Tank (RCDT) via the NIDVS. Temperature and pressure sensors installed on the leak-off line, which indicate in the MCR, detect leakage past the inner O-ring. If the leakage rate from this source is quantified, the leak can be classified as identified leakage.

5.2.5.2.2 PSRVs and Primary Depressurization System Valves

The PSRVs and primary depressurization system (PDS) valves are normally closed. The PSRVs have an associated water collector inside the PZR to form a water seal. The inlet piping to the PDS valves forms a loop seal. Temperature sensors are mounted on the inlet nozzles of the PSRVs and the inlet pipe of PDS valves. Due to ambient cooling, the inlet nozzles and pipes to these valves are normally at a lower temperature than the PZR steam space. Leakage through a PSRV or PDS valve seat will result in an increasing temperature in the PSRV nozzle or PDS valve inlet pipe. A decrease in the temperature difference between PZR steam space and the valves' inlet pipe or nozzles will generate an alarm in the MCR to warn of potential valve leakage. Temperature sensors are also mounted on the discharge pipes of the spring-loaded pilot valves and the solenoid-operated pilot valves for the PSRVs in order to detect leakage through these valves. High temperature in these discharge pipes will also generate an alarm in the MCR. Leakage through the PSRVs, their respective pilot valves, and the PDS valves is directed to the PRT where it is collected and quantified.

5.2.5.3 Detecting and Monitoring Intersystem Leakage

Substantial intersystem leakage from the RCS to its connected auxiliary systems is not anticipated. However, the possibility of intersystem leakage across passive barriers and through closed valves still exists. Intersystem leakage is identified by increasing level, temperature, flow, or pressure in the connected systems. Intersystem leakage is also detected through relief valve actuation or increasing radioactivity in the connected systems.

5.2.5.3.1 Safety Injection System / Residual Heat Removal System

Pressure and temperature sensors in the safety injection system / residual heat removal system (SIS/RHRS) suction lines detect leakage past their RCS isolation valves. Pressure and temperature indication and alarms are provided in the MCR to identify such system leakage.

SIS/RHRS accumulators are isolated from the RCS by check valves downstream of normally open isolation valves. Leakage past these check valves is identified by accumulator pressure and level indications and alarms in the MCR.

5.2.5.3.2 Steam Generator Tubes

A potential identified leakage path for the RCS is through the steam generator tubes into the shell side of the steam generator. Identified leakage from the steam generator primary side is detected by:

- Condenser air removal system discharge noble gas radiation monitors.
- Steam generator blowdown radiation monitors.
- Main steam line ^{16}N radiation monitors.

These monitors indicate in the MCR. These measurements are supplemented by process sampling and laboratory analysis.

5.2.5.3.3 Component Cooling Water System

Leakage from the RCS to the component cooling water system (CCWS) is identified by these methods:

- Radiation monitors, which detect contamination of the system, indicate and alarm in the MCR.
- Monitoring of the CCWS surge tank level and discharge flow from selected components. Surge tank level indication is provided in the MCR.
- Leakage through the LHSI heat exchanger tubes into the CCWS is identified by temperature sensors in the heat exchanger inlet and outlet piping, which indicate and alarm in the MCR.
- Leakage from the RCP thermal barriers to the CCWS is detected by pressure, temperature, and flow sensors downstream of the barriers, which indicate and

alarm in the MCR. In the unlikely event of a thermal barrier tube rupture, CCWS flow to the thermal barrier automatically isolates.

- Leakage from the letdown line heat exchangers to the CCWS is detected by radiation monitors and flow sensors which indicate and alarm in the MCR. In the unlikely event of a tube rupture, CCWS flow to the letdown line heat exchanger automatically isolates.

These methods are supplemented by radiation monitors, process sampling, and laboratory analysis, which indicate increased CCWS system activity from small leaks. Section 9.2.2 and Section 11.5 further address the control of RCS leakage into the CCWS.

5.2.5.4 Inspection and Testing Requirements

The leakage detection systems are designed to permit operability testing and calibration during plant operation. Refer to Chapter 16 (SR 3.4.14) for surveillance requirements. Periodic testing of the floor drainage system verifies that it is free of blockage.

5.2.5.5 Instrumentation Requirements

The leakage detection systems provide data to the instrumentation and control systems for indication, alarm, and archival. Operators in the MCR are provided with the leakage rate (gpm) from each detection system and a common leakage equivalent (gpm) from both identified and unidentified sources. Alarms indicate that leakage has exceeded predetermined limits. The instrumentation system is described in Section 7.1.

5.2.5.5.1 RCDT Indications

The RCDT collects continuous flow during operation from PZR degassing and the RCP seals' leakoff. This flow is quantified from tank level and pump run time indications and a baseline normal in-leakage rate is established. Changes in this rate indicate leakage from additional components whose discharge is routed to the RCDT. Such leakage can be identified through indications from these components and, once quantified, can be monitored as identified leakage.

The additional monitored leakage connections that discharge to the RCDT include the PSRV valve body drains, the reactor vessel O-ring seal leakoff, RCP static seal (main flange) leakoff, valve stem packing leakage, and safety valve discharge lines from the combined RCP #1 seal return line, the four RCP thermal barrier return lines, the CVCS letdown line, and the CVCS charging line. Additional equipment and component drain connections to the RCDT are used only during shutdown or during startup operations and are isolated from the RCDT by a closed manual valve, or are disconnected and flanged, during power operation and are not expected to affect RCPB leakage monitoring efforts.

5.2.5.5.2 Reactor Building Sump Level

During normal operation the Reactor Building sump collects water from the reactor building floor drains and the Reactor Building annular space floor drain sump. Sump level and automatic pump operation for both sumps are indicated in the MCR to allow prompt identification of any unidentified leakage in the Reactor Building.

5.2.6 References

1. ASME Boiler and Pressure Vessel Code, Section III," Rules for Construction of Nuclear Facility Components," The American Society of Mechanical Engineers, 2004.

2. ASME Code for Operation and Maintenance of Nuclear Power Plants, The American Society of Mechanical Engineers, 2004.

3. EPRI Report 1014986, "Pressurized Water Reactor Primary Water Chemistry Guidelines," Volume 1, Revision 6, Electric Power Research Institute, December 2007.

4. ASTM A-262, "Standard Practices for Detecting Susceptibility to Intergranular Attack in Austenitic Stainless Steels," American Society for Testing and Materials International, December 2002.

5. EPRI Report MRP-111, "Materials Reliability Program, Resistance to Primary Water Stress Corrosion Cracking of Alloys 690, 52/52M, and 152 in Pressurized Water Reactors," Electric Power Research Institute, March 2004.

6. SRM-SECY-04-0032, "Programmatic Information Needed for Approval of a Combined License Without Inspections, Tests, Analyses and Acceptance Criteria," May 2004.

7. ANP-10263P-A, "Codes and Methods Applicability Report for U.S. EPR," AREVA NP Inc., August 2007.

8. NRC Branch Technical Position 5-2, "Overpressure Protection of Pressurized Water Reactors While Operating at Low Temperature, Standard Review Plan," U.S. Nuclear Regulatory Commission, March 2007.

Table 5.2-1—ASME Code Cases

Code Case Number	Title
N-60-5	Material for Core Support Structures Section III, Division 1
N-71-18	Additional Materials for Subsection NF, Class 1, 2, 3, and MC Supports Fabricated by Welding, Section III, Division 1
N-284-1 [1]	Metal Containment Shell Buckling Design Methods, Section III, Division 1, Class MC
N-319-3	Alternate Procedure for Evaluation of Stresses in Butt Welding Elbows in Class 1 Piping, Section III, Division 1
N-729-1	Alternative Examination Requirements for PWR Reactor Vessel Upper Heads with Nozzles Having Pressure-Retaining, Partial-Penetration Welds

NOTES:

1. See Section 3.8 for use.

Table 5.2-2—Material Specifications for RCPB Components
Sheet 1 of 3

Component	Material
RCPB Piping	
Reactor coolant piping & surge line	ASME SA-182 Grade F304 (see Notes 3 & 4) ASME SA-336 Grade F304 (see Notes 3 & 4)
Reactor coolant piping & surge line fittings & nozzles	ASME SA-182 Grade F304 (see Notes 3 & 4) ASME SA-336 Grade F304 (see Notes 3 & 4)
Reactor coolant piping other than loop & surge line	ASME SA-213 Grade TP304L (Seamless) (see Note 3 & 4) ASME SA-312 Grade TP304L (Seamless) (see Note 3 & 4) ASME SA-312 Grade TP316LN (Seamless) (see Note 3 & 4)
Reactor coolant piping fittings & nozzles other than loop & surge line fittings & nozzles	ASME SA-182 Grade F304L (see Note 3) ASME SA-182 Grade F316LN (see Notes 3 & 4) ASME SA-403 Grade WP304L Class S (see Notes 3 & 4) ASME SA-403 Grade WP316LN Class S (see Notes 3 & 4)
Steam Generators	
Pressure boundary forgings (including shells, heads, tubesheet, nozzles, & openings)	ASME SA-508 Grade 3 Class 2 (see Note 1)
Small nozzles	ASME SA-105 (see Note 6)
Secondary nozzle safe ends (except emergency feedwater nozzle safe end)	ASME SA-508 Grade 3 Class 1 (see Note 1)
Emergency feedwater nozzle safe end	ASME SA-403 Grade WP316L (Seamless) (see Notes 3 & 4) ASME SA-182 Grade F316L (see Note 3)
Inlet & outlet nozzle safe ends	ASME SA-182 Grade F316 (see Notes 3 & 4) ASME SA-336 Grade F316 (see Notes 3 & 4)
Tubes	ASME SB-163 Alloy 690 (see Note 2)
Openings covers (for manways, inspection holes, & handholes)	ASME SA-533 Type B Class 2 (see Note 1)
Openings studs (for manways, inspection holes, & handholes)	ASME SA-193 Grade B16 (see Note 1) ASME SA-193 Grade B7 (see Note 1)
Primary manway studs	ASME SA-193 Grade B16 (see Note 1)
Openings nuts (for manways, inspection holes, & handholes)	ASME SA-193 Grade B16 (see Note 1) ASME SA-193 Grade B7 (see Note 1)
Pressurizer	
Upper head	ASME SA-508 Grade 3 Class 2 (see Note 1)
Bottom head	ASME SA-508 Grade 3 Class 2 (see Note 1)
Cylindrical shells	ASME SA-508 Grade 3 Class 2 (see Note 1)
Manway	ASME SA-508 Grade 3 Class 2 (see Note 1)

Table 5.2-2—Material Specifications for RCPB Components
Sheet 2 of 3

Component	Material
Manway cover	ASME SA-533 Type B Class 2 (see Note 1)
Surge nozzle	ASME SA-508 Grade 3 Class 2 (see Note 1)
Safety valve nozzles	ASME SA-508 Grade 3 Class 2 (see Note 1)
Spray nozzles	ASME SA-508 Grade 3 Class 2 (see Note 1)
Venting nozzle	ASME SA-508 Grade 3 Class 2 (see Note 1)
Primary depressurization system valve nozzle	ASME SA-508 Grade 3 Class 2 (see Note 1)
Safe ends: • Spray nozzle • Surge nozzle • Safety valve nozzle • Primary depressurization system valve nozzle Nozzles: • Temperature measurement • Level measurement • Sample	ASME SA-182 Grade F316 (see Notes 3 & 4) ASME SA-336 Grade F316 (see Notes 3 & 4)
Heater sleeves	ASME SA-182 Grade F316 (see Notes 3 & 4) ASME SA-336 Grade F316 (see Notes 3 & 4)
Vent nozzle safe ends	ASME SA-182 Grade F316 (see Notes 3 & 4)
Vent manway nozzle	ASME SA-182 Grade F316 (see Notes 3 & 4)
Valve pilot nozzle	ASME SA-182 Grade F316 (see Notes 3 & 4)
Manway studs	ASME SA-193 Grade B16 (see Note 1)
Manway nuts	ASME SA-194 Grade 16 (see Note 1)
Reactor Coolant Pump	
Pressure forgings	ASME SA-182M Grade F304 (see Notes 3 & 4)
Cooler tubes	ASME SA-213M Grade TP316 (see Notes 3 & 4)
Support stand flange – integral part of casing closure bolted assembly	ASME SA-216M Grade WCC
Pressure casting	ASME SA-351M Grade CF3 (see Notes 3 & 5)
Bolting	ASME SA-453M Grade 660 Class B (see Note 7)
Thermowell	ASME SA-479M Type 304 (see Notes 3 & 4)
Flange – integral part of pressure boundary casing closure bolted assembly	ASME SA-508M Grade 3 Class 2 (see Note 1)

Table 5.2-2—Material Specifications for RCPB Components
Sheet 3 of 3

Component	Material
Pressure boundary stud bolts & nuts	ASME SA-540M Grade B24 Class 1 (see Note 1)
Pressure boundary casing closure stud & nuts	ASME SA-540M Grade B24 Class 3 (see Note 1)
Shaft seal pressure boundary parts	ASME SA-705M Type 630 H1150 (see Note 7)
Pressure boundary welds	SFA 5.4 E308L SFA 5.4 E316L SFA 5.9 ER316L
Control Rod Drive Mechanism	
Flange, connection piece, head, loose flange	ASME SA-479 Grade 347 (see Note 3)
Latch housing	ASME SA-479/SA-182 Grade F6NM (see Note 1) (UNS S41500)
Seamless tube	ASME SA-312 Grade TP347 (Seamless) (see Note 3)
Bolt	ASME SA-453 Grade 660 (see Note 7)
Nut	ASME SA-437 Grade B4C (see Note 1)
Welding filler material	SFA 5.4 E347 SFA 5.9 ER347 SFA 5.14 ERNiCrFe-7 SFA 5.14 ERNiCrFe-7A
Pressurizer Safety Relief Valves	
A vendor for the PSRV has not been chosen for the U.S. EPR	

Notes on Table 5.2-2

1. Quenched and tempered
2. Solution annealed and thermally treated
3. Solution annealed and rapidly cooled
4. Carbon content not exceeding 0.03 wt%
5. Silicon not greater than 1.5% and niobium restricted to trace elements
6. Annealed, normalized, normalized and tempered, or quenched and tempered.
7. Solution Treatment and Hardening.

Table 5.2-3—Reactor Coolant Water Chemistry - Control Parameters

Control Parameter	Normal Operating Conditions
Lithium (pH control)	0.39 to 4.0 mg/kg
Hydrogen	17 to 28 cc(STP)/kg (1.5 to 2.5 mg/kg)
Dissolved Oxygen	< 0.100 mg/kg
Chloride	< 0.150 mg/kg
Fluoride	< 0.150 mg/kg
Sulfate	< 0.150 mg/kg
Total Boron and Boron 10	As required for reactivity control

Next File