

2.1 Nuclear Steam Supply Systems

2.1.1 Reactor Pressure Vessel System

Design Description

The Reactor Pressure Vessel (RPV) System consists of (1) the RPV and its appurtenances, supports and insulation, excluding the Loose Parts Monitoring System, and (2) the reactor internal components enclosed by the vessel, excluding the core (fuel assemblies, control rods, in-core nuclear instrumentation and neutron sources), reactor internal pumps (RIPs), and control rod drives (CRDs). The RPV System is located in the primary containment.

The reactor coolant pressure boundary (RCPB) portion of the RPV and its appurtenances (referred to in this section as the RPV pressure boundary) act as a radioactive material barrier during plant operation.

Certain reactor internals support the core, flood the core during a loss-of-coolant accident (LOCA) and support safety-related instrumentation. Other RPV internals direct coolant flow, separate steam, hold material surveillance specimens, and support instrumentation utilized for plant operation.

The RPV System provides guidance and support for the CRDs. It also distributes sodium pentaborate solution when injected from the Standby Liquid Control (SLC) System.

The RPV System restrains the CRD to prevent ejection of the control rod connected with the CRD in the event of a failure of the RCPB associated with the CRD housing weld. A restraint system is also provided for each RIP in order to prevent the RIP from becoming a missile in the event of a failure of the RCPB associated with the RIP casing weld.

The RPV System is shown on Figures 2.1.1a, 2.1.1b and 2.1.1c; key dimensions and the acceptable variations in these dimensions are presented in Table 2.1.1a. The RPV System parameters (break areas) used in LOCA analyses are identified in Table 2.1.1b. The principal design parameters for the RPV System are listed in Table 2.1.1c.

Reactor Pressure Vessel, Appurtenances, Supports and Insulation

The RPV, as shown in Figures 2.1.1a and 2.1.1b, is a vertical, cylindrical vessel of welded construction with removable top head and head closure bolting and seals. The main body of the installed RPV has a cylindrical shell, flange, bottom head, RIP casings, penetrations (including inserted housings), brackets, nozzles, and the shroud support, which has a pump deck forming the partition between the RIP suction and discharge. The shroud support is an assembly consisting of a short vertical cylindrical shell, a horizontal annular pump deck plate and vertical shroud support legs.

The CRD housings are inserted through and welded to the CRD penetrations in the reactor vessel bottom head. The CRDs are mounted into the CRD housings. The in-core housings are inserted through and connected to the bottom head.

For an RPV System that requires to be instrumented for flow-induced vibration (FIV) testing, a flanged nozzle is provided in the top head for bolting of the flange associated with the test instrumentation.

The integral reactor vessel skirt supports the vessel on the Reactor Pressure Vessel Pedestal. The vessel skirt does not have openings connecting the upper and lower drywell regions. Anchor bolts extend from the pedestal through the flange of the skirt. RPV stabilizers are provided in the upper portion of the RPV to resist horizontal loads. Lateral supports for the CRD housings and in-core housings are provided.

A restraint system is provided to prevent a RIP from being a missile in case of a postulated failure in the casing weld with the bottom head penetration. The restraint system is connected to the lugs on the RPV bottom head and the RIP motor cover.

The RPV insulation is supported from the reactor shield wall surrounding the vessel. Insulation for the upper head and flange is supported by a steel frame independent of the vessel and piping.

The RPV pressure boundary and the supports (RPV skirt, stabilizer and CRD housing/in-core housing lateral supports) are classified as Seismic Category I. These components are ASME Code Class 1 vessel and supports, respectively. The shroud support and a portion of the CRD housings inside the RPV are classified as Seismic Category I and ASME Code Class CS structures.

The following ASME materials (or their equivalents) are used in the RPV pressure boundary: SA-533, Type B, Class 1 (plate); SA-508, Class 3 (forging); SA-508, Class 1 (forging); SB-166 (UNS N06600, bar); SB-167 (UNS N06600, seamless pipe); SB-564 (UNS N06600, forging); SA-182 or SA-336, Grade/Class F316L (maximum carbon 0.020%, forging) or F316 (maximum carbon 0.020% and nitrogen from 0.060 to 0.120%, forging); and SA-540, Grade B23 or B24 (bolting).

A stainless steel weld overlay is applied to the interior of the RPV cylindrical shell and the steam outlet nozzles. Other nozzles do not have cladding. The bottom head is clad with Ni-Cr-Fe alloy. The RIP penetrations are clad with Ni-Cr-Fe alloy or, alternatively, stainless steel. The RIP motor casings are clad with stainless steel only in the stretch tube region and around the bottom of the RIP motor casings where they interface with the motor cover closures.

The materials of the low alloy plates and forging used in construction of the RPV pressure boundary are melted using vacuum degassing to fine grain practice and are supplied in quenched and tempered condition.

Electroslag welding is not applied for the RPV pressure boundary welds. Preheat and interpass temperatures employed for welding of the RPV pressure boundary low alloy steel meet or exceed the values given in ASME Code Section III, Appendix D. Post-weld heat treatment at 593°C minimum is applied to these low-alloy steel welds.

The RPV pressure boundary welds are given an ultrasonic examination in addition to the radiographic examination performed during fabrication. The ultrasonic examination method, including calibration, instrumentation, scanning sensitivity, and coverage, is based on the requirements imposed by ASME Code Section XI, Appendix I. Acceptance standards also meet the requirements of ASME Code Section XI.

The fracture toughness tests of the RPV pressure boundary ferritic materials, weld metal and heat-affected zone (HAZ) are performed in accordance with the requirements for ASME Code Section III, Class 1 vessel. Both longitudinal and transverse specimens are used to determine the minimum upper-shelf energy (USE) level of the core beltline materials. The minimum USE level for base and weld metal in the core beltline is initially at least 10.4 kg-m. Separate, unirradiated baseline specimens are used to determine the transition temperature curve of the core beltline base materials, weld metal, and HAZ.

For the RPV material surveillance program, Charpy V-notch and tensile specimens are manufactured from the same material used in the reactor beltline region. To represent those RPV pressure boundary welds that are in the beltline region, Charpy V-notch specimens of weld metal and HAZ material, and tensile specimens of weld metal are manufactured from sample welds. The specimen capsules contain the specimens and temperature monitors. The surveillance specimen holders having brackets welded to the vessel cladding in the core beltline region are provided to hold four specimen capsules and a neutron dosimeter.

Reactor Pressure Vessel Internals

The major reactor internal components in the RPV System are:

(1) Core Support Structures:

Shroud, shroud support and a portion of CRD housings inside the RPV (both integral to the RPV), core plate, top guide, fuel supports (orificed fuel supports and peripheral fuel supports), and control rod guide tubes (CRGTs). The core support structures are classified as Seismic Category I and ASME Code Class CS structures.

(2) Other Reactor Internals:

- (a) Feedwater spargers, shutdown cooling (SDC) and low pressure core flooders (LPFL) spargers for the Residual Heat Removal (RHR) System, high pressure core flooders (HPCF) spargers and couplings, and a portion of the in-core

housings inside the RPV and in-core guide tubes (ICGTs) with stabilizers. These components are classified as Seismic Category I and safety-related.

- (b) Surveillance specimen holders, shroud head and steam separators assembly and the steam dryer assembly. These components are classified as non-safety-related.

A general assembly of these reactor internal components is shown in Figures 2.1.1a, 2.1.1b, and 2.1.1c.

The shroud support, shroud, and top guide make up a cylindrical assembly that provides a partition to separate the upward flow of coolant through the core from the downward recirculation flow. This partition separates the core region from the downcomer annulus.

The core plate consists of a circular plate with round openings and is stiffened with a rim and beam structure. The core plate provides lateral support and guidance for the CRGTs, ICGTs, peripheral fuel supports, and startup neutron sources. The last two items are also supported vertically by the core plate.

The top guide consists of a circular plate with square openings for fuel assemblies and with a cylindrical side forming an upper shroud extension. Each opening provides lateral support and guidance for four fuel assemblies or, in the case of peripheral fuel, less than four fuel assemblies. Holes are provided in the bottom, where the sides of the openings intersect, to anchor the in-core instrumentation detectors and startup neutron sources.

The fuel supports are of two types: (1) peripheral and (2) orificed. The peripheral fuel supports are located at the outer edge of the active core and are not adjacent to control rods. Each peripheral fuel support supports one peripheral fuel assembly and has an orifice to provide coolant flow to the fuel assembly. Each orificed fuel support holds four fuel assemblies and has four orifices to provide coolant flow distribution to each fuel assembly. The control rods pass through cruciform openings in the center of the orificed fuel supports. This locates the four fuel assemblies surrounding a control rod.

The CRGTs pass through holes in the core plate, have four holes under the core plate and rest on top of the CRD housings. Each CRGT guides the lower end of a control rod and supports an orificed fuel support such that the orifices of the orificed fuel support align with the holes in the CRGT for coolant flow. The lower end of the CRGT is supported by the CRD housing, which, in turn, transmits the weight of the guide tube, fuel supports, and fuel assemblies to the reactor vessel bottom head.

The CRGT base is provided with a device for coupling the CRD with it. The CRD is restrained from ejection, in the case of failure of the weld between a CRD housing and CRD penetration, by the coupling of the CRD with the CRGT base; in this event, the flange at the top of the guide tube contacts the core plate and acts to restrain the ejection. The coupling will also prevent

ejection if the housing fails beneath the weld; in this event, the guide tube remains supported on the intact upper housing.

There are six feedwater spargers, three for each of the two feedwater lines. Each sparger is connected to an RPV feedwater nozzle at the double thermal sleeve fitted with the safe end (straight piece) of the nozzle. Feedwater flow enters the middle of the spargers, which are located above the RPV downcomer annulus, and is discharged inward.

Two spargers are provided for two loops of the RHR System; both spargers function as SDC and LPFL spargers. Each sparger is connected to a thermal sleeve fitted with the safe end of each SDC and LPFL inlet nozzle.

Two HPCF spargers with couplings are provided for the two loops of the HPCF System to direct high pressure coolant flow to the upper end of the core during emergency core cooling. One of the HPCF spargers also distributes sodium pentaborate solution when injected from the SLC System via the connecting HPCF line. The spargers are located inside the cylindrical portion of the top guide. Each sparger is connected via an HPCF coupling to a thermal sleeve fitted with the safe end of each HPCF inlet nozzle.

The ICGTs house the in-core neutron flux monitoring instrumentation assemblies, pass through holes in the core plate, and rest on top of the in-core housings. Two levels of stabilizer latticework give lateral support to the ICGTs. The ICGT stabilizers are connected to either the shroud or the shroud support.

The surveillance specimen holders having brackets welded to the vessel cladding in the core beltline region are provided to hold the specimen capsules and a neutron dosimeter.

The shroud head and steam separators assembly includes the connecting standpipes and forms the top of the core discharge mixture plenum. The steam dryer assembly removes moisture from the wet steam leaving the steam separators. The extracted moisture flows down the dryer vanes to the collecting troughs, then flows through tubes into the downcomer annulus.

Cobalt-base material is only used for hard surfacing of areas in HPCF coupling. The wrought austenitic stainless steel used for the RPV internals is limited to a maximum of 0.02% carbon content. Stainless steel materials are supplied in solution heat-treated condition. Furnace sensitized stainless steel material is not used. Electroslag welding is not applied for structural welds of stainless steel.

The RPV internals are designed to withstand the effects of FIV.

Inspections, Tests, Analyses and Acceptance Criteria

Table 2.1.1d provides a definition of the inspections, tests, and/or analyses, together with associated acceptance criteria, which will be undertaken for the Reactor Pressure Vessel System.

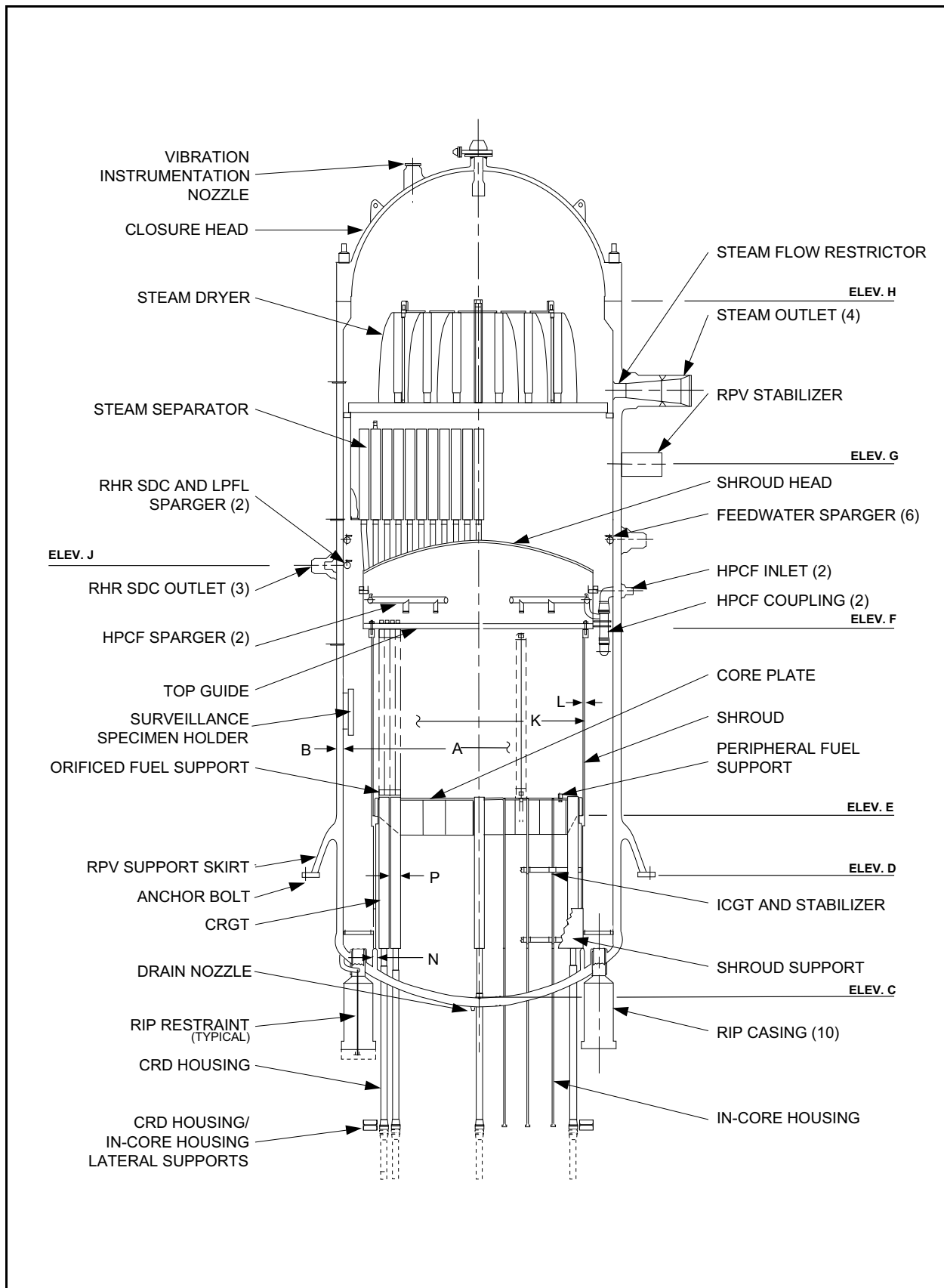


Figure 2.1.1a Reactor Pressure Vessel System Key Features

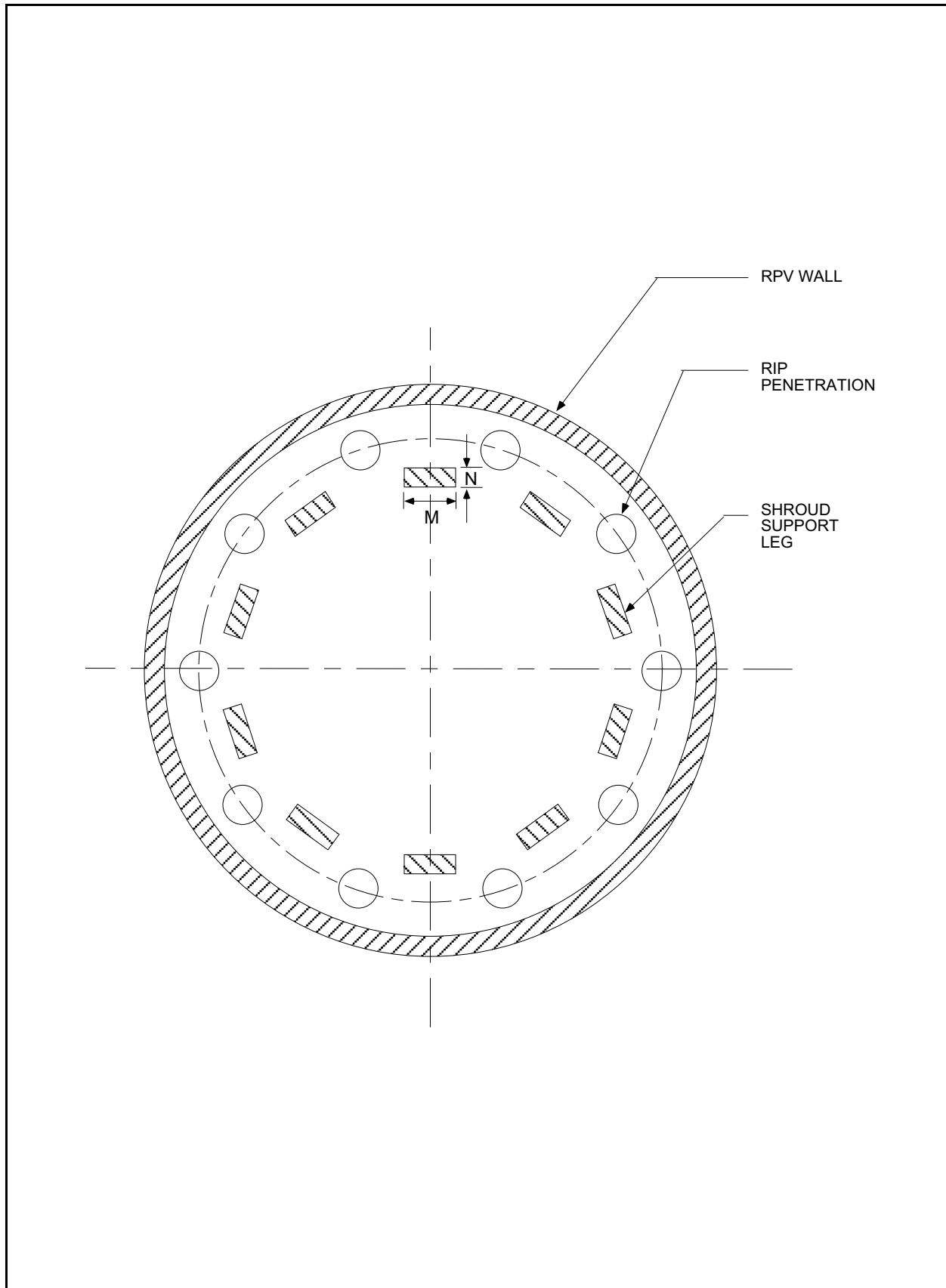


Figure 2.1.1b Pump Penetration and Shroud Support Leg Arrangement

NOTES:

1. THE ARRANGEMENT IS SHOWN FOR QUARTER CORE ONLY. ~~MIRROR~~ SYMMETRY APPLIES. THE REACTOR INTERNALS ACCOMMODATE THE SHOWN CORE ARRANGEMENT; THE CORE IS NOT A PART OF THE RPV SYSTEM.

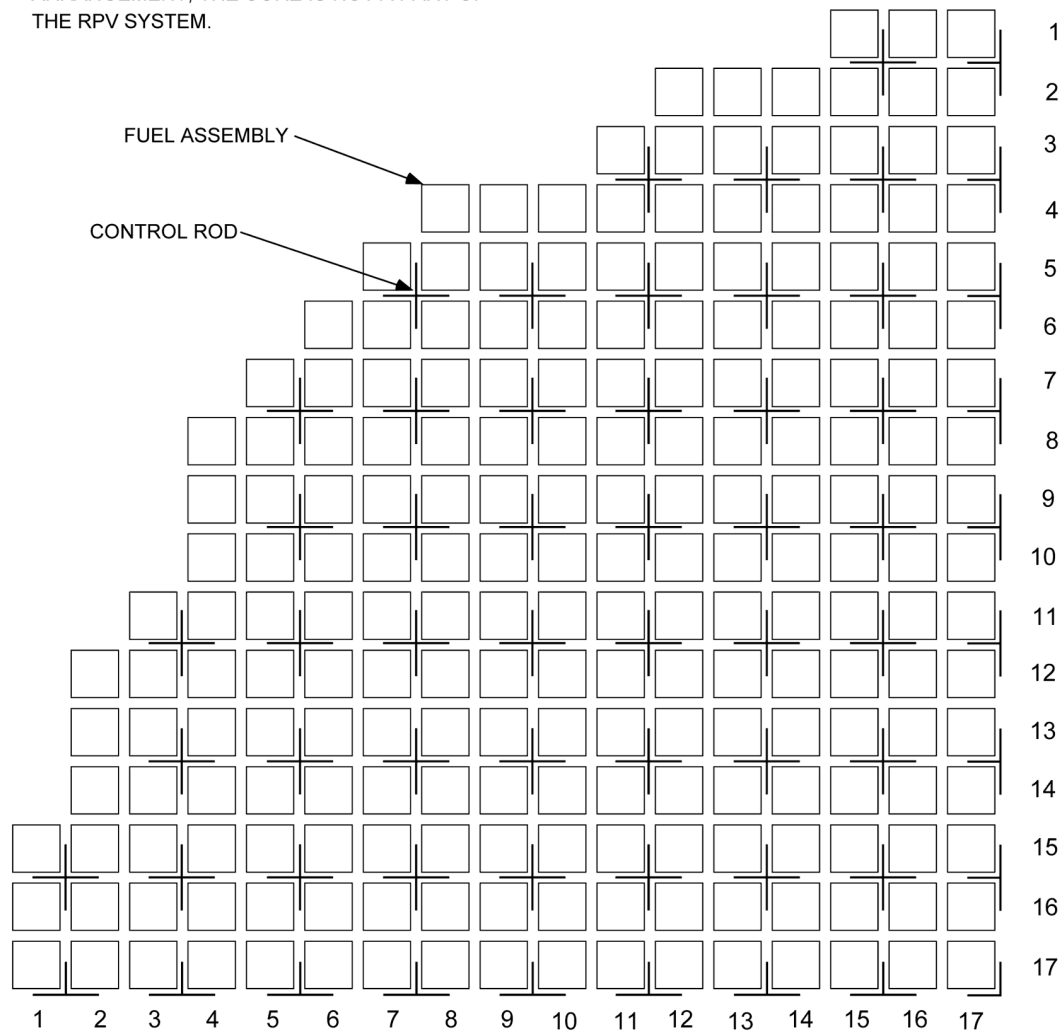
**Figure 2.1.1c Core Arrangement**

Table 2.1.1a Key Dimensions of RPV System Components and Acceptable Variations

Description	Dimension/ Elevation (Figure 2.1.1a)	Nominal Value (mm)	Acceptable Variation (mm)
RPV inside diameter (inside cladding)	A	7112.0	±51.0
RPV wall thickness in beltline (without cladding)	B	174.0	+20.0/−4.0
RPV bottom head inside invert, Elevation	C	0.0	Reference 0.0
RPV support skirt bottom, Elevation	D	3250.0	±75.0
Core plate support/Top of shroud middle flange, Elevation	E	4695.2	±15.0
Top guide support/Top of shroud top flange, Elevation	F	9351.2	±20.0
RPV stabilizer connection, Elevation	G	13,766.0	±20.0
Top of RPV flange, Elevation	H	17,703.0	±65.0
RHR SDC/CUW Outlet Nozzle, Elevation	J	10,921.0	±40.0
Shroud outside diameter	K	5600.7	±25.0
Shroud wall thickness	L	57.2	±10.0
Shroud support legs (Fig. 2.1.1b)	MxN	662.0x153.0	±20.0 for M ±10.0 for N
Control rod guide tube outside diameter	P	273.05	±5.0

Table 2.1.1b RPV System Parameters Used in LOCA Analyses

Line	Inspection Location	Postulated Break Area, mm²
Steamline	Flow restrictor throat diameter in a steam outlet nozzle.	98,480
Feedwater	Inside diameters of flow nozzles on the spargers of a feedwater line for the total flow area.	83,890
RHR Injection	Inside diameters of flow nozzles on an SDC and LPFL sparger for the total flow area.	20,530
High Pressure Core Flooder	Inside diameters of flow nozzles on an HPCF sparger for the total flow area.	9200
RHR Shutdown Cooling	Inside diameter of an RHR SDC outlet nozzle at the safe end weld.	79,150
Drain	Inside diameter of the bottom head hole for the drain outlet nozzle, near the inside surface of the head and below the hole chamfer.	2030
Note: The areas calculated from the inspections shall not exceed the postulated break areas by 5%.		

Table 2.1.1c Principal Design Parameters for RPV System

Description	Value
RCPB design pressure (MPaG)	8.62
RCPB design temperature (°C)	302
Number of fuel assemblies	872
Number of control rods	205
Number of internal pumps	10

Table 2.1.1d Reactor Pressure Vessel System

Inspections, Tests, Analyses and Acceptance Criteria		
Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The basic configuration of the RPV System is as defined as Section 2.1.1.	1. Inspections of the as-built RPV System will be conducted.	1. The RPV System conforms with the basic configuration defined in Section 2.1.1.
2. The RPV pressure boundary defined in Section 2.1.1 is designed to meet the ASME Code Class 1 vessel requirements.	2. Inspections of the ASME Code required documents will be conducted.	2. An ASME Code Certified Stress Report exists for the RPV pressure boundary components.
3. The ASME Code components of the RPV System retain their pressure boundary integrity under internal pressure that will be experienced during service.	3. A hydrostatic test will be conducted on those code components of the RPV System required to be hydrostatically tested by the ASME Code.	3. The results of the hydrostatic test of the ASME Code components of the RPV System conform with the requirements in the ASME Code, Section III.
4. The materials selection and materials testing requirements for the RPV System are as defined in Section 2.1.1.	4. Inspections of the as-built RPV System will be conducted.	4. The RPV System conforms with the materials selection and materials testing requirements defined in Section 2.1.1.
5. The fabrication process and examination process requirements for the RPV System are as defined in Section 2.1.1.	5. Inspections of the as-built RPV System will be conducted.	5. The RPV System conforms with the fabrication process and examination process requirements defined in Section 2.1.1.
6. The material surveillance commitments for the reactor pressure vessel core beltline materials are as defined in Section 2.1.1.	6. Inspections of the as-built RPV System will be conducted for implementation of the material surveillance commitments.	6. The material surveillance program for the reactor pressure vessel core beltline materials conforms with the commitments defined in Section 2.1.1.
7. The RPV internals withstand the effects of FIV.	7. A vibration type test will be conducted on the prototype RPV internals of an ABWR. A flow test and post-test inspections will be conducted on the as-built RPV internals.	7. A vibration type test report exists and concludes that the prototype RPV internals have no damage or loose parts as a result of the vibration type test. The as-built RPV internals have no damage or loose parts.

2.1.2 Nuclear Boiler System

Design Description

General System Description

The primary functions of the Nuclear Boiler System (NBS) are:

- (1) Deliver steam from the Reactor Pressure Vessel (RPV) to the Main Steam (MS) System.
- (2) Provide containment isolation of the main steamlines (MSLs) and the feedwater (FW) lines.
- (3) Deliver feedwater from the Condensate, Feedwater, and Condensate Air Extraction (CFCAE) System to the RPV.
- (4) Provide overpressure protection of the reactor coolant pressure boundary (RCPB).
- (5) Provide automatic depressurization of the RPV in the event of a loss-of-coolant accident (LOCA) where the RPV does not depressurize rapidly and the high pressure makeup systems fail to adequately maintain the water level in the RPV.
- (6) Provide instrumentation to monitor the drywell pressure and RPV pressure, metal temperature, and water level.

Figures 2.1.2a, 2.1.2b, 2.1.2c, 2.1.2d, and 2.1.2e show the basic system configuration and scope. Figure 2.1.2f shows the NBS control interfaces.

The NBS equipment shown on Figures 2.1.2a, 2.1.2b, 2.1.2c, 2.1.2d, and 2.1.2e is classified as safety-related except for the non-safety-related part of the MSL drains, equipment associated with the power actuated relief mode of the SRVs, the SRV discharge pipe temperature sensors, and the non-safety-related instruments shown on Figure 2.1.2e.

Main Steam Lines

The MSLs direct steam from the RPV to the MS System. The NBS contains only the portion of the MSLs from their connection to the RPV to the boundary with the MS System, which occurs at the seismic interface located downstream of the outboard main steam isolation valves (MSIVs). Figures 2.1.2a and 2.1.2b show the general configuration of the MSLs and the MSL drain lines. The MSL drain lines provide a flow path for the MSIV leakage during an accident.

The combined volume of the steamlines, from the RPV to the main steam turbine stop valves and turbine bypass valves, is greater than or equal to 113.2 m³.

Each MSL has a flow limiter. The MSL flow limiter consists of a flow restricting venturi which is located in each RPV MSL outlet nozzle. The restrictor limits the coolant blowdown rate from

the RPV, in the event that a MSL break occurs outside the containment, to a flow rate equal to or less than 200% of rated steam flow at 7.07 MPaG upstream pressure. The throat diameter of each MSL flow limiter is less than or equal to 355 mm.

The pneumatic-operated valve in the MSL drain line shown in Figure 2.1.2.b opens, if either electric power to the valves actuating solenoid is lost, or pneumatic pressure to the valve is lost.

The MSLs and the MSL drain lines are located in the drywell and the steam tunnel.

Main Steam Isolation Valves

Two isolation valves are located in a horizontal run of each of the four main steamlines; one valve is inside of the drywell, and the other is near the outside of the primary containment pressure boundary.

The MSIV closing time is equal to or greater than 3 seconds and less than or equal to 4.5 seconds when N₂ or air is admitted to the MSIV actuator. The MSIVs are capable of closing within 3 to 4.5 seconds under differential pressure, fluid flow and temperature conditions. When all the MSIVs are closed, the combined leakage through the MSIVs for all four MSLs is less than or equal to 66.1 liters per minute at standard temperature (20°C) and pressure (one atmosphere absolute pressure) with the differential pressure across the MSIV equal to, or greater than 0.17 MPa.

The MSIV's primary actuation mechanism for opening and closing is pneumatic. Springs close the MSIV if pneumatic pressure to the MSIV actuator is lost.

Feedwater Lines

The FW lines direct feedwater from the CFCAE System to the RPV. The NBS contains only the portion of the FW lines from the seismic interface located upstream of the motor-operated valves (MOVs) to their connections to the RPV. Figure 2.1.2c shows the portion of the FW lines within the NBS.

Isolation of each FW line is accomplished by two containment isolation valves consisting of one check valve inside the drywell and one positive closing check valve outside the containment. The FW line isolation check valves are qualified to withstand a FW line break outside the primary containment. The FW line upstream of the outboard isolation valve contains an MOV and a seismic interface restraint.

Safety/Relief Valves

The safety/relief valves (SRVs) are located on the MSLs between the RPV and the inboard MSIV. These valves protect against overpressurization of the RCPB. Figures 2.1.2a, 2.1.2b and 2.1.2d show the general configuration of the SRVs and the SRV discharge lines.

The rated capacity of the SRVs is sufficient to prevent a rise in pressure within the RPV of more than 110% of the design pressure (9.48 MPaG) for design basis events.

The SRV discharge lines are sized so that critical flow conditions occur through the valve. Each SRV has its own discharge line. The SRV discharge lines terminate at quenchers located below the surface of the suppression pool.

The SRVs provide three main protection functions:

- (1) Overpressure safety operation: The valves function as spring-loaded safety valves and open to prevent RCPB overpressurization. The valves are self-actuated by inlet steam pressure.

The following table identifies the SRV spring set pressures and flow capacities. The opening time for the SRVs, from the time the pressure exceeds the valve set pressure to the time the valve is fully open, is less than or equal to 0.3 seconds.

Set Pressures and Capacities				
SRVs	Number[*] of Valves	Nameplate Spring Set Pressure (MPaG)[†]	ASME Rated Capacity at 103% Spring Set Pressure	Used For ADS
			(kg/h each)[‡]	
J, P	2	7.92	395,000	
B, G, M, S	4	7.99	399,000	
D, E, K, U	4	8.06	402,000	
C, H, N, T	4	8.13	406,000	X
A, F, L, R	4	8.20	409,000	X

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

† Spring set pressure tolerances as permitted by the ASME Boiler and Pressure Vessel Code, Section III.

‡ Minimum capacity per the ASME Boiler and Pressure Vessel Code, Section III.

- (2) Overpressure relief operation: The valves are opened using a pneumatic actuator upon receipt of an automatic or manually initiated signal.

For overpressure relief valve operation (power-actuated mode), reactor vessel pressure sensors generate a high pressure trip signal which is used to initiate opening the SRVs. Valve opening is initiated when an electrical signal is received at the solenoid valve associated with power actuated relief (Figure 2.1.2d). The SRV relief mode opening time from the receipt of signal at the valve actuator to the full ASME lift position is less than or equal to 0.25 seconds when the SRV inlet pressure is at or above 6.89 MPaG.

The SRV pneumatic operator is so arranged that, if it malfunctions, it does not prevent the SRV from opening when steam inlet pressure reaches the spring lift setpoint. Each SRV is provided with its own pneumatic accumulator and inlet check valve for power actuated relief as shown in Figure 2.1.2d.

The SRVs are either DC powered, or powered from uninterruptible AC.

- (3) Automatic depressurization system (ADS) operation: The ADS valves open automatically or manually in the power actuated mode when required during a loss-of-coolant accident (LOCA). Eight of the eighteen SRVs are designated as ADS valves and are capable of operating from either ADS LOCA logic or overpressure relief logic signals. The above table identifies the ADS SRVs.

The ADS accumulator capacity can open the SRV with the drywell pressure at design pressure following failure of the pneumatic supply to the accumulator.

The SRVs can be operated individually in the power-actuated mode by remote manual switches located in the main control room. They are provided with position sensors which provide positive indication of SRV disk/stem position.

Automatic Depressurization System

As shown in Figure 2.1.2f, the NBS channel measurements are provided for the Safety System Logic and Control (SSLC) for signal processing, setpoint comparisons, and generating trip signals. Except for the pump running permissive, the SSLC uses a two-out-of-four voting logic for ADS initiation. The ADS logic is automatically initiated when a low reactor water level signal is present. If the RPV low water level signal is present concurrently with high drywell pressure signal, both the main ADS timer (less than or equal to 29 seconds) and the high drywell pressure bypass timer (less than or equal to 8 minutes) are initiated. Absent a concurrent high drywell pressure signal, only the ADS high drywell pressure bypass timer is initiated. Upon the time out of the ADS high drywell pressure bypass timer, concurrent with RPV low water level signal, the main ADS timer is initiated, if not already initiated. The main timer continues to completion and times out only in the continued presence of an RPV low water level signal. Upon time out of the main ADS timer, concurrent with positive indication by pump discharge pressure of at least one RHR or one HPCF pump running, the ADS function is initiated.

Signals from all four divisions for low reactor water level and high drywell pressure and Division I control logic signal actuate one set of pilots, and sensors from all four divisions for low reactor water and high drywell pressure and Division II control logic signal actuate the second set of pilots, either of which initiates the opening of the ADS SRVs.

ADS initiation is accomplished by redundant trip channels arranged in two divisionally separated logics that control two separate solenoid-operated pneumatic pilots on each ADS SRV. Either pilot can operate the ADS valve. These pilots control the pneumatic pressure

applied by the accumulators and the High Pressure Nitrogen Gas Supply (HPIN) System. The DC power for the logic is obtained from the SSLC Divisions I and II.

For anticipated transient without scram (ATWS) mitigation, the ADS has an automatic and manual inhibit of the automatic ADS initiation. The average power range monitors (APRMs) ATWS permissive signal from the Neutron Monitoring System is combined with the reactor water level signal such that ADS automatic initiation is inhibited unless both power and level are below their setpoints. There are main control room switches for the manual inhibit of automatic initiation of ADS.

The ADS can also be initiated manually. On a manual initiation signal, concurrent with positive indication of at least one RHR or one HPCF pump is running, the ADS function is initiated.

NBS Instrumentation

The NBS contains the instrument lines and instrumentation for monitoring the reactor pressure and water level. For drywell pressure, turbine inlet pressure, main condenser vacuum, and RPV metal temperature, the NBS contains the sensors. Figure 2.1.2e shows the drywell pressure and RPV instrumentation in the NBS.

The mechanical portion of each division of the safety-related NBS instrumentation located in the Reactor Building is physically separated from the other divisions.

The reactor vessel outside surface (metal) temperatures are measured at the head flange and the bottom head locations.

Figure 2.1.2e shows the water level instrumentation. The instruments that sense the water level are differential pressure devices calibrated for specific RPV pressure and temperature conditions. Instrument zero for the RPV water level ranges is the top of the active fuel. The RPV water level instrumentation considers the effects of dissolved non-condensable gasses in the RPV water level instrumentation lines.

With the exception of turbine inlet pressure sensor and main condenser vacuum sensor located in the Turbine Building, the NBS instrumentation is located in the drywell, the steam tunnel and the Reactor Building.

Other Provisions

The NBS equipment identified as safety-related is classified as Seismic Category I except for the American Society of Mechanical Engineers (ASME) Class 3 equipment shown on Figure 2.1.2c. The non-safety-related section of the feedwater lines between the seismic interface restraint and the motor-operated valves shown in Figure 2.1.2c is classified as Seismic Category I. The MSL drain lines from the MSLs to the Main Condenser are seismically analyzed to withstand the Safe Shutdown Earthquake (SSE).

Figures 2.1.2a, 2.1.2b, 2.1.2c, 2.1.2d and 2.1.2e show the ASME Boiler and Pressure Vessel Code classes.

The divisional equipment in the NBS is powered from its respective Class 1E divisions as shown in Figures 2.1.2b, 2.1.2d, and 2.1.2e. In the NBS, independence is provided between Class 1E divisions, and also between Class 1E divisions and non-Class 1E equipment.

The NBS has the following displays and controls in the main control room:

- (1) Parameter displays for the instruments shown on Figures 2.1.2b and 2.1.2e. This includes the reactor vessel pressure, reactor vessel water level, drywell pressure, main condenser vacuum, and turbine inlet pressure.
- (2) Controls and status indication for the active safety-related components shown on Figures 2.1.2b, 2.1.2c (excluding the inboard FW line check valves, and the ASME Boiler and Pressure Vessel Code Class 2 check valves), and 2.1.2d.
- (3) Manual system level initiation capability for the ADS.
- (4) Manual capability to inhibit automatic initiation of the ADS.

NBS components with displays and control interfaces with the Remote Shutdown System (RSS) are shown on Figures 2.1.2a and 2.1.2e.

The safety-related electrical equipment (including instrumentation and controls) shown on Figures 2.1.2b, 2.1.2c, 2.1.2d, and 2.1.2e located in the containment, steam tunnel and Reactor Building, is qualified for a harsh environment.

The MOVs shown on Figure 2.1.2b (except for the ASME Boiler and Pressure Vessel Code Class 2 MOV) have an active safety-related function to close, and perform this function under differential pressure, fluid flow, and temperature conditions.

The check valves (CVs) shown on Figures 2.1.2c and 2.1.2d (ADS pneumatic CVs only) have the safety--related functions to open, close, or both open and close under system pressure, fluid flow, and temperature conditions.

Inspections, Tests, Analyses and Acceptance Criteria

Table 2.1.2 provides a definition of the inspections, tests and/or analyses, together with associated acceptance criteria, which will be undertaken for the NBS.



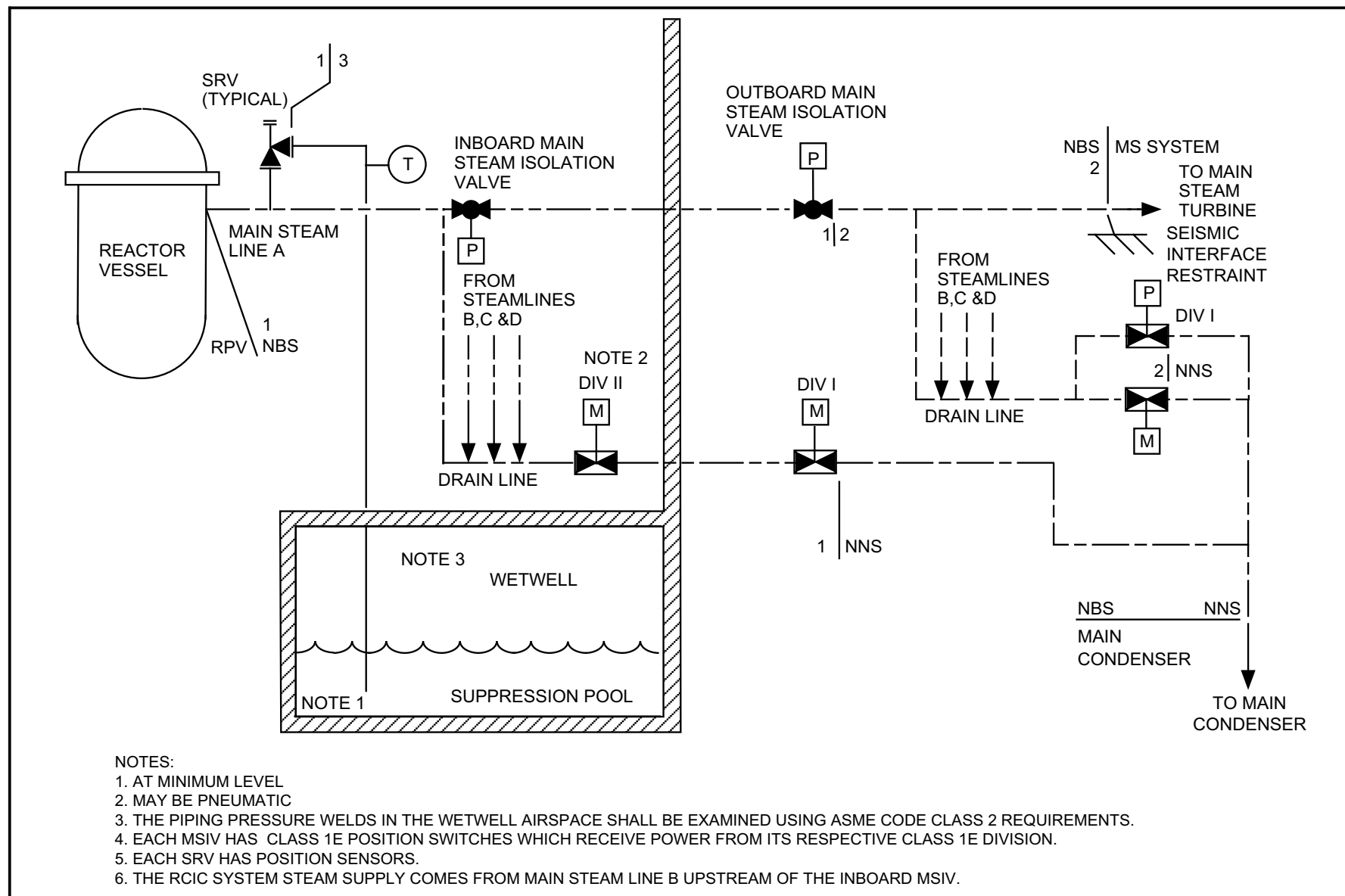


Figure 2.1.2b NBS Steamline

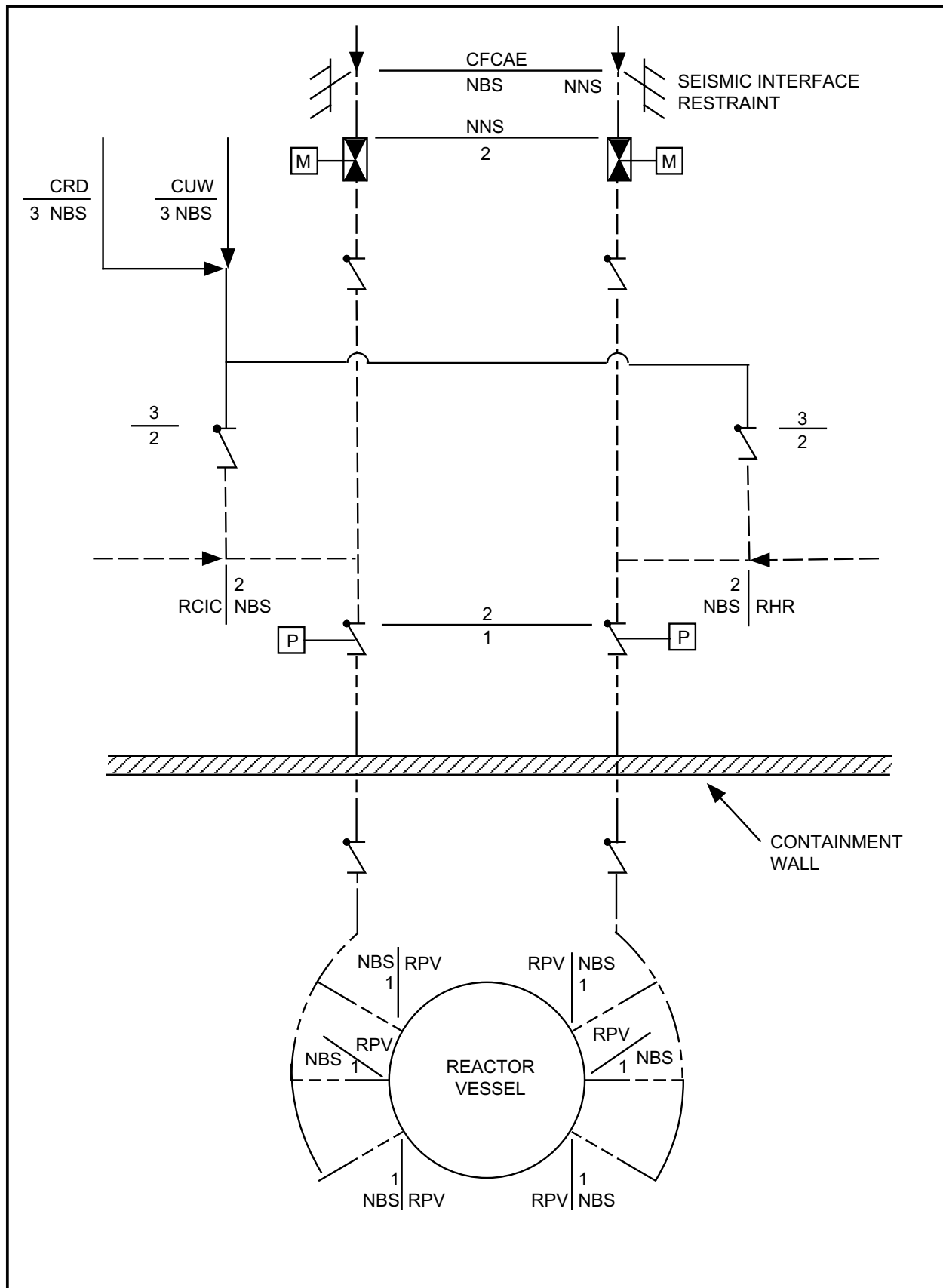


Figure 2.1.2c NBS Feedwater Line

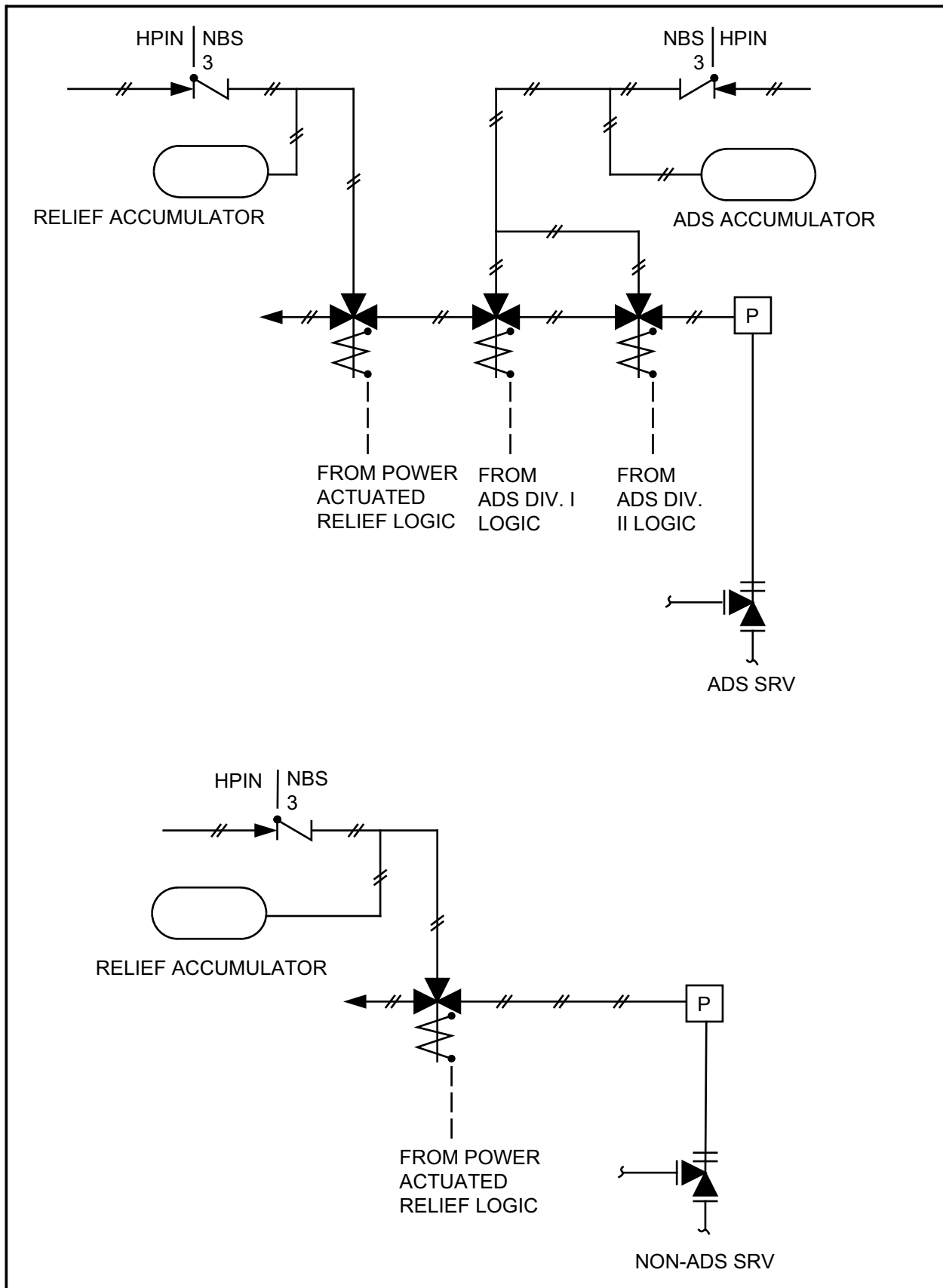


Figure 2.1.2d NBS Safety/Relief Valve Pneumatic Lines

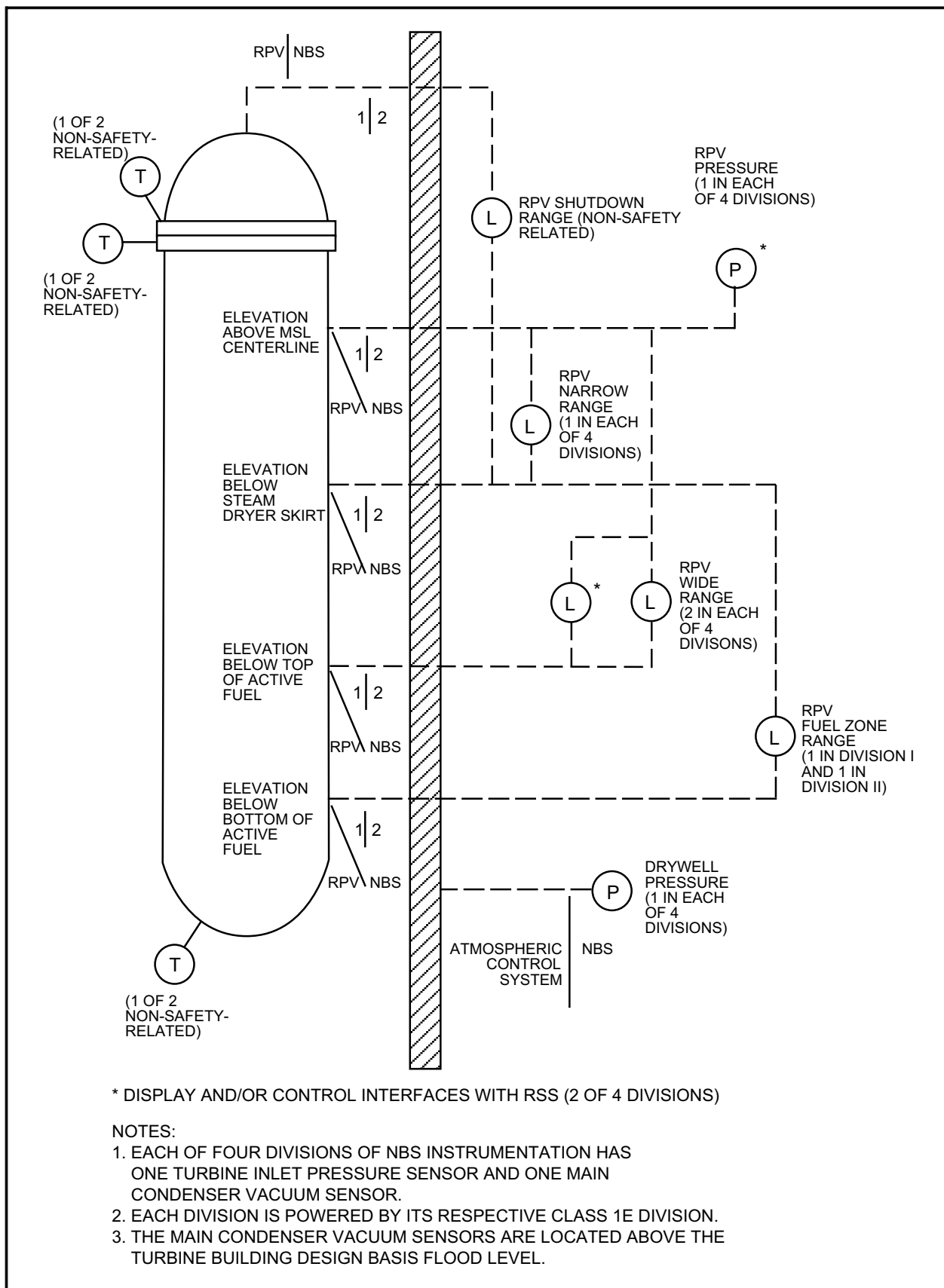


Figure 2.1.2e NBS Drywell Pressure and Reactor Vessel Instrumentation

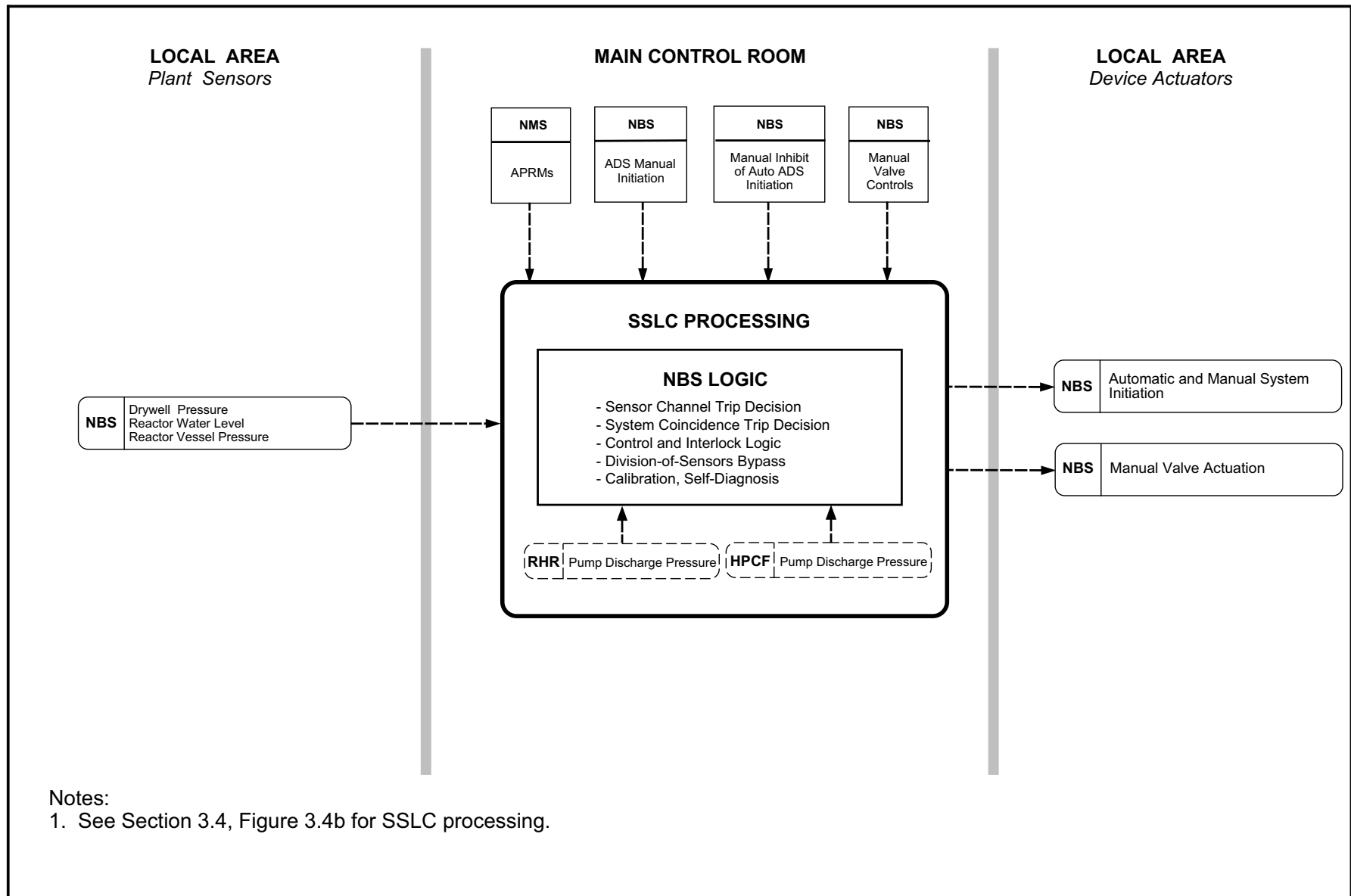


Figure 2.1.2f Nuclear Boiler System Control Interface Diagram

Table 2.1.2 Nuclear Boiler System

Inspections, Tests, Analyses and Acceptance Criteria		
Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The basic configuration of the NBS is shown in Figures 2.1.2a, 2.1.2b, 2.1.2c, 2.1.2d, 2.1.2e, and 2.1.2f.	1. Inspections will be conducted for the NBS System.	1. The as-built NBS conforms with the basic configuration shown in Figures 2.1.2a, 2.1.2b, 2.1.2c, 2.1.2d, 2.1.2e, and 2.1.2f.
2. The ASME Code components of the NBS System retain their pressure boundary integrity under internal pressures that will be experienced during service.	2. A hydrostatic test will be conducted on those Code components of the NBS required to be hydrostatically tested by the ASME Code.	2. The results of the hydrostatic test of the ASME Code components of the NBS conform with the requirements in the ASME Code, Section III
3. The combined volume of the four main steamlines (MSLs) and branch lines from the RPV to the main steam turbine stop valves and turbine bypass valves is greater than or equal to 113.2 m ³ .	3. Analyses will be performed using as-built dimensions of the steamlines to determine the combined steamline volume.	3. The combined steamline volume is greater than or equal to 113.2 m ³ .
4. The throat diameter of each MSL flow limiter is less than or equal to 355 mm.	4. Inspections of the as-built MSL flow limiters will be conducted.	4. The throat diameter of each MSL flow limiter is less than or equal to 355 mm.
5. The pneumatic-operated valve in the MSL drain line shown in Figure 2.1.2b opens if either electric power to the valve actuating solenoid is lost, or pneumatic pressure to the valve is lost.	5. Tests will be conducted on the as-built MSL drain valve.	5. The MSL pneumatic drain line valve shown in Figure 2.1.2b opens when either electric power to the valve actuating solenoid is lost, or pneumatic pressure to the valve is lost.
6. MSIV closing time is equal to or greater than 3 seconds and less than or equal to 4.5 seconds when N ₂ or air is admitted into the MSIV actuator. The MSIVs are capable of closing within 3 to 4.5 seconds under differential pressure, fluid flow and temperature conditions.	6. <ul style="list-style-type: none"> a. Tests of the as-built MSIV will be conducted under preoperational differential pressure, fluid flow, and temperature conditions. b. Tests, or type tests, of an MSIV will be conducted under design basis differential pressure, fluid flow and temperature conditions. 	6. The MSIV closing time is equal to or greater than 3 and less than or equal to 4.5 seconds.

Table 2.1.2 Nuclear Boiler System (Continued)

Inspections, Tests, Analyses and Acceptance Criteria		
Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
7. When all MSIVs are closed, the combined leakage through the MSIVs for all four MSLs is less than or equal to 66.1 liters per minute at standard temperature (20°C) and pressure (one atmosphere absolute pressure) with the differential pressure across the MSIV equal to, or greater than, 0.17 MPa.	7. Test and analysis will be conducted on the as-built MSIVs to determine the leakage.	7. When all MSIVs are closed, the combined leakage through the MSIVs for all four MSLs is less than or equal to 66.1 liters per minute at standard temperature (20°C) and pressure (one atmosphere absolute pressure) with the differential pressure across the MSIV equal to, or greater than, 0.17 MPa.
8. Springs close the MSIV if pneumatic pressure to the MSIV actuator is lost.	8. Tests will be conducted on the as-built MSIV.	8. The MSIV closes when pneumatic pressure is removed from the MSIV actuator.
9.	9.	9.
a. The SRV spring set pressure and flow capacities are given in Section 2.1.2. The opening time for the SRVs from the time the pressure exceeds the valve set pressure to the time the valve is fully open, is less than or equal to 0.3 seconds.	a. Analysis and tests (at a test facility) will be conducted in accordance with the ASME Code.	a. The SRVs have the capacities and set pressures shown on Section 2.1.2. The opening time for the SRVs from the time the pressure exceeds the valve set pressure to the time the valve is fully open is less than or equal to 0.3 seconds.
b. The SRV relief mode opening time from the receipt of signal at the valve actuator to the full ASME lift position is less than or equal to 0.25 seconds when the SRV inlet pressure is at or above 6.89 MPaG.	b. Tests of the SRVs will be conducted at a test facility.	b. The SRV relief mode opening time from the receipt of signal at the valve actuator to the full ASME lift position is less than or equal to 0.25 seconds when the SRV inlet pressure is at or above 6.89 MPaG.

Table 2.1.2 Nuclear Boiler System (Continued)

Inspections, Tests, Analyses and Acceptance Criteria		
Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
10. The ADS accumulator can open the SRV with the drywell pressure at design pressure following failure of the pneumatic supply to the accumulator.	10. An analysis and/or type test will be performed to demonstrate the capacity of the SRV ADS accumulators.	10. Either: a. The SRV ADS accumulators have the capacity to lift the stem of the SRVs to the full open position one time with the drywell pressure at, or above the drywell design pressure, or b. The SRV ADS accumulators have the capacity to lift the stem of the SRVs to the full open position five times with the drywell at atmospheric pressure, and an analysis that shows that five SRV lifts at atmospheric pressure demonstrates the capability to open one time with the drywell at the drywell design pressure.
11. For overpressure relief valve operation, reactor vessel pressure sensors generate a high pressure trip signal which is used to initiate opening of the SRVs.	11. Tests will be conducted on the power actuated relief logic using simulated input signal to cause trip conditions.	11. The valve solenoid receives an initiation signal.

Table 2.1.2 Nuclear Boiler System (Continued)

Inspections, Tests, Analyses and Acceptance Criteria		
Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
12. The ADS logic is automatically initiated when a low reactor water level signal is present.	12. Tests will be conducted using simulated input signals for each NBS process variable to cause trip conditions in two, three, and four instrument channels of the same process variable associated with each of the two ADS logic divisions.	12. a. Upon receipt of a low water level signal, concurrent with a high drywell pressure signal, at the input to the ADS initiation logic, the following occurs: (1) The main ADS timer initiates and continues to time out in the continued presence of the RPV low water level signal. The time delay for the main ADS timer is less than or equal to 29 seconds. (2) Upon time out of the main ADS timer, a concurrent signal that represents positive indication of at least one RHR or HPCF pump running, an ADS actuation signal is generated to the associated ADS valve solenoids.

Table 2.1.2 Nuclear Boiler System (Continued)

Inspections, Tests, Analyses and Acceptance Criteria		
Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
12. (Continued)	12. (Continued)	<p>b. Upon receipt of a low water level signal, in the absence of a high drywell pressure signal, at the input to the ADS initiation logic, the following occurs:</p> <p>(1) The ADS high drywell pressure bypass timer initiates. The time delay for the ADS high drywell pressure bypass timer is less than or equal to 8 minutes.</p> <p>(2) Upon time out of the ADS high drywell pressure bypass timer, concurrent with an RPV low water level signal, the main ADS timer initiates and continues to time out in the continued presence of the RPV low water level signal.</p> <p>(3) Upon time out of the main ADS timer, concurrent with a pump discharge pressure signal that represents positive indication of at least one RHR or HPCF pump running, an ADS actuation signal is generated to the associated ADS valve solenoids.</p>

Table 2.1.2 Nuclear Boiler System (Continued)

Inspections, Tests, Analyses and Acceptance Criteria		
Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
13. For ATWS mitigation, the ADS has an automatic and manual inhibit of the automatic ADS initiation.	13. a. The tests defined in item 12a will be conducted with a simulated APRM ATWS permissive signal present. b. The test defined in 12a will be conducted with the ADS manual inhibit device set to inhibit.	13. a. ADS actuation does not occur. b. ADS actuation does not occur.
14. The ADS can be initiated manually.	14. Tests will be conducted by initiating each ADS division manually, concurrent with a simulated RHR or HPCF pump running signal.	14. Upon receipt of a manual initiation signal, an ADS actuation signal is generated to the associated ADS valve solenoids.
15. The RPV water level instrumentation considers the effects of dissolved non-condensable gasses in the RPV water instrument lines.	15. Analyses of the as-built RPV water level instrumentation will be performed using available test data and/or operating experience.	15. An analysis output exists which concludes that the RPV water level instrumentation considers the effects of dissolved non-condensable gasses in the RPV water level instrument lines.
16. The mechanical portion of each division of the safety-related NBS instrumentation located in the Reactor Building is physically separated from the other divisions.	16. Inspections of the as-built NBS instrumentation will be conducted.	16. The mechanical portion of each NBS instrumentation division is physically separated from the other divisions by structural and/or fire barriers.
17. The MSL drain lines from the MSLs to the main condenser are seismically analyzed to withstand the SSE.	17. An inspection of the stress report containing the dynamic analysis of the piping will be conducted.	17. A stress report exists. This report documents that a dynamic seismic analysis has been performed.

Table 2.1.2 Nuclear Boiler System (Continued)

Inspections, Tests, Analyses and Acceptance Criteria		
Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
18. The divisional equipment in the NBS is powered from its respective Class 1E divisions as shown in Figures 2.1.2b, 2.1.2d and 2.1.2e. In the NBS, independence is provided between Class 1E divisions, and between Class 1E divisions and non-Class 1E equipment.	18. a. Tests will be performed in the NBS by providing a test signal in only one Class 1E division at a time. b. Inspection of the as-installed Class 1E divisions in the NBS will be performed.	18. a. The test signal exists only in the Class 1E division under test in the NBS. b. Physical separation or electrical isolation exists between Class 1E divisions in the NBS. Physical separation or electrical isolation exists between Class 1E divisions and non-Class 1E equipment.
19. Main control room displays and controls provided for the NBS are as defined in Section 2.1.2.	19. Inspections will be performed on the main control room displays and controls for the NBS.	19. Displays and controls exist or can be retrieved in the main control room as defined in Section 2.1.2.
20. RSS displays and controls provided for the NBS are defined in Section 2.1.2.	20. Inspections will be performed on the RSS displays and controls for the NBS.	20. Displays and controls exist on the RSS as defined in Section 2.1.2.
21. MOVs designated in Section 2.1.2 as having an active safety function will close under differential pressure, fluid flow, and temperature conditions.	21. Tests of installed valves for closing will be conducted under preoperational differential pressure, fluid flow, and temperature conditions.	21. Upon receipt of an actuating signal, each MOV closes.
22. The CVs designated in Section 2.1.2 as having an active safety-related function open, close, or both open and closes, under system pressure, fluid flow, and temperature conditions.	22. Tests of the installed valves for opening, closing, or both opening and closing, will be conducted under system preoperational pressure, fluid flow, and temperature conditions.	22. Based on the direction of the differential pressure across the valve, each CV opens, closes, or both opens and closes, depending upon the valve's safety function.

2.1.3 Reactor Recirculation System

Design Description

The Reactor Recirculation System (RRS) is an arrangement of 10 variable speed reactor internal pumps (RIP) with motors mounted in the bottom of the RPV. The RRS circulates coolant through the reactor core at variable flow rates. The motor cooling heat exchangers are located inside the RPV pedestal adjacent to the RIP motors. Figure 2.1.3 shows the basic system configuration and scope.

Individual RIPs and motors provide at least 6912 m³/h flow with a total developed head (TDH) of at least 32.6m with 10 RIPs operating and 8291 m³/h with a TDH of at least 35.8m with 9 RIPs operating, with water at 278°C and 7.25 MPa or less. The individual RIPs, and motors have a dry rotating inertia of not less than 17.5 kg·m² and not more than 26.5 kg·m².

Figure 2.1.3 shows the ASME Code class for the RRS piping and components. The motor cover and its nuts and bolts are classified as safety-related, Seismic Category I, ASME Code Class 1 components.

The RIP motor cooling is provided by an auxiliary impeller mounted on the bottom of the motor rotor, which circulates water through the RIP motor and its cooling heat exchanger. The heat exchangers are cooled by the Reactor Building Cooling Water System (RCW).

Each RIP includes an anti-rotation-device (ARD) which prevents reverse RIP motor rotation by reverse flow-induced torque of equal to or less than 7.55 kN·m when there is no motor power.

RIP maintenance during reactor shutdown requires a temporary plug to be installed in the RIP diffuser when the RIP impeller, shaft and motor are temporarily removed. The temporary RIP diffuser plug cannot be removed unless the RIP motor housing bottom cover is in place.

Inspections, Tests, Analyses and Acceptance Criteria

Table 2.1.3 provides a definition of the inspections, tests, and/or analyses, together with associated acceptance criteria, which will be undertaken for the RRS.

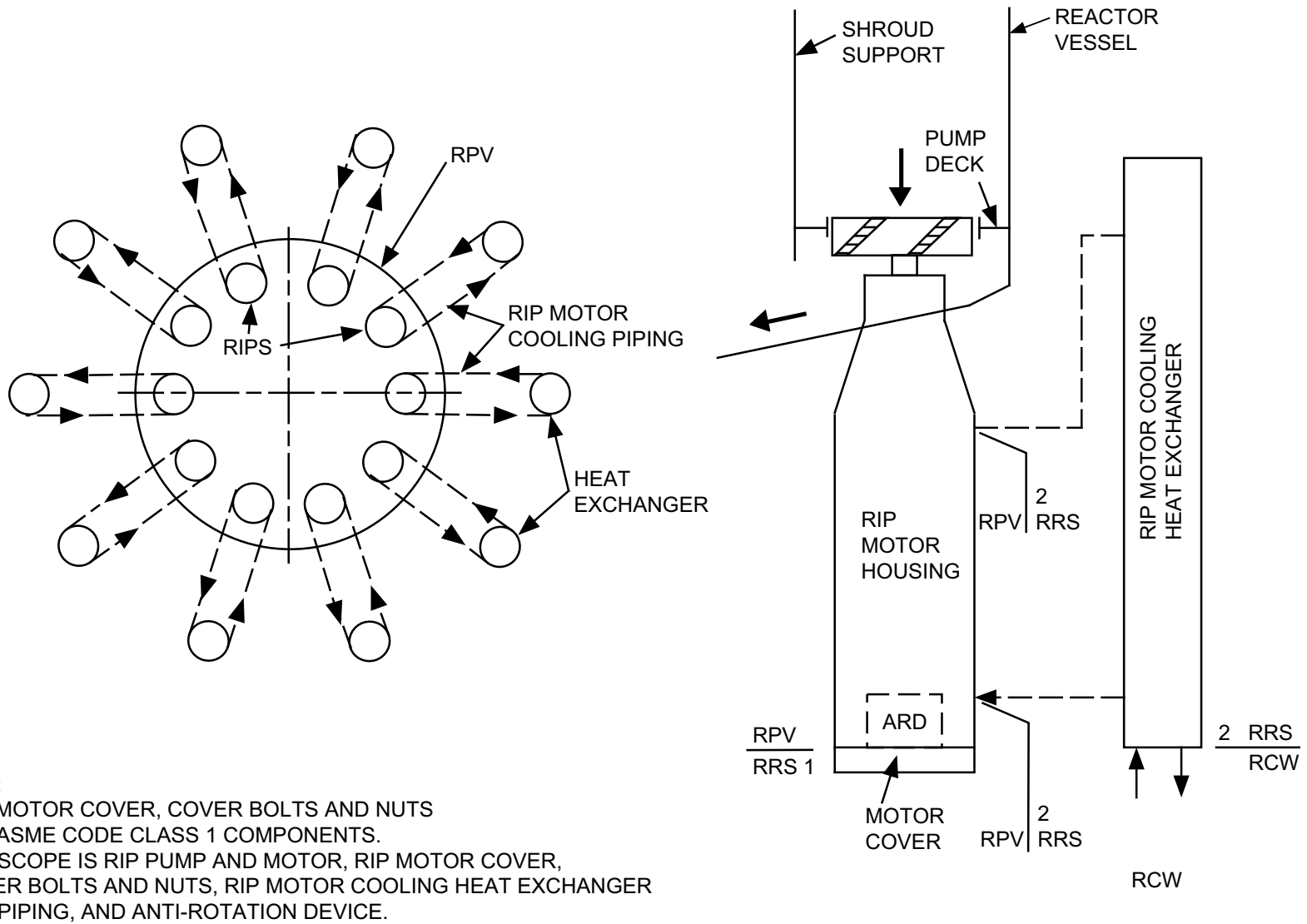


Figure 2.1.3 Reactor Recirculation System

Table 2.1.3 Reactor Recirculation System

Inspections, Tests, Analyses and Acceptance Criteria		
Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The basic configuration of the RRS is shown on Figure 2.1.3.	1. Inspections of the as-built system will be conducted.	1. The as-built RRS conforms with the basic configuration shown in Figure 2.1.3.
2. The ASME components of the RRS retain their pressure integrity under internal pressures that will be experienced during service.	2. A hydrostatic test will be conducted on those Code components of the RRS required to be hydrostatically tested by the ASME Code.	2. The results of the hydrostatic test of the ASME components of the RRS conform with the requirements in the ASME Code, Section III.
3. Individual RIPs and motors provide at least 6912 m ³ /h flow with a total developed head (TDH) of at least 32.6m with water at least 278°C and 7.25 MPa or less, during 10 RIPs operation. During 9 RIPs operation, the individual RIP provides at least 8291 m ³ /h with a TDH of at least 35.8m at the same temperature and pressure conditions.	3. Tests will be conducted on the individual RIP in a test facility which includes a calibrated flow element and a RIP section which is geometrically the same as the RPV bottom plenum region including the RIP differential pressure measurement taps. The RIP performance data will be obtained for rated reactor conditions and minimum to rated RIP speed.	3. Individual RIPs and motors provide at least 6912 m ³ /h flow with a total developed head (TDH) of at least 32.6m with water at least 278°C and 7.25 MPa or less, during 10 RIPs operation. During 9 RIPs operation, the individual RIP provides at least 8291 m ³ /h with a TDH of at least 35.8m at the same temperature and pressure conditions.
4. The individual RIPs and motors have a dry rotating inertia of ≥ 17.5 and ≤ 26.5 kg·m ² .	4. Tests will be conducted on a RIP and motor rotating assembly in a test facility.	4. RIP and motor dry rotating inertia is ≥ 17.5 and ≤ 26.5 kg·m ² .
5. Each RIP includes an ARD which prevents reverse RIP motor rotation by reverse flow induced torque of ≤ 7.55 kN·m when there is no motor power.	5. Tests will be conducted on each ARD in a test facility.	5. Each ARD prevents RIP motor rotation in the reverse direction with a reverse torque of ≤ 7.55 kN·m.
6. The temporary RIP diffuser plug cannot be removed unless the RIP motor housing bottom cover is in place.	6. Tests of a RIP diffuser plug will be conducted in a test facility by simulating conditions associated with plug removal with the motor housing bottom cover removed.	6. The temporary RIP diffuser plug cannot be removed unless the RIP motor housing bottom cover is in place.