5.2 Integrity of Reactor Coolant Pressure Boundary

This section discusses the measures to provide and maintain the integrity of the reactor coolant pressure boundary (RCPB) during plant operation. Section 50.2 of 10 CFR 50 defines the reactor coolant pressure boundary as vessels, piping, pumps, and valves that are part of the reactor coolant system (RCS), or that are connected to the reactor coolant system up to and including the following:

- The outermost containment isolation valve in system piping that penetrates the containment
- The second of two valves closed during normal operation in system piping that does not penetrate containment
- The reactor coolant system overpressure protection valves

The design transients used in the design and fatigue analysis of ASME Code Class 1 and Class CS components, supports, and reactor internals are provided in subsection 3.9.1. The loading conditions, loading combinations, evaluation methods, and stress limits for design and service conditions for components, core support structures, and component supports are discussed in subsection 3.9.3.

The term reactor coolant system, as used in this section, is defined in Section 5.1. The AP600 reactor coolant pressure boundary is consistent with that of 10 CFR 50.2.

5.2.1 Compliance with Codes and Code Cases

5.2.1.1 Compliance with 10 CFR 50.55a

Reactor coolant pressure boundary components are designed and fabricated in accordance with the ASME Boiler and Pressure Vessel Code, Section III. A portion of the chemical and volume control system inside containment that is defined as reactor coolant pressure boundary uses an alternate classification in conformance with the requirements of 10 CFR 50.55a(a)(3). Systems other than the reactor coolant system connecting to the chemical and volume control system have required isolation and are not classified as reactor coolant pressure boundary. The alternate classification is discussed in Section 5.2.1.3. The quality group classification for the reactor coolant pressure boundary components is identified in subsection 3.2.2. The quality group classification is used to determine the appropriate sections of the ASME Code or other standards to be applied to the components.

The edition and addenda of the ASME Code applied in the design and manufacture of each component are the edition and addenda established by the requirements of the Design Certification. The use of editions and addenda issued subsequent to the Design Certification is permitted or required based on the provisions in the Design Certification.

[The baseline used for the evaluations done to support this safety analysis report and the Design Certification is the 1989 Edition, 1989 Addenda, except as follows:

The criteria below are used in place of those in paragraph NB-3683.4(c)(1) and Footnote 11 to Figures NC/ND-3673.2(b)-1 of the 1989 Addenda to the 1989 Edition of ASME Code, Section III. This criteria is based on the criteria included in the 1989 Edition of the ASME Code, Section III.

For girth fillet welds between the piping and socket welded fittings, valves and flanges, and slip on flanges in ASME III Class 1, 2, and 3 piping, the primary stress indices and stress intensification factors are as follows:

Primary Stress Indices

B_1 = 0.75

B_2 = 1.5

Stress Intensification Factor

i = $2.1*(t_n/C_x)$, *but not less than 1.3*

C_x = *fillet weld leg length based on ASME III 1989 Edition, Figures NC/ND-4427-1, sketches (c-1), (c-2), and (c-3). For unequal leg length, use smaller leg length for* C_x.]*

Seismic Integrity of the CVS System Inside Containment

To provide for the seismic integrity and pressure boundary integrity of the non-safety related (B31.1, Piping Class D) CVS piping located inside containment and designated as reactor coolant pressure boundary, a seismic analysis will be performed and a CVS Seismic Analysis Report prepared with a faulted stress limit equal to the smaller of 4.5 S_h and 3.0 S_y and based on the following additional criteria :

Additional loading combinations and stress limits for nonsafety-related chemical and volume control system piping systems and components inside containment

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*NRC Staff approval is required prior to implementing a change in this material; see DCD Introduction Section 3.5.

Condition	Loading Combination[3]	Equations (ND3650)	Stress Limit
Level D	PMAX[1] + DW + SSE + SSES	9	Smaller of 4.5 S_h or 3.0 S_y
	SSES	F_{AM}/A_M[4]	1.0 S_h
	TNU + SSES	i (M1 + M2)/Z[2]	3.0 S_h

Notes:

1. For earthquake loading, PMAX is equal to normal operating pressure at 100% power.
2. Where: M1 is range of moments for TNU, M2 is one half the range of SSES moments,
   M1 + M2 is larger of M1 plus one half the range of SSES, or full range of SSES.
3. See Table 3.9-3 for description of loads.
4. F_{AM} is amplitude of axial force for SSES; A_M is nominal pipe metal area.

Component supports, equipment, and structural steel frame are evaluated to demonstrate that they do not fail under seismic loads. Design methods and stress criteria are the same as for corresponding Seismic Category I components. The functionality of the chemical and volume control system does not have to be maintained to insure structural integrity of the components.

Fabrication, examination, inspection, and testing requirements as defined in Chapters IV, V, VI, and VII of the ASME B31.1 Code are applicable and used for the B31.1 (Piping Class D) CVS piping systems, valves, and equipment inside containment.

5.2.1.2 Applicable Code Cases

ASME Code Cases used in the AP600 are listed in Table 5.2-3. In addition, other ASME Code Cases found in Regulatory Guides 1.84 and 1.85, as discussed in Section 1.9, in effect at the time of the Design Certification may be used for pressure boundary components. Use of Code Cases approved in revisions of the Regulatory Guides issued subsequent to the Design Certification may be used by the Combined License applicant using the process outlined above for updating the ASME Code edition and addenda. Use of any Code Case not approved in Regulatory Guides 1.84 and 1.85 on Class 1 components is authorized as provided in 50.55a(a)(3) and the requirements of the Design Certification.

The use of any Code Case conditionally approved in Regulatory Guides 1.84 and 1.85 used on Class 1 components meets the conditions established in the Regulatory Guide.

5.2.1.3 Alternate Classification

The Code of Federal Regulations, Section 10 CFR 50.55a requires the reactor coolant pressure boundary be class A (ASME Boiler and Pressure Vessel Code Section III, Class 1). Components which are connected to the reactor coolant pressure boundary that can be isolated from the reactor coolant system by two valves in series (both closed, both open, or one closed and the other open) with automatic actuation to close can be classified as class C (ASME Section III, class 3) according to 50.55a.

A portion of the chemical and volume control system inside containment is not classified as safety related. The classification of the AP600 reactor coolant pressure boundary deviates from the requirement that the reactor coolant pressure boundary be classified as safety related and be constructed using the ASME Code, Section III as provided in 10 CFR 50.55a. The safety-related classification of the AP600 reactor coolant pressure boundary ends at the third isolation valve between the reactor coolant system and the chemical and volume control system. The nonsafety-related portion of the chemical and volume control system inside containment provides purification of the reactor coolant and includes heat exchangers, demineralizers, filters and connecting piping. For a description of the chemical and volume control system, refer to subsection 9.3.6. The portion of the chemical and volume control system between the inside and outside containment isolation valves is classified as Class B and is constructed using the ASME Code, Section III.

The nonsafety-related portion of the chemical and volume control system is designed using ANSI B31.1 and ASME Code, Section VIII for the construction of the piping, valves, and components. The nonsafety-related portion of the CVS inside containment is analyzed seismically. The methods and criteria used for the seismic analysis are similar to those used of seismic Category II pipe and are defined in the subsection 5.2.1.1. The chemical and volume control system components are located inside the containment which is a seismic Category I structure.

The alternate classification of the nonsafety-related purification subsystems satisfies the purpose of 10 CFR 50.55a that structures, systems, and components of nuclear power plants which are important to safety be designed, fabricated, erected, and tested to quality standards that reflect the importance of the safety functions to be performed.

The AP600 chemical and volume control system is not required to perform safety-related functions such as emergency boration or reactor coolant makeup. Safety-related core makeup tanks are capable of providing sufficient reactor coolant makeup for shutdown and cooldown without makeup supplied by the chemical and volume control system. Safe shutdown of the reactor does not require use of the chemical and volume control system makeup. AP600 safe shutdown is discussed in Section 7.4.

The isolation valves between the reactor coolant system and the chemical and volume control system are active safety-related valves that are designed, qualified, inspected and tested for the isolation requirements. The isolation valves between the reactor coolant system and chemical and volume control system are designed and qualified for design conditions that include closing against blowdown flow with full system differential pressure. These valves are qualified for adverse seismic and environmental conditions. The valves are subject to inservice testing including operability testing.

The potential for release of activity from a break or leak in the chemical and volume control system is minimized by the location of the purification subsystem inside containment and the design and test of the isolation valves. Chemical and volume control system leakage inside containment is detectable by the reactor control leak detection function as potential reactor coolant pressure boundary leakage. This leakage must be identified before the reactor coolant leak limit is reached. The nonsafety-related classification of the system does not impact the need to identify the source of a leak inside containment.

5.2.2 Overpressure Protection

Reactor coolant system and steam system overpressure protection during power operation are provided by the pressurizer safety valves and the steam generator safety valves, in conjunction with the action of the reactor protection system. Combinations of these systems provide compliance with the overpressure protection requirements of the ASME Boiler and Pressure Vessel Code, Section III, Paragraphs NB-7300 and NC-7300, for pressurized water reactor systems.

Low temperature overpressure protection is provided by a relief valve in the suction line of the normal residual heat removal (RNS) system. The sizing and use of the relief valve for low temperature overpressure protection is consistent with the guidelines of Branch Technical Position RSB 5-2.

5.2.2.1 Design Bases

Overpressure protection during power operation is provided for the reactor coolant system by the pressurizer safety valves. This protection is afforded for the following events to envelope those credible events that could lead to overpressure of the reactor coolant system if adequate overpressure protection were not provided:

- Loss of electrical load and/or turbine trip
- Uncontrolled rod withdrawal at power
- Loss of reactor coolant flow
- Loss of normal feedwater
- Loss of offsite power to the station auxiliaries

The sizing of the pressurizer safety valves is based on the analysis of a complete loss of steam flow to the turbine, with the reactor operating at 102 percent of rated power. In this analysis, feedwater flow is also assumed to be lost. 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 valves. The reactor is maintained at full power (no credit for direct reactor trip on turbine trip and for reactivity feedback effects), and steam relief through the steam generator safety valves is considered. The total pressurizer safety valve capacity is required to be at least as large as the maximum surge rate into the pressurizer during this transient.

This sizing procedure results in a safety valve capacity well in excess of the capacity required to prevent exceeding 110 percent of system design pressure for the events previously listed. The discharge of the safety valve is routed through a rupture disk to containment atmosphere. The rupture disk is to contain leakage past the valve. The rupture disk pressure rating is substantially less than the set pressure of the safety valve. See subsection 5.4.11 for additional information on the safety valve discharge system. Subsection 5.4.5 describes the connection of the safety valves to the pressurizer.

Administrative controls and plant procedures aid in controlling reactor coolant system 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 of operation when operation with a water solid pressurizer is possible, a relief valve in the residual heat removal system provides low-temperature overpressure protection for the reactor coolant system. The valve is sized to prevent overpressure during the following credible events with a water-solid pressurizer:

- Makeup/letdown flow mismatch
- Inadvertent actuation of the pressurizer heaters
- Loss of residual heat removal with reactor coolant system heatup due to decay heat and pump heat
- Inadvertent start of one reactor coolant pump
- Inadvertent hydrogen addition

Of those events the makeup/letdown flow mismatch is the limiting mass input condition. Inadvertent start of an inactive reactor coolant pump is the limiting heat input condition to size the relief valve. The flow rate postulated for mass input condition is based on the flow from two makeup pumps at the set pressure of the relief valve. The heat input condition is based on a 50-degree temperature difference between the reactor coolant system and the steam generator secondary side.

The set pressure for the normal residual heat removal system relief valve is established based on the lower value of the normal residual heat removal system design pressure and the low-temperature pressure limit for the reactor vessel based on ASME Code, Section III, Appendix G, analyses. The pressure-temperature limits for the reactor vessel, based on expected material properties and the vessel design, are discussed in subsection 5.3.3.

The capacity of the residual heat removal relief valve can maintain the pressure in the reactor coolant system and the residual heat removal system to a pressure less than the lesser of 110 percent of the design pressure of the normal residual heat removal system or the pressure limit from the Appendix G analyses for the limiting event.

Overpressure protection for the steam system is provided by steam generator safety valves. The capacity of the steam system safety valves limits steam system pressure to less than 110 percent of the steam generator shell side design pressure. See Section 10.3 for details.

Section 10.3 discusses the steam generator relief valves and connecting piping.

5.2.2.2 Design Evaluation

The relief capacities of the pressurizer safety valves, steam generator safety valves, and the normal residual heat removal system relief valve are determined from the postulated overpressure transient conditions in conjunction with the action of the reactor protection system. An overpressure protection report is prepared according to Article NB-7300 of Section III of the ASME Code. WCAP-7907 (Reference 1) describes the analytical model used in the analysis of the overpressure protection system and the basis for its validity.

Chapter 15 includes a design description of certain initiating events and describes assumptions made, method of analysis, conclusions, and the predicted response of the AP600 to those events. The performance characteristics of the pressurizer safety valves is included in the analysis of the response. The incidents evaluated include postulated accidents not included in the compilation of credible events used for valve sizing purposes.

Subsection 5.4.9 discusses the capacities of the pressurizer safety valves and residual heat removal system relief valve used for low temperature overpressure protection. The setpoints and reactor trip signals which occur during operational overpressure transients are discussed in subsection 5.4.5. The set pressure for the relief valve in the normal residual heat removal system is based on a sizing analysis performed to prevent the reactor coolant system pressure from exceeding the applicable reactor vessel pressure-temperature (P/T) limits (subsection 5.3.3). The limiting mass and energy input transients are assumed for the sizing analysis. Results of this are presented in WCAP-14837 (Reference 7).

5.2.2.3 Piping and Instrumentation Diagrams

The connection of the pressurizer safety valves to the pressurizer is incorporated into the pressurizer safety and relief valve module and is discussed in subsection 5.4.9. The pressurizer safety and relief valve module configuration appears in the piping and instrumentation drawing for the reactor coolant system (Figure 5.1-5). The normal residual heat removal system (subsection 5.4.7) incorporates the relief valve for low-temperature overpressure protection. The valves which isolate the normal residual heat removal system from the reactor coolant system do not have an autoclosure interlock. Figure 5.4-6 shows a simplified sketch of the normal residual heat removal system. Figure 5.4-7 shows the piping and instrumentation drawing for the residual heat removal system.

Section 10.3 discusses the safety valves for the main steam system. Figure 10.3.2-1 shows the piping and instrumentation drawing for the main steam system.

5.2.2.4 Equipment and Component Description

Subsection 5.4.9 discusses the design and design parameters for the safety valves providing operating and low-temperature overpressure protection. The pressurizer safety valves are ASME Boiler and Pressure Vessel 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 as described in subsections 3.9.1, 3.9.2, and 3.9.3.

The relief valve included in the normal residual heat removal system provides containment boundary function since it is connected to the piping between the containment isolation valves for the system. Containment isolation requirements are discussed in subsection 6.2.3. Based on the containment boundary function, the relief valve is a ASME Code Class 2 component and is analyzed to the appropriate requirements.

In addition to the testing and analysis required for ASME Code requirements, the pressurizer safety valves are of a type which has been verified to operate during normal operation, anticipated transients, and postulated accident conditions. The verification program (Reference 2) was established by the Electric Power Research Institute to address the requirements of 10 CFR 50.34 (f)(2)(x). These requirements do not apply to relief valves of the size and type represented by the relief valve on the normal residual heat removal system.

Section 10.3, discusses the equipment and components that provide the main steam system overpressure protection.

5.2.2.5 Mounting of Pressure Relief Devices

Subsection 5.4.9 describes the design and installation of the pressure relief devices for the reactor coolant system. Section 3.9 describes the design basis for the assumed loads for the primary- and secondary-side pressure relief devices. Subsection 10.3.2, discusses the main steam safety valves and the power-operated atmospheric steam relief valves.

5.2.2.6 Applicable Codes and Classification

The requirements of the ASME Boiler and Pressure Vessel Code, Section III, Paragraphs NB-7300 (Overpressure Protection Report) and NC-7300 (Overpressure Protection Analysis), are met.

Piping, valves, and associated equipment used for overpressure protection are classified according to the classification system discussed in subsection 3.2.2. These safety-class designations are delineated in Table 3.2-3.

5.2.2.7 Material Specifications

See subsection 5.2.3 for the material specifications for the pressurizer safety valves. The piping in the pressurizer safety and relief valve module up to the safety valve is considered reactor coolant system. See subsection 5.2.3 for material specifications. The discharge piping is austenitic stainless steel. Subsection 5.4.7 specifies the materials used in the normal residual heat removal system.

5.2.2.8 Process Instrumentation

Each pressurizer safety valve discharge line incorporates a main control room temperature indicator and alarm to notify the operator of steam discharge due to either leakage or actual valve operation.

5.2.2.9 System Reliability

ASME Code safety valves and relief valves have demonstrated a high degree of reliability over many years of service. The in-service inspection and testing required of safety valves and relief valves (subsections 3.9.6 and 5.4.8 and Section 6.6) provides assurance of continued reliability and conformance to setpoints. The assessment of reliability, availability, and maintainability which is done to evaluate the estimated availability for the AP600 includes estimates for the contribution of safety valves and relief valves to unavailability. These estimates were based on experience for operating units.

5.2.2.10 Testing and Inspection

Subsections 3.9.6 and 5.4.8 and Section 6.6 discuss the preservice and in-service testing and inspection required for the safety valves and relief valves. The testing and inspection requirements are in conformance with industry standards, including Section XI of the ASME Code.

5.2.3 Reactor Coolant Pressure Boundary Materials

5.2.3.1 Materials Specifications

Table 5.2-1 lists material specifications used for the principal pressure-retaining applications in Class 1 primary components and reactor coolant system piping. Material specifications with grades, classes or types are included for the reactor vessel components, steam generator components, reactor coolant pump, pressurizer, core makeup tank, and the passive residual heat removal heat exchanger. Table 5.2-1 lists the application of nickel-chromium-iron alloys in the reactor coolant pressure boundary. The use of nickel-chromium-iron alloy in the reactor coolant pressure boundary is limited to Alloy 690. Alloy 600 may be used for cladding or buttering. Steam generator tubes use Alloy 690 in the thermally treated form. Nickel-chromium-iron alloys are used where corrosion resistance of the alloy is an important consideration and where the use of nickel-chromium-iron alloy is the choice because of the coefficient of thermal expansion.

Subsection 5.4.3 defines reactor coolant piping. See subsection 4.5.2 for material specifications used for the core support structures and reactor internals. See appropriate sections for internals of other components. Engineered safeguards features materials are included in subsection 6.1.1. The nonsafety-related portion of the chemical and volume control system inside containment in contact with reactor coolant is constructed of or clad with corrosion resistant material such as Type 304 or Type 316 stainless steel or material with equivalent corrosion resistance. The materials are compatible with the reactor coolant. The nonsafety-related portion of the chemical and volume control system is not required to conform the process to requirements outlined below.

Table 5.2-1 material specifications are the materials used in the AP600 reactor coolant pressure boundary. The materials used in the reactor coolant pressure boundary conform to the applicable ASME Code rules. Cast austenitic stainless steel does not exceed a ferrite content of 30 FN.

The welding materials used for joining the ferritic base materials of the reactor coolant pressure boundary conform to or are equivalent to ASME Material Specifications SFA 5.1, 5.2, 5.5, 5.17, 5.18, and 5.20. They are qualified to the requirements of the ASME Code, Section III.

The welding materials used for joining the austenitic stainless steel base materials of the reactor coolant pressure boundary conform to ASME Material Specifications SFA 5.4 and 5.9. They are qualified to the requirements of the ASME Code, Section III.

The welding materials used for joining nickel-chromium-iron alloy in similar base material combination and in dissimilar ferritic or austenitic base material combination conform to ASME Material Specifications SFA 5.11 and 5.14. They are qualified to the requirements of the ASME Code, Section III.

5.2.3.2 Compatibility with Reactor Coolant

5.2.3.2.1 Chemistry of Reactor Coolant

The reactor coolant system chemistry specifications conform to the recommendation of Regulatory Guide 1.44 and are shown in Table 5.2-2.

The reactor coolant system water chemistry is selected to minimize corrosion. Routinely scheduled analyses of the coolant chemical composition are performed to verify that the reactor coolant chemistry meets the specifications. Other additions, such as those to reduce activity transport and deposition, may be added to the system.

The chemical and volume control system (CVS) provides a means for adding chemicals to the reactor coolant system. The chemicals perform the following functions:

- Control the pH of the coolant during prestartup testing and subsequent operation
- Scavenge oxygen from the coolant during heatup
- Control radiolysis reactions involving hydrogen, oxygen, and nitrogen during power operations following startup

Table 5.2-2 shows the normal limits for chemical additives and reactor coolant impurities for power operation.

The pH control chemical is lithium hydroxide monohydrate, enriched in the lithium-7 isotope to 99.9 percent. This chemical is chosen for its compatibility with the materials and water chemistry of borated water/stainless steel/zirconium/nickel-chromium-iron systems. In addition, lithium-7 is produced in solution from the neutron irradiation of the dissolved boron in the coolant. The lithium-7 hydroxide is introduced into the reactor coolant system via the charging flow. The concentration of lithium-7 hydroxide in the reactor coolant system is maintained in the range specified for pH control.

During reactor startup from the cold condition, hydrazine is used as an oxygen-scavenging agent. The hydrazine solution is introduced into the reactor coolant system in the same manner as described for the pH control agent.

The reactor coolant is treated with dissolved hydrogen to control the net decomposition of water by radiolysis in the core region. The hydrogen reacts with oxygen introduced into the reactor coolant system by the radiolysis effect of radiation on molecules. Hydrogen makeup is supplied to the reactor coolant system by direct injection of high pressure gaseous hydrogen, which can be adjusted to provide the correct equilibrium hydrogen concentration. Subsection 1.9.1 indicates the degree of conformance with Regulatory Guide 1.44, "Control of the Use of Sensitized Stainless Steel."

Boron, in the chemical form of boric acid, is added to the reactor coolant system for long-term reactivity control of the core.

Suspended solid (corrosion product particulates) and other impurity concentrations are maintained below specified limits by controlling the chemical quality of makeup water and chemical additives and by purification of the reactor coolant through the chemical and volume control system.

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 corrosion-resistant cladding on surfaces exposed to the reactor coolant. The corrosion resistance of the cladding material is at least equivalent to the corrosion resistance of Types 304 and 316 austenitic stainless steel alloys or nickel-chromium-iron alloy, martensitic stainless steel, and precipitation-hardened stainless steel. These clad materials may be subjected to the ASME Code-required postweld heat treatment for ferritic base materials.

Ferritic low-alloy and carbon steel nozzles have safe ends of either stainless steel-wrought materials, stainless steel-weld metal analysis A-8, or nickel-chromium-iron alloy-weld metal F-Number 43. The latter buttering material requires further safe ending with austenitic stainless steel base material after completion of the postweld heat treatment when the nozzle is larger than a 4-inch nominal inside diameter and/or the wall thickness is greater than 0.531 inch.

Austenitic stainless steel and nickel-chromium-iron alloy base materials with primary pressure-retaining applications are used in the solution-annealed or thermally treated conditions. These heat treatments are as required by the material specifications.

During later fabrications, these materials are not heated above 800°F other than locally by welding operations. The solution-annealed surge line material is subsequently formed by hot-bending followed by a resolution-annealing heat treatment.

Components using stainless steel sensitized in the manner expected during component fabrication and installation operate satisfactorily under normal plant chemistry conditions in pressurized water reactor (PWR) systems because chlorides, fluorides, and oxygen are controlled to very low levels. Subsection 1.9.1 indicates the degree of conformance with Regulatory Guide 1.44, "Control of the Use of Sensitized Stainless Steel."

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

5.2.3.2.3 Compatibility with External Insulation and Environmental Atmosphere

In general, materials that are used in principal pressure-retaining applications and are subject to elevated temperature during system operation are in contact with thermal insulation that covers their outer surfaces.

The thermal insulation used on the reactor coolant pressure boundary is either reflective stainless steel-type, fibrous insulation enclosed in stainless steel cans, or made of compounded materials that yield low leachable chloride and/or fluoride concentrations.

The compounded materials in the form of blocks, boards, cloths, tapes, adhesives, cements, etc, are silicated to provide protection of austenitic stainless steels against stress corrosion that may result from accidental wetting of the insulation by spillage, minor leakage, or other contamination from the environmental atmosphere. Subsection 1.9.1 indicates the degree of conformance with Regulatory Guide 1.36, "Nonmetallic Thermal Insulation for Austenitic Stainless Steel."

In the event of coolant leakage, the ferritic materials will show increased general corrosion rates. Where minor leakage is considered possible based on service experience (such as valve packing, pump seals, etc), only materials compatible with the coolant are used. Table 5.2-1 shows examples. Ferritic materials exposed to coolant leakage can be readily observed as part of the inservice visual and/or nondestructive inspection program to confirm the integrity of the component for subsequent service.

5.2.3.3 Fabrication and Processing of Ferritic Materials

5.2.3.3.1 Fracture Toughness

The fracture toughness properties of the reactor coolant pressure boundary components meet the requirements of the ASME Code, Section III, Subarticle NB-2300. Those portions of the reactor coolant pressure boundary that meet the requirements of ASME Code, Section III, Class 2 per the criteria of 10 CFR 50.55a, meet the fracture toughness requirements of the ASME Code, Section III, Subarticle NC-2300. The fracture toughness properties of the reactor coolant pressure boundary components also meet the requirements of Appendix G of 10 CFR 50.

The fracture toughness properties of the reactor vessel materials are discussed in Section 5.3.

Limiting steam generator and pressurizer reference temperatures for a nil ductility transition (RT_{NDT}) temperatures are guaranteed at 10°F for the base materials and the weldments.

These materials meet the 50-foot-pound absorbed energy and 35-mils lateral expansion requirements of the ASME Code, Section III, at 70°F. The actual results of these tests are provided in the ASME material data reports which are supplied for each component and submitted to the owner at the time of shipment of the component.

Temperature instruments and Charpy impact test machines are calibrated to meet the requirements of the ASME Code, Section III, Paragraph NB-2360.

Westinghouse has conducted a test program to determine the fracture toughness of low-alloy ferritic materials with specified minimum yield strengths greater than 50,000 psi to demonstrate compliance with Appendix G of the ASME Code, Section III. In this program, fracture toughness properties were determined and shown to be adequate for base metal plates and forgings, weld metal, and heat-affected zone metal for higher-strength ferritic materials used for components of the reactor coolant pressure boundary. WCAP-9292 (Reference 3) documents the program results.

5.2.3.3.2 Control of Welding

Welding is conducted using procedures qualified according to the rules of Sections III and IX of the ASME Code. Control of welding variables (as well as examination and testing) during procedure qualification and production welding is performed according to ASME Code requirements.

The practices for storing and handling welding electrodes and fluxes comply with ASME Code, Section III, Paragraph NB-2400.

Subsection 1.9.1 indicates the degree of conformance of the ferritic materials components of the reactor coolant pressure boundary with Regulatory Guides 1.31, "Control of Ferrite Content in Stainless Steel Welds"; 1.34, "Control of Electroslag Weld Properties"; 1.43, "Control of Stainless Steel Weld Cladding of Low-Alloy Steel Components"; 1.50, "Control of Preheat Temperature for Welding of Low-Alloy Steel"; and 1.71, "Welder Qualification for Areas of Limited Accessibility."

5.2.3.4 Fabrication and Processing of Austenitic Stainless Steel

Subsections 5.2.3.4.1 through 5.2.3.4.5 address Regulatory Guide 1.44, "Control of the Use of Sensitized Stainless Steel," and present the methods and controls to avoid sensitization and to prevent intergranular attack (IGA) of austenitic stainless steel components. Also, subsection 1.9.1 indicates the degree of conformance with Regulatory Guide 1.44.

5.2.3.4.1 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, accepted methods designed to minimize contamination that could lead to stress corrosion cracking. The procedures covering these controls are stipulated in process specifications. Tools used in abrasive work operations on austenitic stainless steel, such as grinding or wire brushing, do not contain and are not contaminated with ferritic carbon steel or other materials that could contribute to intergranular cracking or stress-corrosion cracking.

These process specifications supplement the equipment specifications and purchase order requirements of every individual austenitic stainless steel component or system procured for the AP600, regardless of the ASME Code classification.

The process specifications define these requirements and follow the guidance of ASME NQA-2.

Subsection 1.9.1 indicates the degree of conformance of the austenitic stainless steel components of the reactor coolant pressure boundary with Regulatory Guide 1.37, "Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants."

5.2.3.4.2 Solution Heat Treatment Requirements

The austenitic stainless steels listed in Tables 5.2-1 are used in the final heat-treated condition required by the respective ASME Code, Section II, materials specification for the particular type or grade of alloy.

5.2.3.4.3 Material Testing Program

Austenitic stainless steel materials of product forms with simple shapes need not be corrosion-tested provided that the solution heat treatment is followed by water quenching. Simple shapes are defined as plates, sheets, bars, pipe, and tubes, as well as forgings, fittings, and other shaped products that do not have inaccessible cavities or chambers that would preclude rapid cooling when water-quenched. This characterization of cavities or chambers as inaccessible is in relation to the entry of water during quenching and is not a determination of the component accessibility for inservice inspection.

When testing is required, the tests are performed according to a process specification following the guidelines of ASTM A 262, Practice A or E.

5.2.3.4.4 Prevention of Intergranular Attack of Unstabilized Austenitic Stainless Steels

Unstabilized stainless steels can be subject to intergranular attack if the steels are sensitized, if certain species are present, such as chlorides and oxygen, and if they are exposed to a stressed condition. In the reactor coolant system, reliance is placed on the elimination or avoidance of these conditions. This is accomplished by the following:

- Control of primary water chemistry to provide a benign environment
- Use of materials in the final heat-treated condition and the prohibition of subsequent heat treatments from 800°F to 1500°F
- Control of welding processes and procedures to avoid heat-affected zone sensitization
- Confirmation that the welding procedures used for the manufacture of components in the primary pressure boundary and the reactor internals do not result in the sensitization of heat-affected zones

Further information on each of these steps is provided in the following paragraphs.

The water chemistry in the reactor coolant system is controlled to prevent the intrusion of aggressive elements. In particular, the maximum permissible oxygen and chloride concentrations are 0.005 ppm and 0.15 ppm, respectively. Table 5.2-2 lists the recommended reactor coolant water chemistry specifications.

The precautions taken to prevent the intrusion of chlorides into the system during fabrication, shipping, and storage are stipulated in the appropriate process specifications. The use of hydrogen overpressure precludes the presence of oxygen during operation.

The effectiveness of these controls has been demonstrated by both laboratory tests and operating experience. The long-term exposure of severely sensitized stainless steels to reactor coolant environments in early Westinghouse pressurized water reactors has not resulted in any sign of intergranular attack. WCAP-7477 (Reference 4) describes the laboratory experimental findings and reactor operating experience. The additional years of operations since Reference 4 was issued have provided further confirmation of the earlier conclusions that severely sensitized stainless steels do not undergo any intergranular attack in Westinghouse pressurized water reactor coolant environments.

Although there is no evidence that pressurized water reactor coolant water attacks sensitized stainless steels, it is good metallurgical practice to avoid the use of sensitized stainless steels in the reactor coolant system components.

Accordingly, measures are taken to prohibit the use of sensitized stainless steels and to prevent sensitization during component fabrication. The wrought austenitic stainless steel stock used in the reactor coolant pressure boundary is used in one of the following conditions:

- Solution-annealed and water-quenched

- Solution-annealed and cooled through the sensitization temperature range within less than about 5 minutes

Westinghouse has verified that these practices will prevent sensitization by performing corrosion tests on wrought material as it was received.

The heat-affected zones of welded components must, of necessity, be heated into the sensitization temperature range (800°F to 1500°F). However, severe sensitization (that is, continuous grain boundary precipitates of chromium carbide, with adjacent chromium depletion) can be avoided by controlling welding parameters and welding processes. The heat input and associated cooling rate through the carbide precipitation range are of primary importance. Westinghouse has demonstrated this by corrosion-testing a number of weldments.

The heat input in austenitic pressure boundary weldments is controlled by the following:

- Limiting the maximum interpass temperature to 350°F
- Exercising approval rights on welding procedures
- Requiring qualification of processes

5.2.3.4.5 Retesting Unstabilized Austenitic Stainless Steels Exposed to Sensitization Temperatures

If during the course of fabrication, steel is inadvertently exposed to the sensitization temperature range, the material may be tested according to a process specification, following the guidelines of ASTM A 262, to verify that it is not susceptible to intergranular attack. Testing is not required for the following:

- Cast metal or weld metal with a ferrite content of 5 percent or more
- Material with a carbon content of 0.03 percent or less
- Material exposed to special processing, provided the following:
  - Processing is properly controlled to develop a uniform product
  - Adequate documentation exists of service experience and/or test data to demonstrate that the processing will not result in increased susceptibility to intergranular attack

If such material is not verified to be not susceptible to intergranular attack, the material is resolution- annealed and water-quenched or rejected.

5.2.3.4.6 Control of Welding

The following paragraphs address Regulatory Guide 1.31, "Control of Ferrite Content in Stainless Steel Weld Metal." They present the methods used, and the verification of these methods, for austenitic stainless steel welding.

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

Also, it has been well documented that delta ferrite is one of the mechanisms for reducing the susceptibility of stainless steel welds to hot cracking. The minimum delta ferrite level below which the material will be prone to hot cracking lies between 0 and 3 percent delta ferrite.

The following paragraphs discuss welding processes used to join stainless steel parts in components designed, fabricated, or stamped according to the ASME Code, Section III, Classes 1 and 2, and core support components. Delta ferrite control is appropriate for the preceding welding requirements, except where no filler metal is used or where such control is not applicable, such as the following: electron beam welding; autogenous gas shielded tungsten arc welding; explosive welding; welding using fully austenitic welding materials

The fabrication and installation specifications require welding procedures and welder qualification according to Section III of the ASME Code. They also include the delta ferrite determinations for the austenitic stainless steel welding materials used for welding qualification testing and for production processing.

Specifically, the undiluted weld deposits of the "starting" welding materials must contain at least 5 percent delta ferrite. (The equivalent ferrite number may be substituted for percent delta ferrite.) This is determined by chemical analysis and calculation using the appropriate weld metal constitution diagrams in Section III of the ASME Code or magnetic measurement by calibrated instruments.

When new welding procedure qualification tests are evaluated for these applications, including repair welding of raw materials, they are performed according to the requirements of Sections III and IX of the ASME Code.

The results of the destructive and nondestructive tests are recorded in the procedure qualification record, in addition to the information required by Section III of the ASME Code.

The welding materials used for fabrication and installation welds of austenitic stainless steel materials and components meet the requirements of Section III of the ASME Code. For applications using austenitic stainless steel welding material, the material conforms to ASME weld metal analysis A-8, Type 308, 308L, 316, or 316L.

Bare weld filler metal, including consumable inserts, used in inert gas welding processes conforms to ASME SFA 5.9. The metal is procured to contain not less than 5 FN or more than 13 FN delta ferrite according to Section III of the ASME Code. Weld filler metal materials used in flux-shielded welding processes conform to ASME SFA 5.4 or 5.9. They are procured in a wire-flux combination to be capable of providing not less than 5 FN or more than 13 FN delta ferrite in the deposit, according to Section III of the ASME Code.

Welding materials are tested using the welding energy inputs employed in production welding.

Combinations of approved heats and lots of welding materials are used for welding processes. The welding quality assurance program includes identification and control of welding material by lots and heats as appropriate. Weld processing is monitored according to approved inspection programs that include review of materials, qualification records, and welding parameters. Welding systems are also subject to the following:

- Quality assurance audit, including calibration of gauges and instruments
- Identification of welding materials
- Welder and procedure qualifications
- Availability and use of approved welding and heat-treating procedures

- Documentary evidence of compliance with materials, welding parameters, and inspection requirements

Fabrication and installation welds are inspected using nondestructive examination methods according to Section III of the ASME Code rules.

To verify the reliability of these controls, Westinghouse has completed a delta ferrite verification program, described in WCAP-8324-A (Reference 5). This program has been approved as a valid approach to verify the Westinghouse hypothesis and is considered an acceptable alternative for conformance with the NRC Interim Position on Regulatory Guide 1.31. The regulatory staff's acceptance letter and topical report evaluation were received on December 30, 1974. The program results, which support the hypothesis presented in WCAP-8324-A (Reference 5), are summarized in WCAP-8693 (Reference 6).

Subsection 1.9.1 indicates the degree of conformance of the austenitic stainless steel components of the reactor coolant pressure boundary with Regulatory Guides 1.34, "Control of Electroslag Weld Properties," and 1.71, "Welder Qualification for Areas of Limited Accessibility."

5.2.3.4.7 Control of Cold Work in Austenitic Stainless Steels

The use of cold worked austenitic stainless steels is limited to small parts including pins and fasteners where proven alternatives are not available and where cold worked material has been used successfully in similar applications. Cold work control of austenitic stainless steels in pressure boundary applications is provided by limiting the hardness of austenitic stainless steel raw material and controlling the hardness during fabrication by process control of bending, cold forming, straightening or other similar operation. Grinding of material in contact with reactor coolant is controlled by procedures. Ground surfaces are finished with successively finer grit sizes to remove the bulk of cold worked material.

5.2.3.5 Threaded Fastener Lubricants

The lubricants to be used on threaded fasteners which maintain pressure boundary integrity in the reactor coolant and related systems and in the steam, feed, and condensate systems; threaded fasteners used inside those systems; and threaded fasteners used in component structural support for those systems are specified in the design specification. Field selection of thread lubricants is not permitted. The thread lubricants are selected based on experience and test data which show them to be effective, but not to cause or accelerate corrosion of the fastener. Where leak sealants are used on threaded fasteners or can be in contact with the fastener in service, their selection is based on satisfactory experience or test data. Selection considers possible adverse interaction between sealants and lubricants. Lubricants containing molybdenum sulphide are prohibited.

5.2.4 Inservice Inspection and Testing of Class 1 Components

Preservice and inservice inspection and testing of ASME Code Class 1 pressure-retaining components (including vessels, piping, pumps, valves, bolting, and supports) within the reactor coolant pressure boundary are performed in accordance with Section XI of the ASME Code including addenda according to 10 CFR 50.55a(g). This includes all ASME Code Section XI mandatory appendices.

The specific edition and addenda of the Code used to determine the requirements for the inspection and testing plan for the initial and subsequent inspection intervals is to be delineated in the inspection program. The Code includes requirements for system pressure tests and functional tests for active components. The requirements for system pressure tests are defined in Section XI, IWA-5000. These tests verify the pressure boundary integrity in conjunction with inservice inspection. Section 6.6 discusses Classes 2 and 3 component examinations.

Subsection 3.9.6 discusses the in-service functional testing of valves for operational readiness. Since none of the pumps in the AP600 are required to perform an active safety function, the operational readiness test program for pumps is controlled administratively.

In conformance with ASME Code and NRC requirements, the preparation of inspection and testing programs is the responsibility of the combined license applicant of each AP600. A preservice inspection program (nondestructive examination) and a preservice test program for valves for the AP600 will be developed and submitted to the NRC. The in-service inspection program and in-service test program will be submitted to the NRC by the combined license applicant. These programs will comply with applicable in-service inspection provisions of 10 CFR 50.55a(2).

The preservice programs provide details of areas subject to examination, as well as the method and extent of preservice examinations. In-service programs detail the areas subject to examination and the method, extent, and frequency of examinations. Additionally, component supports and snubber testing requirements are included in the inspection programs.

5.2.4.1 System Boundary Subject to Inspection

ASME Code Class 1 components (including vessels, piping, pumps, valves, bolting, and supports) are designated AP600 equipment Class A (see subsection 3.2.2). 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 programs.

5.2.4.2 Arrangement and Inspectability

ASME Code Class 1 components are designed so that access is provided in the installed condition for visual, surface and volumetric examinations specified by the baseline ASME

Code Section XI (1989 Edition, 1989 Addenda) and mandatory appendices. See subsection 5.2.1.1 for a discussion of the baseline ASME Code Edition and Addenda. Design provisions, in accordance with Section XI, Article IWA-1500, are incorporated in the design processes for Class 1 components.

The AP600 design activity includes a design for inspectability program. The goal of this program is to provide for the inspectability access and conformance of component design with available inspection equipment and techniques. Factors such as examination requirements, examination techniques, accessibility, component geometry and material selection are used in evaluating component designs. Examination requirements and examination techniques are defined by inservice inspection personnel. Inservice inspection review as part of the design process provides component designs that conform to inspection requirements and establishes recommendations for enhanced inspections.

Considerable experience is utilized in designing, locating, and supporting pressure-retaining components to permit preservice and in-service inspection required by Section XI of the ASME Code. Factors such as examination requirements, examination techniques, accessibility, component geometry, and material selections aid in establishing the designs. The inspection design goals are to eliminate uninspectable components, reduce occupational radiation exposure, reduce inspections times, allow state-of-the-art inspection system, and enhance flaw detection and the reliability of flaw characterization.

As one example of component geometry that reduces inspection requirements, the reactor pressure vessel has no longitudinal welds requiring in-service inspection. No Quality Group A (ASME Code Class 1) components require in-service inspection during reactor operation.

Removable insulation and shielding are provided on those piping systems requiring volumetric and surface examination. Removable hangers and pipe whip restraints are provided as necessary and practical to facilitate inservice inspection. Working platforms are provided in areas requiring inspection and servicing of pumps and valves. Permanent or temporary working platforms, scaffolding, and ladders facilitate access to piping and component welds. The components and welds requiring in-service inspection allow for the application of the required in-service inspection methods. Such design features include sufficient clearances for personnel and equipment, maximized examination surface distances, two-sided access, favorable materials, weld-joint simplicity, elimination of geometrical interferences, and proper weld surface preparation.

Some of the ASME Class 1 components are included in modules fabricated offsite and shipped to the site. (See subsection 3.9.1.5.) The modules are designed and engineered to provide access for in-service inspection and maintenance activities. The attention to detail engineered into the modules before construction provides the accessibility for inspection and maintenance. Relief from Section XI requirements should not be required for Class 1 pressure retaining components in the AP600. Future unanticipated changes in the ASME Code, Section XI requirements could, however, necessitate relief requests. Relief from the inspection requirements of ASME Code, Section XI will be requested when full compliance is not practical according to the requirements of 10 CFR 50.55a(g)(5)(iv). In such cases, specific information will be provided which identifies the applicable Code requirements, justification for the relief request, and the inspection method to be used as an alternative.

Space is provided to handle and store insulation, structural members, shielding, and other materials related to the inspection. Suitable hoists and other handling equipment, lighting, and sources of power for inspection equipment are installed. The integrated head package provides for access to inspect the reactor vessel head and the weld of the control rod drive mechanisms to the reactor vessel head. Closure studs, nuts, and washers are removed to a dry location for direct inspection.

5.2.4.3 Examination Techniques and Procedures

The visual, surface, and volumetric examination techniques and procedures agree with the requirements of Subarticle IWA-2200 and Table IWB-2500-1 of the ASME Code, Section XI. Qualification of the ultrasonic inspection equipment, personnel and procedures is in compliance with Appendix VII of the ASME Code, Section XI. The liquid penetrant method or the magnetic particle method is used for surface examinations. Radiography, ultrasonic, or eddy current techniques (manual or remote) are used for volumetric examinations.

The reactor vessel is designed so that the reactor pressure vessel (RPV) inspections can be performed primarily from the vessel internal surfaces. These inspections can be done remotely using existing inspection tool designs to minimize occupational radiation exposure and to facilitate the inspections. Access is also available for the application of inspection techniques from the outside of the complete reactor pressure vessel. Reactor pressure vessel welds are examined to meet the requirements of Regulatory Guide 1.150 as defined in subsection 1.9.1.

5.2.4.4 Inspection Intervals

Inspection intervals are established as defined in Subarticles IWA-2400 and IWB-2400 of The ASME Code, Section XI. The interval may be extended by as much as one year so that inspections are concurrent with plant outages. It is intended that in-service examinations be performed during normal plant outages such as refueling shutdowns or maintenance shutdowns occurring during the inspection interval.

5.2.4.5 Examination Categories and Requirements

The examination categories and requirements are established according to Subarticle IWB-2500 and Table IWB-2500-1 of the ASME Code, Section XI. The preservice examinations comply with IWB-2200.

5.2.4.6 Evaluation of Examination Results

Examination results are evaluated according to IWA-3000 and IWB-3000, with flaw indications according to IWB-3400 and Table IWB-3410-1. Repair procedures, if required, are according to IWB-4000 of the ASME Code, Section XI.

5.2.4.7 System Leakage and Hydrostatic Pressure Tests

System pressure tests comply with IWA-5000 and IWB-5000 of the ASME Code, Section XI. These system pressure tests are included in the design transients defined in Subsection 3.9.1. This subsection discusses the transients included in the evaluation of fatigue of Class 1 components due to cyclic loads.

5.2.5 Detection of Leakage Through Reactor Coolant Pressure Boundary

The reactor coolant pressure boundary (RCPB) leakage detection monitoring provides a means of detecting and to the extent practical, identifying the source and quantifying the reactor coolant leakage. The detection monitors perform the detection and monitoring function in conformance with the requirements of General Design Criteria 2 and 30 and the recommendations of Regulatory Guide 1.45. Leakage detection monitoring is also maintained in support of the use of leak-before-break criteria for high-energy pipe in containment. See subsection 3.6.3 for the application of leak-before-break criteria.

Leakage detection monitoring is accomplished using instrumentation and other components of several systems. Diverse measurement methods including level, flow, and radioactivity measurements are used for leak detection. The equipment classification for each of the systems and components used for leak detection is generally determined by the requirements and functions of the system in which it is located. There is no requirement that leak detection and monitoring components be safety-related. See Figure 5.2-1 for the leak detection approach. The descriptions of the instrumentation and components used for leak detection and monitoring include information on the system.

To satisfy position 1 of Regulatory Guide 1.45, reactor coolant pressure boundary leakage is classified as either identified or unidentified leakage. Identified leakage includes:

- Leakage from closed systems such as reactor vessel seal or valve leaks that are captured and conducted to a collecting tank
- Leakage into auxiliary systems and secondary systems (intersystem leakage) (This leakage is not considered to be part of the 10 gpm limit identified leakage in the bases of the technical specification 3.4.8. This additional leakage must be considered in the evaluation of the reactor coolant inventory balance.)

Other leakage is unidentified leakage.

5.2.5.1 Collection and Monitoring of Identified Leakage

Identified leakage other than intersystem leakage is collected in the reactor coolant drain tank. The reactor coolant drain tank is a closed tank located in the reactor cavity in the containment. The tank vent is piped to the gaseous radwaste system to prevent release of radioactive gas to the containment atmosphere. For positions 1 and 7 of Regulatory Guide 1.45, the liquid level in the reactor coolant drain tank and total flow pumped out of the reactor coolant drain tank are used to calculate the identified leakage rate. The identified leakage rate is automatically calculated by the plant computer. A leak as small as 0.1 gpm can be detected in one hour. The design leak of 10 gpm will be detected in less than a minute. These parameters are available in the main control room. The reactor coolant drain tank, pumps, and sensors are part of the liquid radwaste system. The following sections outline the various sources of identified leakage other than intersystem leakage.

5.2.5.1.1 Valve Stem Leakoff Collection

Valve stem leakoff connections are not provided in the AP600.

5.2.5.1.2 Reactor Head Seal

The reactor vessel flange and head flange are sealed by two concentric seals. Seal leakage is detected by two leak-off connections: one between the inner and outer seal, and one outside the outer seal. These lines are combined in a header before being routed to the reactor coolant drain tank. An isolation valve is installed in the common line. During normal plant operation, the leak-off valves are aligned so that leakage across the inner seal drains to the reactor coolant drain tank.

A surface-mounted resistance temperature detector installed on the bottom of the common reactor vessel seal leak pipe provides an indication and high temperature alarm signal in the main control room indicating the possibility of a reactor pressure vessel head seal leak. The temperature detector and drain line downstream of the isolation valve are part of the liquid radwaste system.

The reactor coolant pump closure flange is sealed with a welded canopy seal and does not require leak-off collection provisions.

Leakage from other flanges is discussed in subsection 5.2.5.3, Collection and Monitoring of Unidentified Leakage.

5.2.5.1.3 Pressurizer Safety Relief and Automatic Depressurization Valves

Temperature is sensed downstream of each pressurizer safety relief valve and each automatic depressurization valve mounted on the pressurizer by a resistance temperature detector on the discharge piping just downstream of each valve. High temperature indications (alarms in the main control room) identify a reduction of coolant inventory as a result of seat leakage through one of the valves. These detectors are part of the reactor coolant system. This leakage is drained to the reactor coolant drain tank during normal plant operation and vented to containment atmosphere or the in-containment refueling water storage tank during accident conditions. This identified leakage is measured by the change in level of the reactor coolant drain tank.

5.2.5.1.4 Other Leakage Sources

In the course of plant operation, various minor leaks of the reactor coolant pressure boundary may be detected by operating personnel. If these leaks can be subsequently observed, quantified, and routed to the containment sump, this leakage will be considered identified leakage.

5.2.5.2 Intersystem Leakage Detection

Substantial intersystem leakage from the reactor coolant pressure boundary to other systems is not expected. However, possible leakage points across passive barriers or valves and their detection methods are considered. In accordance with position 4 of Regulatory Guide 1.45, auxiliary systems connected to the reactor coolant pressure boundary incorporate design and administrative provisions that limit leakage. Leakage is detected by increasing auxiliary system level, temperature, flow, or pressure, by lifting the relief valves or increasing the values of monitored radiation in the auxiliary system.

The normal residual heat removal system and the chemical and volume control system, which are connected to the reactor coolant system, have potential for leakage past closed valves. For additional information on the control of reactor coolant leakage into these systems, see subsections 5.4.7 and 9.3.6 and the intersystem LOCA discussion in subsection 1.9.5.1.

5.2.5.2.1 Steam Generator Tubes

An important potential identified leakage path for reactor coolant is through the steam generator tubes into the secondary side of the steam generator. Identified leakage from the steam generator primary side is detected by one, or a combination, of the following:

- High condenser air removal discharge radioactivity, as monitored and alarmed by the turbine island vent discharge radiation monitor

- Steam generator secondary side radioactivity, as monitored and alarmed by the steam generator blowdown radiation monitor

- Secondary side radioactivity, as monitored and alarmed by the main steam line radiation monitors

- Radioactivity, boric acid, or conductivity in condensate as indicated by laboratory analysis

Details on the radiation monitors are provided in Section 11.5, Radiation Monitoring.

5.2.5.2.2 Component Cooling Water System

Leakage from the reactor coolant system to the component cooling water system is detected by the component cooling water system radiation monitor, by increasing surge tank level, by high flow downstream of selected components, or by some combination of the preceding. Refer to Section 11.5, Radiation Monitoring, and subsection 9.2.2, Component Cooling Water System.

5.2.5.2.3 Passive Residual Heat Removal Heat Exchanger Tubes

A potential identified leakage path for reactor coolant is through the passive residual heat removal heat exchanger into the in-containment refueling water storage tank. Identified leakage from the passive residual heat removal heat exchanger tubes is detected as follows:

- High temperature in the passive residual heat removal heat exchanger, as monitored and alarmed by temperature detectors in the heat exchanger inlet and outlet piping, alerts the operators to potential leakage. The location of these instruments is selected to provide early indication of leakage considering the potential for thermal stratification. The alarm setpoint is selected to provide early indication of leakage.

- The operator then closes the passive residual heat removal heat exchanger inlet isolation valve and observes the pressure indication inside the passive residual heat removal heat exchanger. If pressure remains at reactor coolant system pressure, then tube leakage is not present, and the high passive residual heat removal heat exchanger temperature is indicative of leakage through the outlet isolation valves.

- If the operator observes a reduction in pressure, then passive residual heat removal heat exchanger tube leakage is present. The operator then observes the change in the reactor coolant system inventory balance when the passive residual heat removal heat exchanger inlet isolation valve is closed. The difference in the reactor coolant system leakage when the isolation valve is closed identifies the passive residual heat removal heat exchanger tube leakage rate.

5.2.5.3 Collection and Monitoring of Unidentified Leakage

Position 3 of Regulator Guide 1.45 identifies three diverse methods of detecting unidentified leakage. AP600 use two of these three and adds a third method. To detect unidentified leakage inside containment, the following diverse methods may be utilized to quantify and assist in locating the leakage:

- Containment Sump Level
- Reactor Coolant System Inventory Balance
- Containment Atmosphere Radiation

Other methods that can be employed to supplement the above methods include:

- Containment Atmosphere Pressure, Temperature, and Humidity
- Visual Inspection

The reactor coolant system is an all-welded system, except for the connections on the pressurizer safety valves, reactor vessel head, pressurizer and steam generator manways, and reactor vessel head vent, which are flanged. During normal operation, variations in airborne radioactivity, containment pressure, temperature, or specific humidity above the normal level signify a possible increase in unidentified leakage rates and alert the plant operators that corrective action may be required. Similarly, increases in containment sump level signify an increase in unidentified leakage. The following sections outline the methods used to collect and monitor unidentified leakage.

5.2.5.3.1 Containment Sump Level Monitor

In conformance with position 2 of Regulatory Guide 1.45, leakage from the reactor coolant pressure boundary and other components not otherwise identified inside the containment will condense and flow by gravity via the floor drains and other drains to the containment sump.

A leak in the primary system would result in reactor coolant flowing into the containment sump. Leakage is indicated by an increase in the sump level. The containment sump level is monitored by two seismic Category I level sensors, in accordance with position 6 of Regulatory Guide 1.45. The level sensors are powered from a safety-related Class 1E electrical source. These sensors remain functional when subjected to a safe shutdown earthquake in conformance with the guidance in Regulatory Guide 1.45. The containment sump level and sump total flow sensors located on the discharge of the sump pump are part of the liquid radwaste system.

Failure of one of the level sensors will still allow the calculation of a 0.5 gpm in-leakage rate within 1 hour. The data display and processing system (DDS) computes the leakage rate and the plant control system (PLS) provides an alarm in the main control room if the average change in leak rate for any given measurement period exceeds 0.5 gpm for unidentified leakage. The minimum detectable leak is 0.03 gpm. Unidentified leakage is the total leakage minus the identified leakage. The leakage rate algorithm subtracts the identified leakage directed to the sump.

To satisy positions 2 and 5 of Regulatory Guide 1.45, the measurement interval must be long enough to permit the measurement loop to adequately detect the increase in level that would correspond to 0.5 gpm leak rate, and yet short enough to ensure that such a leak rate is detected within an hour. The measurement interval is less than or equal to 1 hour.

When the sump level increases to the high level setpoint, one of the sump pumps automatically starts to pump the accumulated liquid to the waste holdup tanks in the liquid radwaste system. The sump discharge flow is integrated and available for display in the control room, in accordance with position 7 of Regulatory Guide 1.45.

Procedures to identify the leakage source upon a change in the unidentified leakage rate into the sump include the following:

- Check for changes in containment atmosphere radiation monitor indications,
- Check for changes in containment humidity, pressure, and temperature,
- Check makeup rate to the reactor coolant system for abnormal increases,
- Check for changes in water levels and other parameters in systems which could leak water into the containment, and
- Review records for maintenance operations which may have discharged water into the containment.

5.2.5.3.2 Reactor Coolant System Inventory Balance

Reactor coolant system inventory monitoring provides an indication of system leakage. Net level change in the pressurizer is indicative of system leakage. Monitoring net makeup from the chemical and volume control system and net collected leakage provides an important method of obtaining information to establish a water inventory balance. An abnormal increase in makeup water requirements or a significant change in the water inventory balance can indicate increased system leakage.

The reactor coolant system inventory balance is a quantitative inventory or mass balance calculation. This approach allows determination of both the type and magnitude of leakage. Steady-state operation is required to perform a proper inventory balance calculation. Steady-state is defined as stable reactor coolant system pressure, temperature, power level, pressurizer level, and reactor coolant drain tank and in-containment refueling water storage tank levels. The reactor coolant inventory balance is done on a periodic basis and when other indication and detection methods indicate a change in the leak rate. The minimum detectable leak is 0.13 gpm.

The mass balance involves isolating the reactor coolant system to the extent possible and observing the change in inventory which occurs over a known time period. This involves isolating the systems connected to the reactor coolant system. System inventory is determined by observing the level in the pressurizer. Compensation is provided for changes in plant conditions which affect water density. The change in the inventory determines the total reactor coolant system leak rate. Identified leakages are monitored (using the reactor coolant drain tank) to calculate a leakage rate and by monitoring the intersystem leakage. The unidentified leakage rate is then calculated by subtracting the identified leakage rate from the total reactor coolant system leakage rate.

Since the pressurizer inventory is controlled during normal plant operation through the level control system, the level in the pressurizer will be reasonably constant even if leakage exists. The mass contained in the pressurizer may fluctuate sufficiently, however, to have a significant effect on the calculated leak rate. The pressurizer mass calculation includes both the steam and water mass contributions.

Changes in the reactor coolant system mass inventory are a result of changes in liquid density. Liquid density is a strong function of temperature and a lesser function of pressure. A range of temperatures exists throughout the reactor coolant system all of which may vary over time. A simplified, but acceptably accurate, model for determining mass changes is to assume all of the reactor coolant system is at $T_{Average}$.

The inventory balance calculation is done by the data display and processing system with additional input from sensors in the protection and safety monitoring system, chemical and volume control system, and liquid radwaste system. The use of components and sensors in systems required for plant operation provides conformance with the regulatory guidance of position 6 in Regulatory Guide 1.45 that leak detection should be provided following seismic events that do not require plant shutdown.

5.2.5.3.3 Containment Atmosphere Radioactivity Monitor

Leakage from the reactor coolant pressure boundary will result in an increase in the radioactivity levels inside containment. The containment atmosphere is continuously monitored for airborne gaseous radioactivity. Air flow through the monitor is provided by the suction created by a vacuum pump. Gaseous and N_{13}/F_{18} concentration monitors indicate radiation concentrations in the containment atmosphere.

N_{13} and F_{18} are neutron activation products which are proportional to power levels. An increase in activity inside containment would therefore indicate a leakage from the reactor coolant pressure boundary. Based on the concentration of N_{13}/F_{18} and the power level, reactor coolant pressure boundary leakage can be estimated.

The N_{13}/F_{18} monitoring system will detect a 0.5 gpm leak when the reactor is operating at a power range higher than 20 percent. The N_{13}/F_{18} monitor is seismic Category I. Conformance with the position 6 guidance of Regulatory Guide 1.45 that leak detection should be provided following seismic events that do not require plant shutdown is provided by the seismic Category 1 classification. Safety-related Class 1E power is not required since loss of power to the radiation monitor is not consistent with continuing operation following an earthquake.

At full power, the minimum detectable leak is 0.082 gpm when the radionuclide concentration in containment reaches equilibrium. The N_{13}/F_{18} monitor can detect a 0.5 gpm leak when the plant is above 20 percent power and the concentration of radiogas in-containment is at equilibrium.

Radioactivity concentration indication and alarms for loss of sample flow, high radiation, and loss of indication are provided. Sample collection connections permit sample collection for laboratory analysis. The radiation monitor can be calibrated during power operation.

5.2.5.3.4 Containment Pressure, Temperature and Humidity Monitors

Reactor coolant pressure boundary leakage increases containment pressure, temperature, and humidity, values available to the operator through the plant control system.

An increase in containment pressure is an indication of increased leakage or a high energy line break. Containment pressure is monitored by redundant Class 1E pressure transmitters. For additional discussion see subsection 6.2.2, Passive Containment Cooling System.

The containment average temperature is monitored using temperature instrumentation at the inlet to the containment fan cooler as an indication of increased leakage or a high energy line break. This instrumentation as well as temperature instruments within specific areas including steam generator areas, pressurizer area, and containment compartments are part of the containment recirculation cooling system.

An increase in the containment average temperature combined with an increase in containment pressure indicate increased leakage or a high energy line break. The individual compartment area temperatures can assist in identifying the location of the leak.

Containment humidity is monitored using temperature-compensated humidity detectors which determine the water-vapor content of the containment atmosphere. An increase in the containment atmosphere humidity indicates release of water vapor within the containment. The containment humidity monitors are part of the containment leak rate test system.

The humidity monitors supplement the containment sump level monitors and are most sensitive under conditions when there is no condensation. A rapid increase of humidity over the ambient value by more than 10 percent is indication of a probable leak.

Containment pressure, temperature and humidity can assist in identifying and locating a leak. They are not relied on to quantify a leak.

5.2.5.4 Safety Evaluation

Leak detection monitoring has no safety-related function. Therefore, the single failure criterion does not apply and there is no requirement for a nuclear safety evaluation. The containment sump level monitors and the containment atmosphere monitor are seismic Category I. The components used to calculate reactor coolant system inventory balance are both safety-related and nonsafety-related components. The containment sump level monitors are powered from the Class 1E dc and UPS system (IDS). Measurement signals are processed by the data display and processing system and the plant control system (PLS).

5.2.5.5 Tests and Inspections

To satisfy position 8 of Regulatory Guide 1.45, periodic testing of leakage detection monitors verifies the operability and sensitivity of detector equipment. These tests include installation calibrations and alignments, periodic channel calibrations, functional tests, and channel checks in conformance with regulatory guidance.

5.2.5.6 Instrumentation Applications

The parameters tabulated below satisfy position 7 of Regulatory Guide 1.45 and are provided in the main control room to allow operating personnel to monitor for indications of reactor coolant pressure boundary leakage. The containment sump level, containment atmosphere radioactivity, reactor coolant system inventory balance, and the flow measurements are provided as gallon per minute leakage equivalent.

Parameter	System(s)	Alarm or Indication
Containment sump level and sump total flow	WLS	Both
Reactor coolant drain tank level and drain tank total flow	WLS	Both
Containment atmosphere radioactivity	PSS	Both
Reactor coolant system inventory balance parameters	PCS, PXS, RCS, VCS, WLS	Both
Containment humidity	VUS	Indication
Containment atmospheric pressure	PCS	Both
Containment atmosphere temperature	VCS	Both
Reactor vessel head seal leak temperature	WLS	Both
Pressurizer safety relief valve leakage temperature	RCS	Both
Steam generator blowdown radiation	BDS	Both
Turbine island vent discharge radiation	TDS	Both
Component cooling water radiation	CCS	Both
Main steam line radiation	SGS	Both
Component cooling water surge tank level	CCS	Both

5.2.5.7 Technical Specification

Limits which satisfy position 9 of Regulatory Guide 1.45 for identified and unidentified reactor coolant leakage are identified in the technical specifications, Chapter 16. LCO 3.4.8 addresses leakage limits. LCO 3.4.10 addresses leak detection instrument requirements.

5.2.6 Combined License Information Items

5.2.6.1 ASME Code and Addenda

The Combined License applicant will address in its application the portions of later ASME Code editions and addenda to be used to construct components that will require NRC staff review and approval. The Combined License applicant will address consistency of the design with the construction practices (including inspection and examination methods) of the later ASME Code edition and addenda added as part of the Combined License application. The Combined License applicant will address the addition of ASME code cases approved subsequent to design certification.

5.2.6.2 Plant Specific Inspection Program

The Combined License applicant will provide a plant-specific preservice inspection and inservice inspection program. The program will address reference to the edition and addenda of the ASME Code Section XI used for selecting components subject to examination, a description of the components exempt from examination by the applicable code, and drawings or other descriptive information used for the examination.

5.2.7 References

1. Burnett, T. W. T., et al., "LOFTRAN Code Description," WCAP-7907-P-A (Proprietary), and WCAP-7907-A (Nonproprietary), April 1984.

2. EPRI PWR Safety and Relief Valve Test Program, Safety and Relief Valve Test Report, Interim Report, April, 1982.

3. Logsdon, W. A., Begley, J. A., and Gottshall, C. L., "Dynamic Fracture Toughness of ASME SA-508 Class 2a and ASME SA-533 Grade A Class 2 Base and Heat-Affected Zone Material and Applicable Weld Metals," WCAP-9292, March 1978.

4. Golik, M. A., "Sensitized Stainless Steel in Westinghouse PWR Nuclear Steam Supply Systems," WCAP-7477-L (Proprietary), March 1970, and WCAP-7735 (Nonproprietary), August 1971.

5. Enrietto, J. F., "Control of Delta Ferrite in Austenitic Stainless Steel Weldments," WCAP-8324-A, June 1975.

6. Enrietto, J. F., "Delta Ferrite in Production Austenitic Stainless Steel Weldments," WCAP-8693, January 1976.

7. Carlin, E. L., et al., "AP600 Shutdown Evaluation Report," WCAP-14837, Revision 3, March 1998.

Table 5.2-1 (Sheet 1 of 4)

REACTOR COOLANT PRESSURE BOUNDARY MATERIALS SPECIFICATIONS

Component	Material	Class, Grade, or Type
Reactor Vessel Components		
Head plates (other than core region)	SA-533 or SA-508	GR B, CL 1 or CL 3
Shell courses	SA-508	CL 3
Shell, flange, and nozzle forgings	SA-508	CL 3
Nozzle safe ends	SA-182	F316LN
Appurtenances to the control rod drive mechanism (CRDM)	SB-167 or SA-182	TP690 or F304LN, F316LN
Instrumentation tube appurtenances, upper head	SB-167 or SA-182, SA312, SA376	TP690 or F304LN, F316LN
Closure studs	SA-540	GR B23 or GR B24, CL 3
Monitor tubes and vent pipe	SA-312 or SA-376 or SB-166, SB-167	TP304LN, TP316LN or TP690
Cladding and buttering	SFA 5.4, 5.9, 5.11, and 5.14	308L, 309L, ENiCrFe-7, ERNiCrFe-7, or ERNiCr-3

Table 5.2-1 (Sheet 2 of 4)

REACTOR COOLANT PRESSURE BOUNDARY MATERIALS SPECIFICATIONS

Component	Material	Class, Grade, or Type
Steam Generator Components		
Pressure plates	SA-533	GR B, CL 1
Pressure forgings (including nozzles and tube sheet)	SA-508	CL 3a
Nozzle safe ends	SA-182	F316LN
Channel heads	SA-508	CL 3a
Tubes	SB-163	TP690TT
Cladding and buttering	SFA 5.4, 5.9, 5.11, and 5.14	308L, 309L, ENiCrFe-7, ERNiCrFe-7, or ERNiCr-3
Manway studs/nuts	SA-193, SA-194	GR B7
Pressurizer Components		
Pressure plates	SA-533	GR B, CL 1
Pressure forgings	SA-508	CL 3
Nozzle safe ends	SA-182	F316LN
Cladding and buttering	SFA 5.4, 5.9, 5.11, and 5.14	308L, 309L, ENiCrFe-7, ERNiCrFe-7, or ERNiCr-3
Manway studs/nuts	SA-193, SA-194	GR B7

Table 5.2-1 (Sheet 3 of 4)

REACTOR COOLANT PRESSURE BOUNDARY MATERIALS SPECIFICATIONS

Component	Material	Class, Grade, or Type
Reactor Coolant Pump		
Pressure forgings	SA-182 or SA-336	F304LN, F316LN
Pressure casting	SA-351 or SA-352	CF3A
Tube and pipe	SA-213; SA-376 or SA-312	TP304LN, TP316LN
Pressure plates	SA-240	304LN, 316LN
Closure bolting	SA-193 or SA-540	GR B7 or GR B24,CL 4
Reactor Coolant Piping		
Reactor coolant pipe	SA-376	TP304LN, TP316LN
Reactor coolant fittings, branch nozzles	SA-376, SA-182	TP304LN, TP316LN
Surge line	SA-376	TP304LN, TP316LN
RCP piping other than loop and surge line	SA-312 and SA-376	TP304LN, TP316LN
CRDM		
Latch housing	SA-336	F304LN, F316LN
Rod travel housing	SA-336	F304LN, F316LN
Welding materials	SFA 5.4 or 5.9	308L, 309L,

Table 5.2-1 (Sheet 4 of 4)

REACTOR COOLANT PRESSURE BOUNDARY MATERIALS SPECIFICATIONS

Component	Material	Class, Grade, or Type
Valves		
Bodies	SA-182 or SA-351	F304LN, F316LN or CF3A
Bonnets	SA-182,SA-240 or SA-351	F304LN, F316LN, 304LN, 316LN or CF3A
Discs	SA-182, SA-564 or SA-351	F304LN, F316LN or GR 630 or CF3A
Stems	SA-479 or SA-564	F316, F316LN or GR 630
Pressure retaining bolting	SA-453 or SA-564	GR 660 or GR 630
Pressure retaining nuts	SA-453 or SA-194	GR 6 or TP410
Core Makeup Tank		
Pressure plates	SA-533 or SA-240	GR B, CL 1 or 304L, 304LN, 316L, 316LN
Pressure forgings	SA-508 or SA-182, SA-336	CL 3 or F304L, F316L
Cladding and buttering	SFA 5.4, 5.9, 5.11, and 5.14	308L, 309L, ENiCrFe-7, ERNiCrFe-7, or ERNiCr-3
Passive Residual Heat Removal Heat Exchanger		
Pressure plates	SA-240	304L, 304LN
Pressure forgings	SA-336	F304L, F304LN
Cladding and buttering	SFA 5.4, 5.9, 5.11, and 5.14	308L, 309L, ENiCrFe-7, ERNiCrFe-7, or ERNiCr-3
Tubing	SA-376, SA-213	TP304LN, TP316LN

Table 5.2-2

REACTOR COOLANT WATER CHEMISTRY SPECIFICATIONS

Electrical conductivity	Determined by the concentration of boric acid and alkali present. Expected range is <1 to 40 μmhos/cm at 25°C.
Solution pH	Determined by the concentration of boric acid and alkali present. Expected values range between 4.2 (high boric acid concentration) and 10.5 (low boric acid concentration) at 25°C. Values will be 5.0 or greater at normal operating temperatures.
Oxygen(2)	0.1 ppm, maximum
Chloride(3)	0.15 ppm, maximum
Fluoride(3)	0.15 ppm, maximum
Hydrogen(4)	25 to 50 cm^3 (STP)/kg H_2O
Suspended solids(5)	0.2 ppm, maximum
pH control agent (Li7OH)(6)	Lithium is coordinated with boron per fuel warranty contract.
Boric acid	Variable from 0 to 4000 ppm as boron
Silica(7)	1.0 ppm, maximum
Aluminum(7)	0.05 ppm, maximum
Calcium(7) + magnesium	0.05 ppm, maximum
Magnesium(7)	0.025 ppm, maximum

See notes on following page

Notes:

1. Deleted

2. Oxygen concentration must be controlled to less than 0.1 ppm in the reactor coolant by scavenging with hydrazine prior to plant operation above 200°F. During power operation with the specified hydrogen concentration maintained in the coolant, the residual oxygen concentration will not exceed 0.005 ppm.

3. Halogen concentrations must be maintained below the specified values regardless of system temperature.

4. Hydrogen must be maintained in the reactor coolant for plant operations with nuclear power above 1 MW. The normal operating range should be 30-40 cm^3 (STP) H_2/kg H_2O.

5. Solids concentration determined by filtration through filter having 0.45-μm pore size.

6. The specified lithium concentrations must be established for startup testing prior to heatup beyond 150°F. During cold hydrostatic testing and hot functional testing in the absence of boric acid, the reactor coolant limits for lithium hydroxide must be maintained to inhibit halogen stress corrosion cracking.

7. These limits are included in the table of reactor coolant specifications as recommended standards for monitoring coolant purity. Establishing coolant purity within the limits shown for these species is judged desirable with the current data base to minimize fuel clad crud deposition, which affects the corrosion resistance and heat transfer of the clad.

Table 5.2-3

ASME CODES CASES

Code Case Number	**Title**
N-4-11	Special Type 403 Modified Forgings or Bars, Section III, Division 1, Class 1 and Class CS.
N-20-3	SB-163 Nickel-Chromium-Iron Tubing (Alloys 600 and 690) and Nickel-Iron-Chromium Alloy 800 at a Specified Minimum Yield Strength of 40.0 ksi and Cold Worked Alloy 800 at Yield Strength of 47.0 ksi, Section III, Division 1, Class 1.
N-60-5	Material for Core Support Structures, Section III, Division 1. (a)
N-71-15	Additional Material for Subsection NF, Class 1, 2, 3 and MC Component Supports Fabricated by Welding, Section III Division 1
N-122-2	Stress Indices for integral Structural Attachments Section III, Division 1, Class 1
N-249-11	Additional Materials for Subsection NF, Class 1, 2, 3, and MC Supports Fabricated Without Welding, Section III, Division 1 (b)
N-284 (d)	Metal containment Shell Buckling Design Methods, Section III, Division 1 Class MC
N-318-4	Procedure for Evaluation of the Design of Rectangular Cross Section Attachments on Class 2 or 3 Piping Section III, Division
N-319-1	Alternate Procedure for Evaluation of Stresses in Butt Welding Elbows in Class 1 Piping Section III, Division 1
N-391-1	Procedure for Evaluation of the Design of Hollow Circular Cross Section Welded Attachments on Class 1 Piping Section III, Division 1
N-392-2	Procedure for valuation of the Design of Hollow Circular Cross Section Welded Attachments on Class 2 and 3 Piping Section III, Division 1 (c)
N-474-2	Design Stress Intensities and Yield Strength Values for UNS06690 With a Minimum Yield Strength of 35 ksi, Class 1 Components, Section III, Division 1.
2142	F-Number Grouping for Ni-Cr-Fe, Classification UNS N06052 Filler Metal, Section IX.
2143	F-Number Grouping for Ni-Cr-Fe, Classification UNS W86152 Welding Electrode, Section IX.

(a) Use of this code case will meet the conditions for Code Case N-60-4 in Reg. Guide 1.85 Revision 30.
(b) Use of this code case will meet the conditions for Code Case N-249-10 in Reg. Guide 1.85 Revision 30.
(c) Use of this code case will meet the conditions for Code Case N-392-1 in Reg. Guide 1.84 Revision 30.
(d) Criteria and methods used for evaluation of containment buckling conditions that are not included in Revision 0 of Code Case N-284 are provided in subsection 3.8.2.

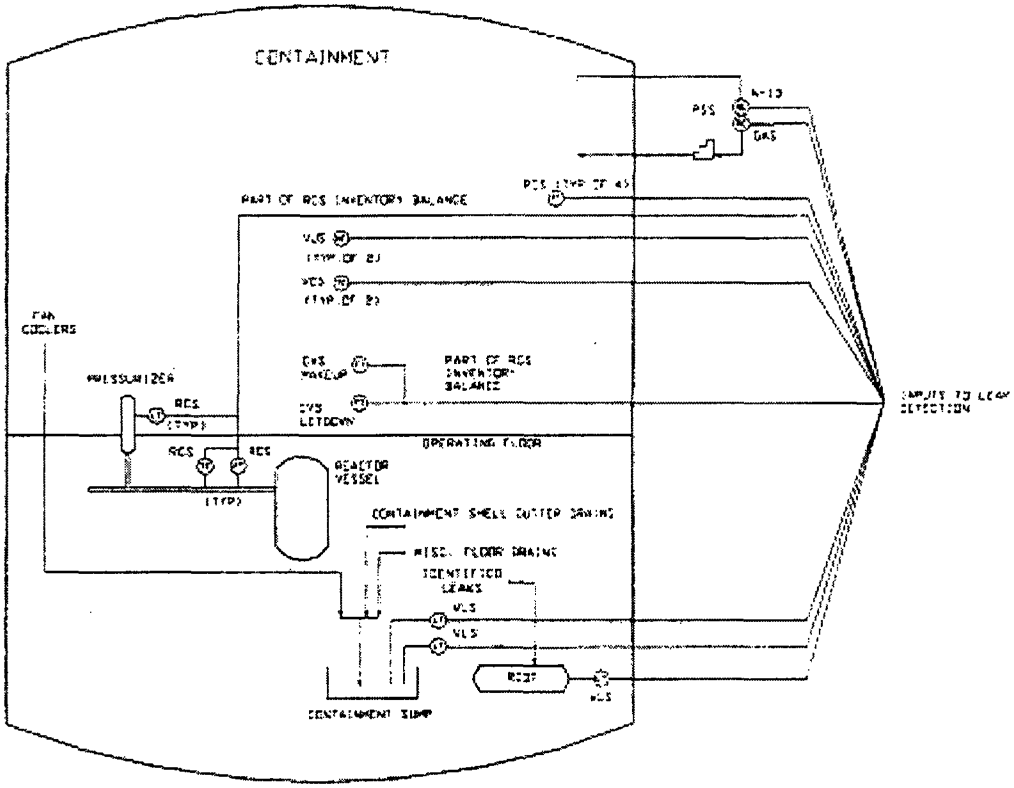

Figure 5.2-1

Leak Detection Approach