5.2 Integrity of Reactor Coolant Pressure Boundary

This section discusses measures employed to provide and maintain the integrity of the reactor coolant pressure boundary (RCPB) for the plant design lifetime.

5.2.1 Compliance with Codes and Code Cases

5.2.1.1 Compliance with 10CFR50, Section 50.55a

Table 3.2-3 shows the ASME Code applied to components. Code edition, applicable addenda, and component dates will be in accordance with 10CFR50.55a.

5.2.1.2 Applicable Code Cases

The reactor pressure vessel and appurtenances and the RCPB piping, pumps, and valves will be designed, fabricated, and tested in accordance with the applicable edition of the ASME Code, including addenda that were mandatory at the order date for the applicable components. Section 50.55a of 10CFR50 requires Code case approval for Class 1, 2, and 3 components. These Code cases contain requirements or special rules which may be used for the construction of pressure-retaining components of Quality Group Classification A, B, and C. The various ASME Code cases that may be applied to components are listed in Table 5.2-1.

Regulatory Guides 1.84, 1.85 and 1.147 provide a list of ASME Design and Fabrication Code cases that have been generically approved by the Regulatory Staff. Code cases on this list may, for design purposes, be used until appropriately annulled. Annulled cases are considered active for equipment that has been contractually committed to fabrication prior to the annulment.

5.2.2 Overpressure Protection

This subsection evaluates systems that protect the RCPB from overpressurization.

5.2.2.1 Design Basis

Overpressure protection is provided in conformance with 10CFR50 Appendix A, General Design Criterion 15. Preoperational and startup instructions are given in Chapter 14.

5.2.2.1.1 Safety Design Bases

The nuclear Pressure Relief System has been designed to:

(1) Prevent overpressurization of the nuclear system that could lead to the failure of the RCPB.

(2) Provide automatic depressurization for small breaks in the nuclear system occurring with maloperation of both the RCIC System and the HPCF System so that the low pressure flooder (LPFL) mode of the RHR System can operate to protect the fuel barrier.

(3) Permit verification of its operability.

(4) Withstand adverse combinations of loadings and forces resulting from normal, upset, emergency, or faulted conditions.

5.2.2.1.2 Power Generation Design Bases

The nuclear Pressure Relief System SRVs have been designed to meet the following power generation bases:

(1) Discharge to the containment suppression pool.

(2) Correctly reclose following operation so that maximum operational continuity is obtained.

5.2.2.1.3 Discussion

The ASME Boiler and Pressure Vessel Code (B&PV) requires that each vessel designed to meet Section III be protected from overpressure under upset conditions.

The SRV setpoints are listed in Table 5.2-3 and satisfy the ASME Code specifications for safety valves because all valves open at less than the nuclear system design pressure of 8.62 MPaG.

The automatic depressurization capability of the nuclear Pressure Relief System is evaluated in Sections 6.3 and 7.3.

The following criteria are used in selection of SRVs:

(1) Must meet requirements of ASME Code, Section III.

(2) Must qualify for 100% of nameplate capacity credit for the overpressure protection function.

(3) Must meet other performance requirements such as response time, etc., as necessary to provide relief functions.

The SRV discharge piping is designed, installed, and tested in accordance with ASME Code Section III.

5.2.2.1.4 Safety/Relief Valve Capacity

SRV capacity of this plant is adequate to limit the primary system pressure, including transients, to the requirements of ASME B&PV Code Section III (Nuclear Power Plant Components), up to and including applicable addenda. The essential ASME requirements which are met by this analysis follow.

It is recognized that the protection of vessels in a nuclear power plant is dependent upon many protective systems to relieve or terminate pressure transients. Installation of pressure-relieving devices may not independently provide complete protection. The safety valve sizing evaluation gives credit for operation of the scram protective system which may be tripped by either one of two sources: a direct or a flux trip signal. The direct scram trip signal is derived from position switches mounted on the MSIVs, the turbine stop valves, or from pressure switches mounted on the dump valve of the turbine control valve (TCV) hydraulic actuation system. The position switches are actuated when the respective valves are closing and following 15% travel of full stroke. The pressure switches are actuated when a fast closure of the TCVs is initiated. Credit is not taken for the power-operated mode. Credit is only taken for the SRV capacity which opens by the spring mode of operation direct from inlet pressure.

The rated capacity of the pressure-relieving devices shall be sufficient to prevent a rise in pressure within the protected vessel of more than 110% of the design pressure (1.10 x 8.62 MPaG=9.48 MPaG) for events defined in Section 15.2.

Full account is taken of the pressure drop on both the inlet and discharge sides of the valves. All combination SRVs discharge into the suppression pool through a discharge pipe from each valve, which is designed to achieve sonic flow conditions through the valve, thus providing flow independence to discharge piping losses.

5.2.2.2 Design Evaluation

5.2.2.2.1 Method of Analysis

The method of analysis is approved by the NRC or developed using criteria approved by the NRC.

5.2.2.2.2 System Design

A parametric study was conducted to determine the required steam flow capacity of the SRVs based on the following assumptions.

5.2.2.2.2.1 Operating Conditions

(1) Operating power = 4005 MWt [102% of nuclear boiler rated (NBR) power].

(2) Vessel dome pressure ≤7.17 MPaG.

(3) Steam flow = 7844 ≤ (102.6% of NBR steam flow).

These conditions are the most severe because maximum stored energy exists at these conditions. At lower power conditions, the transients would be less severe.

5.2.2.2.2.2 Transients

The overpressure protection system is capable of accommodating the most severe pressurization transient. The evaluation of transient behavior based on the core loading shown in Figure 4.3-1 demonstrates that MSIV closure with failure of direct scram (i.e., scram occurs on high flux) is the most severe pressurization transient. Other fuel designs and core loading patterns, including loading patterns similar to those shown in Figure 4.3-2, will not affect the conclusions of this evaluation. Analyses of this event will be performed each cycle and the results provided as information to the USNRC. Table 5.2-2 lists the systems which could initiate during the MSIV closure-flux scram events.

5.2.2.2.2.3 Safety/Relief Valve Transient Analysis Specification

(1) Simulated valve groups:

Spring-action safety mode - 5 groups

(2) Opening pressure setpoint (maximum safety limit):

Spring-action safety mode:

Group 1	8.12 MPaG
Group 2	8.19 MPaG
Group 3	8.26 MPaG
Group 4	8.33 MPaG
Group 5	8.39 MPaG

(3) Reclosure pressure setpoint (% of opening setpoint) both modes:

Maximum safety limit (used in analysis) — 98

Minimum operational limit — 90

The opening and reclosure setpoints are assumed at a conservatively high level above the nominal setpoints. This is to account for initial setpoint errors and any instrument setpoint drift that might occur during operation. Typically, the assumed setpoints in the analysis are at least 1% above the actual nominal setpoints. Conservative SRV response characteristics are also assumed; therefore, the analysis conservatively bounds all SRV operating conditions.

5.2.2.2.2.4 Safety/Relief Valve Capacity

Sizing of the SRV capacity is based on establishing an adequate margin from the peak vessel pressure to the vessel code limit (9.48 MPaG) in response to the reference transients.

The method used to determine total valve capacity is as follows:

Whenever the system pressure increases to the valve spring set pressure of a group of valves, these valves are assumed to begin opening and to reach full open at 102% of the valve spring set pressure. The lift characteristics assumed are shown in Figure 5.2-1.

5.2.2.2.3 Evaluation of Results

5.2.2.2.3.1 Safety/Relief Valve Capacity

The required SRV capacity is determined by analyzing the pressure rise from an MSIV closure with a flux scram transient. Results of this analysis are given in Figure 5.2-2a. The peak vessel bottom pressure calculated is 8.77 MPaG, which is well below the acceptance limit of 9.48 MPaG. The results show that only 12 valves are required to meet the design requirement with adequate margin.

5.2.2.2.3.2 Pressure Drop in Inlet and Discharge

Pressure drop in the piping from the reactor vessel to the valves is taken into account in calculating the maximum vessel pressures. Pressure drop in the discharge piping to the suppression pool is limited by proper discharge line sizing to prevent back pressure on each SRV from exceeding 40% of the valve inlet pressure, thus assuring choked flow in the valve orifice and no reduction of valve capacity due to the discharge piping. Each SRV has its own separate discharge line.

5.2.2.3 Piping and Instrument Diagrams

Figures 5.1-3 and 5.2-3 show the schematic location of the following pressure-relieving devices for:

(1) The reactor coolant system.

(2) The primary side of the auxiliary or emergency systems interconnected with the primary system.

(3) Any blowdown or heat dissipation system connected to the discharge side of the pressure-relieving devices.

Schematic arrangements of the SRVs are shown in Figures 5.2-3 and 5.2-4.

5.2.2.4 Equipment and Component Description

5.2.2.4.1 Description

The nuclear Pressure Relief System consists of SRVs located on the main steamlines between the reactor vessel and the first isolation valve within the drywell. These valves protect against overpressure of the nuclear system.

The SRVs provide three main protection functions:

(1) Overpressure relief operation (the valves are opened using a pnuematic actuator upon receipt of an automatic or manually-initiated signal to reduce pressure or to limit a pressure rise).

(2) Overpressure safety operation (the valves function as safety valves and open to prevent nuclear system overpressurization—they are self-actuated by inlet steam pressure if not already signaled open for relief operation).

(3) Depressurization operation (the ADS valves open automatically as part of the Emergency Core Cooling System (ECCS) for events involving small breaks in the nuclear system process barrier. The location and number of the ADS valves can be determined from Figure 5.1-3.

Chapter 15 discusses the events which are expected to activate the primary system SRVs. The chapter also summarizes the number of valves expected to operate in the safety (steam pressure) mode of operation during the initial blowdown of the valves and the expected duration of this first blowdown. For several of the events, it is expected that the lowest set SRV will reopen and reclose as generated heat decays. The pressure increase and relief cycle will continue with lower frequency and shorter relief discharges as the decay heat drops off.

Remote manual actuation of the valves from the control room is recommended to minimize the total number of these discharges with the intent of achieving extended valve seat life.

The SRV is opened by either of the following two modes of operation:

(1) The safety (steam pressure) mode of operation is initiated when the direct and increasing static inlet steam pressure overcomes the restraining spring and the frictional forces acting against the inlet steam pressure at the main disk and the main disk moves in the opening direction at a faster rate than corresponding disk movements at higher or lower inlet steam pressures. The condition at which this action is initiated is termed the “popping pressure” and corresponds to the set-pressure value stamped on the nameplate of the SRV.

(2) The relief (power) mode of operation is initiated when an electrical signal is received at any of the solenoid valves located on the pneumatic actuator assembly. The solenoid valve(s) will open, allowing pressurized air to enter the lower side of the pneumatic cylinder piston which pushes the piston and the rod upwards. This action pulls the lifting mechanism of the main disk, thereby opening the valve to allow inlet steam to discharge through the SRV until the solenoid valve(s) closes again to cut off pressurized air to the actuator.

The pneumatic operator is so arranged that, if it malfunctions, it will not prevent the valve from opening when steam inlet pressure reaches the spring lift set pressure.

For overpressure SRV operation (self-actuated or spring lift mode), the spring load establishes the safety valve opening setpoint pressure and is set to open at a setpoint designated in Table 5.2-3. In accordance with the ASME Code, the full lift of this mode of operation is attained at a pressure no greater than 3% above the setpoint.

The spring-loaded valves are designed and constructed in accordance with ASME Code Section III, NB-7500, as safety valves with auxiliary actuating devices.

For overpressure relief valve operation (power-actuated mode), valves are provided with pressure-sensing devices which operate at the setpoints designated in Table 5.2-3. When the set pressure is reached, a solenoid air valve is operated, which in turn actuates the pneumatic piston/cylinder and linkage assembly to open the valve.

The maximum opening delay from when the pressure just exceeds the relief setpoint to start of disk motion is 0.5 seconds, of which the time to energize the SRV solenoid shall not exceed 0.4 seconds. When the piston is actuated, the delay time (maximum elapsed time between receiving the overpressure signal at the valve actuator and the actual start of valve motion) will not exceed 0.1 second. The maximum elapsed time between signal to actuator and full-open position of the valve will not exceed 0.25 seconds, with the SRV inlet pressure > 6.89 MPaG and initial SRV pressure < 4% of inlet pressure.

The SRVs can be operated individually in the power-actuated mode by remote manual controls from the main control room.

There is one solenoid provided on each SRV for non-ADS power-actuated operation. The logic for the SRV power-actuated relief function requires two trip signals to open the SRVs. The failure of one pressure transmitter will not cause the SRVs to open. Each SRV is provided with its own pneumatic accumulator and inlet check valve. The accumulator capacity is sufficient to provide one SRV actuation. Subsequent actuations for an overpressure event can be spring actuations to limit reactor pressure to acceptable levels.

The ADS utilizes selected SRVs for depressurization of the reactor as described in Section 6.3. Each of the SRVs utilized for automatic depressurization is equipped with an air accumulator and check valve arrangement. These accumulators assure that the valves can be held open following failure of the air supply to the accumulators. The accumulator capacity is sufficient for one actuation at drywell design pressure or five actuations at normal drywell pressure.

Each SRV discharges steam through a discharge line to a point below minimum water level in the suppression pool. The SRV discharge lines are classified as Quality Group C and Seismic Category I. The SRV discharge lines in the wetwell air space are classified as Quality Group C and Seismic Catgory I, all welds shall be non-destructively examined to the requirements for ASME Boiler and Pressure Vessel Code, Section III, Class 2 piping. SRV discharge piping from the SRV to the suppression pool consists of two parts. The first is attached at one end to the SRV and at its other end to the diaphragm floor penetration, which acts as a pipe anchor.

The second part of the SRV discharge piping extends from the diaphragm floor penetration to the SRV quencher in the suppression pool. Because the diaphragm floor acts as an anchor on this part of the line, it is physically decoupled from the main steam header.

As a part of the preoperational and startup testing of the main steamlines, movement of the SRV discharge lines will be monitored.

The SRV discharge piping is designed to limit valve outlet pressure to approximately 40% of maximum valve inlet pressure with the valve wide open. Water in the line more than about 1/2 of a meter above suppression pool water level would cause excessive pressure at the valve discharge when the valve is again opened. For this reason, two vacuum relief valves are provided on each SRV discharge line to prevent drawing an excessive amount of water into the line as a result of steam condensation following termination of relief operation. The SRVs are located on the main steamline piping rather than on the reactor vessel top head, primarily to simplify the discharge piping to the pool and to avoid the necessity of having to remove sections of this piping when the reactor head is removed for refueling. In addition, valves located on the steamlines are more accessible during a shutdown for valve maintenance.

The ADS automatically depressurizes the nuclear system sufficiently to permit the LPFL mode of the RHR System to operate as a backup for the HPCF. Further descriptions of the operation of the automatic depressurization feature are presented in Section 6.3 and Subsection 7.3.1.

In addition to playing a major role in preventing core damage, depressurization of the RPV (either manually, automatically, or as a result of a LOCA) can help mitigate the consequences of severe accidents in which fuel melting and vessel failure occur. If the RPV were to fail at an elevated pressure (greater than approximately 1.37 MPaG) high pressure melt injection could occur resulting in fragmented core debris being transported into the upper drywell. The resulting heatup of the upper drywell could pressurize and fail the drywell. This failure mechanism is eliminated if the RPV is depressurized. The opening of a single SRV is capable of depressurizing the vessel sufficiently to prevent high pressure melt ejection.

5.2.2.4.2 Design Parameters

The specified operating transients for components within the RCPB are presented in Subsection 3.9.1. Subsection 3.7.1 provides a discussion of the input criteria for design of Seismic Category I structures, systems, and components. The design requirements established to protect the principal components of the reactor coolant system against environmental effects are presented in Section 3.11.

5.2.2.4.3 Safety/Relief Valve

The design pressure and temperature of the valve inlet is 9.48 MPaG at 308°C.

The valves have been designed to achieve the maximum practical number of actuations consistent with state-of-the-art technology.

5.2.2.5 Mounting of Safety/Relief Valves

The safety/relief valves are located on the main steam piping.

The design criteria and analysis methods for considering loads due to the SRV discharge is contained in Subsection 3.9.3.3.

5.2.2.6 Applicable Codes and Classification

The vessel overpressure protection system is designed to satisfy the requirements of Section III of the ASME B&PV Code. The general requirements for protection against overpressure of Section III of the Code recognize that reactor vessel overpressure protection is one function of the reactor protective systems and allows the integration of pressure relief devices with the protective systems of the nuclear reactor. Hence, credit is taken for the scram protective system as a complementary pressure–protection device. The NRC has also adopted the ASME Codes as part of their requirements in the Code of Federal Regulations (10CFR50.55a).

5.2.2.7 Material Specifications

Material specifications for pressure–retaining components of SRVs are in Table 5.2-4.

5.2.2.8 Process Instrumentation

Overpressure protection process instrumentation is listed in Table 4 of Figure 5.1-3.

5.2.2.9 System Reliability

The system is designed to satisfy the requirements of Section III of the ASME Boiler and Pressure Vessel Code. The consequences of failure are discussed in Sections 15.1.4 and 15.6.1.

5.2.2.10 Inspection and Testing

The inspection and testing applicable SRVs utilize a quality assurance program which complies with Appendix B of 10CFR50.

The non-radioactive SRVs are tested at a suitable test facility in accordance with quality control procedures to detect defects and to prove operability prior to installation. The following tests are conducted:

(1) Hydrostatic test at specified test conditions (ASME Code requirement based on design pressure and temperature).

(2) Thermally stabilize the SRV to perform quantitative steam leakage testing at 1.03 MPaG below the SRV nameplate valve with an acceptance criterion not to exceed 0.45 kg/h leakage.

(3) Full flow SRV test for set pressures and blowdown where the valve is pressurized with saturated steam, with the pressure rising to the valve set pressure. (The SRV must be adjusted to open at the nameplate set pressure ±1%, unless a greater tolerance is established as permissible in the overpressure protection report in the valve design specification).

(4) Response time test where each SRV is tested to demonstrate acceptable response time based on system requirements.

The valves are installed as received from the factory. The equipment specification requires certification from the valve manufacturer that design and performance requirements have been met. This includes capacity and blowdown requirements. The setpoints are adjusted, verified, and indicated on the valves by the vendor. Specified manual and automatic initiated signals for power actuation (relief mode) of each SRV are verified during the preoperational test program.

It is not feasible to test the SRV setpoints while the valves are in place. The valves are mounted on 10.36 MPaG primary service rating flanges, and can be removed for maintenance or bench checks and reinstalled during normal plant shutdowns. The valves will be tested to check set pressure in accordance with the requirements of the plant Technical Specifications. The external surface and seating of all SRVs are 100% visually inspected when the valves are removed for maintenance or bench checks. Valve operability is verified during the preoperational test program as discussed in Chapter 14.

5.2.3 Reactor Coolant Pressure Boundary Materials

5.2.3.1 Material Specifications

Table 5.2-4 lists the principal pressure-retaining materials and the appropriate material specifications for the RCPB components.

5.2.3.2 Compatibility with Reactor Coolant

5.2.3.2.1 PWR Chemistry of Reactor Coolant

Not applicable to BWRs.

5.2.3.2.2 BWR Chemistry of Reactor Coolant

A brief review of the relationships between water chemistry variables and RCS materials performance, fuel performance, and plant radiation fields is presented in this section and further information may be obtained from Reference 5.2-9.

The major environment-related materials performance problem encountered to date in the RCS of BWRs has been intergranular stress corrosion cracking (IGSCC) of sensitized austenitic stainless steel. IGSCC in sensitized material adjacent to welds in Type 304 and Type 316 stainless steel piping systems has occurred in the past. Substantial research and development

programs have been undertaken to understand the IGSCC phenomenon and develop remedial measures. For the ABWR, IGSCC resistance has been achieved through the use of IGSCC resistant materials such as Type 316 Nuclear Grade stainless steel and stabilized nickel-based Alloy 600M and 182M.

Much of the early remedy-development work focused on alternative materials or local stress reduction, but recently the effects of water chemistry parameters on the IGSCC process have received increasing attention. Many important features of the relationship between BWR water chemistry and IGSCC of sensitized stainless steels have been identified.

Laboratory studies (References 5.2-1 and 5.2-2) have shown that, although IGSCC can occur in simulated BWR startup environments, most IGSCC damage probably occurs during power operation. The normal BWR environment during power operation is ~280°C water containing dissolved oxygen, hydrogen and small concentrations of ionic and non- ionic impurities (conductivity generally below 0.3 μS/cm at 25°C). It has been well documented that some ionic impurities (notably sulfate and chloride) aggravate IGSCC, and a number of studies have been made of the effects of individual impurity species on IGSCC initiation and growth rates (References 5.2-1 thru 5.2-5). This work clearly shows that IGSCC can occur in water at 280°C with 200 ppb of dissolved oxygen, even at low conductivity (low impurity levels), but the rate of cracking decreases with decreasing impurity content. Although BWR water chemistry guidelines for reactor water cannot prevent IGSCC, maintaining the lowest practically achievable impurity levels will minimize its rate of progression (References 5.2-3 and 5.2-7).

Stress corrosion cracking of ductile materials in aqueous environments is often restricted to specific ranges of corrosion potential*, so a number of studies of impurity effects on IGSCC have been made as a function of either corrosion potential or dissolved oxygen content (the dissolved oxygen content is the major chemical variable in BWR type water that can be used to manipulate the corrosion potential in laboratory tests) (Reference 5.2-8).

As the corrosion potential is reduced below the range typical of normal BWR power operation (+50 to –50 mV_{SHE}), a region of immunity to IGSCC appears at ~ –230 mV_{SHE}. It is apparent that a combination of corrosion potential (which can be achieved in a BWR by injecting usually < 1 ppm hydrogen into the feedwater) plus tight conductivity control (0.2 μS/cm) should permit BWRs to operate in a regime where sensitized stainless steels are immune to IGSCC. In the reactor vessel, the excess hydrogen reacts with the radiolytic oxygen and reduces the electrochemical corrosion potential (References 5.2-9 and 5.2-10). The Reactor Water Cleanup System (CUW), which processes reactor water at a rate of 2% of rated feedwater flow, removes both dissolved and undissolved impurities that enter the reactor water. The removal of dissolved impurities reduces the conductivity into the region of immunity to IGSCC.

* Also called electrochemical corrosion potential (ECP), see Reference 5.2-7.

Since the ABWR has no sensitized stainless steel, IGSCC control by hydrogen injection is not required. However, irradiation assisted stress corrosion cracking (IASCC) can occur in highly irradiated annealed stainless steel and nickel-based alloys. Preliminary in-reactor and laboratory studies (References 5.2-11 and 5.2-13) have indicated that HWC will be useful in mitigating IASCC.

In-reactor and laboratory evidence also indicates that carbon and low alloy steels show improved resistance to environmentally assisted cracking with both increasing water purity and decreasing corrosion potential (Reference 5.2-12).

5.2.3.2.2.1 Fuel Performance Considerations

The uranium oxide powder in nuclear fuel is packed into pellets that are stacked in a zirconium alloy cladding. When there is a breach in the cladding, fission gases and uranium can leak into the coolant. This causes increased operational dose rates and can result in the release of large losses of uranium to the coolant.

Failure modes that have historically caused cladding breaches are (1) pellet to cladding interaction upon extreme power changes, (2) excessive oxidation, and (3) hydriding or crud formation as the zirconium alloy in the cladding interacts with the coolant or impurities in the coolant. These failure modes have successfully been mitigated by the introduction of 10x10 fuel and liner fuel, as well as a proactive approach to coolant chemistry monitoring and verification.

The only fuel failure mechanism resulting in cladding breach is debris fretting. Debris fretting as a failure cause is minimized by the use of multiple debris filters to capture debris before it can enter the fuel bundle.

Fretting failure is also minimized by the use of strict foreign material exclusion procedures during outages.

5.2.3.2.2.2 Radiation Field Buildup

The primary long-term source of radiation fields in most BWRs is Cobalt-60, which is formed by neutron activation of Cobalt-59. Corrosion products are released from corroding and wearing surfaces as soluble, colloidal, and particulate species. The formation of Cobalt-60 takes place after the corrosion products precipitate, adsorb, or deposit on the fuel rods. Subsequent re-entrainment in the coolant and deposition on out-of-core stainless steel surfaces leads to buildup of the activated corrosion products (such as Cobalt-60) on the out-of-core surfaces. The deposition may occur either in a loosely adherent layer created by particle deposition, or in a tightly adherent corrosion layer incorporating radioisotopes during corrosion and subsequent ion exchange. Water chemistry influences all of these transport processes. The key variables are the concentration of soluble Cobalt-60 in the reactor water and the characteristics of surface oxides. Thus, any reduction in the soluble Cobalt-60 concentration will have positive benefits.

As a means to reduce cobalt, the cobalt content has been reduced in alloys to be used in high fluence areas such as fuel assemblies and control rods. In addition, cobalt-based alloys used for guide pads in control rods have been replaced with noncobalt alloys.

The Reactor Water Cleanup (CUW) System, which processes reactor water at a rate of 2% of rated feedwater flow, will remove both dissolved and undissolved impurities which can become radioactive deposits. Reduction of these radioactive deposits will reduce occupational radiation exposure during operation and maintenance of the plant components.

Water quality parameters can have an influence on radiation buildup rates. In laboratory tests, the water conductivity and pH were varied systematically from a high purity base case. In each case, impurities increased the rate of Cobalt-60 uptake over that of the base case. The evidence suggests that these impurities change both the corrosion rate and the oxide film characteristics to adversely increase the Cobalt-60 uptake. Thus, controlling water purity should be beneficial in reducing radiation buildup.

Prefilming of stainless steel in Cobalt-60 free water, steam, or water/steam mixtures also appears to be a promising method to reduce initial radiation buildup rates. As an example, the radiation buildup rates are reduced significantly when samples are prefilmed in high temperature (288°C), oxygenated (200 ppb oxygen) water prior to exposure to Cobalt-60 containing water. Mechanical polishing and electropolishing of piping internal faces should also be effective in reducing radiation buildup.

5.2.3.2.2.3 Sources of Impurities

Various pathways exist for impurity ingress to the primary system. The most common sources of impurities that result in increases in reactor water conductivity are condenser cooling water inleakage, improper operation of ion exchange units, air inleakage, and radwaste recycle. In addition to situations of relatively continuous ingress, such as from low level condenser cooling water inleakage, transient events can also be significant. The major sources of impurities during such events are resin intrusions, organic chemical intrusions, inorganic chemical intrusions, and improper rinse of resins. Chemistry transients resulting from introduction of organic substances into the radwaste system comprised a significant fraction of the transients which have occurred.

The condensate cleanup system has two stages of water treatment. The first stage, high efficiency filters, is effective in removing insoluble solids, such as condensate system insoluble corrosion products. The second stage, the deep bed demineralizers, is effective in removing soluble solids, such as soluble corrosion products and impurities from possible condenser leakage.

The following factors are measured for control or diagnostic purposes to maintain proper water chemistry in the ABWR.

(1) Conductivity

Increasing levels of many ionic impurities adversely influence both the stress corrosion cracking behavior of Reactor Coolant System (RCS) materials, the rate of radiation field buildup and also can affect fuel performance. Therefore, conductivity levels in the reactor water should be maintained at the lowest level practically achievable.

(2) Chloride

Chlorides are among the most potent promoters of IGSCC of sensitized stainless steels and are also capable of inducing transgranular cracking of nonsenitized stainless steels. Chlorides also promote pitting and crevice attack of most RCS materials. Chlorides normally are associated with cooling water inleakage, but inputs via radwaste processing systems have also occurred.

Because chloride is implicated in several different corrosion phenomena, its level in reactor water should be kept as low as practically achievable during power operation.

(3) Sulfate

Recently, sulfate has been found to be more aggressive in promoting IGSCC of sensitized Type 304 stainless steel in BWR-type water (in laboratory tests) than any other ion, including chloride. Sulfates have also been implicated in environment-assisted cracking of high-nickel alloys and carbon and low-alloy steels. Sulfate ingress can result from cooling water inleakage, regenerant chemical inleakage, or resin ingress.

(4) Oxygen

Dissolved oxygen has been identified as a major contributor to IGSCC of sensitized stainless steels and reduction of oxygen content is known to reduce the tendency for pitting and cracks of most plant materials.

During power operation, most of the oxygen content of reactor water is due to the radiolysis of water in the core and, therefore, oxygen control cannot be achieved through traditional chemistry and operational practices. Oxygen control to low, plant-specific levels can be obtained through hydrogen injection. Control of reactor water oxygen during startup/hot standby may be accomplished by utilizing the de-aeration capabilities of the condenser. Independent control of control rod drive (CRD) cooling water oxygen concentration of <50 ppb during power operation is desirable to protect against IGSCC of CRD materials. Carbon steels exhibit minimal general corrosion and release rates in water with a conductivity less than 0.1 μS/cm if the concentration of oxygen is in the range of 20 to 1000 ppb. Regulation of reactor feedwater dissolved oxygen to 20-50 ppb during power operation will minimize corrosion of the condensate and feedwater system and reduce the possibility of

locally increasing reactor water oxygen concentrations. It is important to note that for oxygen concentrations below 30 ppb, the data indicates an increase in the corrosion and corrosion product release for carbon steels.

(5) Iron

High iron inputs into the reactor have been associated with excessive fuel deposit buildup. Proper regulation of feedwater purity and dissolved oxygen levels will minimize iron transport to the reactor. This, in turn, should minimize fuel deposits and may assist in controlling radiation buildup.

(6) Fluoride

Fluoride promotes many of the same corrosion phenomena as chloride, including IGSCC of sensitized austenitic stainless steels, and may also have the potential to cause corrosion of Zircaloy core components. If fluoride is present, it will be measured for diagnostic purposes.

(7) Organics

Organic compounds can be introduced into the RCS via turbine or pump oil leakage, radwaste, or makeup water systems. Of particular concern is the possibility that halogenated organic compounds (e.g., cleaning solvents) may pass through the radwaste systems and enter the RCS, where they will decompose, releasing corrosive halogens (e.g., chlorides and fluorides).

(8) Silica

Silica, an indicator of general system cleanliness, provides a valuable indication of the effectiveness of the CUW System. Silica inputs are usually associated with incomplete silica removal in makeup water or radwaste facilities.

(9) pH

There are difficulties of measuring pH in low conductivity water. Nevertheless, pH of the liquid environment has been demonstrated to have an important influence on IGSCC initiation times for smooth stainless steel specimens in laboratory tests. In addition, pH can serve as a useful diagnostic parameter for interpreting severe water chemistry transients, and pH measurements are recommended for this purpose.

(10) Electrochemical Corrosion Potential

The electrochemical corrosion potential (ECP) of a metal is the potential it attains when immersed in a water environment. The ECP is controlled by various oxidizing

agents, including copper and radiolysis products. At low reactor water conductivities, the ECP of stainless steel should be below –0.23 V_{SHE} to suppress IGSCC.

(11) Feedwater Hydrogen Addition Rate

A direct measurement of the feedwater hydrogen addition rate can be made using the hydrogen addition system flow measurement device and is used to establish the plant-specific hydrogen flow requirements required to satisfy the limit for the ECP of stainless steel (Paragraph 10). Subsequently, the addition rate measurements can be used to help diagnose the origin of unexpected ECP changes.

(12) Recirculation System Water Dissolved Hydrogen

A direct measurement of the dissolved hydrogen content in the reactor water serves as a cross-check against the hydrogen gas flow meter in the injection system to confirm the actual presence and magnitude of the hydrogen addition rate.

(13) Main Steamline Radiation Level

The major activity in the main steamline is Nitrogen-16 produced by a (n, p) reaction with Oxygen-16 in the reactor water. Under conditions of hydrogen water chemistry, the fraction of the Nitrogen-16 that volatilizes with the steam increases with increased dissolved hydrogen. The main steamline radiation monitor readings increase with the hydrogen addition rate. During initial plant testing, the amount of hydrogen addition required to reduce the electrochemical corrosion potential to the desired range is determined at various power levels. Changes in the main steamline radiation monitor readings at the same power level indicate an over-addition (high readings) or under-addition (low readings) of hydrogen.

(14) Constant Extension Rate Test

Constant extension rate tests (CERTs) are accelerated tests that can be completed in a few days, for the determination of the susceptibility to IGSCC. It is useful for verifying IGSCC suppression during initial implementation of hydrogen water chemistry (HWC) or following plant outages that could have had an impact on system chemistry (e.g., condenser repairs during refueling).

(15) Continuous Crack Growth Monitoring Test

This test employs a reversing DC potential drop technique to detect changes in crack length in IGSCC test specimens. The crack growth test can be used for a variety of purposes, including the following:

(a) Initial verification of IGSCC suppression following HWC implementation.

(b) Quantitative assessment of water chemistry transients.

(c) Long-term quantification of the success of the HWC program.

The major impurities in various parts of a BWR under certain operating conditions are listed in Table 5.2-5. The plant operators are encouraged to achieve better water quality by using good operating practice.

Water quality specifications require that erosion-corrosion resistant low alloy steels are to be used in susceptible steam extraction and drain lines. Stainless steels are considered for baffles, shields, or other areas of severe duty. Provisions are made to add nitrogen gas to extraction steamlines, feedwater heater shells, heater drain tanks, and drain piping to minimize corrosion during layup. Alternatively, the system may be designed to drain while hot so that dry layup can be achieved.

Condenser tubes and tubesheet materials are specified in Subsection 10.4.1.2.3.

Erosion-corrosion (E/C) of carbon steel components will be controlled as follows. The mechanism of E/C or, preferably, flow-assisted corrosion is complex and involves the electrochemical aspects of general corrosion plus the effects of mass transfer. Under single-phase flow conditions, E/C is affected by water chemistry, temperature, flow path, material composition and geometry. For wet steam (two phase), the percent moisture has an additional effect on E/C.

The potential deterioration of ABWR carbon steel piping from flow-assisted corrosion due to high velocity single-phase water flow and two-phase steam water flow will be addressed by using the EPRI developed CHECMATE (Chexal Horowitz Erosion Corrosion Methodology for Analyzing Two-phase Environments) computer code. CHECMATE will be used to predict corrosion rates and calculate the time remaining before reaching a defined acceptable wall thickness. Thus, this code will be used to identify areas where design improvements (piping design, materials selection, hydrodynamic conditions, oxygen content, temperature) are required to ensure adequate margin for extended piping performance on the ABWR design.

Water quality specifications for the ABWR require that the condenser be designed and erected so as to minimize tube leakage and facilitate maintenance. Appropriate features are incorporated to detect leakage and segregate the source. The valves controlling the cooling water to the condenser sections are required to be operable from the control room so that a leaking section can be sealed off quickly.

5.2.3.2.2.4 IASCC Considerations

Plant experience and laboratory tests indicate that irradiation assisted stress corrosion cracking (IASCC) can be initiated in solution annealed stainless steel above certain stress levels after exposure to radiation.

Extensive tests have also shown that IASCC has not occurred at fluence levels below $\sim 5 \times 10^{20}$ neutron/cm^2 (E>1.6019 E-13J)) even at high stress levels. Experiments indicate that, as fluence

increases above this threshold of 5×10^{20} neutron/cm^2, there is a decreasing threshold of sustained stress below which IASCC has not occurred. (Examination of top guides in two operating plants which have creviced designs has not revealed any IASCC.)

Reactor core structural components are designed to be below these thresholds of exposure and/or stress to avoid IASCC. In addition, crevices have been eliminated from the top guide design in order to prevent the synergistic interaction with IASCC.

In areas where the 5×10^{20} neutron/cm^2 threshold of irradiation is not practically avoided, the stress level is maintained below the stress threshold. High purity grades of materials are used in control rods to extend their life. Also, Hydrogen Water Chemistry (HWC) introduced in the plant design to control IGSCC may also be beneficial in avoiding IASCC.

5.2.3.2.3 Compatibility of Construction Materials with Reactor Coolant

The construction materials exposed to the reactor coolant consist of the following:

(1) Solution-annealed austenitic stainless steels (both wrought and cast), Types 304, 304L, 316, 316L, XM-19, CF3, CF3A, and CF3M.

(2) Nickel-based alloy (including Niobium Modified Alloy 600 and X-750) and alloy steel.

(3) Carbon steel and low alloy steel.

(4) Some 400-series martensitic stainless steel (all tempered at a minimum of 593°C).

(5) Colmonoy and Stellite hardfacing material (or equivalent).

(6) Precipitation hardening stainless steels, 17-4PH and XM-13 in the H1100 condition.

All of these construction materials are resistant to stress corrosion in the BWR coolant. General corrosion on all materials, except carbon and low alloy steel, is negligible. Conservative corrosion allowances are provided for all exposed surfaces of carbon and low alloy steels.

The requirements of GDC 4 relative to compatability of components with environmental conditions are met by compliance with the applicable provisions of the ASME Code and by compliance with the recommendations of Regulatory Guide 1.44.

Contaminants in the reactor coolant are controlled to very low limits. These controls are implemented by limiting contaminant levels of elements (such as halogens, S, Pb) to as low as possible in miscellaneous materials used during fabrication and installation. These materials (such as tapes, penetrants) are usually completely removed and cleanliness is assured. Lubricants and gaskets are not miscellaneous material. No detrimental effects will occur on any of the materials from allowable contaminant levels in the high purity reactor coolant. Expected radiolytic products in the BWR coolant have no adverse effects on the construction materials.

5.2.3.2.4 Compatibility of Construction Materials with External Insulation

All non-metallic insulation applied to austenitic stainless steel meets Regulatory Guide 1.36.

5.2.3.3 Fabrication and Processing of Ferritic Materials

5.2.3.3.1 Fracture Toughness

Compliance with Code requirements shall be in accordance with the following:

(1) The ferritic materials used for piping, pumps, and valves of the reactor coolant pressure boundary are usually 63.5 mm or less in thickness. Impact testing is performed in accordance with ASME Code Section III, Paragraph NB-2332 for thicknesses of 63.5 mm or less. Impact testing is performed in accordance with NB-2331 for thicknesses greater than 63.5 mm.

(2) Materials for bolting with nominal diameters exceeding 25.4 mm are required to meet both the 0.64 mm lateral expansion specified in NB-2333 and the 6.2 kg-m Charpy V value. The 60.8 N·m requirement of the ASME Code applies to bolts over 100 mm in diameter, starting Summer 1973 Addenda. Prior to this, the Code referred to only two sizes of bolts (≤ 25.4 mm and > 25.4 mm).

(3) The reactor vessel complies with the requirements of NB-2331. The reference temperature (RT_{NDT}) is established for all required pressure-retaining materials used in the construction of Class 1 vessels. This includes plates, forgings, weld material, and heat-affected zone. The RT_{NDT} differs from the nil-ductility temperature (NDT) in that, in addition to passing the drop test, three Charpy V-Notch specimens (transverse) must exhibit 6.9 kg-m absorbed energy and 0.89 mm lateral expansion at 33°C above the RT_{NDT}. The core beltline material must meet 102.0 N·m absorbed upper shelf energy (USE).

(4) Calibration of instrument and equipment shall meet the requirements of ASME Code Section III, Paragraph NB-2360.

5.2.3.3.2 Control of Welding

5.2.3.3.2.1 Regulatory Guide 1.50: Control of Preheat Temperature Employed for Welding of Low-Alloy Steel

Regulatory Guide 1.50 delineates preheat temperature control requirements and welding procedure qualifications supplementing those in ASME Sections III and IX.

The use of low-alloy steel is restricted to the reactor pressure vessel. Other ferritic components in the reactor coolant pressure boundary are fabricated from carbon steel materials.

Preheat temperatures employed for welding of low alloy steel meet or exceed the recommendations of ASME Code Section III, Subsection NB. Components are either held for

an extended time at preheat temperature to assure removal of hydrogen, or preheat is maintained until post-weld heat treatment. The minimum preheat and maximum interpass temperatures are specified and monitored.

All welds were nondestructively examined by radiographic methods. In addition, a supplemental ultrasonic examination was performed.

5.2.3.3.2.2 Regulatory Guide 1.34: Control of Electroslag Weld Properties

For electroslag welding applied to structural joints, the welding process variable specified in the procedure qualification shall be monitored during the welding process.

5.2.3.3.2.3 Regulatory Guide 1.71: Welder Qualification for Areas of Limited Accessibility

Welder qualification for areas of limited accessibility is discussed in Subsection 5.2.3.4.2.3.

5.2.3.3.3 Nondestructive Examination of Tubular Products

Wrought tubular products are supplied in accordance with applicable ASTM/ASME material specifications. Additionally, the specification for the tubular products used for CRD housings specified ultrasonic examination to Paragraph NB-2550 of ASME Code Section III.

These RCPB components meet 10CFR50 Appendix B requirements and the ASME Code requirements, thus assuring adequate control of quality for the products.

5.2.3.3.4 Moisture Control for Low Hydrogen, Covered Arc Welding Electrodes

Suitable identification, storage, and handling of electrodes, flux, and other welding material will be maintained. Precautions shall be taken to minimize absorption of moisture by electrodes and flux.

5.2.3.4 Fabrication and Processing of Austenitic Stainless Steels

5.2.3.4.1 Avoidance of Stress/Corrosion Cracking

5.2.3.4.1.1 Avoidance of Significant Sensitization

When austenitic stainless steels are heated in the temperature range 427°–982°C, they are considered to become “sensitized” or susceptible to intergranular corrosion. The ABWR design complies with Regulatory Guide 1.44 and with the guidelines of NUREG-0313 (Revision 2), to avoid significant sensitization.

For applications where stainless steel surfaces are exposed to water at temperatures above 93°C, low carbon (<0.03%) grade materials are used. For critical applications, nuclear grade (NG) materials (carbon content ≤0.02%) are used. All materials are supplied in the solution heat treated condition. Special sensitization tests are applied to assure that the material is in the annealed condition.

During fabrication, any heating operations (except welding) above 427°C are avoided, unless followed by solution heat treatment. During welding, heat input is controlled. The interpass temperature is also controlled. Where practical, shop welds are solution heat treated. In general, weld filler material used for austenitic stainless steel base metals is Type 308L/316L/309L with an average of 8% (or 8 FN) ferrite content.

5.2.3.4.1.2 Process Controls to Minimize Exposure to Contaminants

Process controls are exercised during all stages of component manufacturing and construction to minimize contaminants. Cleanliness controls are applied prior to any elevated temperature treatment.

Exposure to contaminants capable of causing stress/corrosion cracking of austenitic stainless steel components is avoided by carefully controlling all cleaning and processing materials which contact the stainless steel during manufacture, construction, and installation.

Special care is exercised to insure removal of surface contaminants prior to any heating operations. Water quality for cleaning, rinsing, flushing, and testing is controlled and monitored. Suitable protective packaging is provided for components to maintain cleanliness during shipping and storage.

The degree of surface cleanliness obtained by these procedures meets the requirements of Regulatory Guides 1.37 and 1.44.

5.2.3.4.1.3 Cold-Worked Austenitic Stainless Steels

Cold work controls are applied for components made of austenitic stainless steel. During fabrication, cold work is controlled by applying limits in hardness, bend radii and surface finish on ground surfaces.

5.2.3.4.2 Control of Welding

5.2.3.4.2.1 Avoidance of Hot Cracking

Regulatory Guide 1.31 describes the acceptable method of implementing requirements with regard to the control of welding when fabricating and joining austenitic stainless steel components and systems.

Written welding procedures which are approved by Toshiba are required for all primary pressure boundary welds. These procedures comply with the requirements of Sections III and IX of the ASME Boiler and Pressure Vessel (B&PV) Code and applicable NRC Regulatory Guides.

All austenitic stainless steel weld filler materials were required by specification to have a minimum delta ferrite content of 5 FN (ferrite number), and a maximum of 20 FN, determined

on undiluted weld pads by magnetic measuring instruments calibrated in accordance with AWS Specification A4.2.

Delta ferrite measurements are not made on qualification welds. Both the ASME B&PV Code and Regulatory Guide 1.31 specify that ferrite measurements be performed on undiluted weld filler material pads when magnetic instruments are used. There are no requirements for ferrite measurement on qualification welds.

5.2.3.4.2.2 Regulatory Guide 1.34: Electroslag Welds

See Subsection 5.2.3.3.2.2.

5.2.3.4.2.3 Regulatory Guide 1.71: Welder Qualification for Areas of Limited Accessibility

Regulatory Guide 1.71 requires that weld fabrication and repair for wrought low-alloy and high-alloy steels or other materials such as static and centrifugal castings and bimetallic joints should comply with fabrication requirements of Sections III and IX of the ASME B&PV Code. It also requires additional performance qualifications for welding in areas of limited access.

All ASME Section III welds are fabricated in accordance with the requirements of Sections III and IX of the ASME B&PV Code. There are few restrictive welds involved in the fabrication of BWR components. Welder qualification for welds with the most restrictive access is accomplished by mockup welding. Mockup is examined by sectioning and radiography (or UT).

Acceptance Criterion II.3.b.(3) of SRP Section 5.2.3 is based on Regulatory Guide 1.71. The ABWR design meets the intent of this regulatory guide by utilizing the following alternate approach.

When access to a non-volumetrically examined ASME Section III production weld (1) is less than 305 mm in any direction and (2) allows welding from one access direction only, such weld and repairs to welds in wrought and cast low alloy steels, austenitic stainless steels and high nickel alloys (and in any combination of these materials) shall comply with the fabrication requirements specified in ASME B&PV Code Section III and with the requirements of Section IX invoked by Section III, supplemented by the following requirements:

(1) The welder performance qualification test assembly required by ASME Code Section IX shall be welded under simulated access conditions. An acceptable test assembly will provide both a Section IX welder performance qualification required by this Regulatory Guide.

If the test assembly weld is to be judged by bend tests, a test specimen shall be removed from the location least favorable for the welder. If this test specimen cannot be removed from a location prescribed by Section IX, an additional bend test specimen will be required. If the test assembly weld is to be judged by radiography

or UT, the length of the weld to be examined shall include the location least favorable for the welder.

Records of the results obtained in welder accessibility qualification shall be (1) as certified by the manufacturer or installer, (2) maintained and (3) made accessible to authorized personnel.

Socket welds with a 50.8 mm nominal pipe size and under are excluded from the above requirements.

(2) (a) For accessibility, when more restricted access conditions will obscure the welder’s line of sight to the extent that production welding will require the use of visual aids such as mirrors, the qualification test assembly shall be welded under the more restricted access conditions using the visual aid required for production welding.

(b) Requalification is required when the essential variables listed in ASME Code Section IX are changed.

(3) Surveillance of accessibility qualification requirements will be performed along with normal surveillance of ASME Code Section IX performance qualification requirements.

5.2.3.4.3 Regulatory Guide 1.66: Nondestructive Examination of Tubular Products

For discussion of compliance with Regulatory Guide 1.66, see Subsection 5.2.3.3.3.

5.2.4 Preservice and Inservice Inspection and Testing of Reactor Coolant Pressure Boundary

This subsection describes the preservice and inservice inspection and system pressure test programs for NRC Quality Group A, ASME B&PV Code, Class 1, items.* It describes those programs implementing the requirements of Subsection IWB of the ASME B&PV (ASME Code) Code Section III and ASME B&PV Code Section XI.

The design to perform preservice inspection is based on the requirements of ASME Code Section XI. The development of the preservice and inservice inspection program plans will be the responsibility of the COL applicant and will be based on ASME Code Section XI, Edition and Addenda specified in accordance with 10CFR50, Section 50.55a. For design certification, Toshiba is responsible for designing the reactor pressure vessel for accessibility to perform preservice and inservice inspection. Responsibility for designing other components for preservice and inservice inspection is the responsibility of the COL applicant. The COL

* Items as used in this subsection are products constructed under a Certificate of Authorization (NCA-3120) and material (NCA-1220). See Section III, NCA-1000, footnote 2.

applicant will be responsible for specifying the Edition of ASME Code Section XI to be used, based on the procurement date of the component per 10CFR50, Section 50.55a. The ASME Code requirements discussed in this section are provided for information and are based on the edition of ASME Code Section XI specified in Table 1.8-21.

See Subsection 5.2.6.2 for COL license information.

5.2.4.1 Class 1 System Boundary

5.2.4.1.1 Definition

The Class 1 system boundary for both preservice and inservice inspection programs and the system pressure test program includes all those items within the Class 1 and Quality Group A boundary on the piping and instrumentation drawings (P&IDs). Based on 10 CFR (1-1-90 Edition) and Regulatory Guide 1.26, that boundary includes the following:

(1) Reactor pressure vessel

(2) Portions of the Main Steam System

(3) Portions of the Feedwater System

(4) Portions of the Standby Liquid Control System

(5) Portions of Reactor Water Cleanup System

(6) Portions of the Residual Heat Removal System

(7) Portions of the Reactor Core Isolation Cooling System

(8) Portions of the High Pressure Core Flooder System

Those portions of the above systems within the Class 1 boundary are those items which are part of the Reactor Coolant System (RCS) up to and including any and all of the following:

(1) The outermost containment isolation valve in the system piping which penetrates primary reactor containment.

(2) The second of two valves normally closed during normal reactor operation in system piping which does not penetrate primary reactor containment.

(3) The Reactor Coolant System SRVs,

(4) The main steam and feedwater system, up to and including the outermost containment isolation valve.

5.2.4.1.2 Exclusions

Portions of systems within the reactor coolant pressure boundary (RCPB), as defined in Subsection 5.2.4.1.1, that are excluded from the Class 1 boundary in accordance with 10CFR50, Section 50.55a, are as follows:

(1) Those components where, in the event of postulated failure of the component during normal reactor operation, the reactor can be shut down and cooled down in an orderly manner, assuming makeup is provided by the Reactor Coolant Makeup System (RCMS) only.

(2) Components which are or can be isolated from the RCS by two valves (both closed, both open, or one closed and one open). Each such open valve is capable of automatic actuation, and if the other valve is open, its closure time is such that, in the event of postulated failure of the component during normal reactor operation, each valve remains operable and the reactor can be shut down and cooled down in an orderly manner, assuming makeup is provided by the RCMS only.

The description of portions of systems excluded from the RCPB does not address Class 1 components exempt from inservice examinations under ASME Code Section XI rules. The Class 1 components exempt from inservice examinations are described in ASME Code Section XI, Subsection IWB–1220.

5.2.4.2 Accessibility

All items within the Class 1 boundary are designed to provide access for the examinations required by ASME Section XI, Subsection IWB–2500. Items such as nozzle-to-vessel welds often have inherent access restrictions when vessel internals are installed; therefore, preservice examination shall be performed on these items prior to installation of internals which would interfere with examination.

5.2.4.2.1 Reactor Pressure Vessel Access

Access for examinations of the reactor pressure vessel (RPV) is incorporated into the design of the vessel, biological shield wall and vessel insulation as follows:

(1) RPV Welds Below the Top Biological Shield Wall

The shield wall and vessel insulation behind the shield wall are spaced away from the RPV outside surface to provide access for remotely operated ultrasonic examination devices as described in Subsection 5.2.4.3.2.1. Access for the insertion of automated devices is provided through removable insulation panels at the top of the shield wall and at access ports at reactor vessel nozzles. Platforms are attached to the bioshield wall to provide access for installation of remotely operated nozzle examination devices.

(2) RPV Welds Above Top of the Biological Shield Wall

Access to the RPV welds above the top of the biological shield wall is provided by removable insulation panels. This design provides reasonable access for both automated as well as manual ultrasonic examination.

(3) Closure Head, RPV Studs, Nuts and Washers

The closure head is dry stored during refueling. Removable insulation is designed to provide access for manual ultrasonic examinations of closure head welds. RPV nuts and washers are dry stored and are accessible for surface and visual (VT-1) examination. RPV studs may be volumetrically examined in place or when removed.

(4) Bottom Head Welds

Access to the bottom head to shell weld and bottom head seam welds is provided through openings in the RPV support pedestal and removable insulation panels around the cylindrical lower portion of the vessel. This design provides access for manual or automated ultrasonic examination equipment. Sufficient access is provided to partial penetration nozzle welds (i.e., CRD penetrations, instrumentation nozzles and recirculation internal pump penetration welds) for performance of the visual VT-2, examination during the system leakage, and system hydrostatic examinations.

(5) Reactor Vessel Support Skirt

The weld between the integrally forged vessel support attachment on the lower shell ring and the RPV support skirt will be examined ultrasonically. Sufficient access is provided for either manual or automated ultrasonic examination. Access is provided to the balance of the support skirt for performance of visual, VT-3, examination.

5.2.4.2.2 Piping, Pumps, Valves and Supports

Physical arrangement of piping pumps and valves provides personnel access to each weld location for performance of ultrasonic and surface (magnetic particle or liquid penetrant) examinations and sufficient access to supports for performance of visual, VT-3, examination. Working platforms are provided in some areas to facilitate servicing of pumps and valves. Platforms and ladders are provided for access to piping welds including the pipe-to-reactor vessel nozzle welds. Removable thermal insulation is provided on welds and components which require frequent access for examination or are located in high radiation areas. Welds are located to permit ultrasonic examination from at least one side, but where component geometries permit, access from both sides is provided.

Restrictions: For piping systems and portions of piping systems subject to volumetric and surface examination, the following piping designs are not used:

(1) Valve to valve

(2) Valve to reducer

(3) Valve to tee

(4) Elbow to elbow

(5) Elbow to tee

(6) Nozzle to elbow

(7) Reducer to elbow

(8) Tee to tee

(9) Pump to valve

Straight sections of pipe and spool pieces shall be added between fittings. The minimum length of the spool piece has been determined by using the formula L = 2T + 152 mm, where L equals the length of the spool piece (not including weld preparation) and T equals the pipe wall thickness. Where less than the minimum straight section length is used, an evaluation is performed to ensure that sufficient access exists to perform the required examinations.

5.2.4.3 Examination Categories and Methods

5.2.4.3.1 Examination Categories

The examination category of each item is listed in Table 5.2-8, which is provided as an example for the preparation of the preservice and inservice inspection program plans. The items are listed by system and line number, where applicable. Table 5.2-8 also states the method of examination for each item. The preservice and inservice examination plans will be supplemented with detailed drawings showing the examination areas (Figures 5.2-7a and 5.2-7b).

For the preservice examination, all of the items selected for inservice examination shall be performed once in accordance with ASME Code Section XI, Subsection IWB-2200, including essentially 100% of the pressure retaining welds in all Class 1 components, with the exception of the examinations specifically excluded by ASME Code Section XI from preservice requirements, such as surface or volumetric examinations of welds in lines smaller than NPS 1, volumetric examinations of welds in lines smaller than NPS 4, VT-3 examination of valve body and pump casing internal surfaces (B-L-2 and B-M-2 examination categories, respectively) and the visual VT-2 examinations for categories B-E and B-P. If the as-built design incorporates

external Category B-O control rod drive housing welds, the preservice examination shall be extended to include 100% of the welds in the installed peripheral control rod drive housings only in accordance with IWB-2200.

5.2.4.3.2 Examination Methods

5.2.4.3.2.1 Ultrasonic Examination of the Reactor Vessel

Ultrasonic examination for the RPV will be conducted in accordance with ASME Code Section XI. The design to perform preservice inspection on the reactor vessel shall be based on the requirements of ASME Code Section XI. For the required preservice examinations, the reactor vessel shall meet the acceptance standards of Section XI, Subsection IWB-3510. The RPV shell welds are designed for 100% accessibility for both preservice and inservice inspection. The RPV nozzle-to-shell welds will be 100% accessible for preservice inspection, but might have limited areas that will not be accessible from the outer surface for inservice examination techniques. However, the inservice inspection program for the reactor vessel is the responsibility of the COL applicant and any inservice inspection program relief request will be reviewed by the NRC staff based on the Code Edition and Addenda in effect and inservice inspection techniques available at the time of COL application.

The ultrasonic system for examination of the reactor vessel meets the qualification requirements discussed in Subsection 5.2.4.3.4.

5.2.4.3.2.2 Visual Examination

Visual examination methods VT-1, VT-2 and VT-3 shall be conducted in accordance with ASME Section XI, Subsection IWA-2210. In addition, VT-2 examinations shall meet the requirements of IWA-5240.

Direct visual (VT-1) examinations shall be conducted with sufficient lighting to resolve a 0.8 mm black line on an 18% neutral grey card. Where such examinations are conducted without the use of mirrors or with other viewing aids, clearance (of at least 610 mm of clear space) is provided where feasible for the head and shoulders of a man within a working arm's length (508 mm) of the surface to be examined.

At locations where leakages are normally expected and leakage collection systems are located (e.g., valve stems and pump seals), the visual (VT-2) examination shall verify that the leakage collection system is operative.

Piping runs shall be clearly identified and laid out such that insulation damage, leaks and structural distress will be evident to a trained visual examiner.

5.2.4.3.2.3 Surface Examination

Magnetic particle and liquid penetrant examination techniques shall be performed in accordance with ASME Section XI, Subsections IWA-2221 and IWA-2222, respectively.

Direct examination access for magnetic particle (MT) and penetrant (PT) examination is the same as that required for direct visual (VT-1) examination (Subsection 5.2.4.3.2.1), except that additional access shall be provided as necessary to enable physical contact with the item in order to perform the examination. Remote MT and PT generally are not appropriate as a standard examination process; however, boroscopes and mirrors can be used at close range to improve the angle of vision. As a minimum, insulation removal shall expose the area of each weld plus at least 152 mm from the toe of the weld on each side. Insulation will generally be removed 406 mm on each side of the weld.

5.2.4.3.2.4 Volumetric Ultrasonic Examination

Volumetric ultrasonic examination shall be performed in accordance with ASME Section XI, Subsection IWA-2232. In order to perform the examination, visual access to place the head and shoulders within 508 mm of the area of interest shall be provided where feasible. Twenty three centimeters between adjacent pipes is sufficient spacing if there is free access on each side of the pipes. The transducer dimension has been considered: a 38 mm diameter cylinder, 76 mm long placed with access at a right angle to the surface to be examined. The ultrasonic examination instrument has been considered as a rectangular box, 305 x 305 x 508 mm, located within 12m from the transducer. Space for a second examiner to monitor the instrument shall be provided, if necessary.

Insulation removal for inspection is to allow sufficient room for the ultrasonic transducer to scan the examination area. A distance of 2T plus 152 mm, where T is pipe thickness, is the minimum required on each side of the examination area. The insulation design generally leaves 406 mm on each side of the weld, which exceeds minimum requirements.

5.2.4.3.2.5 Alternative Examination Techniques

As provided by ASME Section XI, Subsection IWA-2240, alternative examination methods, a combination of methods, or newly developed techniques may be substituted for the methods specified for a given item in this section, provided that they are demonstrated to be equivalent or superior to the specified method. This provision allows for the use of newly developed examination methods, techniques, etc., which may result in improvements in examination reliability and reductions in personnel exposure.

5.2.4.3.3 Data Recording

Manual data recording will be performed where manual ultrasonic examinations are performed. Electronic data recording and comparison analyses are to be employed with automated ultrasonic examination equipment. Signals form each ultrasonic transducer will be fed into a data acquisition system in which the key parameters of any reflectors will be recorded. The data to be recorded for manual and automated methods are:

(1) Location

(2) Position

(3) Depth below the scanning surface

(4) Length of the reflector

(5) Transducer data, including angle and frequency

(6) Calibration data

The data so recorded shall be compared with the results of subsequent examinations to determine the behavior of the reflector.

5.2.4.3.4 Qualification of Personnel and Examination Systems for Ultrasonic Examination

Personnel performing examinations shall be qualified in accordance with ASME Section XI, Appendix VII. Ultrasonic examination systems shall be qualified in accordance with industry accepted program for implementation of ASME Section XI, Appendix VIII.

5.2.4.4 Inspection Intervals

The inservice inspection intervals for the ABWR will conform to Inspection Program B as described in Section XI, Subsection IWB-2412. Except where deferral is permitted by Table IWB-2500-1, the percentages of examinations completed within each period of the interval shall correspond to Table IWB-2412-1. An example of the selection of items and examinations to be conducted within the 10-year intervals are described in Table 5.2-8.

5.2.4.5 Evaluation of Examination Results

Examination results will be evaluated in accordance with ASME Section XI, Subsection IWB-3000, with repairs based on the requirements of Subsections IWA-4000 and IWB-4000. Re-examination shall be conducted in accordance with the requirements of IWA-2200. The recorded results shall meet the acceptance standards specified in IWB-3400-1.

5.2.4.6 System Leakage and Hydrostatic Pressure Tests

5.2.4.6.1 System Leakage Tests

As required by Section XI, IWB-2500 for Category B-P, a system leakage test shall be performed in accordance with IWB-5221 on all Class 1 components and piping within the pressure retaining boundary following each refueling outage. For the purposes of the system leakage test, the pressure retaining boundary is as defined in Table IWB-2500-1, Category B-P, Note 1. The system leakage test shall include a VT-2 examination in accordance with IWA-5240. The system leakage test will be conducted approximately at the maximum operating pressure and temperature indicated in the applicable process flow diagram for the system as indicated in Table 1.7-1. The system hydrostatic test (Subsection 5.2.4.6.2) is acceptable in lieu of the system leakage test.

5.2.4.6.2 Hydrostatic Pressure Tests

As required by Section IX, IWB-2500 for Category B-P, the hydrostatic pressure test shall be performed in accordance with ASME Section IWB-5222 on all Class 1 components and piping within the pressure retaining boundary once during each 10-year inspection interval. For purposes of the hydrostatic pressure test, the pressure retaining boundary is defined in Table IWB-2500-1, Category B-P, Note 1. The system hydrostatic test shall include a VT-2 examination in accordance with IWA-5240. For the purposes of determining the test pressure for the system hydrostatic test in accordance with IWB-5222 (a), the nominal operating pressure shall be the maximum operating pressure indicated in the P&ID for the Nuclear Boiler System (Figure 5.1-3).

5.2.4.7 Code Exemptions

As provided in ASME Section XI, IWB-1220, certain portions of Class 1 systems are exempt from the volumetric and surface examination requirements of IWB-2500. These portions of systems are specifically identified in Table 5.2-8.

5.2.5 Reactor Coolant Pressure Boundary and Core Cooling Systems Leakage Detection

5.2.5.1 Leakage Detection Methods

RCPB leakage detection is a primary function of the Leak Detection and Isolation System (LDS). The LDS (Figure 5.2-8) consists of temperature, pressure, radiation and flow sensors with associated instrumentation, power supplies and logic used to detect, indicate, and alarm leakage from the reactor primary pressure boundary and, in certain cases (Subsections 7.3.1.1.2, 7.6.1.3 and 7.7.1.7), to initiate closure of isolation valves to shut off leakage external to the containment. The system is designed to be in conformance with Regulatory Guide 1.45 (for leak detection functions) and IEEE-279 (for isolation function).

Abnormal leakage from the following systems within the primary containment (drywell) and within selected areas of the plant outside the drywell (both inside and outside the reactor building) is detected, indicated, alarmed, and, in certain cases, isolated:

(1) Main steamlines

(2) Reactor Core Isolation Cooling (RCIC) System

(3) High Pressure Core Flooder (HPCF)

(4) Residual Heat Removal (RHR) System

(5) Reactor Water Cleanup (CUW) System

(6) Feedwater System

(7) Coolant systems within the drywell

(8) Reactor pressure vessel

(9) Miscellaneous systems

Leak detection methods (in accordance with Regulatory Guide 1.45) differ for the plant areas inside the drywell as compared to those areas outside the drywell. These areas are considered separately as follows.

5.2.5.1.1 Detection of Leakage Within Drywell

The primary detection method for small unidentified leaks within the drywell includes (1) drywell floor drain sump pump activity and sump level increases, (2) drywell cooler condensate flow rate increases, and (3) airborne gaseous and particulate radioactivity increases. The sensitivity of these primary detection methods for unidentified leakage within the drywell is 3.785 liters/min within one hour. These variables are continuously indicated and/or recorded in the control room. If the unidentified leakage increases to 19 liters/min, the detection instrumentation channel will trip and activate an alarm in the control room to alert the operator.

The secondary detection methods, pressure and temperature of the drywell atmosphere are used to detect gross unidentified leakage. High drywell pressure will alarm and trip the isolation logic, which will result in closure of the containment isolation valves. High drywell temperature is recorded and alarmed only.

The detection of small identified leakage within the drywell is accomplished by monitoring drywell equipment drain sump pump activity and sump level increases. The equipment drain sump level monitoring instruments will activate an alarm in the control room when the total leak rate reaches 114 liters/min.

Equipment drain sump pump activity and sump level increases will be caused primarily from leaks from large process valves through valve stem drain lines.

The determination of the source of other identified leakage within the drywell is accomplished by (1) monitoring the reactor vessel head seal drain line pressure, (2) Not Used, and (3) monitoring temperature in the SRV discharge lines to the suppression pool to detect leakage through each of the SRVs. All of these monitors continuously indicate and/or record in the control room and will trip and activate an alarm in the control room on detection of leakage from monitored components.

Excessive leakage inside the drywell (e.g., process line break or loss-of-coolant accident) is detected by high drywell pressure, low reactor water level, or high steamline flow (for breaks downstream of the flow elements). The instrumentation channels for these variables will trip when the monitored variable exceeds predetermined limits to activate an alarm and trip the isolation logic, which will close appropriate isolation valves.

The alarms, indication and isolation trip functions performed by the foregoing leak detection methods are summarized in Tables 5.2-6 and 5.2-7.

Listed below are the variables monitored for detection of leakage from piping and equipment located within the drywell:

(1) High drywell temperature

(2) Not Used

(3) High flow rate from the drywell floor and equipment drain sumps

(4) High steamline flow rate (for leaks downstream of flow elements in main steamline and RCIC steamline)

(5) High drywell pressure

(6) High fission product releases

(7) Reactor vessel low water level

(8) Reactor vessel head seal drain line high pressure

(9) SRV discharge piping high temperature.

(10) Feedwater lines pressure difference

5.2.5.1.2 Detection of Leakage External to Drywell

The areas outside the primary containment (drywell) that are monitored for primary coolant leakage are (1) the equipment areas in the Reactor Building (R/B), (2) the main steam tunnel, and (3) the main steamline tunnel area in the Turbine Building (T/B). The process piping, for each system to be monitored for leakage, is located in compartments or rooms separated from other systems, so that leakage may be detected by area temperature monitors.

The areas are monitored by thermocouples that sense high ambient temperature in each area. The temperature elements are located or shielded so that they are sensitive to air temperature only and not radiated heat from hot piping or equipment. Increases in ambient temperature will indicate leakage of reactor coolant into the area. These monitors have sensitivities suitable for detection of reactor coolant leakage into the monitored areas of 95 liters/min or less. The temperature trip setpoint will be a function of the room size and the type of ventilation provided. These monitors provide alarm and indication in the control room and will trip the isolation logic to close the appropriate isolation valves (e.g., the main steam tunnel area temperature monitors will close the MSIV, MSL drain isolation valves, and the CUW isolation valves.

Ambient differential temperature monitoring is provided in equipment areas of the reactor building and the R/B MSL tunnel area to monitor for small leaks. The leakage is monitored and alarmed in the control room.

Leakage detection will be provided in the turbine building. The T/B monitors will also alarm and indicate in the control room and trip the isolation logic to close the MSIVs and MSL drain isolation valves when leakage exceeds 95 liters/min.

Large leaks external to the drywell (e.g., process line breaks outside of the drywell) are detected by low reactor water level, high process line flow, high ambient temperatures in the MSL tunnel to the turbine or equipment areas, floor or equipment drain sump activity, high differential flow (CUW only), low steamline pressures or low main condenser vacuum. These monitors provide alarm and indication in the control room and will trip the isolation logic to cause closure of appropriate system isolation valves.

Intersystem leakage detection is accomplished by monitoring radiation of the Reactor Building Cooling Water (RCW) System coolant return lines from the reactor internal pumps (RIPs), Residual Heat Removal (RHR) System, and Reactor Water Cleanup (CUW) System and fuel pool cooling heat exchangers. This monitoring is provided by the Process Radiation Monitoring System. Potential intersystem leakage from the RCPB to RCIC, RHR or HPCF is discussed in response to Question 430.2c.

Listed below are the variables monitored for detection of leakage from piping and equipment located external to the primary containment (drywell):

(1) Within the reactor building:

  (a) Main steamline and RCIC steamline high flow.

  (b) Reactor vessel low water level.

  (c) High flow rate from reactor building sumps outside drywell.

  (d) High ambient temperature or high differential in equipment areas of RCIC, RHR, and the hot portions of the CUW.

  (e) RCIC turbine exhaust line high diaphragm pressure.

  (f) High differential mass flow rate in CUW piping.

  (g) High radiation in the RHR, CUW, and RIP, and FPC reactor building cooling water heat exchanger discharge lines (intersystem leakage).

  (h) RCIC steamline low pressure.

(2) Within steam tunnel (between primary containment and turbine building):

  (a) Not Used

(b) Main steam tunnel high ambient air temperature or high differential temperature.

(3) Within turbine building (outside secondary containment):

(a) Main steamline low pressure.

(b) Low main condenser vacuum.

(c) Turbine building ambient temperature in areas traversed by main steamlines.

5.2.5.2 Leak Detection Instrumentation and Monitoring

5.2.5.2.1 Leak Detection Instrumentation and Monitoring Inside the Drywell

(1) Drywell Floor Drain Sump Monitoring

The drywell floor drain sump collects unidentified leakage such as leakage from control rod drives, floor drains, valve flanges, closed cooling water for reactor services (e.g., RIP motor cooling), condensate from the drywell atmosphere coolers, and any leakage not connected to the drywell equipment drain sump. The sump is equipped with two pumps and special instrumentation to measure sump fillup and pumpout times and provide continuous sump level rate of change monitoring with control room indication and alarm capabilities for excessive fill rate or pumpout frequency of the pumps. The drain sump instrumentation has a sensitivity of detecting reactor coolant leakage of 3.785 liters/min within a 60-minute period. The alarm setpoint has an adjustable range up to 19 liters/min for the drywell floor drain sump. In order to provide an early warning of RCS leakage to the operators, a computer based control based alarm is provided that requires operator action with an 8L/min increase in unidentified leakage over four hours.

(2) Drywell Equipment Drain Sump Monitoring

The drywell equipment drain sump collects only identified leakage from identified leakage sources. This sump monitors leakage from the RPV head flange seal, and other known leakage sources which are piped directly into the drywell equipment drain sump. The number of sump pumps and the types of drain sump instrumentation is the same as that used for the drywell floor drain sump. The monitoring channels measure sump level rate of change and sump fillup and pumpout times, with main control room indication and alarm capabilities. Collection in excess of background leakage would indicate an increase in reactor coolant leakage from an identifiable source.

(3) Drywell Air Cooler Condensate Flow Monitoring

The condensate flow rates from the drywell atmosphere coolers are monitored for high drain flows, which indicate leaks from piping or equipment within the drywell.

This flow is monitored by one channel of flow instrumentation located to measure flow in the common condensate cooler drain line, which drains the condensate from all of the drywell coolers to the drywell floor drain sump. The transmitter and its associated comparator provide main control room flow readout and trip and alarm on high flow conditions approaching the unidentified discharge rate limit. Location of the common header is such that at least a 25% safety margin is available for flow transmitter pressure head requirements.

(4) Drywell Temperature Monitoring

The ambient temperature within the drywell is monitored by four single element thermocouples located equally spaced in the vertical direction within the drywell. An abnormal increase in drywell temperature could indicate a leak within the drywell.

Ambient temperatures within the drywell are recorded and alarmed in the main control room. Air temperature monitoring sensors are located such that they are sensitive to reactor coolant leakage and not to radiated heating from pipes and equipment.

(5) Drywell Fission Product Monitoring

Primary coolant leaks within the drywell are detected by radiation monitoring of continuous drywell atmosphere samples. The fission product radiation monitors provide gross counting of radiation from radioactive particulates, and radioactive gases. The count levels are recorded in the control room and alarmed on abnormally high activity level.

(6) Drywell Pressure Monitoring

Drywell pressure is monitored by pressure transmitters which sense drywell pressure relative to R/B (secondary containment) pressure. Four channels of drywell monitoring are provided by the Nuclear Boiler System (NBS). A pressure rise above the normally indicated values will indicate a possible leak or loss of reactor coolant within the drywell. Pressure exceeding preset values will be alarmed in the main control room and required safety action will be automatically initiated.

(7) Reactor Vessel Head Flange Seal Monitoring

A single channel of pressure monitoring is provided for measurement and control room indication of pressure between the inner and outer reactor head flange seals. High pressure will indicate a leak in the inner O-ring seal. This high pressure is annunciated in the main control room (no isolation). A pressure tap for this measurement is provided by the NBS. Leakage through both inner and outer seals will be detected by other drywell leak detection instrumentation. Any leakage through the inner seal can be directed to the drywell equipment drain sump.

(8) Reactor Recirculation Pump Motor Leakage Monitoring

Excess leakage from the RIP motor casing will be detected by the drywell floor drain sump monitors described in (1) above.

(9) Safety/Relief Valve Leakage Monitoring

SRV leakage is detected by temperature sensors located on each relief valve discharge line such as to detect any valve outlet port flow. Each of the temperature channels includes control room recording and alarm capabilities. The temperature sensors are mounted using thermowells in the discharge piping about half of a meter from the valve body to prevent false indication. The monitoring of this leakage is provided by the NBS.

(10) Not Used

(11) Main Steamline High Flow Monitoring (for leaks downstream of flow elements)

High flow in each main steamline is monitored by four differential pressure transmitters that sense the pressure difference across a flow restrictor in the RPV main steam outlet nozzle. The pressure taps are part of the Nuclear Boiler System. Two sets of taps are provided, each set includes a nozzle tap and a vessel tap. High flow rate in the main steamlines during plant operation could indicate a MSL break. High flow exceeding the preset value in any of the four main steamlines will result in trip of the MSIV isolation logic to close all the MSIVs and the MSL drain valves, and annunciate the high flow in the main control room. Each monitoring channel includes inputs to the process computer.

(12) Reactor Vessel Low Water Level Monitoring

The Nuclear Boiler System provides reactor water level monitoring for the LDS functions and for safety functions of other systems. Sixteen channels of monitoring (four in each division to provide trip signals at four different water levels, i.e., Levels 3, 2, 1.5 and 1) are provided for the LDS functions (e.g., RHR, CUW, MSL and isolations of other portions of the plant). The safety-related performance requirements of the level monitoring channels are a function of the NBS. For additional information on reactor vessel water level instrumentation see Subsection 7.7.1.1.

The impact of noncondensable gases on the accuracy of reactor vessel water level measurements shall be considered in the design of water level instrument piping. The COL applicant will design the water level instrumentation flow control system to provide flow rates determined by the results of the BWR Owners' Group testing, as required in Subsection 5.2.6.3.

(13) RCIC Steamline Flow Monitoring (for leaks downstream of flow elements)

The steam supply line for motive power for operation of the RCIC turbine is monitored for abnormal flow. Four channels of flow measurement are provided for detection of steamline breaks downstream of the flow elements by LDS flow transmitters which sense differential pressure across elbow taps in the RCIC turbine supply steamline. High steam flow exceeding preset values will result in the closure of the RCIC steamline isolation valves, warmup bypass valve, and trip the turbine isolation valve. Isolation trip signals from one division will close the outboard isolation valves, while trip signals from a second division will close the inboard RCIC steamline isolation valve and warmup bypass valve. Any isolation signal to the RCIC logic will also trip the RCIC turbine. LDS measurements are taken as close to the reactor vessel as possible to maximize LDS coverage.

(14) Feedwater Lines Pressure Difference

The feedwater lines are monitored for excessive pressure differences that would indicate a break has occurred in one of the lines. Four channels are provided. A confirmatory high drywell pressure signal is also needed to initiate a trip of condensate pumps.

5.2.5.2.2 Leak Detection Instrumentation and Monitoring External to Drywell

(1) Visual and Audible Inspection

Accessible areas are inspected periodically and the temperature, pressure, sump level and flow indicators discussed below are monitored regularly. Any instrument indication of abnormal leakage will be investigated.

(2) Reactor Building Floor and Equipment Drain Sump Monitoring

Reactor building equipment drain sumps collect the identified leakage from known sources from within enclosed equipment areas. Leakage from unknown or unidentified sources (e.g., RHR Shutdown Cooling System piping, CUW System piping, process instrumentation piping or CRD HCU unit piping) is collected in several R/B floor drain sumps. The number of pumps and the instrumentation used for monitoring both the R/B floor and equipment and equipment drain sumps, are similar to those used for monitoring the drywell floor drain sump as described in Subsections 5.2.5.2.1(1) and 5.2.5.2.1(2). The R/B and equipment drain sump monitoring channels measure sump levels and sump fillup and pumpout times and initiate alarms when setpoints are exceeded.

(3) Reactor Water Cleanup System Differential Flow Monitoring

The suction and discharge flows of the Reactor Water Cleanup (CUW) System are monitored for flow differences between that coming from the reactor and that returning to the reactor or to the main condenser. Temperature compensated flow differences greater than preset values cause alarm and isolation. Bypass time delay interlocks are provided for delaying the isolation signals and prevent isolation initiation during normal CUW surge conditions. Flow in the CUW suction line from the reactor and in the CUW return lines to the reactor and in the blowdown line to the radwaste system is monitored by 12 differential flow transmitters (four for each line). CUW flow measurements are taken as close to the reactor vessel as possible to maximize the degree of coverage of the LDS channels. The outputs of the flow transmitters in the suction line are compared with the outputs from the discharge lines, and alarms in the control room and isolation signals are initiated when higher flow out of the reactor vessel indicates that leaks equal to the established leak rate limits for alarm or isolation may exist. Net flow indication readout is provided in the control room.

(4) Main Steamline Area Temperature Monitors

High temperature in the main steamline tunnel area is detected by thermocouples. Four thermocouples are used for measuring main steam tunnel ambient temperatures and are located in the area of the main steamlines tunnel area. All temperature elements are located or shielded so as to be sensitive to air temperatures and not to the radiated heat from hot equipment. High ambient temperatures will alarm in the control room and provide signals to close the main steamline and MSL drain line isolation valves, and the CUW isolation valves. High ambient temperature in the steam tunnel area can also indicate leakage from the reactor feedwater piping or equipment within the tunnel. Isolation of the feedwater lines, if necessary, may be accomplished by manual closure by the operator of valves located in the feedwater lines in the steam tunnel. Monitoring of the main steamline area outside the steam tunnel and before the inlet to the turbine is provided with sufficient ambient temperature sensors to cover the full length of the steamlines in the turbine building.

The channel signals are combined so as to provide the four divisional trip signals used as inputs to the LDS isolation logic for closure of the MSIVs and MSL drain lines. High ambient T/B temperatures (main steamline areas) will also be indicated in the control room. The T/B temperature elements are located so as not to be sensitive to radiated heat from hot equipment.

(5) Temperature Monitors in Equipment Areas

Dual element thermocouples are installed in the RCIC, RHR and CUW equipment rooms for sensing high ambient temperature in these areas. These elements are located or shielded so that they are sensitive to air temperature only and not to radiated heat from hot equipment. Four ambient temperature channels are provided

in each equipment area. Each of the four channels drive voting logic in two divisions (three divisions for RHR), which provides an alarm signal and a trip signal for that division's isolation logic to close the respective system isolation valves.

(6) Not Used

(7) RCIC Steamline Pressure Monitors

Pressure in the RCIC steamline is monitored by LDS instruments to provide RCIC turbine shutoff and closure of the RCIC isolation valves on low steamline pressure as a protection for the RCIC turbine. This steamline pressure is monitored by four pressure transmitters, each connected to one tap of the two elbows used for RCIC steam flow measurement, and upstream of the RCIC steamline isolation valves (Subsection 5.2.5.2.1(13)). Low pressure is alarmed in the control room and low pressure isolation signals close the same RCIC valves as those closed by the RCIC steam flow monitoring instruments.

(8) RCIC Turbine Exhaust Line Diaphragm Pressure Monitors

Pressure between the rupture disk diaphragms in the RCIC System turbine exhaust vent line is monitored by four channels of pressure instrumentation. The instrumentation channel equipment and piping are provided by the RCIC System as an interface to the LDS. The two logic channels of Division I trip on high pressure to close the inboard RCIC isolation valves, and the channels of Division II trip to close the outboard isolation valves. Either divisional logic channel will also trip the turbine.

(9) Main Steamline Low Pressure Monitoring

Main steamline low pressure is monitored by four pressure transmitters (one in each line) that sense the pressure downstream of the outboard MSIVs. The sensing points are located as close as possible to the turbine stop valves. Low steamline pressure at the points monitored can be an indication of an excessive steamline leak or a malfunction of the Reactor Pressure Control System. The transmitters are provided by the Nuclear Boiler System. The LDS will automatically initiate closure of all MSIVs and the MSL drain valves if pressure at the turbine end of the main steamlines decreases below a preselected value when the reactor mode switch is in the "RUN" position.

(10) Main Condenser Low Vacuum Monitoring

Low main condenser vacuum could indicate that primary reactor coolant is being lost through the main condenser. Four channels of main condenser pressure monitoring are provided by the Nuclear Boiler System. The LDS utilizes the low vacuum signals to trip the MSIV logic on low condenser vacuum and close all MSIVs and the MSL

drain valves. The condenser vacuum trip signals can be bypassed by a manual keylocked bypass switch in the control room during startup and shutdown operations.

(11) Intersystem Leakage Monitoring

Radiation monitors are used to detect reactor coolant leakage into the Reactor Building Cooling Water (RCW) System, which supplies coolant water to the (1) RHR heat exchangers, (2) the reactor internal pumps (RIPs) heat exchangers, (3) the CUW non-regenerative heat exchangers, and (4) the fuel pool cooling heat exchangers. One process sensing channel is provided in each of the three RCW loops to monitor for radiation due to coolant leakage into the RSW. Each channel will alarm on high radiation conditions, indicating process leakage into the RCW System. The PRMS provides the monitoring of this variable. No isolation trip functions are performed by these monitors. Potential intersystem leakage from the RCPB to RCIC, RHR or HPCF System is discussed in response to Question 430.2c.

(12) Large Leaks External to the Drywell

The main steamline high flow monitoring, the reactor vessel low water level monitoring and the RCIC steamline flow monitoring (Subsection 5.2.5.2.1, Paragraphs 11, 12 and 13) can also indicate large leaks from the reactor coolant piping external to the drywell.

5.2.5.2.3 Summary

Tables 5.2-6 and 5.2-7 summarize the actions taken by each leakage detection function. Table 5.2-6 shows that those systems which detect gross leakage initiate immediate automatic isolation action to terminate the gross leakage or minimize loss of reactor coolant. The systems which are capable of detecting small leaks initiate an alarm in the control room as shown in Table 5.2-7. In addition, Table 5.2-6 shows that two or more leakage detection methods are provided for each system or area that is a potential source of leakage. Plant operating procedures will dictate the action an operator is to take upon receipt of an alarm from any of these systems. The operator can manually isolate the violated system or take other appropriate action.

A time delay is provided for CUW differential flow isolation signals to prevent system isolation during CUW surges.

The LDS is a four-divisional channel which is redundantly designed so that failure of any single element within a channel will not interfere with a required detection of leakage or a required isolation. In the four-division LDS, where inadvertent isolation could impair plant performance (e.g., closure of the MSIVs), any single channel or divisional component malfunction will not cause a false indication of leakage and will not cause a false isolation trip. Only one of the four channels will trip and two or more channels are required to trip in order to cause closure of the main steamline isolation valves. The LDS thus combines a very high probability of operating

when needed with a very low probability of operating falsely. The system is testable during plant operation.

5.2.5.3 Indication in the Control Room

Leak detection methods are discussed in Subsection 5.2.5.1. Details of some of the LDS alarms, recordings and other indications in the control room are discussed in Subsections 5.2.5.1.1, 5.2.5.1.2, 5.2.5.2.1 and 5.2.5.2.2. Further details of the LDS control room indications are included in Subsection 7.3.1.1.2.

5.2.5.4 Limits for Reactor Coolant Leakage

5.2.5.4.1 Total Leakage Rate

The total reactor coolant leakage rate consists of all leakage (identified and unidentified) that flows to the drywell floor drain and equipment drain sumps. The total leakage rate limit is well within the makeup capability of the RCIC System (182 m^3/h). The total reactor coolant leakage rate limit is established at 114 liters/min.

The total leakage rate limit is established low enough to prevent overflow of the sumps. The equipment drain sumps and the floor drain sumps, which collect all leakage, are each pumped out by two 10 m^3/h pumps.

If either the total or unidentified leak rate limit is exceeded, an orderly shutdown shall be initiated and the reactor shall be placed in a cold shutdown condition within 36 hours.

5.2.5.4.2 Identified Leakage Inside Drywell

The reactor vessel head flange seal and other seals in systems that are part of the reactor coolant pressure boundary, and from which normal design identified source leakage is expected, are provided with leakoff drains. The reactor vessel head flange is equipped with double seals. The leakage from the reactor vessel head flange inner seal, which discharges to the drywell equipment drain sump, is measured during plant operation. Leakage from the main steam SRVs, discharging to the suppression pool, is monitored by temperature sensors mounted in thermowells in the individual SRV exhaust lines. The thermowells are located several feet from the valve bodies so as to prevent false indication. These temperature sensors transmit signals to the control room for monitoring. Any temperature increase detected by these sensors, that is above the ambient temperatures, indicates SRV leakage.

5.2.5.5 Unidentified Leakage Inside the Drywell

5.2.5.5.1 Unidentified Leakage Rate

The unidentified leakage rate is the portion of the total leakage rate received in the drywell sumps that is not identified as previously described. The unidentified leakage rate limit must be low because of the possibility that most of the unidentified leakage rate might be emitted from a break in the nuclear system process barrier.

An allowance for leakage that does not compromise barrier integrity and is not identifiable is established for normal plant operation.

The unidentified leakage rate limit is established at 19 liters/min to allow time for corrective action before the process barrier could be significantly compromised.

5.2.5.5.2 Margins of Safety

The margins of safety for a detectable flaw to reach critical size are presented in Subsection 5.2.5.5.3.

5.2.5.5.3 Criteria to Evaluate the Adequacy and Margin of Leak Detection System

For process lines that are normally open, there are at least two different methods of detecting abnormal leakage from each system comprising the nuclear system process barrier, located both inside the primary containment (drywell) and external to the drywell, in the reactor building the steam tunnel and the turbine building (Tables 5.2-6 and 5.2-7). The instrumentation is designed so it can be set to provide alarms at established leakage rate limits and isolate the affected system if necessary. The alarm points are determined analytically or based on measurements of appropriate parameters made during startup and preoperational tests.

The unidentified leakage rate limit is based, with an adequate margin for contingencies, on the crack size large enough to propagate rapidly.

The established limit is sufficiently low so that, even if the entire unidentified leakage rate were coming from a single crack in the nuclear system process barrier, corrective action could be taken before the integrity of the barrier would be threatened.

The Leak Detection and Isolation System (LDS) will satisfactorily detect unidentified leakage of 3.785 liters/min within one hour in the drywell.

5.2.5.6 Differentiation Between Identified and Unidentified Leaks

Subsection 5.2.5.1 describes the leak detection methods utilized by the LDS. The ability of the LDS to differentiate between identified and unidentified leakage is discussed in Subsections 5.2.5.4 and 5.2.5.5.

5.2.5.7 Sensitivity and Operability Tests

Sensitivity, including sensitivity tests and response time of the LDS, and the criteria for shutdown if leakage limits are exceeded are covered in Subsections 5.2.5.1.1, 5.2.5.1.2, 5.2.5.2.1(1) and 7.3.1.1.2.

Testability of the LDS is contained in Subsection 7.3.1.1.2(10).

5.2.5.8 Testing and Calibration

Provisions for testing and calibration of the LDS are covered in Chapter 14.

5.2.5.9 Regulatory Guide 1.45: Compliance

This regulatory guide is prescribed to assure that leakage detection and collection systems provide maximum practical identification of leaks from the RCPB.

Leakage is separated into identified and unidentified categories and each is independently monitored, thus meeting Position C.1 requirements.

Leakage from unidentified sources from inside the drywell is collected into the floor drain sump and monitored with an accuracy better than 3.785 liters/min within one hour thus meeting Position C.2 requirements.

By monitoring (1) floor drain sump fillup and pumpout rate, (2) airborne particulates, and (3) air coolers condensate flow rate, Position C.3 is satisfied.

Monitoring of the R/B cooling water heat exchanger coolant return lines for radiation due to leaks within the RHR, RIP, CUW and the Fuel Pool Cooling System heat exchangers satisfies Position C.4 (see Subsection 7.6.1.2 for details).

The floor drain sump monitoring, air particulates monitoring, and air cooler condensate monitoring are designed to detect leakage rates of 3.785 liters/min within one hour, thus meeting Position C.5 requirements.

The fission products monitoring subsystem is qualified for SSE. The containment floor drain sump monitor, air cooler, and condensate flow meter are qualified for SSE, thus meeting Position C.6 requirements.

Leak detection indicators and alarms are provided in the main control room, thus satisfying Position C.7 requirements. Procedures and graphs will be provided by the COL applicant to plant operators for converting the various indicators to a common leakage equivalent, when necessary, thus satisfying the remainder of Position C.7 (see Subsection 5.2.6.1 for COL license information). The LDS is equipped with provisions to permit testing for operability and calibration during the plant operation using the following methods:

(1) Simulation of trip signal.

(2) Comparing channel to channel of the same leak detection method (i.e., area temperature monitoring).

(3) Operability checked by comparing one method versus another (i.e., sump fillup rate versus pumpout rate and particulate monitoring or air cooler condensate flow versus sump fillup rate).

(4) Continuous monitoring of floor drain sump level, and a source of water for calibration and testing is provided.

These satisfy Position C.8 requirements.

Limiting unidentified leakage to 19 liters/min and identified total leakage to 114 liters/min satisfies Position C.9.

5.2.6 COL License Information

5.2.6.1 Conversion of Indications

Procedures and graphs will be provided by the COL applicant to operations for converting the various indicators into a common leakage equivalent (Subsection 5.2.5.9).

5.2.6.2 Plant-Specific ISI/PSI

COL applicants will submit the complete plant-specific ISI/PSI program. Each applicant will submit or address the following:

(1) The PSI program should include reference to the edition and addenda of ASME Code Section XI that will be used for selecting of components for examinations, lists of the components subject to examination, a description of the components exempt from examination by the applicable code, and isometric drawings used for the examination.

(2) Submit plans for preservice examination of the reactor pressure vessel welds to address the degree of compliance with Regulation Guide 1.150.

(3) Discuss the near-surface examination and resolution with regard to detecting service-induced flaws and the use of electronic gating as related to the volume of material near the surface that is not being examined. Discuss how the internal surfaces (e.g., inner radius of a pipe section and reactor vessel internals) will be examined.

(4) Submit an acceptable resolution of the information requested regarding the ISI/PSI program.

(5) Submit all relief requests, if needed, with a supporting technical justification.

5.2.6.3 Reactor Vessel Water Level Instrumentation

The COL applicant will design the reactor vessel water level instrumentation flow control system to provide flow rates determined by the results of the BWR Owners group testing. (See Subsection 5.2.5.2.1(12)).

5.2.7 References

5.2-1 D.A. Hale, “The Effect of BWR Startup Environments on Crack Growth in Structural Alloys”, Trans. of ASME, Vol. 108, January 1986.

5.2-2 F.P. Ford and M. J. Povich, “The Effect of Oxygen/Temperature Combinations on the Stress Corrosion Susceptibility of Sensitized T-304 Stainless Steel in High Purity Water”, Paper 94 presented at Corrosion 79, Atlanta, GA, March 1979.

5.2-3 “BWR Normal Water Chemistry Guidelines: 1986 Revision”, EPRI NP-4946-SR, July 1988.

5.2-4 B.M. Gordon, “The Effect of Chloride and Oxygen on the Stress Corrosion Cracking of Stainless Steels: Review of Literature”, Material Performance, NACE, Vol. 19, No. 4, April 1980.

5.2-5 W.J. Shack, et al, “Environmentally Assisted Cracking in Light Water Reactors: Annual Report, October 1983 - September 1984”, NUREG/CR-4287, ANL-85-33, June 1985.

5.2-6 D.A. Hale, et al, “BWR Coolant Impurities Program”, EPRI, Palo Alto, CA, Final Report on RP2293-2.

5.2-7 K.S. Brown and G.M. Gordon, “Effects of BWR Coolant Chemistry on the Propensity of IGSCC Initiation and Growth in Creviced Reactor Internals Components”, paper presented at the Third International Symposium of Environmental Degradation of Materials in Nuclear Power Systems, ANS-NACE-TMS/AIME, Traverse City, MI, September 1987.

5.2-8 B.M. Gordon et al, “EAC Resistance of BWR Materials in HWC”, Predeeding of the Second International Symposium of Environmental Degradation of Materials in Nuclear Power Systems, ANS, LaGrange Park, IL 1986.

5.2-9 BWR Hydrogen Water Chemistry Guidelines: 1987 Revision EPRI NP-4947-SR, December 1988.

5.2-10 Guideline for Permanent BWR Hydrogen Water Chemistry Installations: 1987 Revision, EPRI NP-5203-SR-A.

5.2-11 M.Kodama, et al, “IASCC Susceptibility of Irradiated Austenitic Stainless Steel under Very Low Dissolved Oxygen”, Seventh International Symposium on Environmental Degradation of Materials in Nuclear Power systems in Breckenridge, p-1121.

5.2-12 B.M. Gordon et al, “Hydrogen Water Chemistry for BWRs- Materials Behavior”, EPRI NP-5080, Palo Alto, CA, March 1987.

5.2-13 M.Kodama, et al, "Effects of Fluence and Dissolved Oxygen on IASCC in Austenitic Stainless Steels", Fifth International Symposium on Environmental Degradation of Materials in Nuclear Power systems in Monterey, p-948.

Table 5.2-1 Reactor Coolant Pressure Boundary Components Applicable Code Cases

Number	Title	Applicable Equipment	Remarks
[N-60-5	*(33)*	*Core Support]**	Accepted per RG 1.84
[N-71-18	*(1)*	*Component Support]**	Conditionally Accepted per RG 1.84
*[N-122-*2	*(2)*	*Piping]**	Accepted per RG 1.84
[N-247	*(3)*	*Component Support]**	Accepted per RG 1.84
[N-249-14	*(4)*	*Component Support]**	Conditionally Accepted per RG 1.84
[N-309-1	*(5)*	*Component Support]**	Accepted per RG 1.84
[N-313	*(6)*	*Piping]**	Accepted per RG 1.84
[N-316	*(7)*	*Piping]**	Accepted per RG 1.84
[N-318-5	*(8)*	*Piping]**	Accepted per RG 1.84
[N-319-3	*(9)*	*Piping]**	Accepted per RG 1.84
[N-391-2	*(10)*	*Piping]**	Accepted per RG 1.84
[N-392-3	*(11)*	*Piping]**	Accepted per RG 1.84
[N-393	*(12)*	*Piping]**	Accepted per RG 1.84
[N-411-1	*(13)*	*Piping]**	Conditionally Accepted per RG 1.84
[N-414	*(14)*	*Component Support]**	Accepted per RG 1.84
[N-430	*(15)*	*Component Support]**	Accepted per RG 1.84
N-236-1	(16)	Containment	Conditionally Accepted Per RG 1.147
N-307-2	(17)	RPV Studs	Accepted per RG 1.147
N-416-3	(20)	Piping	Accepted Per RG 1.147
N-432	(21)	Class 1 Components	Accepted Per RG 1.147
N-435-1	(22)	Class 2 Vessels	Accepted Per RG 1.147
N-457	(23)	Bolt and Studs	Accepted Per RG 1.147
N-463-1	(24)	Piping	Accepted Per RG 1.147
N-460	(25)	Class 1 & 2 Components and Piping	Accepted Per RG 1.147
N-472	(26)	Pumps	Accepted Per RG 1.147
[N-476	*(26a)*	*Component Support]**	Accepted per RG 1.84
N-479-1	(27)	Main Steam System	Accepted Per RG 1.147
N-491	(28)	Component Supports	Accepted Per RG 1.147

Table 5.2-1 Reactor Coolant Pressure Boundary Components Applicable Code Cases

Number	Title	Applicable Equipment	Remarks
N-496	(29)	Bolts and Studs	Accepted Per RG 1.147
N-580-2	(30)	RPV, Reactor Internals, etc.	Approved by ASME Standards Committee (2008)
N-608	(31)	Use of Applicable Code and Addenda, NCA-1140(a)(2)	Accepted per RG 1.84
N-613-1	(32)	Reactor Vessel	Accepted per RG 1.147
N-632	(34)	Containment	Accepted per RG 1.84

Table 5.2-1a Reactor Coolant Pressure Boundary Components Applicable Code Cases

[(1)	*Additional Materials for Subsection NF, Classes 1, 2, 3 and MC Component Supports Fabricated by Welding, Section III, Division 1.]**
[(2)	*Stress Indices for Structure Attachments, Class 1, Section III, Division 1.]**
[(3)	*Certified Design Report Summary for Components Standard Supports, Section III, Division 1, Classes 1, 2, 3 and MC.]**
[(4)	*Additional Material for Subsection NF, Classes 1, 2, 3 and MC Component Supports Fabricated Without Welding, Section III, Division I.]**
[(5)	*Identification of Materials for Component Supports, Section III, Division 1.]**
[(6)	*Alternate Rules for Half-Coupling Branch Connections, Section III, Division 1.]**
[(7)	*Alternate Rules for Fillet Weld Dimensions for Socket Welded Fittings, Section III, Division 1, Classes 1, 2, 3.]**
[(8)	*Procedure for Evaluation of the Design of Rectangular Cross Section Attachments on Class 2 or 3 Piping, Section III, Division 1.]**
[(9)	*Alternate Procedure for Evaluation of Stress in Butt Weld Elbows in Class 1 Piping, Section III, Division 1.]**
[(10)	*Procedure for Evaluation of the Design of Hollow Circular Cross Section Welded Attachments on Class 1 Piping, Section III, Division 1.]**
[(11)	*Procedure for Evaluation of the Design of Hollow Circular Cross Section Welded Attachments on Classes 2 and 3 Piping, Section III, Division 1.]**
[(12)	*Repair Welding Structural Steel Rolled Shapes and Plates for Component Supports, Section III, Division 1.]**
[(13)	*Alternative Damping Values for Seismic Analysis of Classes 1, 2, 3 Piping Sections, Section III, Division 1.]**
[(14)	*Tack Welds for Class 1, 2, 3 and MC Components and Piping Supports.]**
[(15)	*Requirements for Welding Workmanship and Visual Acceptance Criteria for Class 1, 2, 3 and MC Linear-Type and Standard Supports.]**
(16)	Repair and Replacement of Class MC Vessels.
(17)	Revised Examination Volume for Class 1 Bolting, Table IWB-2500-1, Examination Category B-G-1, When the Examinations Are Conducted from the Drilled Hole.
(18)	Not Used
(19)	Not Used
(20)	Alternative Rules for Hydrostatic Testing of Repair or Replacement of Class 2 Piping.
(21)	Repair Welding Using Automatic Or Machine Gas Tungsten-Arc Welding (GTAW) Temperbead Technique.
(22)	Alternative Examination Requirements for Vessels With Wall Thicknesses 2 in. or Less.
(23)	Qualification Specimen Notch Location for Ultrasonic Examination of Bolts and Studs.

Table 5.2-1a Reactor Coolant Pressure Boundary Components Applicable Code Cases (Continued)

(24)	Evaluation Procedures and Acceptance Criteria for Flaws in Class 1 Ferritic Piping That Exceed the Acceptance Standards of IWB-3514-2.
(25)	Alternative Examination Coverage for Class 1 and 2 Welds.
(26)	Use of Digital Readout and Digital Measurement Devices for Performing Pump Vibration Testing.
[*(26a)*	*Class 1, 2, 3, and MC Linear Component Supports—Design Criteria for Single Angle Members Section III, Division I, Subsection NF; SUPP. 1 — NC, May 6, 1989*]*
(27)	Boiling Water Reactor (BWR) Main Steam Hydrostatic Test.
(28)	Alternate Rules for Examination of Class 1, 2, 3 and MC Component Supports of Light-Water-Cooled Power Plants.
(29)	Helical-Coil Threaded Inserts, Section XI, Div. 1.
(30)	Use of Alloy 600 (UNS N066000) with Columbium added, Section III, Div. 1 (SC III File #N96-44) (MC97-86)
(31)	Applicable Code Edition and Addenda, NCA-1140(a)(2), Section III, Division 1
(32)	UT Exam of Penetration Nozzles in Vessels, Category B-D, Item Nos. B3.10 and B3.90, Reactor Nozzle to Vessel Welds, Figs. IWB 2500-7(a), (b), (c), Section XI, Division 1
(33)	Material for Core Support Structures, Section III, Division 1
(34)	Use of ASTM A 572, Grades 50 and 65 for Structural Attachments to Class CC Containment Liners, Section III, Division 2.

* See Subsection 3.9.1.7. The change restriction is limited to the edition of Code Cases in application only to the design of piping and piping supports.

Table 5.2-2 Systems Which May Initiate During Overpressure Event

Systems	Initiating/Trip Signal*
Reactor Protection	Reactor shutdown on high flux
RCIC	ON when reactor water level is at L2
	OFF when reactor water level is at L8
Recirculation System	Four pumps OFF when reactor water level is at L3
	Remaining six pumps OFF when reactor water level is at L2
	Four pumps (the same four tripped at L3)
	OFF when reactor pressure is at 7.76 MPaG
CUW	OFF when reactor water level is at L2
HPCF	ON when reactor water level is at L1.5

* Vessel level trip settings (Figure 5.1-3, Tables 2 and 3).

Table 5.2-3 Nuclear System Safety/Relief Valve Setpoints Set Pressures and Capacities

Number of Valves*	Spring Set Pressure (MPaG)	ASME Rated Capacity at 103% Spring Set Pressure (kg/h each)	Relief Pressure Set Pressure (MPaG)
1	7.92	395,000	7.51
1	7.92	395,000	7.58
4	7.99	399,000	7.65
4	8.06	402,000	7.72
4	8.13	406,000	7.79
4	8.20	409,000	7.85

* Eight of the SRVs serve in the automatic depressurization function.

Table 5.2-4 Reactor Coolant Pressure Boundary Materials

Component	Form	Material	Specification (ASTM/ASME)
Main Steam Isolation Valves			
Valve Body	Cast	Carbon steel	SA352 LCB
Cover	Forged	Carbon Steel	SA350LF2
Poppet	Forged	Carbon Steel	SA350LF2
Valve stem	Rod	Precipitation Hardened Stainless Steel	SA 564 630 (H1100)
Body bolt	Bolting	Low-Alloy steel	SA 540 B23 CL4 or 5
Hex nuts	Bolting Nuts	Low-Alloy steel	SA 194 GR7
Main Steam Safety/Relief Valve			
Body	Forging or Casting	Carbon steel Carbon steel	ASME SA 350 LF2ASME SA 352 LCB
Bonnet (yoke)	Forging or Casting	Carbon steel Carbon steel	ASME SA 350 LF2 ASME SA 352 LCB
Nozzle (seat)	Forging or Casting	Stainless steel or Carbon steel	ASME SA 182 Gr F316 or SA351 CF3 or CF 3M ASME SA 350 LF2 or SA 352 LCB
Body to bonnet stud	Bolting	Low-Alloy steel	ASME SA 193 Gr B7
Body to bonnet nut	Bolting Nuts	Low-Alloy steel	ASME SA 194 Gr 7
Disk	Forging or Casting	NiCrFe Alloy Stainless steel	ASME SB 637 Gr 718 ASME SA 351 CF 3A
Spring washer &	Forging	Carbon steel	ASME SA 105
Adjusting Screw or	Bolting	Alloy steel	ASME SA 193 Gr B6 (Quenched + tempered or normalized & tempered)
Setpoint adjustment assembly	Forgings	Carbon and alloy steel parts	Multiple specifications
Spindle (stem)	Bar	Precipitation-hardened stainless steel	ASTM A564 Type 630 (H 1100)
Spring	Wire or Bellville washers	Steel Alloy Steel	ASTM A304 Gr 4161 N 45 Cr Mo V67
Main Steam Piping (between RPV and the turbine stop valve)			
Pipe	Seamless	Carbon steel	ASME SA 333 Gr. 6
Contour nozzle 250A 10.36 MPaG	Forging	Carbon steel	ASME SA 350 LF 2

Table 5.2-4 Reactor Coolant Pressure Boundary Materials (Continued)

Component	Form	Material	Specification (ASTM/ASME)
Large groove flange	Forging	Carbon steel	ASME SA 250 LF 2
50A special nozzle	Forging	Carbon steel	ASME SA 350 LF2
Elbow	Seamless	Carbon steel	ASME SA 420
Head fitting/penetration piping	Forging	Carbon steel	ASME SA 350 LF2
Feedwater Piping (between RPV and the seismic interface restraint)			
Pipe	Seamless	Carbon steel	ASME SA 333 Gr. 6
Elbow	Seamless	Carbon steel	ASME SA 420
Head fitting/penetration piping	Forging	Carbon steel	ASME SA 350 LF2
Nozzle	Forging	Carbon steel	ASME SA 350 LF2
Recirculation Pump Motor Cover			
Bottom flange (cover)	Forging	Low-Alloy steel	ASME SA 533 Gr. B Class 1 or SA 508 Class 3
Stud	Bolting	Low-Alloy steel	ASME SA 540 CL.3 Gr.B24 or SA 193, B7
Nut	Bolting Nuts	Low-Alloy steel	ASME SA 194 Gr. 7
CRD			
Middle flange	Forging	Stainless steel	SA 182/182M, F304L, F304*, F316L or F316*, or SA 336/336M, F304* or F316*
Spool piece	Forging	Stainless steel	SA 182/182M, F304L, F304*, F316L or F316*, or SA 336/336M, F304* or F316*
Mounting bolts	Bolting	Low-Alloy steel	SA-193/193M, Grade B7
Seal housing	Forging	Stainless steel	SA 182/182M, F304L, F304*, F316L or F316*,or SA 336/336M, F304* or F316*
Seal housing nut	Bar	Stainless steel	SA 564, 630 (H1100)

Table 5.2-4 Reactor Coolant Pressure Boundary Materials (Continued)

Component	Form	Material	Specification (ASTM/ASME)
Reactor Pressure Vessel			
Shells and Heads	Plate Forging	Low-Alloy steel Low alloy steel	SA-533, Type B, Class 1 SA-508, Class 3
Shell and Head Flange	Forging	Low alloy steel	SA-508 Class 3
Flanged Nozzles	Forging	Low alloy steel	SA-508 Class 3
Drain Nozzles	Forging	Carbon steel or Stainless Steel	SA-508 Class 1 or SA 182, F316L* or F316*, SA-336, F316*
Appurtenances/ Instrumentation Nozzles	Forging	Stainless steel	SA-182, Grade F316L*or F316* or SA-336, Class F316*
	Bar, Smls. Pipe	Ni-Cr-Fe (UNS N06600)	Code Case N-580-2
Stub Tubes	Forging	Ni-Cr-Fe (UNS N06600)	Code Case N-580-2
	Bar, Smls. Pipe	Ni-Cr-Fe (UNS N06600)	Code Case N-580-2

* Carbon content is maximum 0.020%.

Table 5.2-5 BWR Water Chemistry

	Concentrations* Parts Per Billion (ppb)					Conductivity	
	Iron	**Copper**	**Chloride**	**Sulfate**	**Oxygen†**	**μS/cm at 25°C**	**pH at 25°C**
Condensate Treatment Effluent and Feedwater	<5	<0.2	—	—	30 - 200	<0.065	
Reactor Water							
(a) Normal Operation			<5	<5	=	<0.3	~7
(b) Shutdown			<100	<100	-	<2.0	5.3 - 8.6
Control Rod Drive Cooling Water	<5	<0.2			30 - 100	≤0.065	

* These values of a parameter represent values beyond which long-term system reliability will be threatened. If a parameter exceeds these values, restore the parameter below these values.

† Some revision of oxygen values may be established after hydrogen water chemistry has been established

Table 5.2-6 LDS Control and Isolation Function vs. Monitored Process Variables

LDS Control & Isolation Functions / Monitored Variables	Reactor Water Level Low	Turbine Inlet SL Press Low	Reactor Pressure High	MSL Flow Rate High		MSL Tunnel Amb. Temp High	Turbine Area Amb. Temp High	Main Condenser Vacuum Low	Drywell Pressure High	RHR Equip Area Temp High	RCIC Equip Area Temp High	RCIC SL Pressure Low	RCIC SL Flow Rate High	RCIC Vent Exhaust Press High	CUW Equip Area Temp HIgh	CUW Differential Flow High	SLCS Pumps Running	LCW Drain Line Radiation High	HCW Drain Line Radiation High	R/B HVAC Exhaust Air Rad High	F/H Exhaust Air Rad High	FW Line Pressure Difference
MSIVs & MSL Drain Line Valves	L1.5	X		X		X	X	X														
CUW Process Lines Isolation	L2		X*			X									X	X	X					
RHR S/C PCV Valves	L3		X							X												
RCIC Steamline Isolation											X	X	X	X								
ATIP Withdrawal	L3								X													
DW RAD Sampling Isolation	L2								X													
SPCU Process Line Isolation	L3								X													
DW LCW Sump Drain Line Isolation	L3								X									X				
DW HCW Sump Drain Line Isolation	L3								X										X			
RCW PCV Valves Isolation	L1								X													
HNCW PCV Valves Isolation	L1								X													
AC System P&V Valves Isolation	L3								X											X	X	
R/B HVAC Air Ducts Isolation	L3								X											X	X	
SGTS Initiation	L3								X											X	X	
Condensate Pump Trip**									X													X

* Head spray valve only
** Both signals must be present

Table 5.2-7 Leakage Sources vs. Monitored Trip Alarms

Leakage Source \ Monitored Plant Variable	Location		Reactor Vessel Water Level Low	Drywell Pressure High	DW Floor Drain Sump High Flow	DW Equip Drain Sump High Flow	DW Fission Products Radiation High	Drywell Temperature High	SRV Discharge Line Temperature High	Vessel Head Flange Seal Pressure High	RB Eq/Fl Drain Sump High Flow	DW Air Cooler Condensate Flow High	MSL or RCIC Steamline Flow High	MSL Tunnel or TB Ambient Area Temp High	Equip Areas Ambient or Diff Temp High	CUW Differential Flow High		Inter-System Leakage (Radiation) High	Feedwater Line Differential Pressure High
Main Steamlines	I		X	X	X		X	X	X			X	X						
		O	X								X		X	X	X				
RCIC Steamline	I		X	X	X		X	X				X	X						
		O	X								X		X		X				
RCIC Water	I																		
		O									X				X				
RHR Water	I		X	X	X		X	X				X							
		O									X				X			Ⓧ	
HPCF Water	I		X	X	X		X	X				X							
		O									X				X				
CUW Water	I		X	X	X		X	X				X							
		O	X								X			X	X	X		Ⓧ	
Feedwater	I		X	X	X		X	X				X							X
		O									X			X					
Recirc Pump Motor Casing	I			X	X		X	X				X						Ⓧ	
		O																	
Reactor Vessel Head Seal	I					X				X									
		O																	
Valve Stem Packing	I					X													
		O									X								
Miscellaneous Leaks	I				X			X											
		O									X							Ⓧ	

I = Inside Drywell Leakage

O = Outside Drywell Leakage

Ⓧ= Reactor coolant leakage in cooling water to RHR Hx, RIP Hx, CUW Non-regen Hx's or to FP cooling Hx.

Table 5.2-8 Examination Categories

Quality Group	System Number	System Title	System Description	P&ID Diagram	Sec. XI Exam Cat.	Items Examined	Exam method
A	B11/B21	Reactor Pressure Vessel/ Nuclear Boiler	Reactor Pressure Vessel	Figure 5.1-3			
			Vessel Shell Welds		B-A	Welds	UT (Note 7)
			Vessel Head Welds		B-A	Welds	UT (Note 7)
			Shell-to-Flange Weld		B-A	Weld	UT
			Head-to-Flange Weld		B-A	Weld	UT, MT
			Nozzles for: Main Steam, Feedwater, SD Outlet, CCS (Fldg.) & SD Inlet, SD - CUW SD Outlet, CCS (Spray) & SD Inlet		B-D	Welds, Inner Radius	UT
			CRD Housing to Middle Flange and Niddle Flange to Spool Piece Bolting		B-G-2	Bolts	VT-1
			Nozzles for CRD, RIP & Instrumentation		B-E	External Surfaces	VT-2 (Note 8)
			Closure Head Nuts		B-G-1	Nuts	MT
			Closure Studs		B-G-1	Studs	UT, MT (Note 9)
			Threads in Flange		B-G-1	Threads	UT
			Closure Washers, Bushings		B-G-1		VT-1
			Reactor Pressure	Figure 5.1-3			
			Vessel Integral Attachments		B-H	Welds	UT or MT (Note 10)
			Vessel Interior		B-N-1	Vessel	VT-3 (Note 11)
			Interior Attachment Welds Within Beltline Region		B-N-2	Welds	VT-1 (Note 12)
			Interior Attachment Welds Beyond Beltline Region		B-N-2	Welds	VT-3 (Note 12)

Table 5.2-8 Examination Categories (Continued)

Quality Group	System Number	System Title	System Description	P&ID Diagram	Sec. XI Exam Cat.	Items Examined	Exam method
A	B21	Nuclear Boiler	Main steamlines A,B,C,D from RPV up to and including SRVs F0I0A thru U and outboard MSIVs F009A B, C & D	Figure 5.1-3			
			Lines 700A-NB-023,-25, - 27, -29, Piping		B-J	Welds (Note 1)	UT,MT
			MSIV F009A,B,C,D F008A,B,C,D		B-M-1	Valve Body (Note 2)	UT
			MSIV F009A,B,C,D F008A,B,C,D		B-M-2	Valve Body Internal Surfaces (Note 3)	VT-3
			Safety/Relief Valves F0I0, A through H F0I0, J through N F0I0 P F0I0 R through U		B-M-2	Valve Body Internal Surfaces (Note 3)	VT-3
			All pressure-retaining components and Piping		B-P	External Surfaces (Note 4)	VT-2
			Integral Attachments		B-K-1	Welds	UT or MT (Note 5)
			Piping and component Supports		F-A	Supports (Note 13)	VT-3
			Bolting		B-G-2	Bolts, Nut & Stud (Note 6)	VT-1
			Main steamlines A,B,C,D drain lines from inboard MSIVs F008A,B,C,D inlet up to and including outboard drain valve F012A,B,C,D	Figure 5.1-3			

Table 5.2-8 Examination Categories (Continued)

Quality Group	System Number	System Title	System Description	P&ID Diagram	Sec. XI Exam Cat.	Items Examined	Exam method
A	B21	Nuclear Boiler (Continued)	Piping		B-J	Welds (Note 1)	MT
			Valves		B-M-2	Internal Surfaces (Note 3)	VT-3
			All pressure retaining components and piping		B-P	External Surfaces (Note 4)	VT-2
			Component and piping supports		F-A	Supports (Note 13)	VT-3
			Integral attachments		B-K-1	Welds	UT or MT (Note 5)
			Bolting		B-G-2	Bolts, Nuts & Studs (Note 6)	VT-1
			Head vent line from RPV nozzle up to and including warmup line to main steamline A and valve F019	Figure 5.1-3			
			Piping		B-J	Welds (Note 1)	MT
			Valves		B-M-2	Internal Surfaces (Note 3)	VT-3
			All pressure-retaining components and Piping		B-P	External Surfaces (Note 4)	VT-2
			Component and piping supports		F-A	Supports (Note 13)	VT-3

Table 5.2-8 Examination Categories (Continued)

Quality Group	System Number	System Title	System Description	P&ID Diagram	Sec. XI Exam Cat.	Items Examined	Exam method
A	B21	Nuclear Boiler (Continued)	Integral Attachments		B-K-1	Welds	UT or MT (Note 5)
			Bolting		B-G-2	Bolts, Nuts & Studs (Note 6)	VT-1
			Feedwater lines from RPV up to and including outer isolation valves F003A,B	Figure 5.1-3			
			Piping		B-J	Welds (Note 1)	UT, MT
			Valves		B-M-2	Valve Body Internal Surfaces (Note 3)	VT-3
			All pressure-retaining components		B-P	External Surfaces (Note 4)	VT-2
			Integral Attachments		B-K-1	Welds	UT or MT (Note 5)
			Bolting		B-G-2	Bolts, Nuts & Studs (Note 6)	VT-1
			Piping and Components		F-A	Supports (Note 13)	VT-3
			All Class A piping 25A and smaller (i.e., valve gland leakoff lines)	Figure 5.1-3	Exempted per IWB-1220 (b) (1)		
			All pressure-retaining components and piping		B-P	External Surfaces (Note 4)	VT-2

Table 5.2-8 Examination Categories (Continued)

Quality Group	System Number	System Title	System Description	P&ID Diagram	Sec. XI Exam Cat.	Items Examined	Exam method
A	C41	SLCS	Injection line from HPCF-B injection line connection up to and including outboard isolation valve F007	Figure 9.3-1			
			40A-SLC-4 piping		B-J	Welds (Note 1)	MT
			Valves		B-M-2	Valve Body Internal Surfaces (Note 3)	VT-3
			Pressure-retaining components and piping		B-P	External Surfaces (Note 4)	VT-2
			Integral attachments		B-K-1	Welds	MT or UT (Note 5)
			Piping and component supports		F-A	Supports (Note 13)	VT-3
			Bolting		B-G-2	Bolts, Nuts & Studs (Note 6)	VT-1
A	E11	RHR	LPFL B & C injection lines from RPV nozzles up to and including injection valves F005B and C	Figure 5.5-10			
			200A-RHR-107 piping 250A-RHR-106 piping 200A-RHR-207 piping 200A-RHR-206 piping		B-J	Welds (Note 1)	UT, MT
A	E11	RHR (Continued)	Valves		B-M-2	Valve Body Internal Surfaces (Note 3)	VT-3

Table 5.2-8 Examination Categories (Continued)

Quality Group	System Number	System Title	System Description	P&ID Diagram	Sec. XI Exam Cat.	Items Examined	Exam method
			All pressure-retaining components and piping		B-P	External Surfaces (Note 4)	VT-2
			Integral attachments		B-K-1	Welds	UT or MT (Note 5)
			Piping and component supports		F-A	Supports (Note 13)	VT-3
			Bolting		B-G-2	Bolts, Nuts & Studs (Note 6)	VT-1
			Shutdown cooling suction lines from RPV nozzles up to and including outboard isolation valves F011A,B,C	Figure 5.4-10			
			350A-RHR-010 piping 350A-RHR-211 piping 350A-RHR-110 piping		B-J	Welds (Note 1)	UT, MT
			Valves		B-M-2	Internal Surfaces (Note 3)	VT-3
			All pressure-retaining components & piping		B-P	External Surfaces (Note 4)	VT-2
			Bolting		B-G-2	Bolts, Nuts & Studs (Note 6)	VT-1
A	E11	RHR (Continued)	Integral Attachments		B-K-1	Welds	UT or MT (Note 5)
			Piping & component Supports		F-A	Supports (Note 13)	VT-3

Table 5.2-8 Examination Categories (Continued)

Quality Group	System Number	System Title	System Description	P&ID Diagram	Sec. XI Exam Cat.	Items Examined	Exam method
			All Class A piping 20A, and 25A in diameter, i.e.: - valve gland leakoff lines - test connections - drain lines - equalizing lines	Figure 5.4-10	Exempted per IWB-1220 (2) (1)		
			All pressure-retaining components and piping		B-P	External Surfaces (Note 4)	VT-2
A	E22	HPCF	HPCF injection lines from RPV nozzles up to and including injection valves F003B,C	Figure 6.3-7			
			200A-HPCF-008 Piping		B-J	Welds (Note 1)	UT, MT
			Valves		B-M-2	Valve Body Internal Surfaces (Note 3)	VT-3
			All pressure-retaining components and piping		B-P	External Surfaces (Note 4)	VT-2
			Integral attachments		B-K-1	Welds	UT or MT (Note 5)
			Components and piping supports		F-A	Supports (Note 13)	VT-3
A	E22	HPCF (Continued)	Bolting		B-G-2	Bolts, Nuts & Studs (Note 6)	VT-1

Table 5.2-8 Examination Categories (Continued)

Quality Group	System Number	System Title	System Description	P&ID Diagram	Sec. XI Exam Cat.	Items Examined	Exam method
			All Class A piping 20A in diameter. i.e: - test connections - valve gland leakoff lines - equalizing lines	Figure 6.3-7	Exempted per IWB-1220(b)(1)		
			All pressure-retaining components and piping		B-P	External Surfaces (Note 4)	VT-2
A	E51	RCIC	RCIC steam supply line from main steamline B up to and including outboard isolation valve F036	Figure 5.4-8			
			150A-RCIC-033		B-J	Welds (Note 1)	UT, MT
			Valves F035, F036		B-L-2	Valve Body Internal Surfaces (Note 3)	VT-3
			All pressure-retaining component and piping		B-P	External Surfaces (Note 4)	VT-2
			Integral attachments		B-K-1	Welds	UT or MT (Note 5)
			Piping and component supports		F-A	Supports (Note 13)	VT-3
			Bolting		B-G-2	Bolts, Nuts & Studs (Note 6)	VT-1

Table 5.2-8 Examination Categories (Continued)

Quality Group	System Number	System Title	System Description	P&ID Diagram	Sec. XI Exam Cat.	Items Examined	Exam method
A	E51	RCIC (Continued)	All Class A piping 20A, 25A in diameter i.e: - valve gland leakoff lines - test connections - drain lines - warmup line	Figures 5.4-8	Exempted per IWB-1220 (b) (1)		
			All pressure-retaining components and piping		B-P	External Surfaces (Note 4)	VT-2
A	G31	CUW	Vessel head spray line from head vent nozzle up to and including outboard isolation valve F017	Figure 5.4-12			
			150A-CUW-24-CS		B-J	Welds (Note 1)	UT & MT
			Valves		B-M-2	Valve Body Internal Surfaces (Note 3)	VT-3
			All pressure-retaining components and piping		B-P	External Surfaces (Note 4)	VT-2
			Integral attachments		B-K-1	Welds	UT or MT (Note 5)
			Piping and component supports		F-A	Supports (Note 13)	VT-3
			Bolting		B-G-2	Bolts, Studs & Nut (Note 6)	VT-1

Table 5.2-8 Examination Categories (Continued)

Quality Group	System Number	System Title	System Description	P&ID Diagram	Sec. XI Exam Cat.	Items Examined	Exam method
A	G31	CUW (Continued)	RPV bottom head drain line from RPV nozzle up to and including valve F001 and outboard isolation valve F003	Figure 5.4-12			
			Branch Connection 65A-CUW-20-55 to 200A-CUW-1-CS		B-F	Weld	UT & PT
			200A-CUW-1-CS piping		B-J	Welds (Note 1)	UT & MT
			65A-CUW-20-SS piping		B-J	Welds (Note 1)	PT
			Valves		B-M-2	Valve Body Internal Surfaces (Note 3)	VT-3
			All pressure-retaining components and piping		B-P	External Surfaces (Note 4)	
			Integral attachments		B-K-1	Welds	UT or MT (Note 5)
			Piping and component supports		F-A	Supports (Note 13)	VT-3
			Bolting		B-G-2	Bolts, Stud & Nut (Note 6)	VT-1
			Suction line from RHR B shutdown cooling suction line up to valve F001 up to RPV bottom head blowdown header to CUW	Figure 5.4-12 Figure 5.4-10			
			200A-CUM-1-CS piping		B-J	Welds (Note 1)	UT & MT

Table 5.2-8 Examination Categories (Continued)

Quality Group	System Number	System Title	System Description	P&ID Diagram	Sec. XI Exam Cat.	Items Examined	Exam method
A	G31	RUCU (Continued)	Valves		B-M-2	Valve Body Internal Surfaces (Note 3)	VT-3
			All pressure-retaining components and piping		B-P	External Surfaces (Note 4)	VT-2
			Integral attachments		B-K-1	Welds	UT or MT (Note 5)
			Piping and component supports		F-A	Supports (Note 13)	VT-3
			Bolting		B-G-2	Bolts, Studs & Nuts (Note 6)	VT-1
			All Class A piping 20A in diameter. i.e: - test connections - valve gland leakoff lines - drain lines - sample lines - instrument lines	Figure 5.4-12	Exempted per IWB-1220(b)(1)		
			All pressure-retaining components and piping		B-P	External Surfaces (Note 4)	VT-2

Table 5.2-8
Examination Categories and Methods

NOTES:

(1) Category B-J: At least 25% of the circumferential piping welds (including branch connection welds) shall be selected for inservice inspection in accordance with the rules of Table IWB-2500-1 for examination category B-J. Welds NPS 4 and larger are examined by both ultrasonic (UT) and magnetic particle (MT) methods. Welds in piping less than NPS 4 are examined by the MT method. The examination includes at least a pipe-diameter length, but not more than 305 mm of each longitudinal weld intersecting the circumferential weld.

(2) Category B-M-1: Valve body welds selected for inservice inspection are limited to at least one valve within each group of valves of the same size and type and performing a similar function in accordance with rules of Table IWB-2500-1 for examination category B-M-1.

(3) Category B-M-2: Valve Bodies selected for inservice inspection are limited to at least one valve within each group of valves of the same size and type and performing a similar function in accordance with the rules of Table IWB-2500-1 for examination category B-M-2. Examination is required only when a valve is disassembled for maintenance, repair or volumetric examination.

(4) Category B-P: Visual examination of the external surfaces of pressure retaining components and piping for inservice inspection is performed in conjunction with the system leakage and system hydrostatic tests in accordance with the rules of Table IWB-2500-1 for examination category B-P.

(5) Category B-K-1: Examination of integral attachments for inservice inspection is limited to those attachments which are external, associated with an NF type component support and which have a base material thickness greater than 16 mm. Ultrasonic (UT) examination may be substituted for magnetic particle (MT) examination for some configurations as specified by Table IWB-2500-1 for examination category B-K-1.

(6) Category B-G-2: All bolts, studs and nuts, 5.1 cm and less in diameter, are examined for inservice inspection in accordance with the rules of Table IWB-2500-1 for examination category B-G-2.

(7) Category B-A: All RPV welds are subject to inservice inspection. For RPV head welds, only the accessible length of each weld is required to be examined.

(8) Category B-E: The visual VT-2 examination is performed in conjunction with the system hydrostatic test.

(9) Category B-G-1: Closure studs are examined ultrasonically only when examined in place or by ultrasonic and magnetic particle when removed.

(10) Category B-H: Examination of integral attachments for inservice inspection is limited to those attachments which are external, associated with an NF type component support and which have a base material thickness greater than 16 mm and the attachment weld joins either directly to the surface of the vessel or to an integrally cast or forged attachment to the vessel. For the reactor vessel support skirt, ultrasonic examination from only one side shall be substituted for the surface examination in accordance with Table IWB-2500-1 for examination category B-H.

(11) Examination Category B-N-1: Areas to be examined shall include the spaces above and below the reactor core that are made accessible from examination by removal of components during refueling outages.

(12) Examination Category B-N-2: Only welds made accessible for examination by removal of components during normal refueling outages are required to be examined.

(13) Category F-A: Supports selected for inservice examination, as described in IWF-2510, shall include 25% of Class 1 piping supports. The total percentage sample shall be comprised of supports from each system where the individual sample sizes are proportional to the total number of non-exempt supports of each type and function within each system. All supports of non-exempt components (i.e., vessels, pumps and valves) shall be subject to inservice examination.

General: The preservice examination includes all of the items in all examination categories with the exception of categories B-E, B-P and the internal surface examination of category B-M-2. The preservice examinations shall include essentially 100% of the pressure retaining welds in non-exempt Class 1 piping and components except examination category B-O, which shall be limited to peripheral control rod drive housings only in accordance with IWB-2200. Preservice examination of supports shall be performed following the initiation of hot functional or power ascension tests.

Table 5.2-9 Not Used

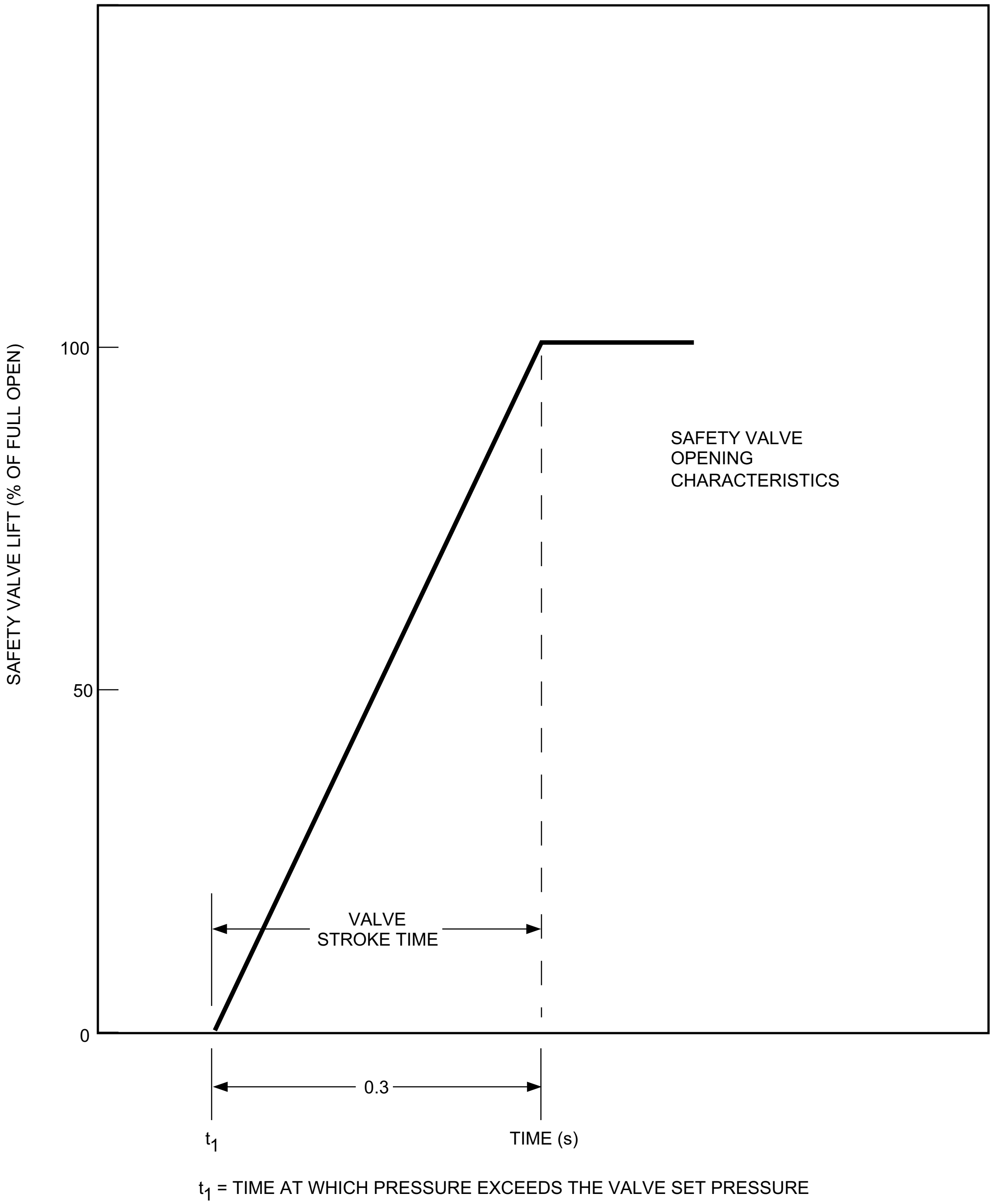


Figure 5.2-1 Safety-Action Valve Lift Characteristics

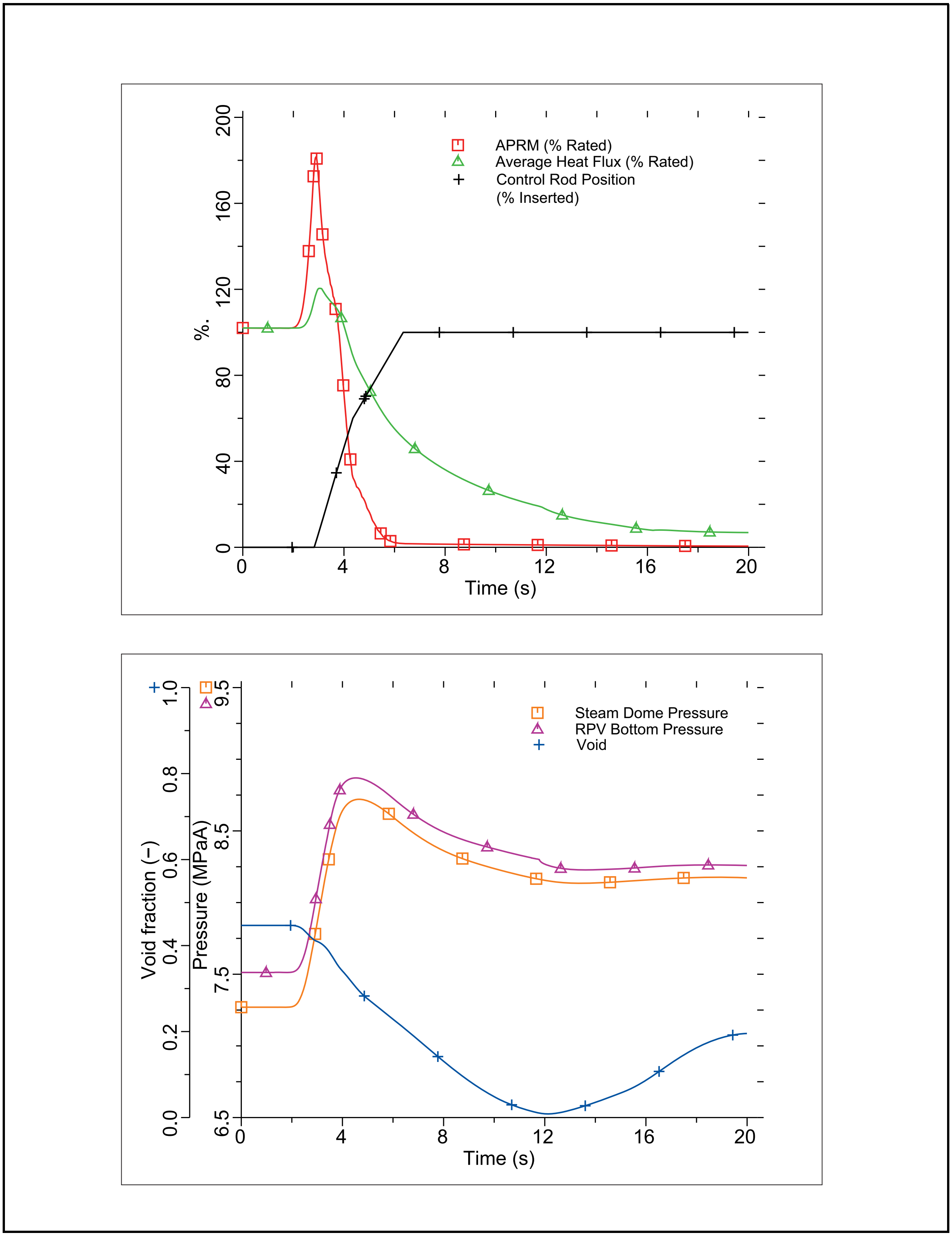
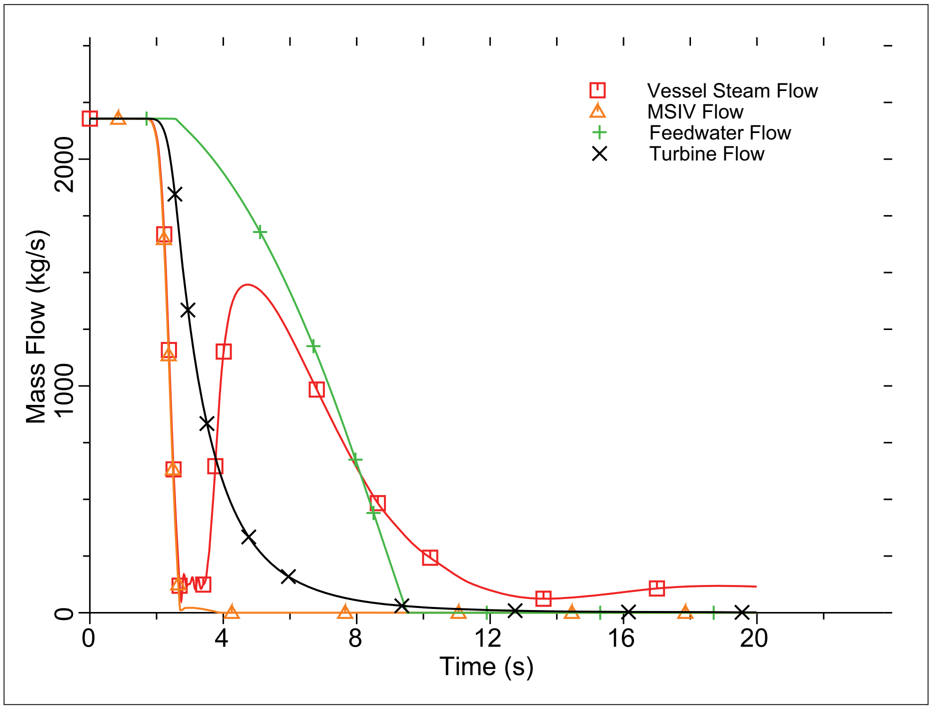


Figure 5.2-2a MSIV Closure with Flux Scram and Installed Safety/Relief Valve Capacity



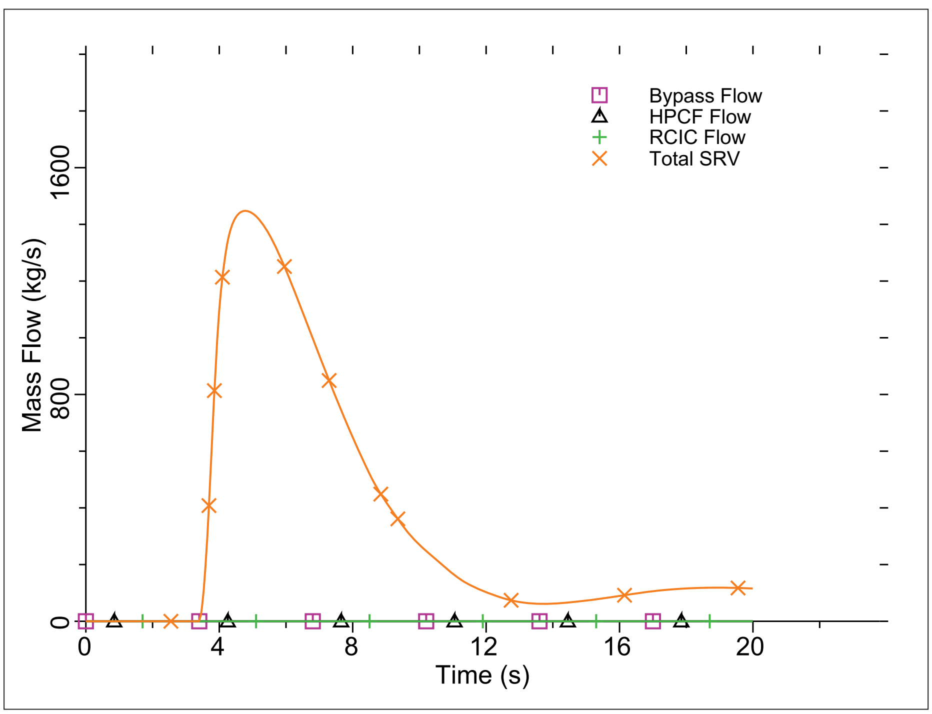


Figure 5.2-2b MSIV Closure with Flux Scram and Installed Safety/Relief Valve Capacity

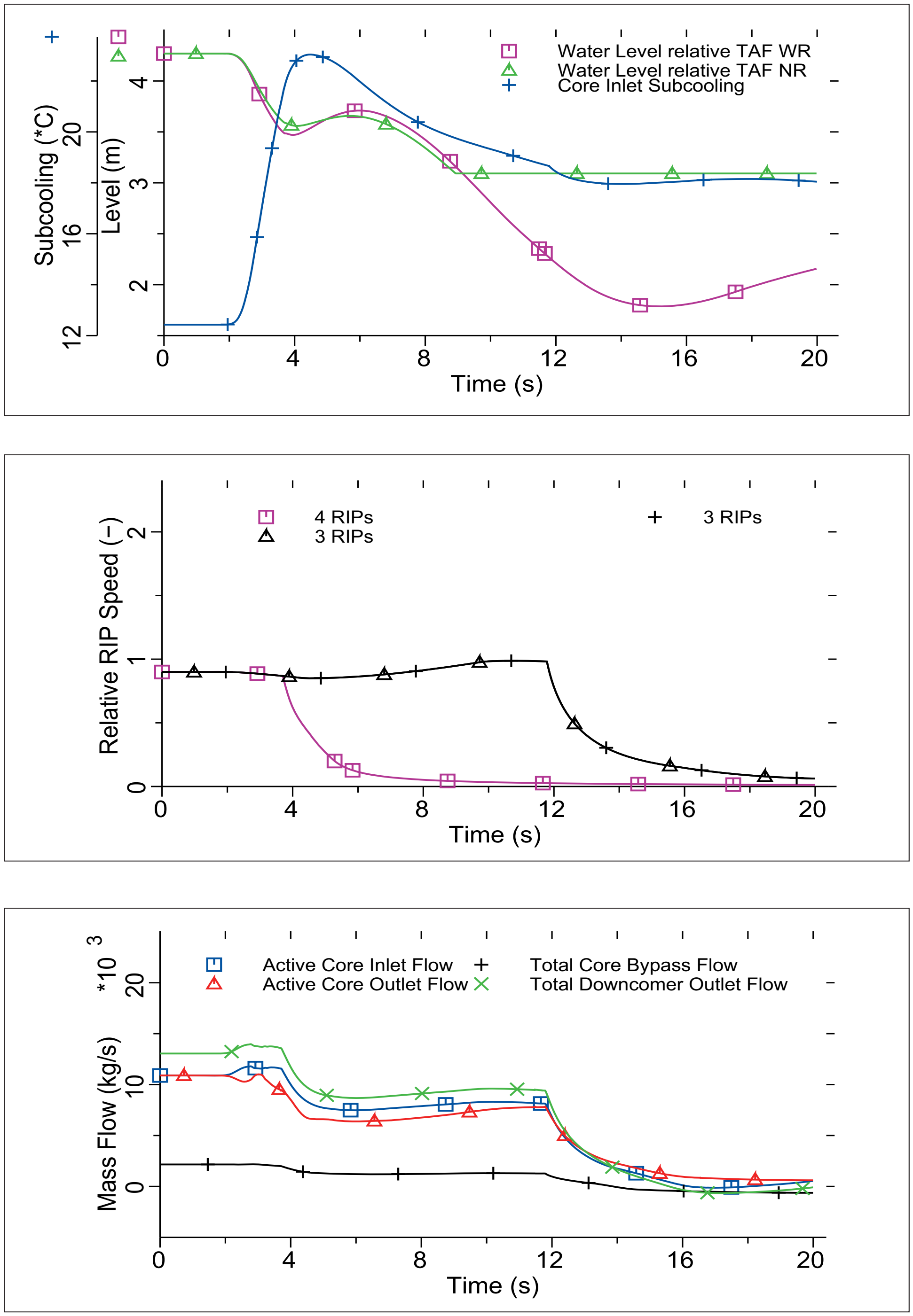


Figure 5.2-2c MSIV Closure with Flux Scram and Installed Safety/Relief Valve Capacity

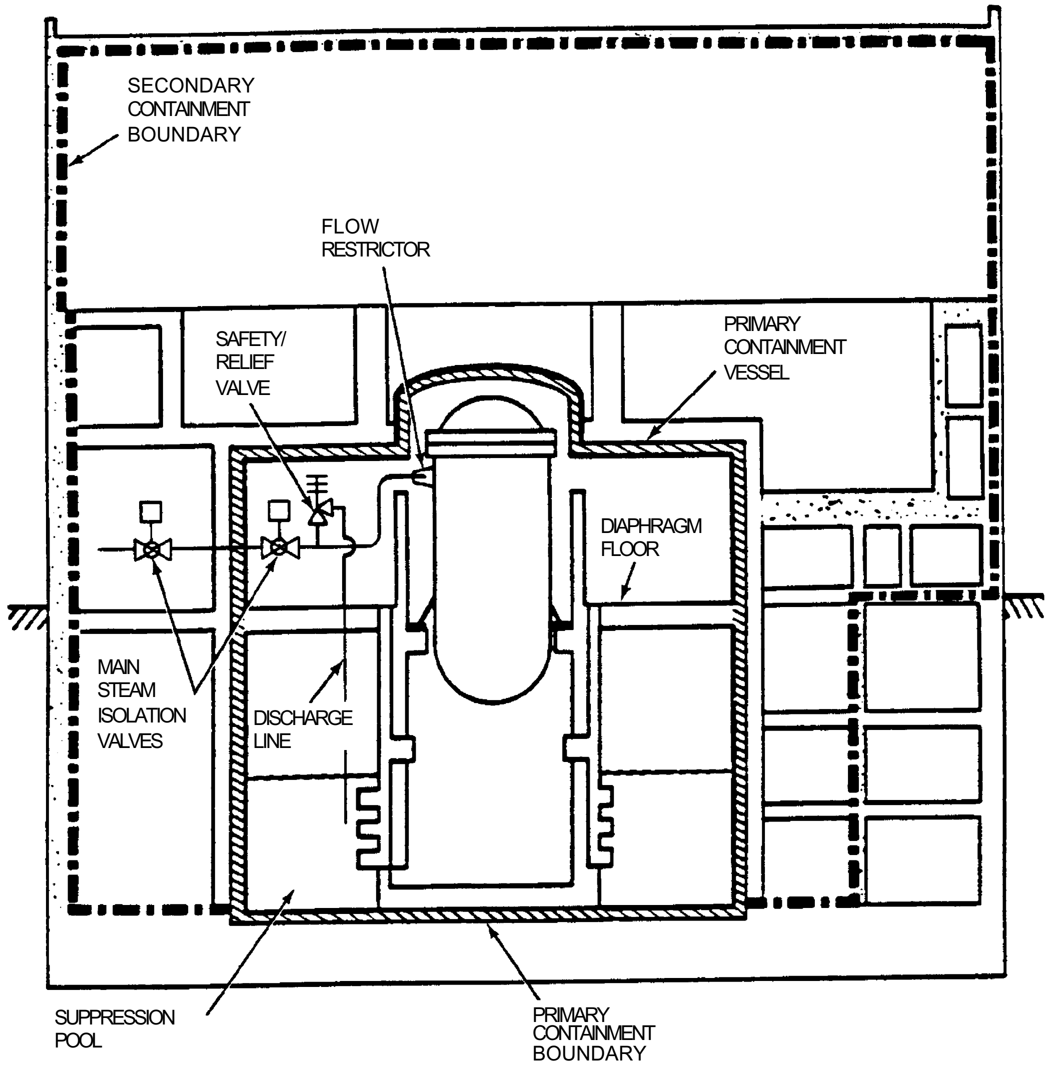


Figure 5.2-3 Safety/Relief Valve Schematic Elevation

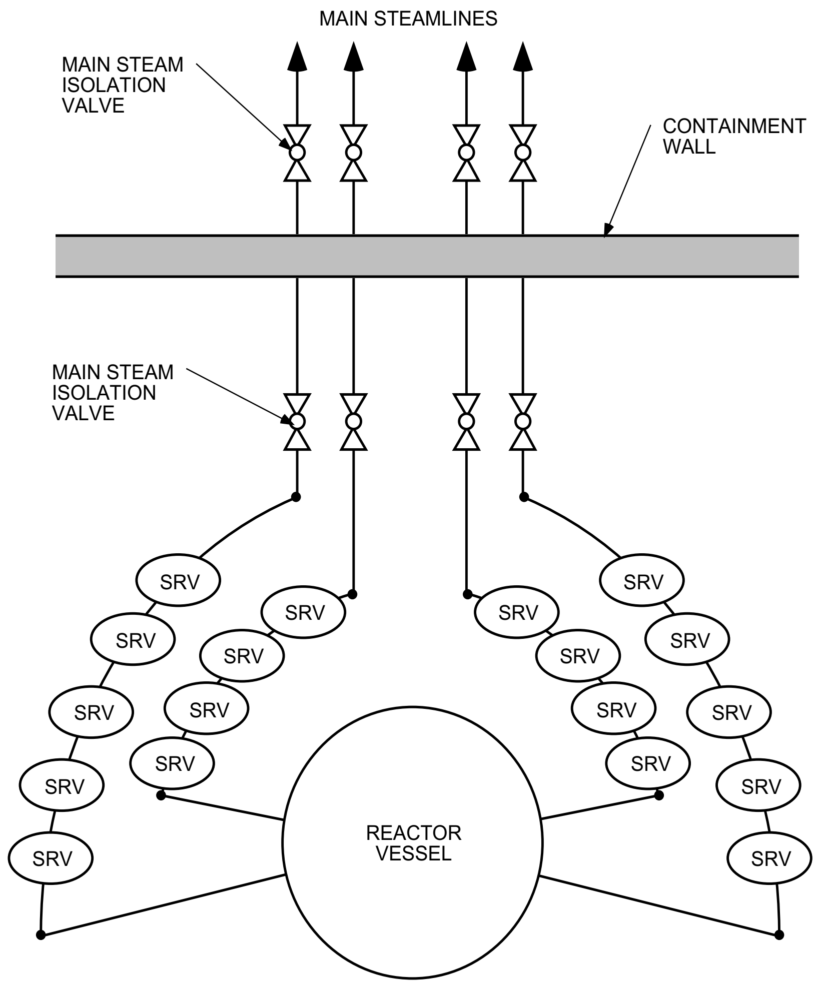


Figure 5.2-4 Safety /Relief Valve and Steamline Schematic

Figure 5.2-5 Not Used

Figure 5.2-6 Not Used

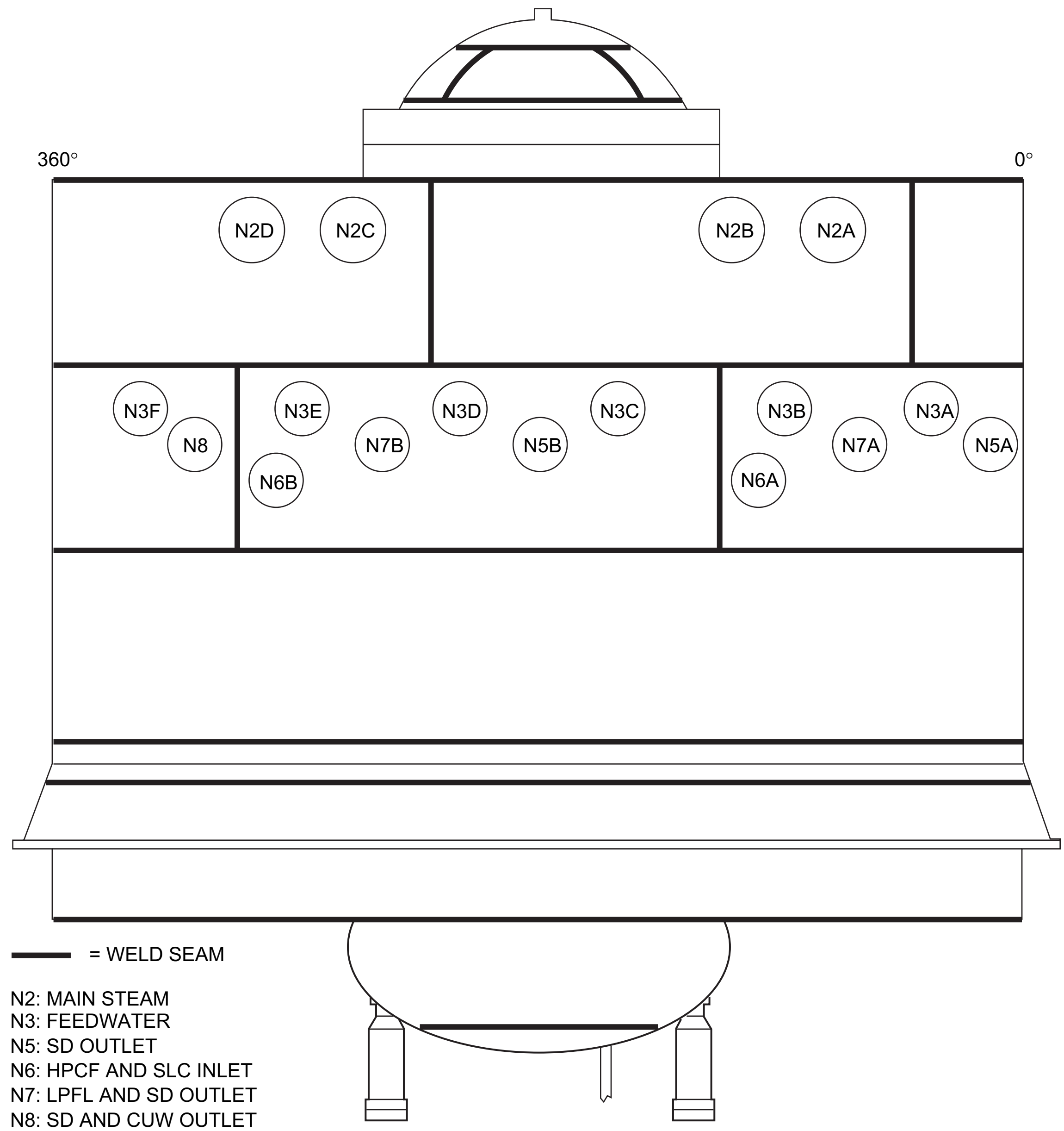


Figure 5.2-7a RPV Examination Areas

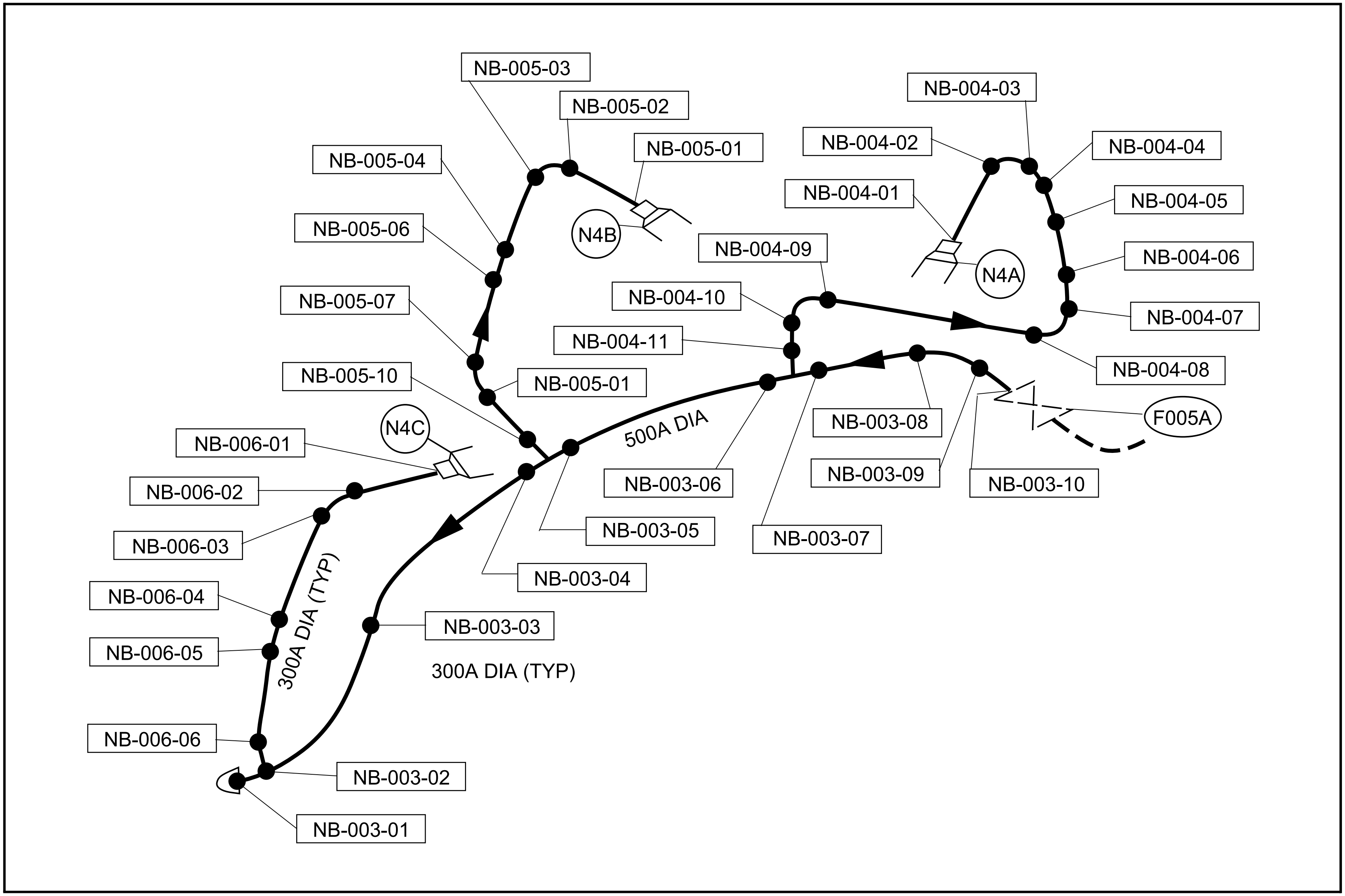


Figure 5.2-7b Typical Piping System Isometric (Feedwater Line from RPV to Valve F005A)

The following figure is located in Chapter 21 :

Figure 5.2-8 Leak Detection and Isolation System IED (Sheets 1 - 10)